NUCLEAR TRAINING COURSE

COURSE 133

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Reactor Boiler and Auxiliaries - Course 133

THE FUNCTION OF A REACTOR

In a nuclear electric generating station heat energy is produced by the fissioning of nuclei, such as those of uranium, in a reactor. Thus the source of heat energy is the reactor which is, therefore, equivalent to the furnace in a fossil fired plant. As in thermal electric generating station, it is necessary to transport this energy from where it is produced to the turbine where it is changed to mechanical energy of rotation.

Consideration will be given, in this series of lessons, to the principles involved in the design of the reactor itself and the systems, such as the moderator and heat transport systems, which are associated with the reactor. Separate consideration will be given to reactor fuel, fuel handling, fuel changing, fuel storage and fuel transfer.

Review of Reactor Theory

Energy is produced in a reactor as the result of the fissioning of the nucleus of a fissile material such as Uranium-235, Uranium-238, Uranium-233 or Plutonium-239. The neutrons produced at fission are used to cause further fission and a chain reaction is thus sustained. When the chain reaction is just being maintained, as in Figure 1, the reactor is just critical, the neutron multiplication factor, $k = 1.0$ and the reactivity, $\delta k = 0$. Under these conditions the reactor power is steady.

To raise reactor power the neutron losses by leakage or radiative capture are reduced and more than one fission neutron is used to produce further fissions so that neutron multiplication occurs. k is, therefore, greater than 1.0 and the reactivity is positive. To reduce power k is reduced below 1.0 by increasing neutron losses so that δk is negative. k is only greater or less than 1.0 while the power is being increased or decreased respectively. When the desired power level is reached, k is then maintained equal to 1.0 to keep the power steady.

Fissioning of U-238 will only occur with neutron energies greater than 1.2 Mev and so U-238 is not considered as a nuclear fuel, although U-238 fissions do contribute to the power produced.

However, radiative capture of neutrons of U-238 forms U-239 which decays to Pu-239, a nuclear fuel. The nuclear fuel U-233 is formed by neutron capture in Thorium-232 and the subsequent decay of the Th-233 formed. U-238 and Th-232 are fertile materials.

Fissioning of U-233, U-235 and Pu-239 will occur with neutrons of all energies but the probability of fission occuring increases as the neutron energy decreases. This increase of fission cross-section with decrease in neutron energy occurs because the slower neutron remains in the vicinity of the nucleus for a longer period of time. Neutrons released at fission have an average energy of 2 Mev whereas the neutrons most likely to cause fission would be thermal neutrons with energies comparable to those of molecules (approximately 0.025 ev). It would, therefore, be more efficient if the fission neutrons were slowed down or moderated before being allowed to cause further fissions.

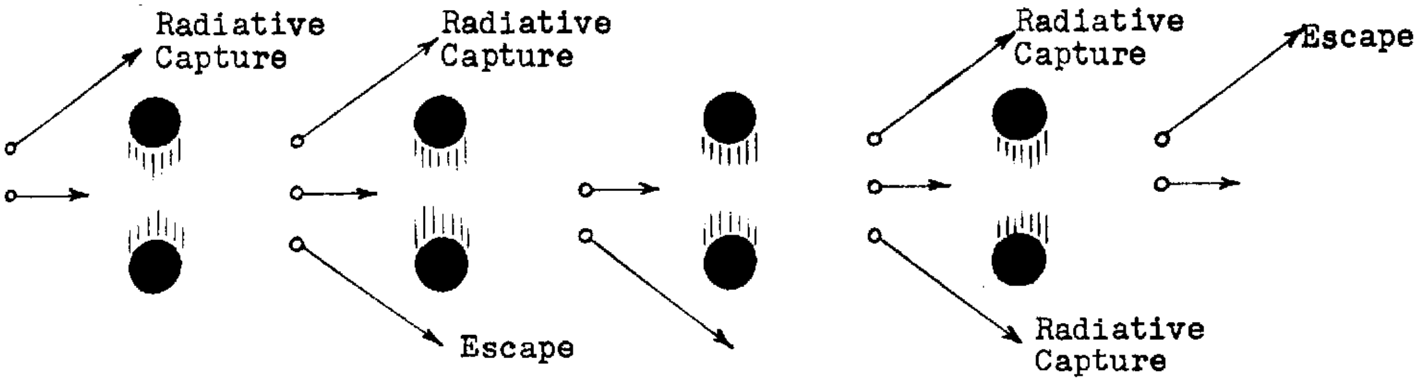


Fig. 1

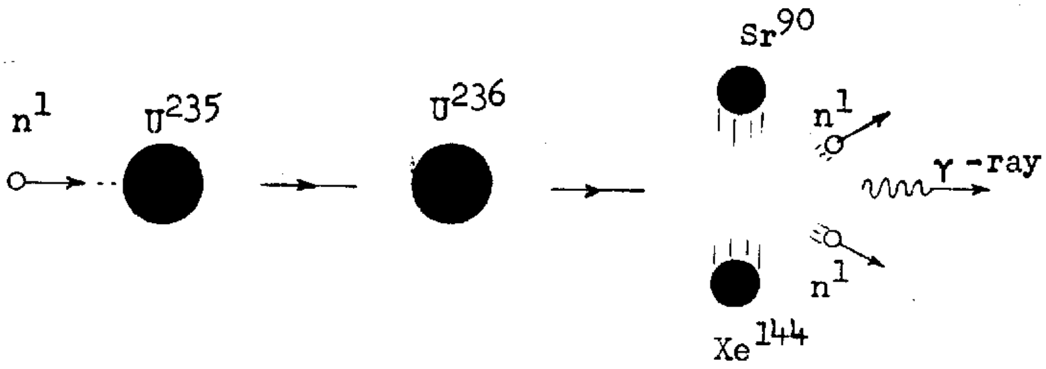


Fig. 2

As shown in Fig. 2, when fission occurs the nucleus divides into two new nuclei called fission products. These are usually unstable and decay into other nuclei, also classified as fission products. If not controlled these fission products could become a radioactive hazard to station personnel. They also absorb neutrons to a greater or lesser degree and are, therefore, known as poisons. The worst poisons that have to be considered are Xenon-135 and Samarium-149.

The Four Factor Formula

A clearer understanding of reactor design requirements and reactor classifications is possible if the factors on which k depends are considered.

k, the neutron multiplication factor is defined by:-

$$k = \frac{\text{Number of neutrons causing fission in any one generation}}{\text{Number of neutrons causing fission in the previous generation}}$$

and the reactivity $\delta k = k-1$.

Now

$$\underline{k = \eta \epsilon pf - \text{neutron leakage}}$$

this relationship being known as the four factor formula.

The neutron leakage depends on the reactor size and shape. The other terms are defined as follows:-

(a) $\underline{\eta}$ represents the number of fast neutrons produced by fission from each thermal neutron <u>captured</u> in the fuel.

It must be remembered that every neutron captured in the fuel does not cause fission since radiative capture is also possible. So η is not the number of neutrons per fission.

If γ is the number of neutrons per fission, then:-

$$\eta = \gamma \times \frac{(\text{neutron captures causing fission})}{(\text{total neutron captures in the fuel})}$$

In pure U-235 for example $\gamma = 2.48$ for thermal neutrons and the fission cross section = 577 barns whereas the radiative capture cross section is 106 barns i.e 577 out of (577+106) neutrons captured in U-235 cause fission.

$$\text{Hence } \eta = 2.48 \times \frac{577}{683} = 2.1 \text{ for thermal neutrons}$$

$$\eta = 2.46 \text{ for fast neutrons}$$

For natural uranium, containing only 0.72% U-235, η = 1.32 for thermal neutrons.

(b) ϵ , the Fast Fission Factor, is the ratio by which the fast neutron production is increased because of fast fissions in U-238. The value of ϵ is not normally more than 1.02 or 1.03 and it will, therefore, be assumed to be 1.0.

(c) p, the Resonance Escape Probability, is the fraction of neutrons which avoid resonance capture in U-238 while they are being slowed down or thermalized. If the neutrons are not thermalized p=1.

(d) The Thermal Utilization Factor, f, is defined as the ratio of neutrons absorbed in the fuel to the total neutrons absorbed in the reactor (i.e in fuel, sheath, structural material, heat transport fluid and moderator if present).

Basic Reactor Design Considerations

The fundamental purpose of a reactor is to continually produce heat energy by fission. It must, therefore, contain fissionable material or fuel. Sufficient fuel must be present so that a chain reaction is sustained or that the reactor is critical. The minimum requirement for this criticality is that k=1.0 or that:-

$\eta \epsilon p f$ - neutron leakage = 1.0

A further objective is to obtain as much energy as possible from the fuel i.e to obtain as high a burn-up as possible. If just sufficient fuel is provided to make k=1.0, then as soon as a little fissile material is used k would become less than 1.0 and the reactor would become subcritical. Thus excess reactivity must be built into the reactor, by providing more fissile material in order to allow for depletion of fuel as it is used up.

As fuel is fissioned, fission product poisons accumulate which absorb neutrons. This decreases either η or f or both and this again reduces k. Hence enough excess reactivity must be available, in the reactor, to compensate for the poison build-up. Some mthod of reactor control must be available to maintain k = 1.0 during steady power operation, despite the excess reactivity built into the reactor and to make k greater or less than 1.0 when increase or decrease in power is required.

Suitable choice of reactor material can reduce non-fission neutron captures, increase the value of f and decreases the amount of fuel required.

A third objective which requires consideration, is the necessity of transporting the heat out of the reactor in a useful form, and therefore, at as high a temperature as possible. A heat transport system must be associated with the reactor which will transport the heat, produced in the fuel, to the boiler or turbine. As will be seen later, there are limitations on the temperature of the heat transport system.

If it is decided to thermalize the fission neutrons in order to increase the fission cross-section, a moderator must be provided, in the reactor, for this purpose. Finally the value of k could be increased by decreasing neutron leakage out of the reactor. A reflector may be used in order to decrease this leakage by reflecting neutrons back into the reactor.

Conversion and Breeding

It has been estimated that roughly 8 x 10^{19} Btu of energy are available from the known coal, oil and gas reserves in the world which can be extracted at no more than twice the present cost. At the estimated rate of consumption, these reserves would be exhausted in about 100 years.

If the U-235 fuel alone is utilized in natural uranium to produce energy an estimated 1.2 x 10^{19} Btu are available. However if all the U-238 fissile material could be used to produce Pu-239 and the Pu-239 used to produce energy by fission, then the available energy resources are estimated at 170 x 10^{19} Btu. A further 7 x 10^{19} Btu would be available if all the Th-232 could be used to produce fissile U-233.

Even if these estimates are only accurate within a factor of 2, there appears to be no immediate urgency to produce a reactor system that could utilize U-238 and Th-232 for energy production. The present requirement is the production of economic power by nuclear fission. However, unless other sources of energy are discovered, serious consideration will have to be given, in the forseeable future, to better utilization of available resources.

In nuclear physics, a reactor, in which the fissile material produced from the fertile material is the same as the fissile material consumed, is defined as a breeder reactor. For example if U-233 is used as a fuel and U-233 is also produced from Th-232, then we have a breeder. In a natural uranium reactor, U-235 is used as a fuel and Pu-239 is produced from U-238. This is a converter system. Frequently the term breeder is used loosely to describe both systems.

The effectiveness of a reactor as a breeder or converter is measured by the CONVERSION FACTOR, c. This is defined as the <u>net</u> production of fissile atoms per fissile atom consumed. For there to be a net gain of fissile material which could be extracted and used in another reactor, the conversion factor must be at least equal to 1.0.

An alternative definition of breeder and converter systems can be made on the basis of this conversion factor: -

A <u>converter</u> is a reactor system in which c is less than 1.0

A <u>breeder</u> is a reactor system in which c is greater than or equal to 1.0

Now η is the number of neutrons produced per neutron absorbed in the fuel and 1.0 of these neutrons must be available to cause further fissions and maintain the chain reaction. So, the maximum possible number of neutrons available for breeding or conversion is $\eta - 1$ and this disregards neutron losses by leakage and absorption in reactor material. If w is the number of neutrons lost from the system, per fission, by leakage or absorption,

$$c = \eta - w - 1$$

For good conversion η must be as large as possible and w kept as small as possible. The following table lists the value of η for fast and thermal neutrons for the three fissile materials.

	U-235	U-233	Pu-239
η for fast neutrons	2.46	2.54	2.88
η for thermal neutrons	2.08	2.31	2.03

It may be seen that, in thermal reactors using U-235 or Pu-239 as fuel, $\eta - 1$ is only just greater than 1.0 and, therefore, net breeding or conversion cannot be achieved. It is however theoretically possible in a thermal reactor using U-233 fuel and Th-232 as the fertile material.

In a fast reactor, on the other hand, breeding or conversion is possible with all three fissile materials and is particularly attractive with Pu-239.

Other quantities of interest when considering breeding possibilities are:-

The BREEDING RATIO defined as the ratio of the number of fissile atoms produced to the number of the <u>same kind</u> consumed.

The BREEDING GAIN, G, which is the number of fissile atoms gained for each one consumed, ie, the excess of the breeding ratio over unity.

The DOUBLING TIME defined as the time taken for the surplus fissile material produced to equal the total quantity in the fuel cycle.

A short doubling time means a rapid increase in the production of surplus fissile material, which is achieved by having a high breeding gain.

The conversion factor or breeding ratio also affects the fuel burnup. If the factor c is high many fissile atoms are produced while fissile atoms are consumed. These new fissile atoms could, in turn, be fissioned to produce energy and, thus, extend the life of the fuel. The higher the value of c the greater the utilization of fertile atoms.

The maximum or ultimate burnup which can be achieved in the fuel is given by:-

$$\text{U.B.} = \frac{100}{1-c} \text{ \% original fissile atoms}$$

If $c = 0.8$ U.B. = 500% original fissile atoms = 35,700 MWD/ton

which means that the number of fertile atoms utilized is four times the number of fissile atoms present.

If $c = 1.0$ all the fertile atoms will be used up. However, there are other factors which limit the fuel burnup and, so, any net fissile material produced would be extracted if c was high enough to make this worth while.

ASSIGNMENT

1. What is the minimum condition for reactor criticality and what are the values of k and δk under this condition?

2. (a) Define the quantities η p and f in the four factor formula.
   (b) Express the minimum condition of reactor criticality in terms of the four factor formula.

3. Give two reasons for the fact that excess reactivity must be built into a reactor.

4. What non-nuclear objective requires consideration during the design of the reactor-boiler system?

5. What factor will necessitate the use of breeder and converter reactors in the future?

6. Explain the difference between a breeder and a converter.

7. Which reactor systems offer the best possibility of net conversion and breeding? Explain your answer.

8. How does the conversion factor or breeding ratio affect the fuel burnup?

A. Williams

Reactor Boiler and Auxiliaries - Course 133

REACTOR CLASSIFICATIONS - FAST & THERMAL REACTORS

Development of nuclear power in various countries has depended on a variety of factors not the least of which is the availability of either water resources or fossil fuel for the production of power economically. No country has unlimited water resources that can be developed and the transmission cost of hydro electric power from the point of generation to the load centre will continue to increase and become a major factor in the overall cost of power. Fossil fuel resources are also limited and the costs of production and transportation of such fuel are continually increasing as the availability decreases. There seems little doubt that nuclear sources of energy will have to be extensively utilized for power production. More engineering design and development is required to demonstrate that nuclear electric generation is really competetive with fossil electric generation.

It would be ideal if a number of reactor concepts could be developed ahead of time so that the most suitable or most economic concept could be chosen when the need arises. However the cost of experimental facilities, and the cost of scientific, technological and industrial development prohibit development on a broad front except in such countries as the United States. Most countries have concentrated on developing one, or perhaps two, reactor concepts.

It was seen, in the previous lesson, that there are many variables involved and, consequently, many types of reactors have been or are being developed. They may be classified in a number of ways and the following classifications will be considered in these lessons:-

(a) Classification on the basis of the energy of the neutrons causing fission.

(b) Classification according to core structure,ie, homogeneous or heterogeneous.

(c) Classification on the basis of the moderator used.

(d) Classification on the basis of the heat transport fluid used.

The classifications are by no means as clear cut as the list above would suggest, since there are some factors common to some or all of the classifications. The discussion will be confined to power reactors as distinct from experimental facilities.

Classification of Power Reactors on the Basis of Neutron Energy

Each fission process produces 2½ new neutrons and, for a chain reaction to be maintained, at least one of these must produce a further fission. So for every 100 neutrons, produced in one neutron generation, at least 40 must cause further fissions so as to produce 40 x 2½ or 100 neutrons in the next generation.

Now the neutrons produced at fission are fast neutrons with an average energy of 2 Mev. If the fissions occur in natural uranium fuel, 99.3% of the nuclei are U-238 which will only fission with neutrons having energies greater than 1.2 Mev. Therefore only half the fission neutrons can cause U-238 fissions. So of 100 neutrons produced at fission, only 50 can cause U-238 fissions. The inelastic scattering cross-section of U-238, at these neutron energies, is 10 times greater than the fission cross-section. So, of the 50 neutrons that could cause fission in U-238, only 5 will do so and 45 will be scattered and lose so much energy that they can no longer cause U-238 fission. The fast fission cross-section in U-235 is only 1.44 barns and, with so little U-235 in natural uranium, U-235 fast fissions can be ignored.

Therefore, of 100 fast neutrons, produced at fission, only 5 will cause further fissions and produce 5 x 2.5 or 12.5 new neutrons. Thus, even if leakage and radiative capture are ignored the chain reaction can not be maintained by fast neutrons in natural uranium. One of two alternatives are available which lead to a power reactor classification as follows:-

(a) Fast Reactors

The U-235 content of the fuel can be increased, ie, the fuel is highly enriched in U-235 with a substantial decrease in U-238. The U-235 fast fissions are thus, considerably increased and a fast reactor results such as the Enrico-Fermi reactor. Some reduction in neutron energy does occur due to inelastic collisions of neutrons with nuclei of the fuel and structural material but most of the fissions are caused by neutrons of energies greater than 0.1 Mev. The curves in Figure 1 show how the mass of U-235 required for the reactor to be critical varies with amount of U-235 enrichment. Curve A refers to a core with fuel only whereas curves B and C apply to cores with the following compositions by volume:-

B - 50% fuel, 33.33% sodium, 16.67% stainless steel

C - 25% fuel, 50% sodium. 25% steel

In each case the core is surrounded by a depleted uranium blanket.

In all cases the critical mass of fissile material required increases rapidly below 15% to 20% U-235 enrichment. To avoid large fuel inventories a practical fast reactor, such as case C above, would require fuel containing at least 20% U-235 by volume. The Enrico-Fermi fuel is an alloy of 28% U-235 with 10% by weight of molybdenum. Incidentally the critical mass of U-235 in a fast reactor is considerable greater than ~~in a fast reactor is considerably greater than~~ in a thermal reactor with the same fuel composition.

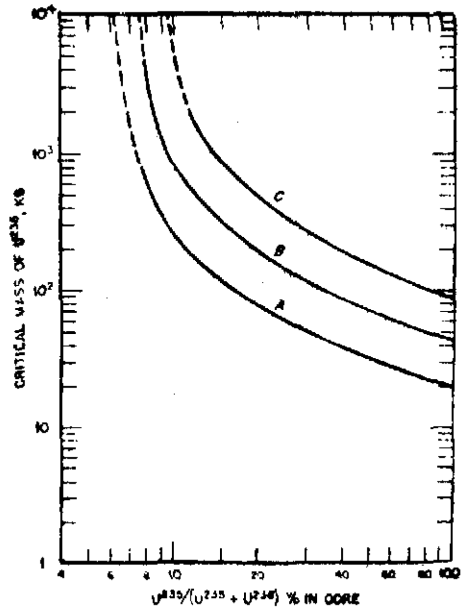


Fig. 1

The highly enriched fuel and absence of moderator results in a small core, the size of the Enrico-Fermi core, for instance, being 31 inch in diameter and 31 inch high with a thermal power production of about 200 MW. Therefore, fast reactors have high power density cores. The average heat flux in the Fermi reactor is 650,000 Btu/hr/ft^2 compared with 100,000 Btu/hr/ft^2 for a pressurized water reactor of similar thermal output. It is therefore essential that a heat transport fluid with good thermal properties be used. The choice is also limited to a non-moderating fluid and liquid metals seem to satisfy both requirements.

The capture cross-sections of most elements for fast neutrons are small and since there is a relatively large mass of U-235 in the reactor, the macroscopic capture cross-sections of structural material and fission products are small compared with the macroscopic fission cross-section of the U-235.

Consequently there is more flexibility in the choice of materials and stainless steel can be used instead of aluminum or zirconium. Fission product poisoning is not significant and for this reason, (and the fact that temperature coefficient of reactivity is low), the excess reactivity required in a fast reactor is small.

A large amount of fuel subdivision is required to provide a large heat transfer area and this increases fabrication problems and manufacturing costs. The core of the Fermi reactor is made up of subassemblies of fuel pins each 0.158 in outer diameter. However the liquid metal heat transport fluid and the use of stainless steel allows for high outlet temperatures and good steam conditions. The heat transport outlet temperature in the Fermi reactor is 800°F giving steam conditions at the turbine of 740°F and 600 psi. This compares with NPD steam conditions of 450°F and 400 psi.

The prompt neutron lifetime is only 10^{-7} secs. compared with 10^{-3} secs. in a thermal reactor but fast reactors can be controlled satisfactorily on delayed neutrons in the same way as thermal reactors. It is true that, if the reactivity becomes equal to the delayed neutron fraction and the reactor becomes prompt critical, the power excusion would be severe. However serious damage would result if a thermal reactor went prompt critical and this is less likely to happen in a fast reactor since the excess reactivity that has to be built into the core is much less than in a thermal reactor.

The chief advantage of the fast reactor concept lies in the possibility of using it for breeding or conversion, ie, for obtaining Pu-239 from U-238 or U-233 from Th-232. As shown in the previous lesson, the values of η-1 (which could be called the BREEDING POTENTIAL) is generally higher, for all three fissile materials, with fast neutrons than with thermal neutrons. The variation of this breeding potential with neutron energy is shown in the following table.

Nuclide	Breeding Potential (η - 1) for neutrons in energy range				
	Thermal	1 to 3000 ev	3 to .10 Kev	0.1 to 0.4 Mev	0.4 to 1.0 Mev
Pu-239	1.03	0.62	0.93	1.61	1.88
U-235	1.08	0.68	0.90	1.12	1.46
U-233	1.31	0.88	1.29	1.30	1.54

It may be seen, from the table, that it is desirable to have as little moderation of the neutrons as possible in order to maintain as high a value of η - 1 as possible. Since the initial breeder must use U-235 as fissile material, it is even more important to keep the neutron energy as high as possible. Therefore a non-moderating heat transport fluid is essential. Even the use of carbide or oxide fuel can cause some moderation.

Fast reactors have another important advantage as far as breeding potential is concerned. Radiative capture in reactor material is lower with fast neutrons than with thermal neutrons. This decreases the value of w in the equation

$$c = \eta - w - 1$$

w is decreased still further by surrounding the core with a blanket of fertile material as shown in the Enrico-Fermi reactor in Figure 2. The blanket then serves as a neutron reflector to decrease leakage and those neutrons that do escape are captured in the fertile material to produce fissile material.

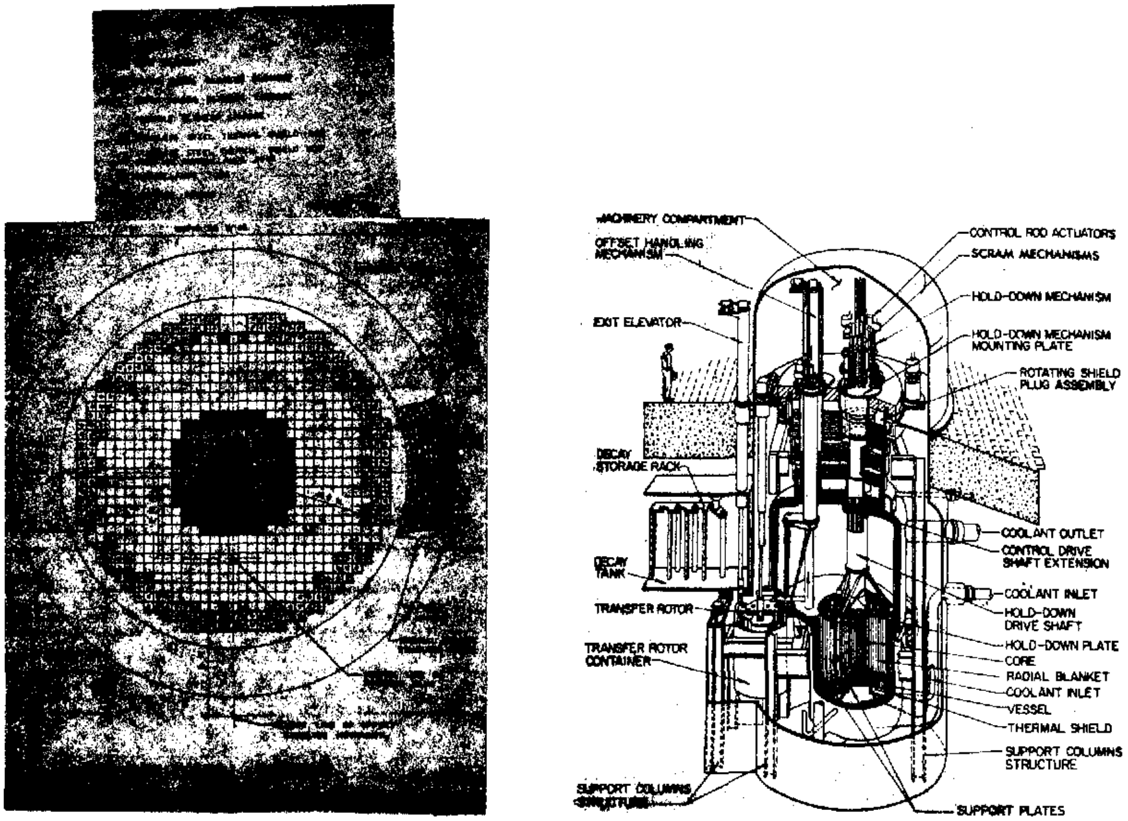


Fig. 2

In order to maintain steady-state conditions in a breeder operating at constant power, the blanket must be processed at such a rate as to provide fissile atoms to compensate for those consumed in the core. It should not be necessary to obtain fissile material from an outside source after the initial loading.

Fast breeder reactors can only be utilized if enrichment and fuel processing facilities are available. Such facilities, on the other hand are only economic if enough reactors can be serviced by them. The cost of fuel and the cost of nuclear power must allow for the capital cost of such facilities.

(b) Thermal Reactors

Since a chain reaction can not be maintained with fast neutrons without considerable enrichment, the alternative is to reduce the neutron energy until the fission cross-section of U-235 is sufficiently increased. If the neutrons are reduced to thermal energies (ie, thermalized), the U-235 fission cross-section is 580 barns whereas the radiative capture cross-section is 106 barns. Thus, even allowing for the low percentage of U-235 in natural uranium, the thermal neutron fission cross-section in natural uranium is 4.2 barns whereas the radiative capture cross-section is 3.5 barns.

Thus, for every 77 neutrons captured in natural uranium about 40 will cause fission and produce 40 x 2½ or 100 new neutrons. For 77 neutrons out of every 100 to be captured, fewer than 23 neutrons can be lost by escape or radiative capture in other reactor material. Under these conditions a chain reaction could be sustained. In thermal reactors the fission neutrons are thermalized by slowing them down in a moderator. Most of the power reactors in existence are thermal reactors.

Thermal reactors have several advantages over the fast reactors:-

(1) They can use any form of fuel from natural uranium to pure fissionable material. This permits a wide range of reactor sizes depending on fuel and enrichment availability.

(2) Since the fuel used is normally natural or slightly enriched uranium, the reactor is relatively large and this simplifies the design of the calandria and associated instrumentation.

(3) Fuel changing is very much simplified because the fuel diameter is greater and the fuel channels are further apart.

(4) The power density is lower in a larger reactor with lower U-235 concentration so that heat removal is less of a problem. This allows for a wider choice of heat transport fluids especially as the fluid can have moderating properties.

However, thermal reactors have the following disadvantages compared with the fast reactor:-

(1) The larger reactor size increases the shielding costs.

(2) The choice of reactor structural material is more limited because of the higher capture cross-section of thermal neutrons. This may limit the temperature of the heat transport fluid and result in poorer steam quality.

(3) There is a high Xenon and Samarium poison load during normal operation which necessitates having enough excess reactivity in the reactor to overcome this.

(4) On reactor shutdown the Xenon load builds up above its normal equilibrium level, resulting in the so called Xenon transient. It is impractical and uneconomical to have sufficient excess reactivity to overcome this transient. Thus, unless the reactor is started up again within 10 or 15 minutes it will be incapable of starting up for several hours, this delay being known as the poison-out time. The poison-override time, during which the reactor can start up after a shutdown, can be extended by booster fuel rods but this necessitates a higher fuel inventory.

ASSIGNMENT

1. Explain why fission neutrons can not maintain a chain reaction in natural uranium.
2. How is the chain reaction maintained in a fast reactor?
3. What are the two requirements for the heat transport fluid in a fast reactor?
4. Give two advantages of fast reactors over thermal reactors.
5. Explain why a fast reactor concept offers breeding or conversion possibilities.
6. Why is a chain reaction maintained in natural uranium by thermal neutrons?
7. List two advantages and two disadvantages of a thermal reactor as compared with a fast reactor.

A. Williams

Reactor Boiler and Auxiliaries - Course 133

REACTOR CLASSIFICATIONS - TYPES OF THERMAL REACTORS

In the previous lesson reactors were classified on the basis of neutron energy and the various advantages and disadvantages of fast and thermal systems were enumerated. It was mentioned that most of the reactor systems, at present in operation, are thermal reactors.

Thermal reactors will now be classified further on the basis of core structure, the moderator used and the heat transport system used. Some reference will be made to the advantages and disadvantages of each type, but some of these considerations will be discussed later when moderator and heat transport system properties are discussed.

Homogeneous and Heterogeneous Thermal Reactors

If neutron leakage from a reactor is ignored the condition for criticality is that:

$$k = \eta \epsilon p f = 1$$

Once the choice of a thermal, as distinct from a fast, reactor has been made, the choice and arrangement of fuel, moderator, heat transport fluid and structural material must be such that this criticality condition is satisfied.

There are two possible arrangements of fuel and moderator:

(a) Homogeneous System

In this type of reactor the fuel and moderator are intimately mixed together. The uranium can be in solution in the moderator or in fine suspension called a slurry. Other possibilities, which have been considered, are uranium tetrafluoride dissolved in a molten salt such as beryllium fluoride or dissolved or suspended in a liquid metal.

In a homogeneous system the fission neutron is immediately in contact with moderator nuclei and would not have enough energy to cause fast fission in U-238. Hence $\epsilon = 1$.

The value of η depends only on the fuel composition. For natural uranium fuel $\eta = 1.32$.

Therefore, ignoring leakage, the condition for criticality is that

$$pf = \frac{1}{1.32} = 0.77$$

i.e., pf = 0.77 is the minimum condition for a chain reaction to be sustained in a homogeneous system.

Both p and f depend on the proportion of fuel to moderator in the homogeneous mixture. The table below shows how p, f and the product pf vary with this ratio of moderator to fuel atoms in a homogeneous mixture of graphite and natural uranium.

$\frac{\text{Mod. atoms}}{\text{Fuel atoms}}$	p	f	pf
200	0.579	0.889	0.515
300	0.643	0.842	0.541
400	0.682	0.800	0.546
500	0.693	0.762	0.528

From the table, it may be seen that the problem is that as the moderator/fuel ratio is increased, p increases, because there is better thermalization of neutrons, but f decreases, because there is more neutron capture in the moderator and less neutron capture in the fuel. The maximum value pf can have is only 0.55 whereas the value must be 0.77 to sustain a chain reaction. So a chain reaction can not be maintained with a homogeneous mixture of graphite and natural uranium.

A similar situation exists with homogeneous systems of natural uranium and either light water or beryllium. With heavy water as moderator the optimum ratio of moderator to fuel gives a maximum value of pf of 0.78, which corresponds to a maximum value of k = 1.03. This does not allow for neutron leakage nor does it allow for build-up of fission product poison. This means that the size of reactor required, with a natural uranium - heavy water homogeneous mixture, would be too large to be practical.

What could be done to make such a system practical? The answer lies in increasing the value of η. For a practical reactor, in which fuel consumption and poison accumulation are allowed for, k would have to be around 1.1 to 1.2. Therefore the value of η required is given by

$$\eta = \frac{1.2}{pf} = \frac{1.2}{0.78} = 1.54 \text{ for } D_2O \text{ moderator}$$

$$\eta = \frac{1.2}{0.55} = 2.18 \text{ for graphite moderator}$$

The only way to increase η is to increase the U-235 concentration (R) in the fuel, i.e., enrich the fuel. Figure 1 shows how the value of η varies with the U-235 concentration in the fuel. The U-235 concentration in natural uranium is 0.00715 (0.715%). For a D_2O moderated homogeneous system to be practical, the U-235 concentration would have to be increased to about 0.02.

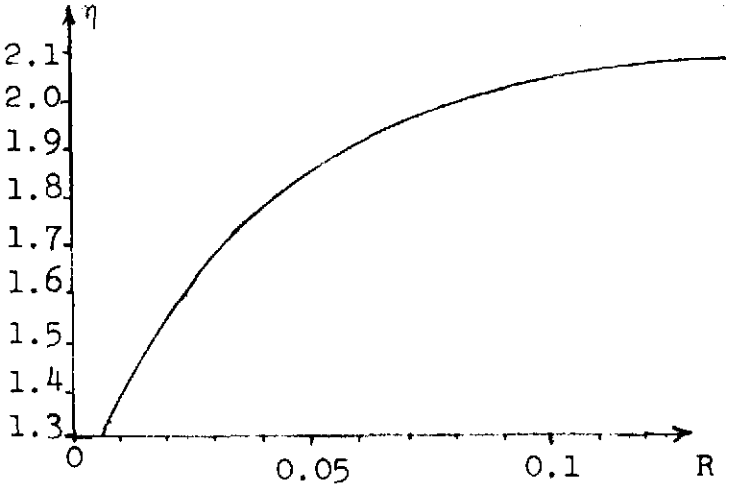


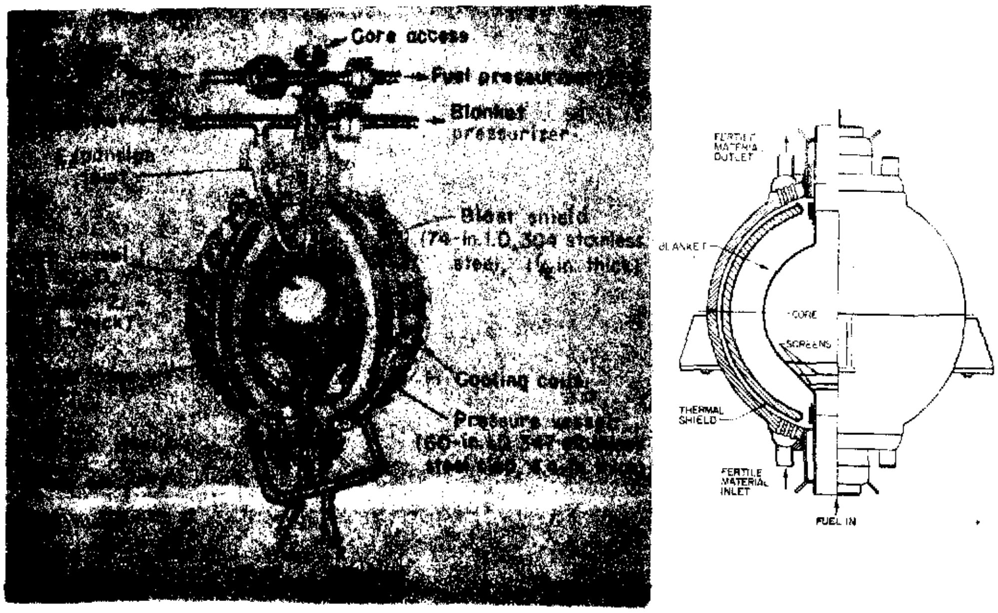
Fig. 1

It can be seen from Figure 2 that a value of η of 2.18 could not be obtained. However, as the enrichment increases, the U-238 content decreases and the value of p consequently increases. This reduces the value of η required for the graphite homogeneous system below the 2.1 value shown above. The decrease in the required value of η is such that, with a graphite moderator, the enrichment required is about a factor of 10 higher than with a D_2O moderator.

The graph shows that there is very little to be gained, in the value of η , by making R greater than 0.1 but, as mentioned above, the value of p increases with larger values of R because the U-238 concentration decreases.

Homogeneous experimental reactors have been operated successfully. HRE-I, containing a solution of highly enriched uranyl sulphate in ordinary water, operated at 1.6 Mwt at Oak Ridge National Laboratory and produced electricity in 1953. HRE-II used a dilute solution of U-235 in heavy water and operated at 5 Mwt until 1961.

What advantages would such a system have? Since the fuel is a fluid, some of the factors that limit burnup are avoided. As shown in Figure 2, the reactor vessel is a simple container with little or no structure to capture neutrons.



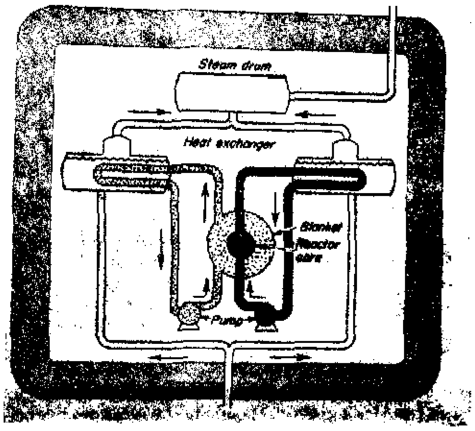


Fig. 3

Fig. 2

The HRE-II reactor vessel, shown in Figure 2, is surrounded by a pressure vessel, which was intended to contain a blanket for breeding. It actually carried heavy water. Figure 3 shows the type of circulation arrangement that would be used for a core and blanket system.

Dimensional changes due to irradiation and fission product gas pressure build-up do not occur. Also, the fuel-moderator mixture is circulated from the reactor through the external boiler and back to the reactor. It is therefore relatively simple to continuously remove fission product poisons. Fabrication costs, which are significant for solid fuel, are of little or no importance in liquid fuel reactors. All these factors offer the possibility of reducing fuel costs. In addition reactor control may be very much simplified since the reactor can be made critical or sub-critical by changing the fuel concentration. Such systems are inherently safe because they have a large negative temperature coefficient of reactivity and are, therefore, self-regulating.

There are, however, several disadvantages and several problems that still have to be solved. Because of the circulation of the fuel outside the reactor, many of the delayed neutrons are born outside the core. It is, therefore, necessary to rely on the large negative temperature coefficient of reactivity in order to have a stable system. There are drawbacks in circulating highly corrosive solutions such as uranyl sulphate and there are still unsolved problems in this area. Another disadvantage is the circulation, outside the reactor, of highly radioactive fission products and the shielding problems associated with this. Radiolytic decomposition of water and the high vapour pressure of aqueous solutions necessitates pressurization to 1000 psig or 2000 psig. The maximum practical temperatures are low (482°F for HRE-II) and so the thermodynamic efficiency of the system is limited.

(b) Heterogeneous Systems

The alternative to using a homogeneous system with enriched fuel, is to use a heterogeneous system. In such a system, shown in Figure 4, the fuel is in the form of lumps or rods arranged in a regular pattern or lattice in the moderator.

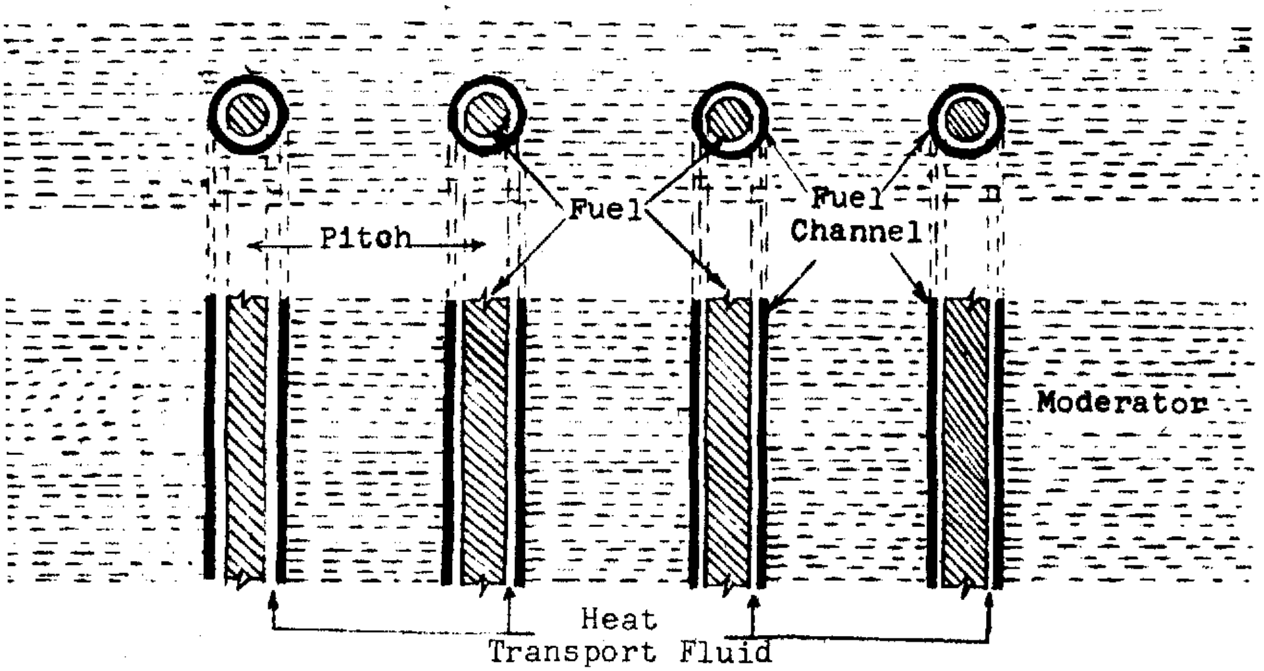


Fig. 4

The rod arrangement, inside fuel channel, is preferred since it fixes the fuel location, makes fuel changing easier and enables the heat to be removed from the fuel. This type of arrangement increases the value of ϵ because the fast neutrons are now released in the fuel and are more likely to cause fast fissions in U-238 before escaping into the moderator. ϵ will now be 1.02 to 1.03.

The value of p is increased because the neutrons, after leaving the fuel, are slowed down in the moderator before entering the next fuel rod. With correct rod separation, (or lattice pitch), most of the neutrons are thermalized before entering the next rod.

The thermal utilization factor, f, is decreased in a heterogeneous system because relatively fewer neutrons are captured in the fuel. However, unless the moderator has a large neutron capture cross section, the decrease in f is small and can still be 0.9 or greater.

Thus pf = 0.85 or more compared with 0.55 with a graphite homogeneous system and 0.78 with a D_2O homogeneous system. Hence, with natural uranium fuel and $\eta = 1.32$,

$$k = 0.85 \times 1.02 \times 1.32 = 1.14 \text{ without leakage.}$$

This value of k makes a chain reaction possible provided the moderator absorption does not decrease f too far. This will be discussed further when moderator properties are considered.

Types of Heterogeneous Reactors

The classification of heterogeneous reactors may be based on the type of moderator used or on the heat transport system employed. The basic requirements and properties of moderators and heat transport systems will be discussed at length later. It is sufficient, for the moment, to list the moderators and heat transport fluids in general use.

The moderator may be:

1. Light water, 2. Heavy water, 3. Graphite, or 4. Organic liquids.

The heat transport system may be:

1. Pressurized light water
2. Pressurized heavy water
3. Boiling light water
4. Boiling heavy water
5. Gases such as CO_2 or helium
6. Liquid metals
7. Steam or fog
8. Organic liquids

There are many possible arrangements or combinations of moderator and heat transport systems but only the combinations actually in use or to be in use will be discussed briefly. A brief reference will be made to their advantages and disadvantages.

(a) Light Water Moderated Reactors

The pressurized - light water reactor and the boiling - light water reactor are the principal types in this category. Recent developments have tended to bring the two concepts closer together.

Some local boiling is now tolerated in the pressurized-water systems and boiling-water reactors operate at elevated pressures, in any case. It is therefore better to designate them as reactors using the indirect cycle and requiring a heat exchanger or boiler to produce steam and reactors using the direct cycle, the steam being produced in the reactor and transported directly to the turbine.

The PWR and BWR types have a heat transport fluid which also acts as the moderator. Hence both moderator and heat transport fluid are pressurized and must be contained in a pressure vessel as shown in Figures 5 and 6.

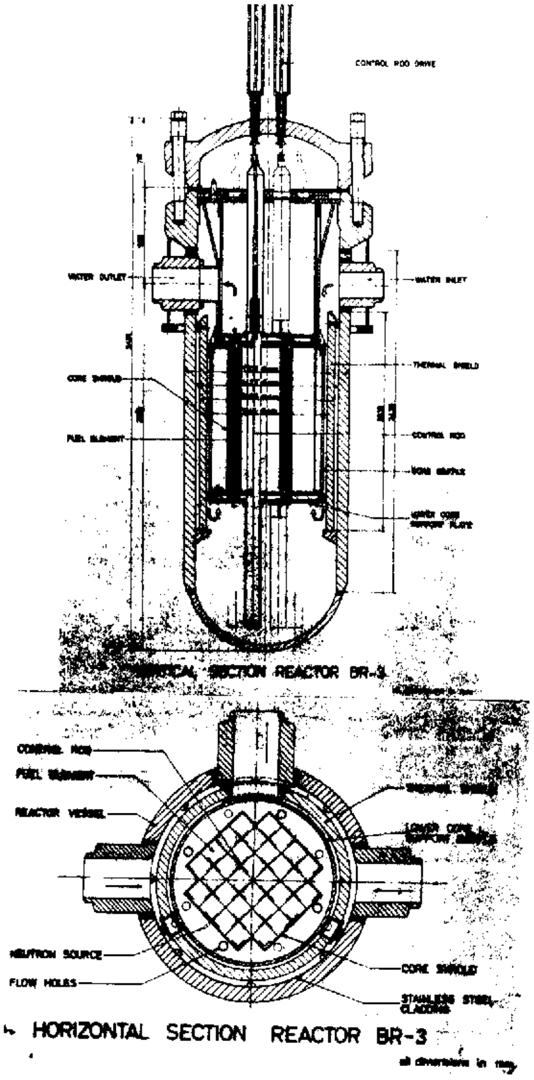


Fig. 5

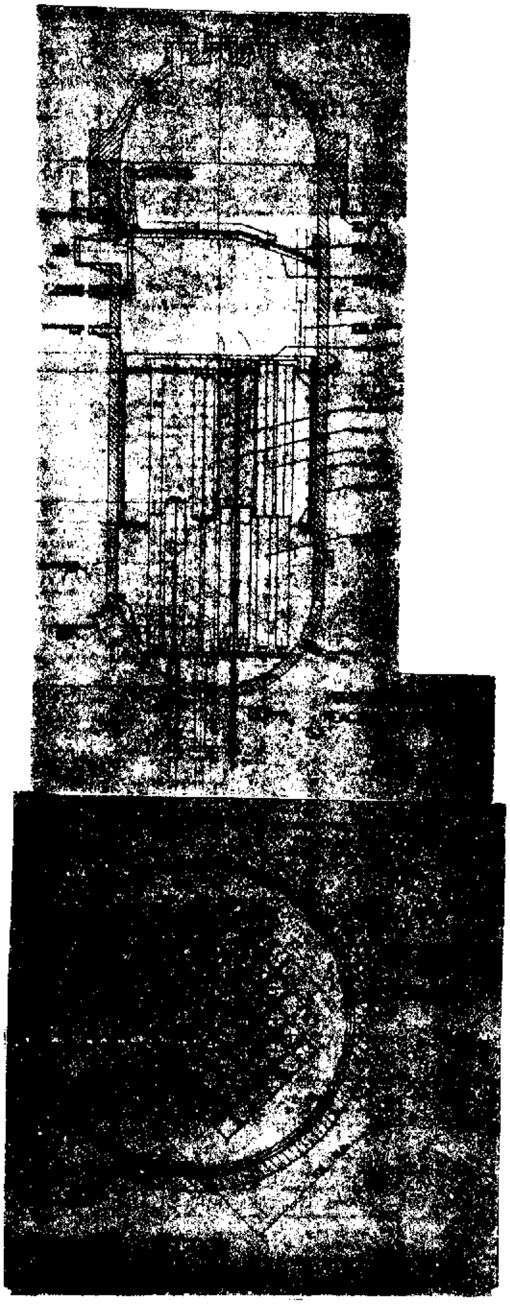


Fig. 6

Figure 5 shows the Belgian BR-3 pressurized water reactor as a typical example of a PWR type. The boiling water reactor, shown in Figure 6, is the Big Rock Point reactor in the U.S. The figures illustrate how very similar the BWR and PWR types actually are.

The use of light water moderator makes it impossible to maintain a chain reaction with natural uranium fuel. Slightly enriched fuel must therefore be used which necessitates an enrichment plant. Typical enrichment used is from 2% to 5% U-235. The critical mass required in such a reactor is around 50 or 60 kgm U-235 compared with 600 or 700 kgm U-235 or more in a fast reactor. Since enriched fuel has to be used it is possible that it might be advantageous to increase the enrichment in order, say, that stainless steel fuel sheaths and fuel channels can be used. This would give better fuel integrity, reduce corrosion and reduce the material and fabrication costs associated with zircalloy.

Because the same fluid is used for both moderator and heat transport medium, only thin tubes are required, as fuel channels, to guide the fluid over the fuel. This tends to decrease the neutron absorption in the reactor structure.

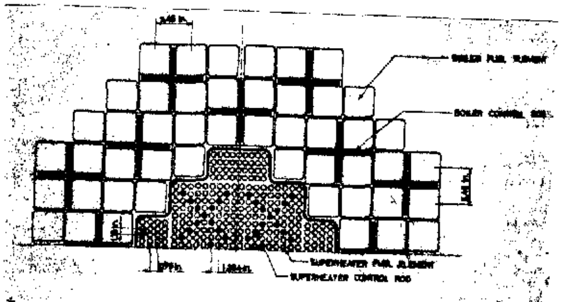
In the case of pressurized water reactors, the heat transport temperature must be kept as high as possible for good thermodynamic efficiency. To prevent boiling, even at modest operating temperatures, the heat transport pressure must be as high as 1000 psi to 2000 psi. The whole reactor vessel must withstand these high pressures. It is estimated that for a 500 Mwe reactor the vessel would have an internal diameter of 12 feet and a wall thickness of 11 inches. It is difficult to fabricate large vessels of this sort with wall thicknesses greater than 9 to 10 inches. This limits the maximum temperatures at which the heat transport systems can operate. It is also difficult to transport such vessel and 350 tons is an accepted maximum weight. This tends to restrict the maximum power. Field fabrication techniques have reduced the severity of the problems.

In the boiling water type the pressure is built up as the steam is produced in the core and the pressure is generally lower. The pressure vessel thicknesses are, therefore, smaller being around 5 inches to 6 inches. Even so, there are limitations on both operating temperatures and reactor power.

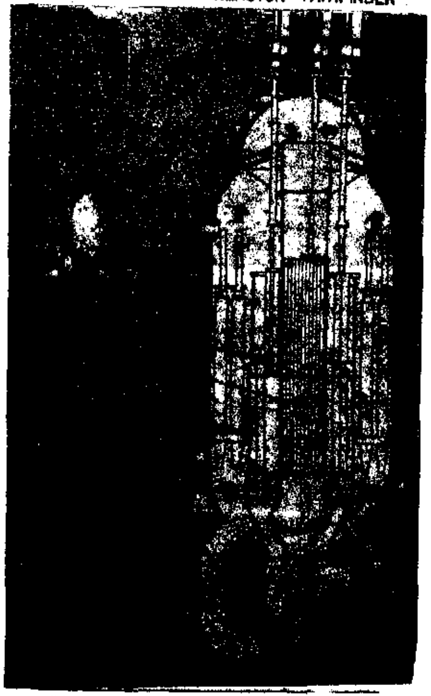
Because the pressure vessels are so thick, severe thermal stresses could be set up by radiation absorption. It is, therefore, necessary to place a thermal shield between the reactor and its containing vessel to reduce the radiation absorption in the vessel.

Finally, refueling at power is impractical with this type of reactor because of the large number of penetrations required through the vessel head. The reactor is sized so that it can operate for six months or so without refueling. The refueling is then carried out with the reactor shut down and depressurized. This means that the dome or head of the reactor vessel must be removable to permit this refueling.

Although no reactor has, as yet, been constructed which uses fog or steam as the heat transport fluid, there are two reactors which have a nuclear superheating section in the core through which steam is passed. These are the Pathfinder and Boiling-Nuclear Superheat (Bonus) reactors, the former in South Dakota and the latter in Puerto Rico. The Pathfinder reactor is illustrated in Figure 7. It may be seen that light water is boiled in the outer core region where the light water acts as moderator and heat transport fluid. The saturated steam passes through steam separators and dryers and the dry steam then passes down through the centre superheater region of the core. In this way steam at 825°F is obtained which leads to savings on turbine size and cost.



CORE ARRANGEMENT REACTOR PATHFINDER



VERTICAL SECTION REACTOR PATHFINDER

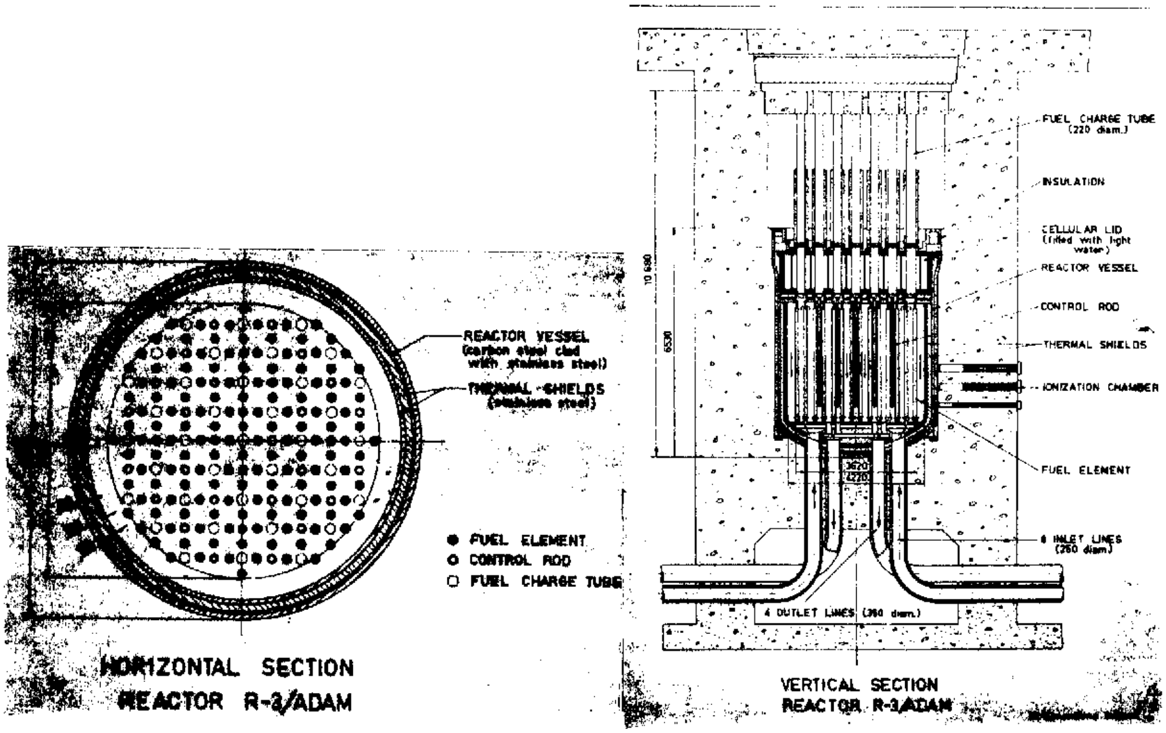
Fig. 7

(b) Heavy Water Moderated Reactors

Heavy water has a much lower neutron capture cross section than light water. The principal advantage of using heavy water as a moderator is, therefore, the neutron economy that can be achieved with it. The thermal utilization factor, f, in the four factor formula, is increased because of lower neutron capture in the moderator. Neutron economy is so much improved that, not only can natural uranium fuel be used, but that this fuel can be used in oxide or carbide form. Thus, there is no longer any need for an enrichment plant. In addition oxide or carbide fuel improve the fuel integrity and the fuel is less susceptible to distortion

Higher conversion factors and breeding ratios are possible because the factor w is smaller. This, combined with the improved fuel integrity, enables high fuel burnups to be obtained. There are promising breeding prospects using the U-233:Th-232 system because the value of η for thermal neutrons is 2.31.

A variety of heat transport fluids can and have been used with heavy water as a moderator. Most of the heavy water moderated reactors which have been constructed or are being designed us a heavy water heat transport fluid. This maintains the good neutron economy and retains the good handling and thermal characteristics of light water. The Swedish R-3/Adam Reactor, shown in Figure 8, is a pressure vessel type of reactor with the moderator and heat transport fluid being intimately mixed and at the same temperature.



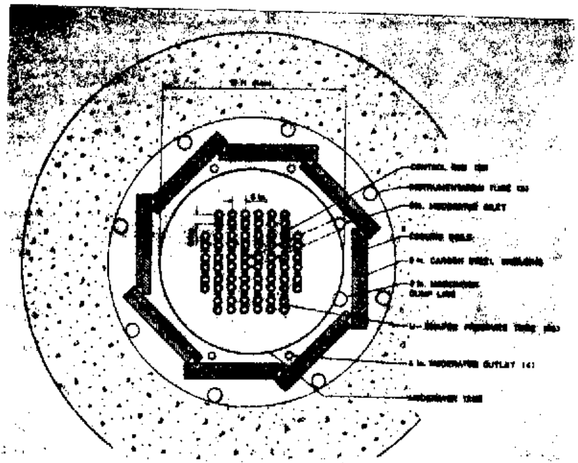
HORIZONTAL SECTION
REACTOR R-3/ADAM

VERTICAL SECTION
REACTOR R-3/ADAM

Fig. 8

The Halden Boiling Heavy Water Reactor, in Norway, uses a similar arrangement but the heavy water is allowed to boil. The only advantages offered by these reactors, over their light water equivalents, are use of natural uranium fuel and good neutron economy. However, they are still subject to the disadvantages and restrictions of the pressure vessel concept.

Thermalization of neutrons requires far fewer elastic collisions in light water than in heavy water. Thus, in heavy water the separation of fuel channels (i.e., the lattice pitch) must be greater than in light water, if the resonance escape probability, p, is to remain high. This larger fuel channel separation requirement has been utilized to advantage in the "pressure tube" reactor concept which avoids the problems inherent in the pressure vessel type of design.



HORIZONTAL SECTION REACTOR CVTR

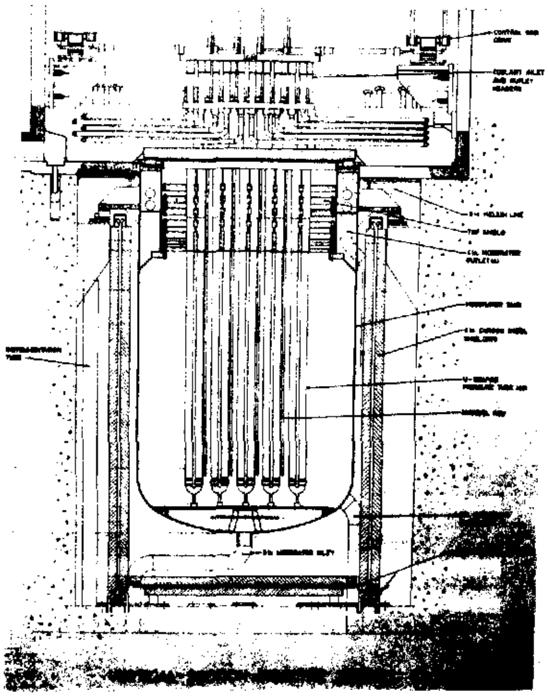


Fig. 9

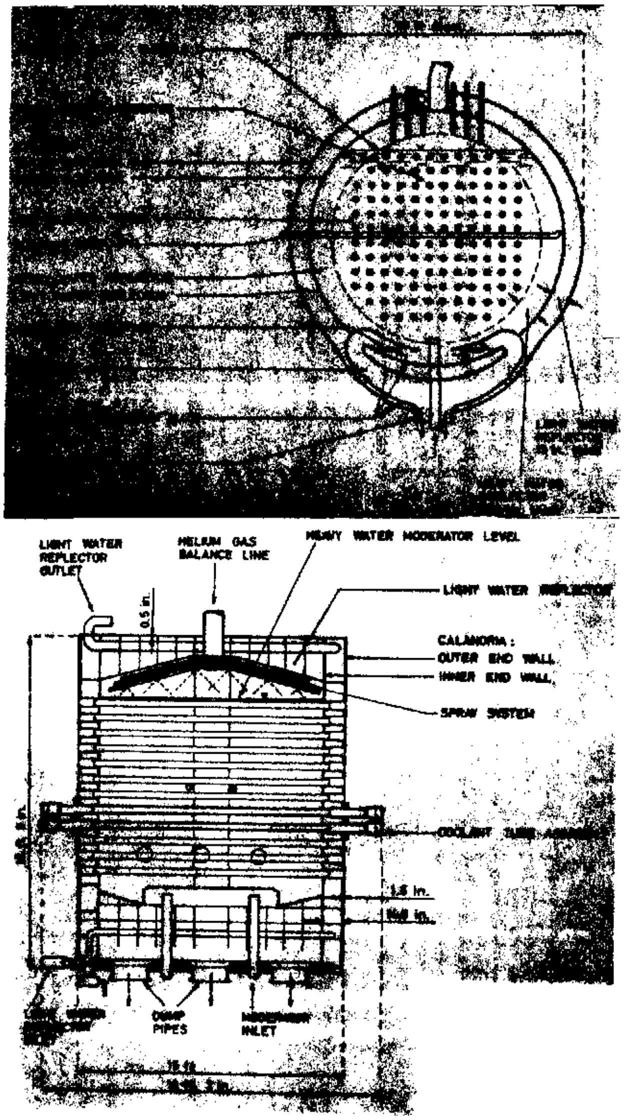


Fig. 10

In the Carolinas Virginia Tube Reactor (CVTR), shown in Figure 9, the moderator is contained in a cylindrical tank surrounding U-shaped pressure tubes containing the fuel and heavy water heat transport fluid. The pressure tubes are vertical and are suspended from a support structure resting on top of the moderator tank. Slightly enriched UO_2 fuel is used, the critical mass required being 11.2 kgm U-235. Four fuel assemblies are replaced every 6 - 8 weeks with the reactor shut down.

The Nuclear Power Demonstration Reactor (NPD), shown in Figure 10, is fairly typical of the Canadian power reactor designs. The moderator is contained in a cylindrical drum or calandria the axis of which is horizontal. Horizontal pressure tubes, containing natural UO_2 fuel and heavy water heat transport fluid, pass through this calandria.

In both these pressure tube reactors, the heat transport fluid only is pressurized. The moderator, being separated from the heat transport fluid, is also at a much lower temperature. The NPD type of reactor has the added advantage that it permits bi-directional refueling at full power. This maintains symmetrical thermal neutron flux distribution throughout the reactor and permits efficient fuel utilization without the expense of shutdowns.

The main economic disadvantage of heavy water reactors has been the high capital cost resulting from the larger size core and the high cost of heavy water. Lower costs of heavy water in the future will decrease the heavy water capital costs involved and decrease the operating costs due to heavy water losses. Some disadvantages remain, however, primarily because of the use of pressurized heavy water as a heat transport fluid. Heavy water leakage is difficult to eliminate altogether although improved engineering design and fabrication in the future will minimize such losses. In the meantime heavy water collection and recovery systems add to the capital cost of the stations. The leaking heavy water contains tritium which presents an added health hazard requiring careful control. Radiolytic decomposition of heavy water is another source of heavy water losses and presents a possible hazard because of deuterium build-up. Recombination units are used to recover the losses and keep the deuterium concentration within acceptable limits.

Other reactor concepts, using heavy water as moderator, have been and are being given consideration. The CANDU-BLW station, to be built near Gentilly, Quebec, is a vertical pressure tube reactor using boiling light water as the heat transport medium. This station is expected to be producing power in 1971. This may well lead to a future concept in which the steam produced in one region of the reactor is superheated in another region. The problem involved is to increase the fuel heat rating without using sheathing and pressure tube material which will necessitate fuel enrichment. Organic liquids have also been proposed as heat transport fluids and the WR-1 reactor at Whiteshell, Manitoba, is designed specifically for engineering studies and development of organic heat transport fluids. Organic liquids would allow much higher temperatures to be achieved with little or no pressurization.

Figure 11 shows a heavy water moderated reactor, using carbon dioxide as heat transport fluid. This HWGCR (Heavy Water Moderated Gas Cooled Reactor) is being constructed in Czechoslovakia and has a heat output of 590 Mwt. It uses natural uranium fuel and has a designed outlet gas temperature of 425°C (797°F). This will enable better steam conditions to be obtained than would be possible with pressurized or boiling water systems.

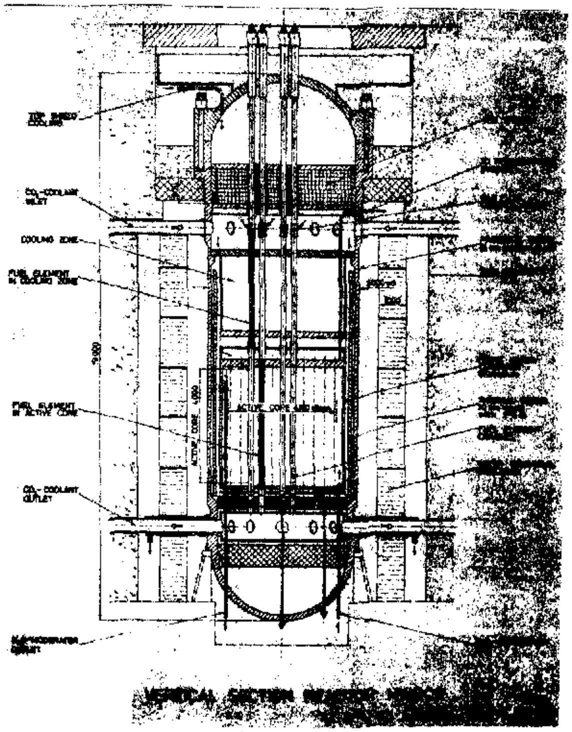


Fig. 11

(c) Graphite Moderated Reactors

The major portion of the nuclear power being produced today is generated in graphite moderated reactors with gas as the heat transport fluid. The Hinkley Point reactor, shown in Figure 12, is a typical example. The fuel used is natural uranium metal clad with Magnox, a magnesium alloy. The heat transport fluid is carbon dioxide gas at a pressure of 150 to 250 psia. A chain reaction could not be sustained using oxide fuel and a graphite moderator. Even with uranium metal, very high grade graphite must be used and 2000 tons of this are required. This introduces structural problems of locating and constraining the graphite and controlling differential expansion between it and the pressure vessel in which it is contained. The Hinkley Point vessel is 67 feet in diameter and this factor alone limits the vessel thickness and consequently the operating pressure. In comparison, the Douglas Point calandria is less than 20 feet in diameter and the Dresden boiling light water reactor vessel is around 12 feet in diameter. The resulting high capital cost is compensated for to some extent by the lower fuel costs.

The temperature of the fuel, and therefore of the CO_2, is also limited because of the alpha to beta phase change that occurs in the uranium metal fuel and because of the temperature limitation on Magnox. Steam temperatures are, therefore, limited to about 740°F. A chemical reaction also occurs between the CO_2 and the graphite at higher temperatures. Gases have poor heat transfer characteristics and have high pumping power requirements, so that up to 20% of gross plant power may be needed for gas circulation.

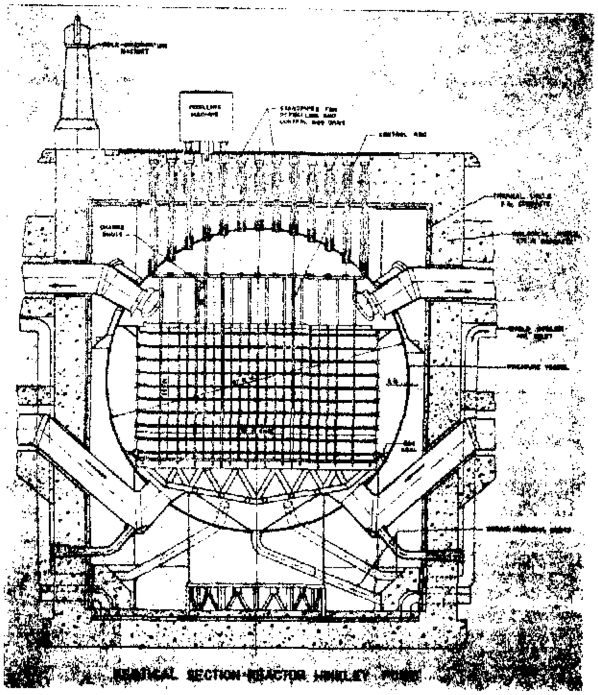

VERTICAL SECTION-REACTOR HINKLEY POINT

Fig. 12

There is considerable incentive to develop satisfactory methods for using carbon dioxide at higher temperatures since this also reduces the pumping power requirements. The Advanced Gas-Cooled Reactor (AGR) in the U.K. used slightly enriched UO_2 fuel in stainless steel cladding with graphite moderator. The CO_2 temperature is 1070°F at the outlet, giving 850°F steam at 650 psi. The use of enriched fuel reduces the diameter of the pressure vessel to 21 feet. Chemical reaction between the CO_2 and the graphite is inhibited by adding methane to the CO_2. The fuel channels also have removable liners.

It would still appear that there are better prospects of higher heat transport temperatures and, therefore, better steam conditions if a fluid other than CO_2 is used as the heat transport medium. The Peach Bottom High Temperature Gas-Cooled Reactor uses an Uranium-Thorium fuel combination and helium as the heat transport fluid. The fuel is in the form of carbides dispersed in graphite and clad in a dense, impervious graphite. The reactor is also graphite moderated. It is hoped to obtain a heat transport temperature of 1380°F, providing 1000°F steam at 1450 psia and a net thermal cycle efficiency of 35%. It is also hoped to have fuel burnups approaching 75,000 Mwd/tonne by Th-232 conversion.

An alternative heat transport fluid has been used in the Hallam Nuclear Power Facility shown in Figure 13.

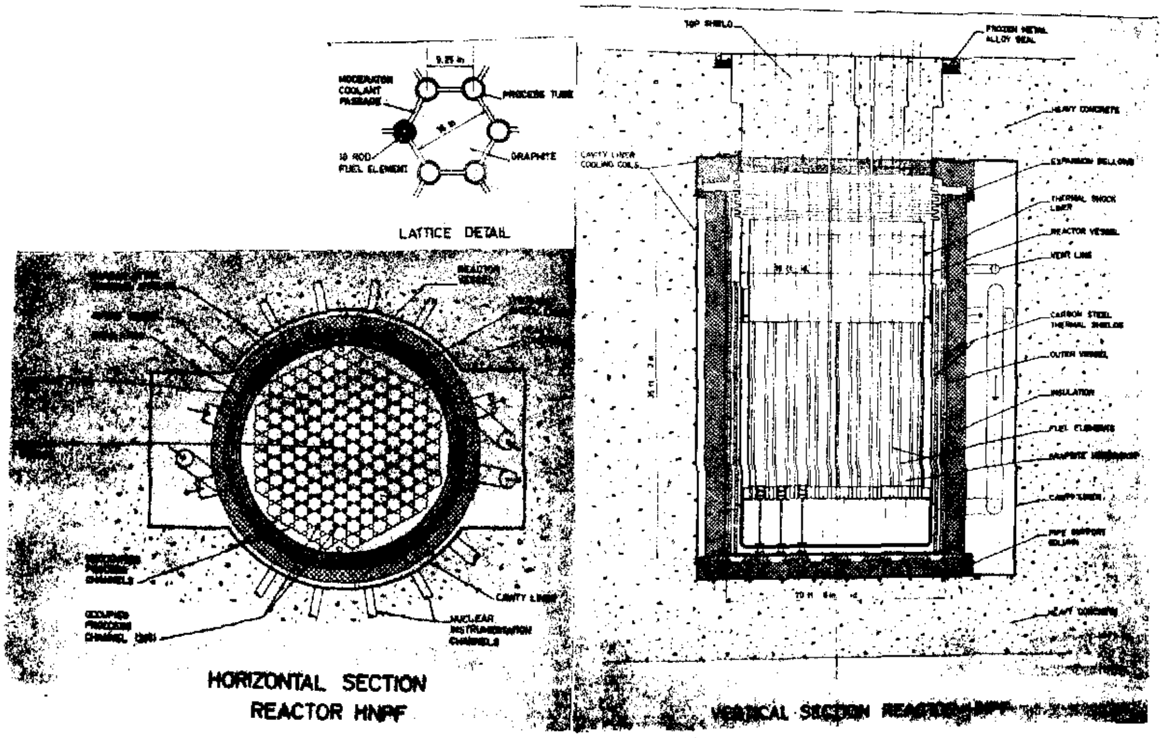


Fig. 13

The core is composed of hexagonal graphite "logs" canned in stainless steel. The fuel channels, containing slightly enriched uranium, are located at the corners of the hexagons. Sodium is used as the heat transport fluid and an intermediate heat exchanger is used to transfer the heat to a secondary sodium circuit which is connected to the boiler. A core outlet temperature of 945°F is achieved and superheated steam at 833°F and 850 psig produced.

(d) Organic Moderated Reactors

The advantages of organic liquids as heat transport fluids have been mentioned already. The moderating properties of an organic, such as diphenyl ($C_{12}H_{10}$) are not very different from those of light water. Both require some fuel enrichment to make a self-sustained chain reaction possible. There seems to be no advantage, then, in using an organic moderator unless both the moderator and the heat transport fluid are of the same material and the pressure vessel concept is used.

The Piqua Nuclear Power Facility, shown in Figure 14, uses this concept. The fuel used is an alloy of 1.94% enriched uranium metal with 3.5% by weight of molybdenum and about 0.1% by weight of aluminum. The combined moderator and heat transport fluid is a mixture of terphenyls. An outlet temperature of 575°F can be obtained with a pressure of only 120 psia. The use of inexpensive materials and the fairly low pressure requirements lead to comparatively low capital costs, which are partially offset by the cost of purification and replacement to make up for radiolytic and thermal decomposition of the organic liquid. Organics are flammable in air at temperatures above 415°F and the highest operating temperatures, regardless of pressure, is about 800°F.

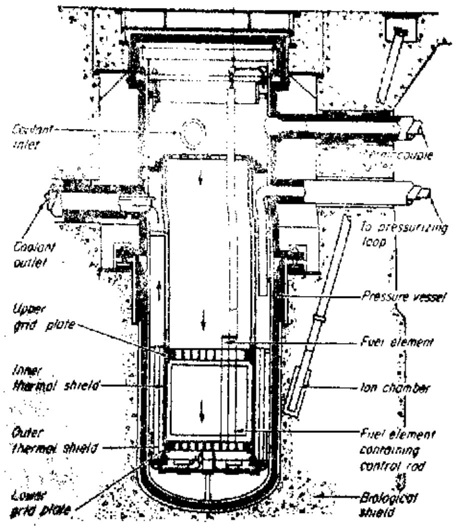


Fig. 14

ASSIGNMENT

1. Explain why fuel enrichment is required in a homogeneous reactor system and indicate the degree of enrichment required with heavy water and graphite moderators.

2. Briefly enumerate the advantages and disadvantages of a homogeneous reactor system.

3. What factors are affected and in what way are they affected by using a heterogeneous, as distinct from a homogeneous, arrangement of fuel and moderator?

4. What similarities exist between pressurized and boiling light water reactor and in what principal ways do they diffe

5. (a) How are some of the basic disadvantages of the boiling water reactors overcome by using a system such as that used in the Pathfinder reactor?

   (b) What basic problem is involved in applying the same principle in a reactor using natural uranium fuel?

6. Briefly state the principal advantages of using heavy water moderated reactors and the principal disadvantages if heavy water is also used as a heat transport fluid.

7. What advantages are inherent in the horizontal pressure tube concept?

8. Specify the problems inherent in the Hinkley Point type of reactor and the reasons for the temperature limitations imposed in such a system.

9. What are the alternative heat transport fluids that could be used in a graphite moderated reactor and what effect would they have on the above limitations?

A. Williams

Reactor Boiler and Auxiliaries - Course 133

REACTOR CONSTRUCTION

As was seen, in the previous lesson, there are many different concepts for power reactors involving different combinations of fuel arrangements, core arrangements, moderators and heat transport fluids. Consequently there are many possible variations in design and construction of power reactors. Some of the associated problems may be common to more than one concept. Other problems may be specific to a particular type of reactor. The major problems will be discussed in this lesson, particularly those involving the choice of reactor structural materials.

Reactor Components

It has already been said that the basic function of a reactor is to contain the required combination of fissile material (or fuel) and moderator to sustain a chain reaction. A further objective is to obtain the maximum burn-up possible from the fuel and a third objective is to transport the heat from the fuel in as useful a form as possible, (i.e. at as high a temperature as possible.)

The principal components that may be required to fulfill these functions or objectives are:-

a. Fuel which includes the fissile material and fertile or alloying materials that may cause dilution of the fissile material.

b. Fuel cladding or sheaths if the fuel is solid, the main purposes of which are to prevent dispersion of fuel throughout the heat transport system, to prevent release of fission products which would contaminate the system and to prevent a possible chemical reaction between the heat transport fluid and the fuel material.

c. Heat transport fluid to transport the heat from the fuel.

d. Fuel channels or tubes to position the fuel in the reactor, and so simplify fuel changing, and to direct the heat transport fluid over the fuel.

e. The moderator, in a thermal reactor, to thermalize the fission neutron.

f. The reactor vessel to contain or position any or all of the above components.

g. Reactivity mechanisms used to vary the excess reactivity in the reactor either for power regulation, to compensate for reactivity changes or to shutdown the reactor.

h. A reflector to reduce neutron leakage which may also be a blanket used for breeding.

It may be necessary to provide some means of thermally insulating the moderator from the heat transport fluid. The reactor vessel may also contain structural material to strenghthen the vessel itself, to support the moderator and reflector or to guide the reactivity mechanism. A thermal shield may be located inside the reactor vessel to reduce heat generation in the vessel.

The components such as the fuel, fuel cladding, heat transport fluid, moderator, reactivity mechanisms and reflector will be discussed in later lessons. In this lesson the discussions will be confined to reactor and reactor vessel structural components.

Structural Material Requirements.

The requirements of a reactor structural material will vary to some extent, with the type of reactor and with its specific purpose in the reactor. In particular it will depend on whether or not the material is inside the reactor core. The reactor core may be defined as the region, within the reactor vessel which contributes directly to the production of energy and the removal of the energy from the fuel. It will, therefore, include all materials or components, such as fuel, cladding, heat transport fluid, fuel channels and moderator, up to the outer boundaries of the moderator or within the reflector, i.e. in the region where neutron conservation or economy is a primary consideration.

Structural material requirements may, however, be stated as follows:

(a) They must have small neutron capture cross-sections, particularly in the core region. This reduces neutron capture in materials other than fuel and increases the thermal utilization factor, f. It also reduces the factor w and, thereby, increases the conversion factor or breeding ratio. The following table lists the thermal neutron absorption cross-sections, σ_a, in structural materials or elements to be found in structural materials.

Material	σ_a (barns)	Material	σ_a (barns)
Magnesium	0.069	Inconnel	~ 4.1
Zirconium	0.185	Monel	~ 4.2
Aluminum	0.24	Nickel	4.6
Aluminum (2S)	~ 0.26	Vanadium	5.0
Aluminum (3S)	~ 0.36	Titanium	5.8
Tin	0.6	Manganese	13.2
Niobium	1.2	Tungsten	19.2
Molybdenum	2.7	Tantalum	21.3
Iron	2.6	Cobalt	37.0
Chromium	3.1	Hafnium	105
Stainless Steel	~ 3.1		
Copper	3.8		

The materials above the first dotted line have absorption cross-sections below 1 barn and they, or their alloys, could be considered for use in a thermal natural uranium reactor. The group below the second dotted line have cross-sections in excess of 10 barns and could not, in general, be considered as core structural materials in a thermal reactor. The materials with intermediate cross-sections might be used in a thermal reactor using enriched fuel.

The absorption cross-sections for fast neutrons are substantially smaller than for thermal neutrons e.g. $\sigma_a = 0.006$ for iron at 1 Mev. Also the quantity of fissile material, relative to structural material, in a fast reactor is much greater than in a thermal reactor. Therefore, the neutron utilization factor tends to be higher. As a result, structural material, which could not be considered in a thermal system, can be used in a fast reactor. Other material properties are then more important and stainless steels, niobium, molybdenum, tantalum and tungsten would appear attractive, the latter two because of their high melting points.

There is, of course, a much wider choice of materials for structural components outside the reactor core, since neutron absorption cross-section is no longer important.

(b) They must be able to resist radiation damage effects. Absorption of radiation and scattering of neutrons in materials causes changes in the physical properties of the material. Changes in strength, ductility and thermal conductivity occur in metals. Organic materials may have their essential properties drastically changed.

This resistance to radiation damage is particularly important in the reactor core because of the higher radiation intensities but radiation damage, particularly decrease in strength, may be serious in the reactor vessel itself. Because of the adverse effects of intense nuclear radiation on most organic materials, reactor structural components are predominantly metallic in construction.

(c) They should not become radioactive as a result of neutron capture or, if they do become radioactive, the radioactive nucleus should have a short half-life and, preferably, emit no gamma radiation. Corrosion or erosion causes small quantities of structural material to be circulated around the moderator and heat transport systems. If these corrosion products become radioactive as they circulate through the reactor and gamma ray emitters result, then the equipment associated with these systems will not be accessible during reactor operation. Shielding will be required to protect personnel from the radiation emitted. The radioactive corrosion products may well plate out on pipes, pumps and other equipment and the presence of long half-life nuclei would cause serious maintenance problems. The radioactive nuclei that would present the most serious problems are those given in the following table:-

Nucleus	Cr-51	Mn-56	Fe-59	Co-60	Zn-65	Zr-95	Mo-99	Ta-182
$t_{\frac{1}{2}}$	28d	2.6h	45d	5.3y	250d	63d	67h	112d
(Mev) γ energy	0.32	2.1	1.3	1.3	1.1	0.75	0.78	1.1

(d) Their mechanical properties, such as tensile strength, impact strength and rupture stress, must be adequate for the operating conditions. The reactor vessel and its internal components constitute a mechanical structure which must support itself under elevated temperature and possibly under elevated pressures.

(e) They must be able to maintain stability under severe thermal stress. Considerable internal heating of reactor components may occur as a result of radiation absorption or slowing down of fast neutrons. The removal of heat from the exterior of such components results in high temperature gradients within the material. Thermal stresses may be particularly severe during reactor startup, power changes or reactor shutdown.

(f) They should have high thermal conductivity so that there is efficient transfer of the heat generated in the components.

(g) They must have good corrosion resistance so that they are not chemically attacked by the moderator or heat transport fluid.

(h) The coefficient of thermal expansion must be low or well matched to that of other materials.

(i) They should have good fabrication characteristics.

(j) They should be readily available in pure form. Small amounts of impurities could cause large increases in the neutron absorption cross-section and impair the ductility.

(k) The cost should be reasonable.

Properties of Possible Structural Materials

From a consideration of absorption cross-section only Magnesium, Aluminum, Tin and Zirconium are attractive as possible materials in a thermal natural uranium reactor. Tin must be rejected, as a structural material, because of other considerations. Magnesium is too reactive chemically although its alloy, Magnox, is used as cladding material in reactors using CO_2 as heat transport fluid.

If enrichment can be tolerated then the most attractive materials are the iron and steel alloys, mainly because of cost. Many other materials can be considered for high temperatures particularly in fast reactors.

The most common materials, Aluminum, Zirconium and steel, will be considered in details and brief reference will be made to some of the other possibilities.

(a) Aluminum and its Alloys

Aluminum has a low thermal neutron capture cross-section. In addition it is relatively cheap and easily produced and fabricated. It is relatively easily rivetted, welded, forged and machined and cast shapes can be produced if proper care is taken to prevent porosity. It has good corrosion resistance at low temperatures but this resistance decreases even at moderately elevated temperatures. It is particularly susceptible to galvanic corrosion and pitting if small particles of foreign material are present. So care must be taken to keep aluminum surfaces free from other metals, particularly copper.

The pH of water in contact with the metal must be carefully controlled between 5.5 and 7.0, since it is subject to acidic and basic corrosion.

The thermal conductivity of aluminum is particularly good (around 121 Btu/hr-ft-°F). Its coefficient of linear expansion is 14.8×10^{-6} per °F between 20°C and 400°C. It has moderate mechanical strength (13000 psi with 30 - 35% elongation) and it is not subject to appreciable radiation damage. It melts at 1220°F.

Both the mechanical strength and the corrosion resistance can be substantially improved by alloying with silicon, iron manganese etc. The addition of copper, however, reduces the corrosion resistance. Most aluminum alloys suffer severe reduction in tensile strength at temperatures as low as 400°F. There is therefore a temperature limitation on the use of aluminum or its alloys.

(b) Zirconium and its Alloys

Zirconium has a unique combination of properties which makes it a useful metal in reactor construction. In the pure state it has a thermal neutron absorption cross-section of only 0.185 barns which is even lower than that of aluminum. It occurs with hafnium in nature and it has similar properties to hafnium. This makes the removal of the hafnium difficult and expensive and it is imperative that it be removed since it has an absorption cross-section of 105 barns.

The melting point of Zirconium is 3355°F.its thermal conductivity is around 12 Btu/hr-ft^2-°F (about one-tenth of that of aluminum) and its coefficient of linear expansion about 3×10^{-6} per °F. It has good structural properties compared with aluminum (50,000 - 60,000 psi tensile strength with 24 - 30% elongation). As a pure metal it is ductile and easily fabricated.

Annealed Zirconium is significantly affected by neutron irradiation the yield and tensile strengths being increased about 10,000 psi. Cold-worked zirconium, however, is not significantly affected. The strength of zirconium decreases rapidly with temperature but it is still greater than that of aluminum.

Alloys of zirconium are generally used in reactors but the alloying elements have to be carefully chosen so as not to increase the neutron absorption cross-section. Thus, the addition of nickel, iron or chromium would greatly diminish the usefulness of zirconium as a reactor material. Tin, which has a small absorption cross-section, could be used as an alloying element. Alloys of zirconium, such as zircalloy-2, 3Z1 and zircalloy-4, have better room-temperature and elevated-temperature strength than pure zirconium. These alloys are much superior to aluminum in the intermediate temperature range but do not have sufficient strength for operation at high temperatures. For instance, at 900°F, the creep strength of zirconium alloys is only one-fifth of that of stainless steels.

Pure zirconium has outstanding corrosion resistance to high temperature water and steam. Zirconium suffers less attack in water at 400°F than stainless steel. It also has good resistance to liquid sodium up to 1100°F. Small amounts of impurities can seriously impair its corrosion resistance to high temperature water, nitrogen, oxygen and carbon being particularly harmful. Fortunately, alloying elements such as tantalum, niobium and tin, offset the detrimental effects of nitrogen, carbon and oxygen and improve the corrosion resistance.

Although Zircalloy-2, has excellent corrosion resistance, one problem, associated with it, occurs as a result of its retention of hydrogen. The hydrogen forms a brittle hydride which *precipitates, during cooling from 600°F, in the form of thin* plates. The plates cause brittleness and the whole effect is known as hydrogen embrittlement. Thus the effect of the hydrogen absorption is not noticed except at lower temperatures. Consequently there are restrictions on fuel changing at low temperatures. Another alloy, Zircalloy-4, has been developed in which nickel has been eliminated in an attempt to prevent hydrogen embrittlement.

Another problem recently encountered with the zircalloys is the accelerated creep rate believed to occur in high fast neutron fluxes. A new zirconium-niobium alloy has now been developed which, it is hoped, will not be subject to this accelerated creep. The pressure tubes of the Gentilly reactor will be made of this material.

The main disadvantages of using zirconium alloys are the high costs involved and the temperature limitations. These materials cannot be used above 750°F (400°C).

(c) Carbon Steels

Carbon steel is an important reactor material for use in pressure vessels and other components where corrosion resistance and low neutron absorption are not important, but where an ability to withstand thermal stresses is desirable. Where some corrosion resistance may be required, the vessel may be clad on the inside with a protective layer such as of stainless steel.

Steels for thick-walled pressure vessels could be more accurately described as high strength, low-alloy steels rather than as carbon steels. Small amounts of alloying materials, such as manganese, silicon or molybdenum, may be present to improve mechanical properties. Tensile strengths up to 100,000 psi and yield strengths up to 50,000 psi, with 20% or so elongation, may then be obtained.

Prolonged exposure of low-alloy steels to high fast neutron fluxes (the equivalent of 1 year's exposure to a flux of 10^{13} neutrons/cm^2/sec) can cause an appreciable increase in the tensile strength and an appreciable decrease in the ductility. What is, perhaps, more important is the increase in the brittle-to-ductile transition temperature. Although loss of ductility reduces the ability of the steel to accommodate thermal stresses, the increase in the transition temperature could be much more serious. After prolonged exposure to fast neutron irradiation, a drop in temperature might cause the vessel to suffer brittle fracture because of this increase in the transition temperature. The effects of radiation damage may be annealed out at higher temperatures but too little is really known about this to rule out a brittle fracture possibility.

(d) Stainless Steels

Stainless steels are a familiar group of materials, being known for their excellent corrosion resistance and their strength retention at high temperatures. Since the thermal efficiency of the system increases with heat transport temperature, the use of stainless steel is an attractive possibility. However, a neutron absorption cross-section of 3 barns, limits the use of stainless steel in thermal reactors to reactors using enriched uranium fuel. They are particularly suited for use in fast reactors.

Stainless steel melting points are in the range 2500 - 2750°F. Their thermal conductivities vary from about 7 to about 21 Btu/hr ft^2-°F and are generally rather low. This poor heat conductivity tends to give rise to high thermal stresses and indicates the unsuitability of stainless steels for thick structural components. They have excellent mechanical properties at high temperatures, the tensile strength of 1200°F being 50,000 psi. Thus, they are far superior in this respect to aluminum and zirconium and their alloys. They show substantial increase in hardness due to fast neutron irradiation.

Stainless steels are readily fabricated into a variety of shapes and they can also be cast. As has been said they are very corrosion resistant at room temperature and will maintain this resistance to 1600°F or 1800°F depending on the chromium content. They are not generally desirable for containing liquid metals but they are particularly resistant to liquid sodium or Sodium-potassium alloy.

The nickel used in stainless steel manufacture should have a low cobalt content because of the high absorption cross-section of cobalt and the formation of radioactive Co-60. This, however, tends to increase the price of the resulting steel.

Chlorine causes stress-corrosion cracking of stainless steel, particularly under stagnant conditions. The corrosion resistance of the steel is adversely affected. It is, therefore, imperative that stainless steel be kept free of any chlorine. It should be noted that chlorine can be introduced from chlorides in solvents sponges, organic gaskets and packings and in some types of insulations.

(e) High-temperature Materials

From the standpoint of neutron economy, most high-temperature materials would find use mainly in fast reactors. Titanium, vanadium, inconnel and the Hastelloys (nickel-molybdenum alloys) are examples of such materials. There is a special interest in titanium because of its high strength-to-weight ratio up to 842°F. It makes it attractive for use where weight-saving is important. Many of the alloys in this group, such as vitallium (chromium-cobalt - nickel - molybdenum), are much superior to stainless steel at high temperature. However, they are generally expensive and may also present fabrication problems.

Design and Construction Considerations

The design and constructional considerations required and the choice of materials for a reactor will be largely determined by the type of reactor being built. Such considerations will, therefore, be discussed as they pertain to each general type of reactor.

(a) Fast Reactors

No moderator is used in fast reactors and the only neutron slowing down process is the inelastic scattering in fuel, structural material and heat transport fluid. Highly enriched fuel is required and the core is basically fuel subassemblies with heat transport fluid flowing between them. The structural material in the core forms a relatively small percentage of the total core material, eg, a typical core composition would be 25% fuel, 50% heat transport fluid and 25% structural material and cladding. Neutron losses, in a fast reactor, are rather insensitive to the amount of heat transport fluid and structural material used. For instance in the Enrico-Fermi reactor less than 0.1% of the neutrons are absorbed in the sodium and only about 1% are absorbed in structural material. So choice of core material is limited more by their neutron slowing down capabilities than by neutron absorption considerations.

Because of the absence of moderator the core is small and the power density in the core is high. To permit operation at high temperatures without high pressures, and also in order to avoid slowing down of neutrons, liquid metals appear to be the most practical heat transport fluids. The amount of fuel subdivision and the size of heat transport passages involved is illustrated in Fig. 1. The heat transport passage may well be as small as 1/8 inch in diameter or less and plugging of these passages could easily occur by material deposition. Such plugging does not seem to be a problem with liquid metal heat transport fluid, since any dissolved metal from structural components deposits preferentially on cold rather than hot surfaces. This is another factor in favour of liquid metals.

The structural material used must have corrosion resistance to liquid metals. With sodium, many materials have good corrosion resistance at the temperature involved, provided the oxygen content is low. Low-carbon iron, chromium iron, stainless steel, nickel and Inconnel and zirconium are all *compatible with sodium and* the final choice would be made on the basis of cost and mechanical properties at elevated temperatures. In this type of core the problem of mechanical strength is more important for the core-supporting components. In the Enrico-Fermi reactor core the structural materials are either zirconium or, where greater strength is required, stainless steel. The reactor vessel, which is only pressurized to 110 psig, is made from stainless steel. Thermal stresses in the vessel are prevented by surrounding the core with a stainless steel thermal shield.

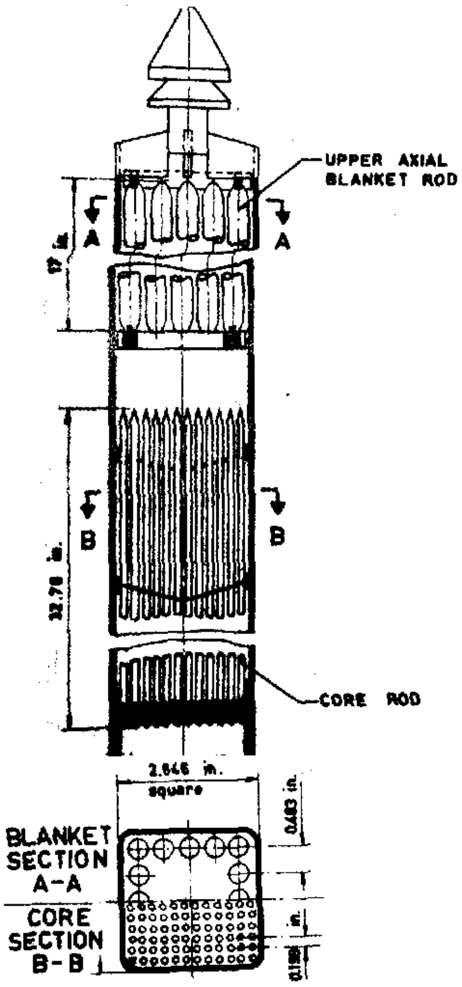


Fig. 1

(b) Homogeneous Reactors

In thermal homogeneous reactors, the reactor is essentially a large pot which is just an enlargement in the circulating system. There are no fuel elements and, consequently, no superfluous structural materials are required. If the reactor is a one region system, with a reactor core but no blanket, then choice of material is only required for the reactor vessel, which is entirely outside the neutron conservation region. 2000 psig pressurization is required and the most suitable material for the vessel would be a low-alloy steel particularly in view of the thermal stresses involved. However, the major problem involved is corrosion. If a uranyl sulphate solution in heavy water is used, the reactor vessel would have to be lined with stainless steel. Where phosphoric acid is used, as the solvent, gold or platinum lining would be required.

If the core is surrounded by a blanket which is also pressurized, the core and blanket would be separated by the core vessel. Since both regions are more or less equally pressurized, the core vessel would not be required to withstand a pressure differential of any significance and corrosion resistance and thermal neutron economy would be the major considerations. Under such circumstances an alloy of zirconium such as Zircalloy-2 would be the most suitable material. For highly corrosive fuel and blanket, such as solutions in phosphoric acid, gold or platinum cladding or lining would be required. The blanket would be contained in an outer pressure vessel the material for which would be chosen on the same basis as for a one region system.

(c) Thermal Heterogeneous Reactors

Heterogeneous reactors can be divided, structurally, into two main types, the pressure vessel type and the pressure tube type.

In the pressure vessel arrangement, illustrated in Figure 2, the reactor core and associated equipment are enclosed in a large vessel. The heat transport fluid enters the vessel, passes through the core to remove the heat produced in the fuel and then passes out of the vessel to a boiler or directly to the turbine.

If the heat transport fluid is light water, as in the pressurized or boiling light water reactors in Figures 5 and 6 of the previous lesson, light water is also the moderator. Fuel enrichment is required and if uranium oxide is used as fuel, the enrichment will be higher. The fuel channels are only required to guide the heat transport fluid over the fuel and do not have to withstand any pressurization. The tubes can, therefore, be thin and this helps to conserve neutrons. Since the operating temperatures are not excessive, the core structural material can be made of a Zirconium alloy for further neutron economy. Many reactors of this type have stainless steel structural components. The thicknesses required are then smaller but the enrichment required increases.

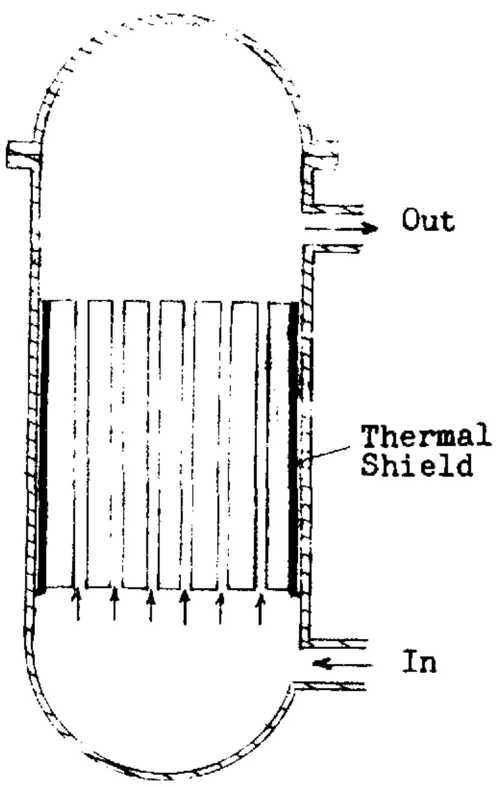


Fig. 2

If the reactor has a nuclear superheat section, like the Pathfinder reactor, the structural material in the superheat region of the core would be made of stainless steel.

In the Swedish R-3/ADAM reactor which is a pressure vessel type using heavy water, all core structural components are made of zirconium alloys. Stainless steel could be used but fuel enrichment would then be required. This would negate the main advantage of using heavy water ,ie, that of conserving neutrons and being able to use natural uranium and in the oxide form.

In graphite moderated reactors, using gas as the heat transport fluid, the graphite forms its own core structure and little other structural components are required except to support the core. Where such components are required, they would be made of magnesium or aluminum alloys. If higher temperatures are attained as in the Advanced Gas-Cooled Reactor or in the HNPF reactor, which uses sodium as the heat transfer fluid, stainless steel structural components are used for more reliable high temperature strength.

In all cases the best material for the pressure vessel is low-alloy steel. In some cases the inside wall of the vessel is lined with stainless steel to reduce corrosion. The core may also be surrounded by a thermal shield to reduce thermal stresses in the vessel. The thermal shield may be of carbon steel, stainless steel clad carbon steel, or stainless steel depending on the amount of corrosion to be expected.

In the pressure tube design only the heat transport fluid is pressurized, the moderator being separated from it. The reactor vessel, or calandria, is then merely a low pressure container for the moderator, as shown in Figure 3.

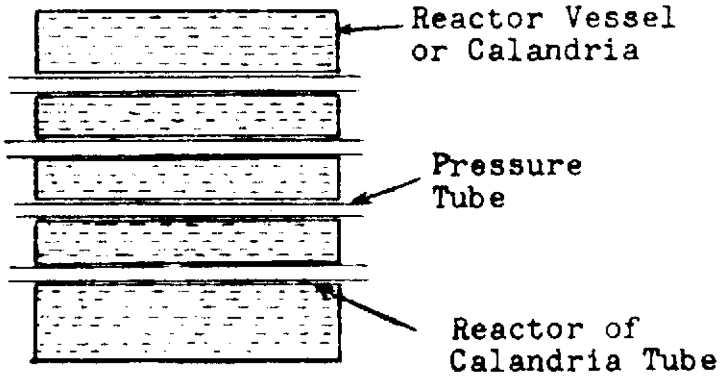


Fig. 3

The fuel is contained in channels which run horizontally or vertically through the vessel. These channels, which contain the heat transport fluid under pressure, are known as pressure tubes.

Heat is generated in the moderator by radiation absorption and neutron scattering but, because the moderator is not pressurized, it must be kept cool. The moderator can also receive heat, by convection, radiation and conduction, from the heat transport fluid and this increases the moderator cooling requirements. Transfer of heat from the pressure tubes is minimized by placing the pressure tubes inside calandria or reactor tubes. The space between them is either insulated or cooled by flow of gas. The arrangement is shown enlarged in Figure 4.

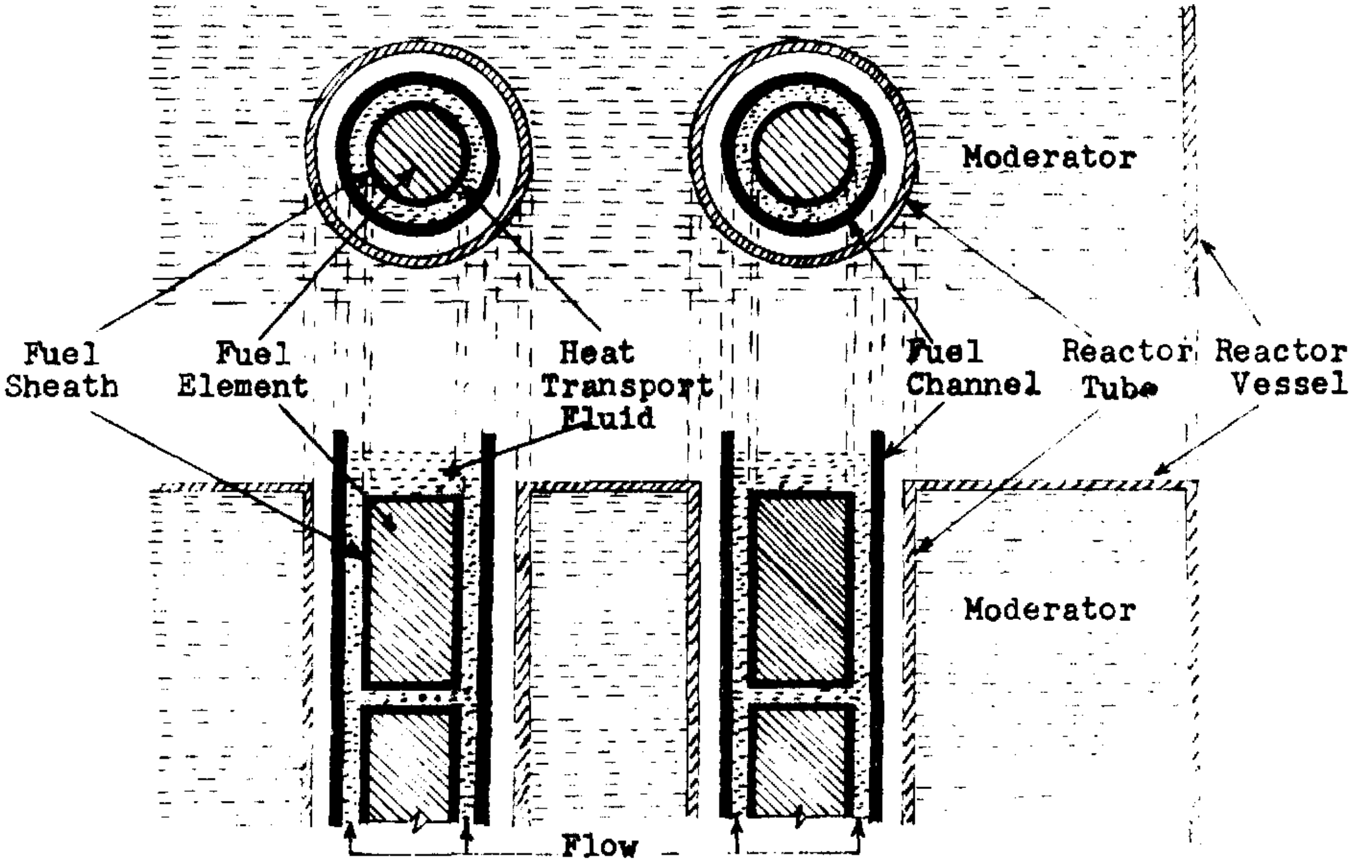


Fig. 4

The thickness of the fuel channels must now be sufficient to withstand the heat transport pressure and this introduces additional neutron absorbing material into the core. Still more such material is to be found in the reactor tubes. The choice of material for these tubes is, therefore, even more important than in the pressure vessel design. This is particularly so in the case of the reactors of this type which have been or are being designed at the present time, since they all use the natural uranium-heavy water concept in which neutron economy is of prime importance.

The reactor tube material will depend on the material used for the calandria since they are welded or rolled to the calandria wall. In a small power reactor, such as NPD, the core is surrounded by a light water reflector, which is separated from the moderator by the inner wall of a double-walled calandria. The inner wall is, therefore, part of the core and is made of aluminum alloy rather than the more expensive zirconium. Thus, the whole calandria is made of aluminum and so are the reactor tubes. The pressure tubes, being at a much higher temperature and pressurized, must be made out of zircalloy. All other core penetrations, required for reactivity control etc, are made of aluminum.

The Douglas Point and Pickering reactors are bigger and use only the outer region of the D_2O moderator as a reflector. The calandria then becomes a single-walled structure entirely outside the core. It can, therefore be made of stainless steel for added strength and corrosion resistance. The reactor tubes then have to be made of zircalloy but they would not be required to be as thick as the pressure tubes. Aluminum and stainless steel are not compatible. **All core penetration must also be of zircalloy.**

ASSIGNMENT

1. Why may material requirements for core structural components differ from those for structural components external to the core?

2. (a) On the basis of neutron capture cross-sections, what structural materials are suitable for a thermal reactor?

   (b) What other common material could be considered if the fuel was enriched?

   (c) Why is there a wider choice of structural material for a fast reactor?

3. State two nuclear requirements, other than low neutron capture, for reactor structural materials, and, briefly, explain their significance.

4. Explain why aluminum appears, initially, to be attractive as a structural material and why it is not, in fact, that attractive.

5. Enumerate the factors that make zirconium, or its alloys, useful as reactor materials. What disadvantage does it have?

6. (a) What material would be used for a reactor pressure vessel and why?

6. (b) Why would the same material not be used for pressure tubes?

7. What is the most likely material to be used for core structural components in a fast reactor and what factors would decide this?

8. Briefly compare the core structural material requirement for a pressure vessel and a pressure tube type of reactor.

A. Williams

Reactor Boiler and Auxiliaries - Course 133

MODERATOR REQUIREMENTS

The advantages of slowing down and thermalizing neutrons before allowing them to cause fission, were considered during the discussions of reactor classifications. It was seen that the fission cross-section for thermal neutrons is so much greater than the radiative capture cross-section that the high fuel enrichment, required in fast reactors, is no longer necessary. In heterogeneous thermal reactor systems, little or no enrichment is required.

The slowing down of fission neutrons to thermal energies takes place in two stages:-

(a) Inelastic scattering by the heavier nuclei, such as U-238, which are already present in the fuel. During this stage the neutron energy is only reduced to about 0.1 Mev and so, further slowing down of the neutrons is required.

(b) Further slowing down of neutrons, below 0.1 Mev, occurs by elastic scattering of the neutrons by the lighter nuclei of the moderator.

The basic requirements of moderators will now be discussed at greater length and the suitability of substances, as moderators, will be considered.

Nuclear Considerations

The first requirement of a moderator is that it should slow down neutrons, to thermal energies, quickly. The energy loss at each collision, between the neutron and moderator nuclei, should be as large as possible. This decreases the resonance capture and increases the probability of fission. It also decreases the chances of radiative capture generally. It has already been established that the lighter nuclei are the most effective in slowing down neutrons rapidly. As may be seen, from Table 1, which follows later in the lesson, a neutron would be slowed down to thermal energies after 18 collisions with hydrogen nuclei whereas it would require 25 collisions with deuterium nuclei, 43 with helium, 105 with boron and 114 with carbon.

The effectiveness of a material in slowing down neutrons, by elastic scattering, is measured by the Average Logarithmic Energy Decrement (ξ). This quantity, ξ, is the average decrease in the natural logarithm of the neutron energy per collision.

The number of collisions (N) required to thermalize a fission neutron is then given by $N = \frac{18.2}{\xi}$

This is not the only factor that determines the suitability of a material as a moderator. Consideration must also be given to the following:-

(a) A nucleus may be very effective in slowing down a neutron when a neutron collides with it, but this is of no value at all unless these scattering collisions take place. This leads to two further requirements, which are:-

  (i) The moderator cannot be a gas since the density of the nuclei in a gas is much too small for frequent scattering collisions to take place. This means that the two best substances, hydrogen and deuterium gases, cannot be used as moderators. They are available, in liquid form, combined with oxygen, in water (H_2O) and heavy water (D_2O). So the moderator must be a solid or a liquid.

  (ii) The moderator substance must have a large scattering cross-section to ensure frequent scattering collisions.

The overall efficiency of a substance, for slowing down neutrons, is measured by the SLOWING DOWN POWER, ($\xi \Sigma_s$). This slowing down power takes the frequency of the scattering collisions into consideration as well as the energy loss at each collision. From the table, it is clear that light water (H_2O) is the most effective slowing down medium followed by beryllium, heavy water (D_2O), carbon and boron. The slowing down power of the gas, helium, is shown for comparison.

(b) Not only must the moderator be effective as a slowing down medium but it must, also, have a small capture cross-section. Neutrons are slowed down to decrease radiative captures compared to fission captures and, therefore, the whole object of moderation would be defeated if the moderator nuclei themselves, capture neutrons.

The MODERATING RATIO, $\frac{\xi \Sigma_s}{\Sigma_a}$ is the ratio of the slowing down power to the absorption cross-section. This quantity is, therefore, a better indication of the overall suitability of a material as a moderator.

TABLE 1

Material	Number of Collisions to Thermalize	Slowing-down Power	Moderating Ratio
Hydrogen	18		
Deuterium	25		
Beryllium	86	0.176	159
Boron	105	0.06	0.0009
Carbon	114	0.064	170
Helium	43	1.6×10^{-5}	83
H_2O	19	1.53	72
D_2O	35	0.170	21,000
Zinconium Hydride		0.8	56
Terphenyl		0.73	80

The above table shows that, although H_2O has the highest slowing down power, its moderating ratio is lower than that of carbon and beryllium. The moderating ratio of H_2O is so low that it can not be used as a moderator to sustain a chain reaction unless it is used with enriched fuel. The superiority of D_2O, however, is very apparent and it can be seen why D_2O is the only moderator which could be used in a natural uranium homogeneous system.

(c) The substance used as a moderator must be very pure. It is usually used, in a reactor, in larger amounts than any other material e.g. the volume of carbon, in a graphite moderated reactor, is 70 - 80 times that of the fuel. A very small amount of impurity in a moderator can substantially increase its capture cross-section. The addition of 1 boron atom to every million graphite atoms would increase the capture cross-section of graphite by 25%.

> For the same reason the isotopic purity of D_2O must be kept high. The addition of 0.25% H_2O to pure D_2O more than doubles the capture cross-section. The isotopic purity of D_2O is, therefore, kept at between 99.75% and 99.8% D_2O by weight. A 1% decrease in the isotopic purity of the NPD heavy water would cause a 25mk decrease in the reactivity. This would result in a loss of fuel burnup estimated to be worth $175,000 per year.

Two other quantities that are closely connected with the moderator properties are the SLOWING DOWN LENGTH and the DIFFUSION LENGTH of the neutrons. As shown in Figure 1 a neutron travels an erratic zig-zag path from A to B while it is being thermalized. The slowing down length, L_s, is a measure of the "crow flight" or direct distance AB travelled by the neutron while being thermalized.

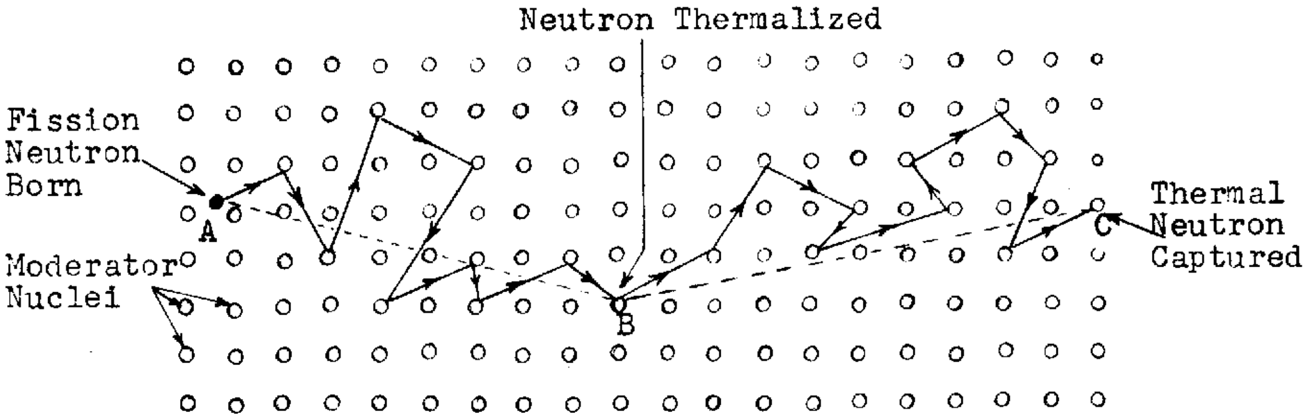


Fig. 1

$$L_s^2 = \frac{1}{6} \text{ (average value of } AB^2)$$

After being thermalized at B, the thermal neutron continues to move or diffuse through the reactor before being captured. It follows the zig-zag path BC. The diffusion length, L, of a thermal neutron is a measure of the "crow flight" distance BC travelled by the thermalized neutron before being captured.

$$L^2 = \frac{1}{6} \text{ (average value of } BC^2)$$

The slowing down length determines the extent to which fast neutrons escape from the reactor and it is, therefore, a factor in determining the critical size of the reactor. It also determines how much moderator is required between fuel channels to thermalize the neutrons and keep p, the resonance escape probability, as high as possible, i.e. it determines the reactor pitch. Ideally, then, L_s should be as small as possible.

The diffusion length determines how fast the thermal neutrons are lost by capture. It also determines the extent to which thermal neutrons escape or leak out of the reactor but since the majority of thermal neutrons are captured in the fuel, or in core material the former factor is the more important of the two. Therefore, L, should be as large as possible in order to have a high value of f, the thermal utilization factor.

The slowing down lengths and the diffusion lengths for possible moderator materials are given in the following table.

TABLE 2

Material	L(cm)	L_s(cm)
Water (H_2O)	2.8	5.6
Heavy Water (99.84% D_2O)	100	11
Beryllium (ρ = 1.85)	21	9.2
Beryllium Oxide (ρ = 3.0)	29	10
Graphite (ρ = 1.6)	64	19

The temperature of the moderator may well affect the values of L, L_s and other nuclear parameters. An increase in temperature causes a decrease in density which, in turn, causes an increase in both L and L_s. This tends to increase neutron leakage and to increase the value of f. An increase in temperature also causes an increase in the molecular energy so that the thermal energy of the neutrons will be increased. Thus the energy of the neutrons entering the fuel will be higher. In uranium fuel this would decrease the U-235 fission captures and increase the U-238 resonance captures i.e. decrease p. As plutonium builds up in the fuel the increase in neutron energy increase plutonium fission captures. Neutron economy is therefore, improved by keeping the moderator at a moderate temperature, particularly with fresh fuel. This is of particular significance in heavy water moderated reactors where neutron economy is of such importance.

Non-nuclear Considerations

Numerous chemical and physical properties of possible moderator materials must be considered in addition to the nuclear properties considered above. The following are some of the more important considerations:-

(a) The material should be chemically inert to its environment. In the case of a liquid it should not cause corrosion of the containing system nor cause scale formation. Corrosion products can cause flow reductions and they also become radioactive and may lead to contamination.

Solid moderators should not oxidize, even at elevated temperatures, nor should they react with the heat transport fluid.

(b) Liquid moderators should have as high a boiling point as possible particularly if a common moderator -- heat transport fluid is being used. Little or no pressurization is then required.

Solid moderators must have high melting points.

(c) Where solid moderators are also used as structural materials they should have good, thermal conductivities, low thermal expansion, high tensile and compressive strengths and have good resistance to thermal shock and creep.

These desirable characteristics must be maintained at elevated temperatures.

(d) Materials used as moderators should be non-toxic since additional costs are involved in the production, fabrication and containment of toxic materials.

(e) The material should be resistant to radiation damage. In particular, solids should not distort or expand under irradiation nor should their desirable physical and mechanical properties be affected.

Radiation should not cause dissociation, cross-linking, tar or coke formation or any excessive changes in the desirable physical properties of liquids.

(f) It must be readily available at a reasonable cost.

ASSIGNMENT

1. (a) Of what significance are the "Average Logarithmic Energy Decrement", the "Slowing Down Power" and the "Moderating Ratio" in determining the suitability of a material as a moderator?

   (b) Helium gas requires only 43 collisions to thermalize a neutron compared with 114 with carbon. Why is helium gas not considered as a moderator?

   (c) The slowing down power of boron is 0.06 whereas that of carbon is 0.064. Which would be considered the better moderator and why?

2. (a) Why is the purity of a moderator material such an important factor?

   (b) Of what significance is this factor in the choice of the graphite used in a reactor?

3. (a) Of what significance are the "Slowing Down Length" and the "Diffusion Length" in reactor design and, ideally, what values should they have?

A. Williams

Reactor Boiler and Auxiliaries - Course 133

MODERATOR PROPERTIES AND COMPARISON

The only substances which could be considered as possible moderators are light water (H_2O), heavy water (D_2O), carbon (as graphite), beryllium, beryllium oxide, beryllium carbide, and hydrogen compounds such as the hydrocarbons or zirconium hydride. The properties of these materials will now be considered and their suitabilities as moderators discussed.

Carbon (Graphite)

The first reactor built used carbon in the form of graphite, as the moderator. It has been used extensively since as both moderator and reflector, particularly in the United Kingdom. Graphite occurs in considerable quantity in nature but the impurities contained in natural graphite affect neutron economy to such an extent that it is unsuitable as a moderator. Reactor-grade graphite is, therefore, produced artificially by mixing petroleum coke with a filler (coal-tar pitch), extruding the mixture into bars and baking the bars in gas or electric furnaces at temperatures up to 1500°C. This carbonizes the pitch and sets the binder and produces what is known as industrial carbon. The bulk density is then increased by impregnating with pitch under vacuum and re-baking. This is followed by a reheating treatment in an electric furnace at 2700°C to graphitize the product. Careful elimination of all neutron absorbers, in the raw material, is required if reactor-grade graphite is required. Even so the presence of such materials as boron is still likely and the extruded pieces have to be graded according to impurity content. The highest quality graphite is then used in the centre of the reactor and the lowest quality graphite on the outside.

The density of artificially produced graphite is much lower than that of natural graphite, its specific gravity being 1.6 compared with a theoretically attainable value of 2.25. This increases the volume of graphite moderator required. Table 1 below lists some important properties of artificial reactor-grade graphite at ordinary temperatures. These are typical values only and can vary considerably depending on the method of production and the raw materials used.

TABLE 1

Property	Parallel to extrusion axis	Perpendicular to extrusion axis
Coefficient of thermal expansion.	0.7×10^{-6} per°F	1.5×10^{-6} per°F
Thermal conductivity (Btu/hr-ft-°F)	1.4×10^{4}	1.1×10^{4}
Tensile strength	2000 psi	700 psi
Compressive strength	6000 psi	6000 psi
Sublimation temperature	6600 °F	

The advantages of using graphite as a moderator are:-

(1) Its moderating ratio is second only to heavy water and it can, therefore, be used as a moderator in a heterogeneous system using natural uranium metal as fuel. However the moderating ratio is not large enough to enable it to be used with oxide or carbide fuel.

(2) Graphite, of reactor grade,is readily available at reasonable cost.

(3) It has good mechanical properties and can be used as a reactor structural material without having to be contained. The fuel channels themselves can be holes in the graphite structure or can be supported by the graphite. Its use as a structural material is enhanced by the fact that it is easily machined to close tolerances.

(4) It has a high thermal conductivity so that heat produced in the graphite, by radiation absorption, is easily dissipated and removed.

(5) Graphite has excellent resistance to thermal shock.

(6) It has excellent high-temperature properties, its tensile strength, for instance, increasing up to 4500°F. This factor combined with its thermal shock resistance, makes it potentially valuable in a high temperature reactor.

(7) Little oxidation occurs below 750°F or so and no chemical reaction occurs at such temperatures between graphite and carbon dioxide, used as a heat transport fluid.

The use of graphite as a moderator has the following disadvantages:

(1) It has a relatively long slowing down length so that fuel channels must be fairly widely separated for effective moderation. This factor combined with its relatively small moderating ratio results in a large size core compared with cores using light or heavy water as moderator. The United Kingdom reactors have cores 40 to 50 ft in diameter and up to 25 ft in height. Such massive graphite structures pose problems of support and containment and the large steel pressure vessel required sets limits on the possible vessel thickness and, consequently, on heat transport fluid pressure. The replacement of steel by prestressed concrete has, to some extent, eliminated these problems.

(2) The small moderating ratio precludes the use of uranium dioxide fuel unless the fuel is enriched. Severe limitations are also placed on the materials that can be used in the core for such things as fuel sheaths. These factors result in moderate heat transport temperatures and low fuel burnups.

(3) The coefficient of thermal expansion of graphite parallel to the extrusion axis differs from that perpendicular to it. Care must therefore be exercised in allowing for this differential expansion when the graphite blocks are assembled.

(4) There is not decrease in strength of graphite under irradiation. However, dimensional changes occur. These dimensional changes also differ parallel to and perpendicular to the extrusion axis. The graphite expands perpendicular to this axis but it may expand or contract, in a direction parallel to the axis, depending on the irradiation exposure. These effects must be allowed for in the design of the reactor to prevent distortion of the fuel channels during operation.

The thermal conductivity decreases under irradiation and can become a factor of 40 or more lower than it was before reactor startup. This could have a marked effect on heat dissipation and removal.

Irradiation of graphite causes an accumulation of stored energy in the graphite lattice, which is sometimes called Wigner energy. If this was allowed to go unchecked, it would continue to accumulate until a very unstable condition is reached. At this point all the energy is suddenly released causing the graphite temperature to rise to perhaps 1800°F or more.

These radiation effects are less marked at temperatures above 660°F or so. However, it is the practice in graphite moderated reactors to operate at fairly low temperatures. This permits the temperature to be raised, at predetermined intervals, to anneal the radiation damage and release the stored energy. The temperature rise resulting from this energy release is controlled by adjusting the heat transport flow. The Windscale reactor was seriously damaged by local overheating which occurred during such a periodic energy release.

(5) Even carbon dioxide reacts with graphite at higher temperatures which again tends to limit the heat transport temperature. The addition of methane to the carbon dioxide helps to inhibit the reaction and raise the temperature limitation.

Graphite is also attacked by liquid metals so that unclad graphite cannot be used with liquid metal heat transport fluids.

Beryllium

Beryllium is the only light metal with a high melting point. The specific gravity varies from 1.81 to 1.86, the higher valves being due to the presence of beryllium oxide, which has a specific gravity of 3.0. The mineral beryl is not found in large deposits and its occurrence is erratic and difficult to predict. Beryllium billets or slabs can be made by melting the metal in a beryllium oxide crucible and pouring into a graphite mold. However, cracks develop very easily and the cast material is not easily fabricated.

It is difficult to machine without causing surface damage. Hence the principle form of beryllium suitable for commercial fabrication is that made by vacuum pressing the hot powder at around 2000°F. The resulting material can be machined quite easily and this is, frequently, all the treatment it receives. However it can be fabricated by extruding, rolling or forging at about 2000°F.

Some of the properties of beryllium, produced by hot-pressing, are listed in Table 2 below.

TABLE 2

Tensile strength	45,000 psi
Melting point	2,340 °F
Coeff. of expansion	9.5×10^{-6} per °F
Thermal conductivity	2.85×10^{4} Btu/hr-ft-°F)

Beryllium has a slightly smaller moderating ratio than carbon and, like graphite, it could only be used with natural uranium metal fuel. However, it has a much better slowing down power and a shorter slowing down length. It would therefore lend itself to the design of smaller reactors with slightly enriched fuel. It is particularly attractive as a moderator because it is a metal and it would appear particularly suitable as a moderator in a space vehicle power unit.

It does not suffer serious attack in air below 1100°F but could not be recommended, in air, above this temperature. The corrosion rate of pure beryllium in deaerated, deionized water at 600°F is relatively small and decreases with time due to the formation of a protective oxide film. It has good resistance to molten sodium in the absence of oxygen. In addition it has a high thermal conductivity.

The use of beryllium in reactor construction has been limited for the following reasons:-

(1) It is an expensive material.

(2) It is brittle and its ductility may be zero under certain conditions.

(3) It is extremely toxic and this complicates its fabrication procedures.

(4) Helium gas is formed in the material by the following reactions:-

$$ {}_4\mathrm{Be}^9 + {}_0n^1 = {}_2\mathrm{He}^6 + {}_2\mathrm{He}^4 \quad (\alpha) $$

$$ {}_4\mathrm{Be}^9 + {}_0n^1 = \underset{\substack{\downarrow \\ 2\,{}_2\mathrm{He}^4\ (2\alpha)}}{{}_4\mathrm{Be}^8} + 2\,{}_0n^1 $$

This gas may collect as bubbles and cause local swelling.

An alternative material to beryllium metal is beryllium oxide (BeO). Apart from its good moderating properties, beryllium oxide has the advantages of having a high melting point (4620°F), low vapour pressure in a dry atmosphere and excellent thermal shock resistance for a ceramic material. It also has a very high thermal conductivity for a ceramic (2.1×10^4 Btu/hr-ft-°F) and is stable and inert to most materials. Its tensile strength (15,000 psi) and its compressive strength (114,000 psi) at ordinary temperatures are far superior to graphite.

However, one reason why it has not been more widely employed is that, when hot, it vapourizes in moist air due to the formation of the hydroxide. Like beryllium metal it is toxic. Its strength and thermal conductivity decrease with increase in temperature. The change in thermal conductivity is so drastic that, at 1800°F, it is only one-tenth of the value quoted above. It has question*able resistance to thermal stresses. Unlike the metal, it is* dimensionally unstable under irradiation, the dimensional changes noted being of the order of 1%.

Beryllium carbide (Be_2C) is another potential moderating material, having a specific gravity of 2.44. It is quite stable under irradiation. However, it is also toxic and is very reactive with water and water vapour even at ordinary temperatures. It also reacts with oxygen and nitrogen at elevated temperatures and it would, therefore, have to be clad to overcome this problem. It has poor resistance to thermal shock.

Light Water (H_2O)

Many reactors, currently in operation, use light water as the moderator. In most cases the water also serves as the heat transport fluid and even as shielding material. However, only its properties pertinent to its use as a moderator will be discussed here. The additional requirements of water as a heat transport fluid or as a shielding material are discussed elsewhere.

Light water is attractive as a moderator because of its low cost, its excellent slowing down power and its small slowing down length. However its moderating ratio is too low to permit it to be used with natural uranium fuel even in metal form. Some slight enrichment of the fuel is required. However, if enrichment facilities are available, the small slowing down length makes it possible to design a reactor of relatively small size. The fact that water can be used as both moderator and heat transport fluid may be considered an added advantage. A single fluid system only is then required and the fuel channels in the reactor need only be thin tubes required to guide the water over the fuel. This helps to decrease neutron capture in the core material. However, it also means that a pressure vessel concept has to be used with all its inherent disadvantages.

Water melts at 32°F and will not, therefore, solidify during reactor shutdown. It boils at 212°F and requires pressurizing at elevated temperatures if boiling is to be prevented. This is not a disadvantage if the moderator is separated from the heat transport fluid and, consequently kept at a low temperature. Water has a relatively poor thermal conductivity (0.32 Btu/hr-ft-°F) but resonable heat transfer rates can be obtained with water.

The water must be free from impurities since these not only capture neutrons but they also become radioactive as a result of neutron capture. This may result in high radiation fields near the moderator system and may also cause permanent contamination of the system. It is relatively easy to circulate water through filters and demineralizers to remove such impurities. However radioactive nuclei (O-19 and N-16) are still produced as a result of neutron capture in oxygen nuclei. This prevents access to the moderator system during reactor operation but, because of the short half-lives of these radioactive nuclei, access to the moderator system is possible shortly after reactor shutdown. Water treatment is also important to minimize corrosion and scale formation. Careful control of pH is required if aluminum or carbon steel is used in the moderator system, otherwise the choice of material would be limited to stainless steel or zirconium. Chemical or heat treatment of the system may also be required in order to form a protective oxide layer on the pipe surfaces and this is normally done during commissioning.

Water does not decompose at elevated temperature but it does however, decompose when irradiated according to the equation:-

$$H_2O = H + OH$$

The H and OH may recombine thus:-

$$H + OH = H_2O$$

in which case there is no net decomposition, or they may combine as follows:-

$$H + H = H_2$$

$$OH + OH = H_2O_2$$

The hydrogen peroxide can then decompose according to:-

$$H_2O_2 = H_2O + 1/2\ O_2$$

The overall result is the formation of hydrogen, oxygen and hydrogen peroxide. Dissolved impurities in the water increases the severity of the radiolytic damage.

There are several consequences of this radiolytic decomposition. The oxygen formed may dissolve in the water and cause increased corrosion. If the hydrogen concentration is allowed to increase, it could become an explosion hazard in the presence of oxygen. If the gases produced are allowed to accumulate, a steady state is eventually reached when the rate of recombination is equal to the rate of dissociation. Should the gas concentrations, at this point, be tolerable no further action is required. However, if the concentrations are excessive, recombination units must be provided to recombine the hydrogen and oxygen to form water.

Heavy Water (D_2O)

Heavy water has an exceptionally high moderating ratio and a long diffusion length. Its slowing down length is shorter than that of graphite and about twice that of light water. Therefore, the fuel channel separation with heavy water should be larger than with light water and smaller than with graphite. However, in D_2O moderated reactors, advantage is frequently taken of the low neutron capture in heavy water. The fuel channel separation is made larger than the minimum required for the neutron of average energy. This ensures that most of the neutrons become thermalized before entering the fuel and this increases the resonance escape probability (p). The reactor is then said to be overmoderated and the reactor pitch may then be equal to or greater than the pitch in a graphite moderated reactor. However, the size of the D_2O reactor is still considerably less than the graphite reactor. The main advantage of using heavy water as the moderator is that it enables natural uranium oxide fuel to be used in a reactor the size of which is still considerably smaller than the graphite moderated natural uranium reactor. The oxide fuel improves the fuel integrity and enables high fuel burnups to be achieved without enrichment.

Heavy water, as a moderator, has similar physical characteristics to light water. The same problems occur with radiolytic decomposition and it is even more important to ensure that the D_2O is free from impurities. The same corrosion problems can also occur. However, there are some factors involved which are peculiar to heavy water. These are:-

(1) The high cost of heavy water which increases the initial capital cost of the reactor.

(2) The limitation that is placed on core material in order to maintain neutron economy and to ensure that natural UO_2 fuel can be used and high burnups achieved. Core structural material must be aluminum, zirconium or their alloys.

(3) The necessity of keeping heavy water losses to a minimum. Since radiolytic decomposition involves a loss of heavy water, recombination units are required to recover it. The heavy water system should, wherever possible, be of welded construction. D_2O leak detection systems are required. Since leakage can not be entirely avoided, (e.g. valve stems, pump seals etc), heavy water collection systems are required to recover the water with a minimum of downgrading. Heavy water recovery systems are necessary in heavy water areas where leakages are possible and the ventilation systems for these areas are closed circulation systems with dryers to remove D_2O vapour from the air. All these requirements tend to increase the initial capital cost of the station.

(4) Tritium is produced in heavy water by neutron capture in deuterium:-

$$_1H^2 + {_0n^1} = {_1H^3} + \gamma$$

The tritium, (H^3 or $_1T^3$), has a 12 year half-life and, therefore, accumulates in the water. It is a beta particle emitter and is absorbed into body tissue like hydrogen, in the form of water (TDO). It is, therefore, a total body emitter, with a 12 day biological half-life and presents a serious health hazard.

Heavy water leakage must be kept to a minimum *because of the tritium hazard as well as for economic* reasons. Care must be taken to avoid leakage of tritiated vapour into accessible areas of the station. *Handling of tritiated heavy water requires special* techniques and precautions to avoid tritium intake into the body. Before entry can be allowed into areas containing heavy water vapour, the area must be purged, and the heavy water loss accepted, or special protective equipment must be worn to prevent tritium intake. Over a long period of time the tritium concentration may become prohibitive necessitating replacing the tritiated water.

(5) Heavy water freezes at 38.9°F and there is a possibility that the heavy water would freeze, during a reactor shutdown, in, say, a heat exchanger where it is cooled by river water or lake water in the winter.

Organic Materials

Organic compounds, such as the polephenyls have moderating properties very similar to light water. The moderating ratio of the terphenyls is 80 compared with 72 for H_2O, whereas the slowing down power is 0.73 compared with 1.53 for H_2O. Like water, they can be used in reactors of relatively small size, although fuel enrichment is required. The following table lists some of the physical properties of organic compounds.

TABLE 3

	Isopropyl Diphenyl	Diphenyl	Santo-wax R	Santo-wax O-M
Melting point (°F)	-40 to +28	157	140-293	100
Boiling point (°F)	552	491	700	650
Vapour pressure at 800°F (psia)	190	220	38	57
Specific gravity at 600°F	0.78	0.87	0.86	0.75
Specific heat at 600°F (Btu/lb-°F)	0.6	0.61	0.60	0.60
Thermal conductivity (Btu/hr-ft.-°F)	0.066	0.066	0.066	0.066

As may be seen, from this table, the santowaxes, which are mixtures of terphenyls, have high boiling points and low vapour pressures at temperatures above their boiling points. They would require very little pressurization up to 800°F. However this property really enhances their suitability as a heat transport fluid, unless they were used as a combined heat transport and moderator fluid in a pressure vessel type of reactor. In fact there seems little advantage in using organic liquids as moderators unless this type of arrangement is used, particularly as they have a fairly high melting point and could not, therefore, be cooled too much. Used in this way, they have the following additional advantages:-

(1) No chemical reactions between organics and fuel, fuel cladding or water.

(2) Negligible corrosion rates permitting the use of standard materials for fuel channels and cladding provided that they are suitable otherwise.

(3) Require only conventional "hot oil" pumps and circuit equipment.

(4) No radioactivity induced by neutron capture in the organic liquids themselves but only in impurities.

(5) Fair high temperature stability up to 800°F.

(6) They are fairly cheap (17 - 20¢ per lb).

Organic liquids do, on the other hand, have the following limitations:-

(1) Most are solid at room temperature and, so, trace-heating or preheating is required in the system.

(2) Organic compounds suffer radiation damage at all temperatures but the damage is greater at elevated temperatures. The effect of radiation is to produce gases, such as hydrogen, and cause polymerization of the molecules which results in the formation of tars or coke or varnish and cause increases in the viscosity. Continual purification, by distillation, is, therefore, required. The concentration of the higher polymers is kept at around 30%. This decreases the rate of radiolytic damage. A degassing system is also required.

From the point of view of radiation damage, the terphenyls are more stable than the diphenyls. Even so, it has been estimated that, for a 240 Mwt reactor with organic moderator and heat transport fluid, the rate of radiolytic damage would be more than 1 lb per Mw Hr at 675°F and that 260 lb/hr of organic make-up would be required to replace the damaged material.

(3) Organic liquids have much lower heat-transfer coefficients than such liquids as water and liquid sodium. The heat transfer properties can be improved by placing lateral fins on the fuel elements but this is really a heat transport problem.

Zirconium Hydride ($Zr\ H_2$)

This compound offers interesting and unusual moderator possibilities. It is prepared by heating zirconium metal in hydrogen gas at about 660°F. When fully hydrided, the hydrogen atom density in it approaches that in water. Consequently, the hydride has good moderating properties, particularly as the zirconium has a small neutron capture cross-section. Its moderating ratio is somewhat lower than light water but its slowing down power is second only to light water. The use of zirconium hydride as a moderator with enriched fuel would, therefore, result in a relatively small reactor.

The material is stable at temperatures below 1000°F. In powder form it has poor thermal conductivity and would require special cooling, in reactors operating at moderate or high power levels, to keep its temperature down. In the TRIGA (Training, Research, Isotope, General Atomics) reactor the hydride is incorporated with the uranium in the fuel elements. The thermal conductivity of the material then approaches that of zirconium metal.

ASSIGNMENT

1. (a) By considering any of the appropriate nuclear parameters compare the suitability of H_2O, D_2O, beryllium, graphite and terphenyls for use of moderators with natural uranium metal and natural uranium oxide.

(b) Also compare the sizes and fuel channel separation that can be obtained with these moderating materials.

(c) Why does the actual pitch in a D_2O moderated reactor frequently not correspond to what would be expected under 1b?

2. (a) Why, in the case of a liquid moderator, would corrosion be undesirable and how might it limit the choice of core structural material?

   (b) How is corrosion controlled in water moderated reactors?

3. (a) What advantages does a solid moderator have over a liquid moderator?

   (b) What additional physical properties must they have under these circumstances?

   (c) How do graphite, beryllium oxide and beryllium metal compare as far as these properties are concerned?

4. How does nuclear radiation affect the properties of

   (a) Graphite

   (b) Beryllium

   (c) Water

   (d) Organics

   and how are the effects of the property changes reduced or minimized?

5. Name three factors, that are peculiar to the use of heavy water as a moderator and indicate, briefly, their effects on station design, capital cost or operating costs.

A. Williams

Reactor Boiler and Auxiliaries - Course 133

MODERATOR SYSTEM & EQUIPMENT CONSIDERATIONS

With solid moderators, such as graphite or beryllium, no circulation of the moderator is possible. Furthermore, any heat removal, from the moderator, that may be required is accomplished by conduction through to the heat transport system. Therefore, no system, as such, is associated with solid moderators.

In reactors using light water or organic liquids as the moderator, the same material is also used as the heat transport fluid, and the two systems are common. In such cases the pertinent considerations are those appropriate to the heat transport system. Some heavy water moderated reactors, such as the Halden reactor in Norway and the R3/ADAM reactor in Sweden, are also pressure vessel reactors which are in the same category.

It is only when the moderator system is separated from the heat transport system that the moderator system and equipment requires particular consideration. Such a separation only exists in the pressure tube type of reactor using heavy water as the moderator and the discussions in this lesson will be confined to this type of system. Many of the principles involved would apply equally well to other fluids if they were used, as moderators, in a pressure tube reactor.

Basic Requirements

The main purpose of the heavy water moderator system is to maintain the reactor critical by thermalizing the neutrons released at fission. In order that it continues to do this effectively, the desirable neutron properties of the moderator must be maintained. This will involve treatment of the moderator in such a way that:

(a) The corrosion product concentration is kept to a minimum.

(b) The temperature is prevented from rising.

Removal of impurities, in the form of corrosion products or other dissolved or suspended solids, is also desirable to avoid them becoming radioactive and preventing access to the system components, especially during reactor shutdown. It is also easier to maintain low corrosion product concentration if corrosion in the system is prevented as far as possible. This means that:

(a) The materials used in the system must be compatible with heavy water.

(b) The pH of the water must be carefully controlled to suit the materials used.

For the reactor to remain critical, it may be necessary to be able to vary the moderator level in the reactor, particularly if the reactor is to be partially or completely regulated by varying the moderator level. A protective system will operate to shutdown or trip the reactor quickly, should certain variables, such as reactor neutron power, exceed some predetermined level. Reactor shutdown is conveniently accomplished by removing the moderating material or by inserting neutron absorbers into the core. It may, therefore, be necessary to empty the moderator from the reactor into a storage tank, as the primary and only method of shutting down the reactor or as a backup to some other means of reactor shutdown.

Since the moderator is contained in the reactor vessel, or calandria, thermal stresses in the vessel can only be avoided or controlled by keeping the moderator temperature within certain limits. It is also necessary to prevent the moderator from boiling and this must be done without applying much pressure to the system so as to avoid having to use a pressure vessel and high pressure piping and components.

The moderator is very likely to be used for other purposes as well. It may well be used to cool absorber rods, booster rods, and even calandria support rods. It is likely to be used for spray cooling in the calandria or dump tank when this is required. Provisions may be necessary for heavy water exchange between the moderator and the heat transport systems either on a normal operational basis or under emergency conditions. The moderator system would also be the source of water for dousing sprays in such areas as the reactor vault and fuelling machine rooms. Heavy water is also supplied to the blower seals or jet exhausters that would be required if a cover gas system is used.

The basic requirements of the moderator system can, therefore, by summarized as follows:

(a) Means must be provided to remove heat from the heavy water moderator so that its temperature may be controlled.

(b) Some method is required to maintain the purity and/or pH of the heavy water.

(c) A storage tank is required with provisions for transferring heavy water from the calandria to the storage tank and from the storage tank to the calandria. The same equipment would enable the moderator level in the calandria to be raised or lowered.

(d) The necessary piping and valves must be installed so that heavy water can be transferred from the moderator system to the auxiliary systems such as spray cooling, absorber and booster rod cooling and the dousing sprays. Exchange of heavy water between the moderator and heat transport systems must be arranged and the necessary piping, valves and connections installed.

(e) Provisions are required for removal and addition of heavy water from or to the system and for the addition of liquid neutron poisons.

(f) The piping and equipment in the system must be made of a material compatible with heavy water and the calandria material. Alternatively the surfaces in contact with the heavy water must be compatible.

(g) Heavy water leakage must be kept to a minimum. Pipe and fitting joints, valve stem packing and seals must be such as to ensure this. Not only does reduction of heavy water losses result in lower operating costs, but it also reduces the tritium hazard to station personnel.

(h) The arrangement of equipment and pipes in the system must be such as to minimize heavy water holdup. This reduces the capital cost of the heavy water required. However, this should not result in such a congested layout that maintenance of equipment and heavy water handling becomes very difficult. Accidental losses of heavy water are then more likely and the savings on heavy water holdup might be partly offset by increased operating costs.

(i) Choice of equipment must be such as to ensure maximum reliability of the system at all times.

Moderator Heat Removal

Heat is generated in the moderator by neutron scattering and radiation absorption. Additional heat is transferred, to the moderator, from the heat transport system. The moderator may also be used to cool booster and control rods. The total heat produced in or transferred to the moderator is about 7% of the total heat produced in the reactor. Thus, in a 20 Mwe station, such as NPD G.S., requiring 82 Mw of thermal generation, the total heat produced in or transferred to the moderator is 5.6 Mw. In a 200 Mwe station, such as Douglas Point G.S., this increases to 50 Mw of heat. If this heat was allowed to accumulate in the moderator it would soon boil. It is, therefore, necessary to remove this heat and keep the moderator at a temperature below its boiling point. The moderator can not, therefore, be allowed to be stagnant in the reactor vessel but must be circulated through a circuit external to the reactor in which the heat is removed.

The simplest type of heat removal circuit, shown in Figure 1, consists of a pump, a heat exchanger or cooler and the connecting pipes. A considerable amount of heat must be removed from the moderator, particularly in a 500 Mwe unit. It seems uneconomical, therefore, to transfer this heat to cooling water, which is later discharged to a lake or river. Unfortunately the moderator temperature is only 120°F to 180°F and it is not practical to use heat energy, at such a low temperature, for, say, building heating. However, serious consideration has been given to using this moderator heat for turbine condensate reheat. The heat exchanger would then replace a low pressure feedwater heater normally heated by steam extracted from the turbine. However, the moderator heat has, to date, been merely removed in a conventional heat exchanger.

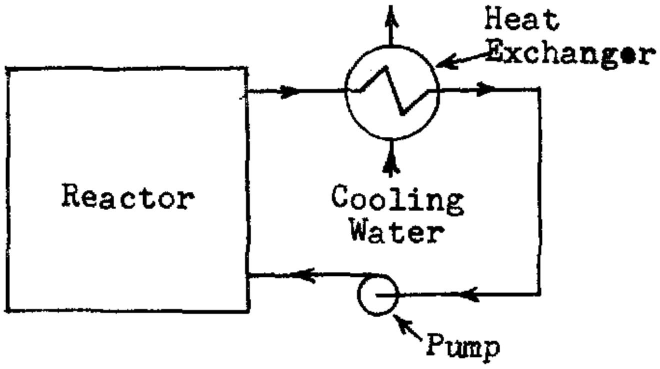


Fig. 1

The basic circuit is rarely as simple as that shown in Fig. 1. Greater reliability is normally ensured by having more than one pump, operating in parallel, as shown in Fig. 2. Multiple heat exchangers may also be used as in Fig. 3.

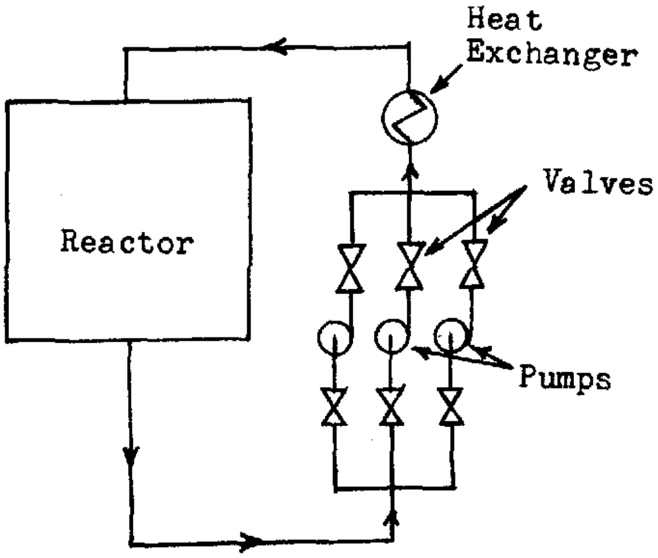


Fig. 2

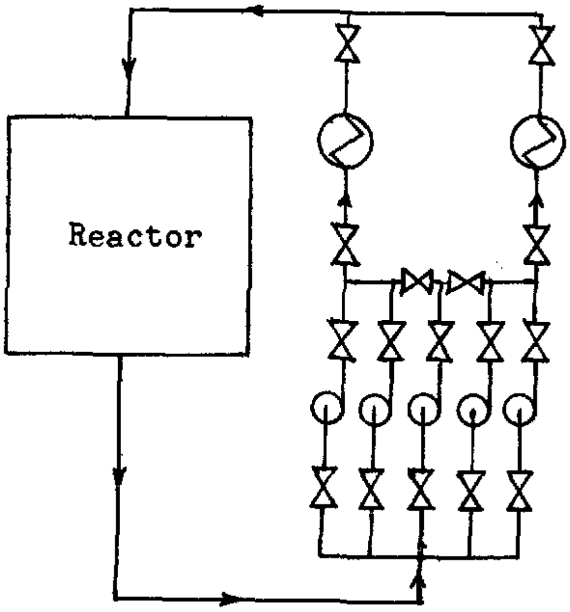


Fig. 3

In Fig. 2, two of the three pumps may be required for normal operation with the third acting as a standby pump in case one pump becomes defective. In Fig. 3, two pairs of pumps may be normally operating with the centre pump acting as a standby for either pair. On reactor shutdown, the moderator sprays are still cooling the calandria walls and tubes and the booster and control rod cooling is still required. For example, in a 200 Mwe unit, 4 Mw of heat must still be removed from the moderator when the reactor is shut down and 5 Mw must be removed if the booster rod has been recently used. So, in Fig. 3, only two of the five pumps may be required and only one heat exchanger. In Fig. 2 the different heat removal requirements would be allowed for by changing the cooling water flow.

It may now be seen that valves are required in the circuit for pump isolation and interconnection and, possibly for heat exchanger isolation.

The requirements for the various pieces of equipment in the circuit are as follows:

(a) Pumps

Parallel pump operation is required in this type of system. To reduce the number of pumps which are required, large flow, medium discharge pressure, centrifugal pumps are used. These are located at a point low enough in the system to provide the required net positive suction head. They must not be placed at a point where the suction static head is so high that special pump casings would be needed to withstand the pressure. Pumps with drooping head:flow characteristic curves should not be used for parallel operation. The pump curve should rise continually from full capacity to zero flow. As shown in Fig. 4, a drooping pump curve intersects a given head at two different points. The pump with the continually rising curve will operate steadily at the one point but the pump with the drooping curve will tend to "hunt" between the two points at the same head. This is likely to cause serious damage to the system.

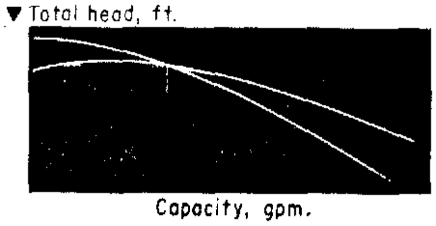


Fig. 4

The hydrostatic pressure in the moderator system is low compared with that in the heat transport system. However, it is high enough to cause heavy water leakage problems. Heavy water leakage in the pumps can be minimized by the use of one of two types of pumps. Canned rotor pumps as shown in Fig. 5, are pumps in which the motor is inside the pressure boundary of the system. A thin layer of metal separates the oil immersed stator windings from the system fluid. The leakage problems, in such a pump are similar to those encountered with flanges and are much less severe than with the alternative shaft-sealed pumps.

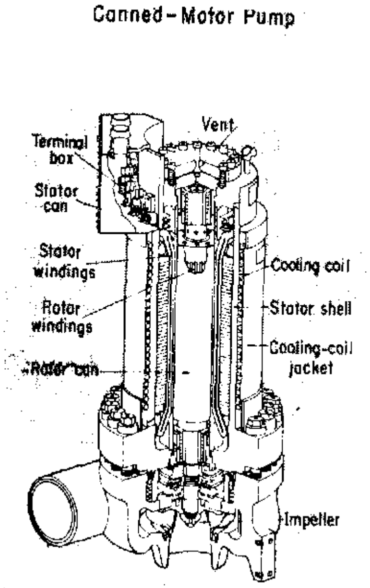


Fig. 5

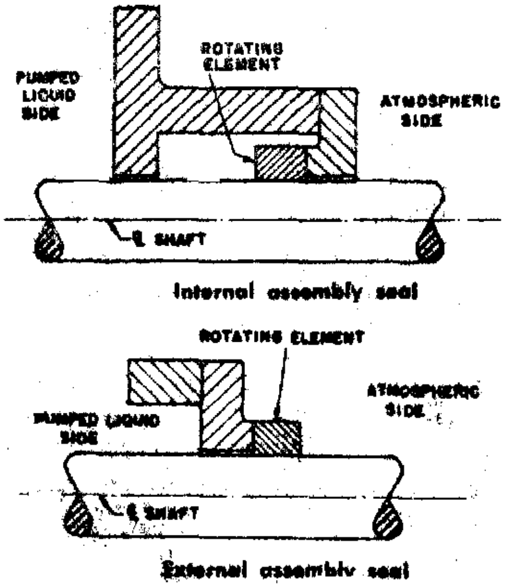


Fig. 6

However, canned rotor pumps are also much more expensive. The mechanical shaft seal pump uses a sealing device which forms a running seal between flat, precision-finished surfaces. The seal is made between a hard face made of stellite and a carbon face as shown in Fig. 6. The rotating sealing surface is carried on the pump shaft and this bears against the other stationary face. Double seals may be used, the space between the seals being bled off to a collection system. However in a moderator pump it may be sufficient to back up the seal with a labyrinth restriction and a bleed-off to the collection system, or simply allow any leakage through the seal to pass to the collection system. Such a collection system recovers the heavy water with little or no downgrading and also prevents the spread of radioactive contamination.

The parts of the pumps which are in contact with the heavy water should be of stainless steel.

(b) Heat Exchangers

Tube-and-shell type heat exchangers are suitable in this type of system. The U-tube type heat exchanger, shown in Fig. 7, simplifies expansion problems and reduce the number of tube sheet joints.

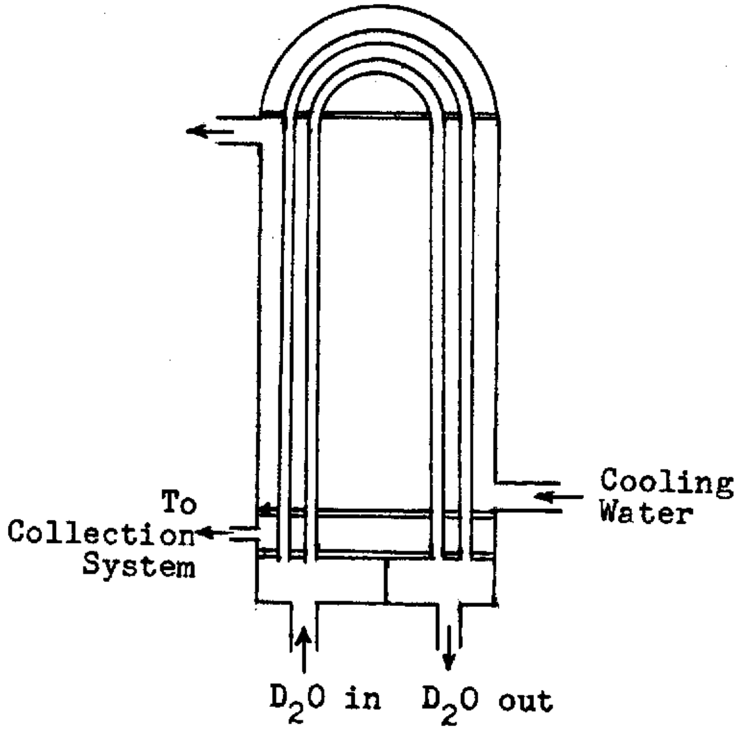


Fig. 7

Heavy water holdup is reduced by passing the heavy water through the tubes and the cooling water on the shell side. Double tube sheets are used to reduce leakage and the space between the tube sheets is drained to the collection system.

The moderator pumps maintain the heavy water at a higher pressure than the cooling water. This ensures that any leakage that does occur will be from the heavy water to the cooling water. The loss of D_2O incurred while a leakage point was located would perhaps cost a few thousand dollars whereas the downgrading of the whole of the moderator as a result of cooling water inleakage would involve hundreds of thousands of dollars in upgrading costs and unit down time. A system for detecting radioactive nuclei, in the cooling water leaving the heat exchanger, would indicate D_2O leakage into the cooling water.

The tube material would probably be inconnel or some copper-nickel alloy, whereas the shell would be made of carbon steel.

In order to maintain moderator cooling at all times the cooling water supply has to be very reliable. Should the normal process water supply fail an alternative source of cooling water must be available.

(c) Valves

Moderator system operating conditions are generally below 200°F and 100 psig. The radiation levels, around the equipment, are probably only moderate. The valve sizes required may be as large as 10" or 12" or as small as ½". The choice of valve will depend partly on the function of the valve, ie, a globe valve might be used where flow control is required whereas a gate valve might be preferred where it is used for isolation. Other factors, such as prices, heavy water holdup and friction losses, are also considered before a particular type of valve is chosen.

In NPD G.S. all the large isolating valves are gate valves but for Douglas Point G.S. a calculation, based on the above factors, was made to compare gate and globe valves with butterfly valves. The comparison indicated a significant economic advantage of the wafer type butterfly valves in the larger sizes. Consequently, rubber-lined wafer type butterfly valves were chosen for valves 6 inches and larger. Development tests showed that the leakage past the vane was acceptable and that the operation of the valve was satisfactory. The shorter length also made them more suitable for use in a congested layout.

Saunders rubber or neoprene diaphragm seat valves are generally used, in smaller pipelines, where the radiation damage to the diaphragm does not prohibit their use. If the radiation damage to the diaphragm is likely to be severe all metal diaphragm or globe valves would be used.

Consideration must be given to heavy water leakage along the valve stem. Metal bellows, of the type shown in Fig. 8 can be used to seal the valve stem but these are not really required in low pressure systems. A more likely approach is the use of double stem packing seals with an inter-seal drain to a closed collection system. It may not even be necessary to have a stem leakage connection. In Douglas Point G.S., for instance the smaller diaphragm valves have V-notches bonnets with a stem O-ring outer seal. The bonnets have vent plugs through which periodic leakage checks can be made in the inerspace between the diaphragm and the stem O-ring seal.

The material, in the valve, in contact with the heavy water must either be stainless steel or a compatible material like rubber. Teflon has very poor radiation resistance and should only be used as a stem packing material in areas where the radiation fields are low or when the packing can be periodically checked and changed.

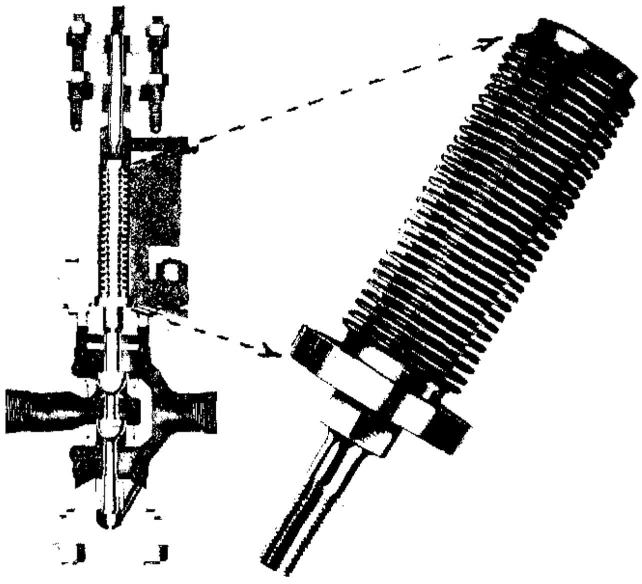


Fig. 8

(d) Piping

In order to limit heavy water losses pipe joints must be welded wherever possible. Where two different materials meet at the joint or where the connection requires rapid or repeated breaking, flange connections are used. Weldneck flanges are preferred in larger pipe sizes, but slip-on flanges are used in locations of restricted clearance. The flange gasket material must not deteriorate due to contact with water or due to thermal cycling. The gasket material must also be radiation resistant. Natural rubber or Buna N rubber gaskets are generally suitable in the moderator system. Screwed or threaded connections have been known to loosen up because of vibration or thermal cycling. They should, therefore, be avoided if at all possible and should not be used unless they are sealed by back welding.

The piping and equipment must be arranged so as to reduce heavy water holdup as much as possible. The model of the Douglas Point moderator system layout, Fig. 9, shows how congested such a layout can become. It is clear from the figure that such congestion

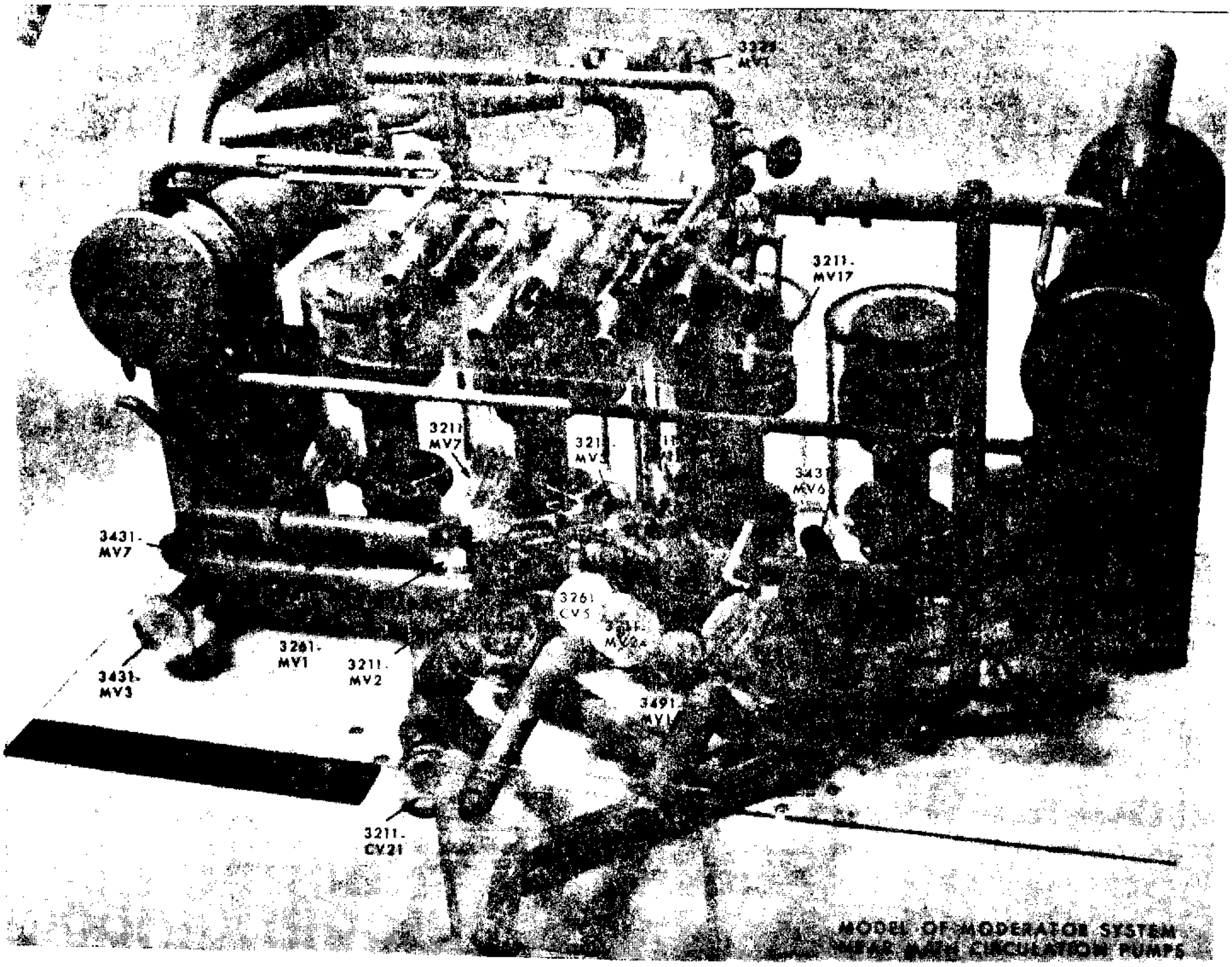


Fig. 9

could result in serious maintenance difficulties and would increase the possibility of equipment damage and accidental heavy water spillage. Such congestion at Douglas Point has already caused many problems resulting in high maintenance costs. It is estimated that, fairly early in the commissioning of Douglas Point, the increase in maintenance costs had already equalled the saving in the cost of heavy water. It may also be seen that there are likely to be many flange connections in the system. A leak detection system should therefore be installed. The general floor area below the equipment could be enclosed in a weir and moisture detectors placed inside the weir.

It is possible that the valves supplied in the system may not be sufficient to provide the isolation required for all maintenance work. Alternative isolation would, then, have to be provided. Ice plugs can be used for this purpose on small pipe or pipes in which the fluid velocity is low. For larger pipes or where the fluid velocity will not allow an ice plug to form, a gas bag inflated in the pipe can be used, provided the system is at low pressure. Such a gas bag might well be preferred to an ice plug because of the inherent hazards with ice plug formation and the effect of low pipe temperatures on welds. Isolation needs must therefore be anticipated and means provided of inserting the deflated gas bag into the pipes. Connections are also required for draining and venting equipment or various sections of piping.

If the calandria material is stainless steel, the pipes will also be of stainless steel. Aluminum pipe will generally be used with an aluminum calandria although stainless steel pipes can be used if flanged connections are used. Mild steel and aluminum would not be used in the same system since mild steel can only be used in a system where the pH is around 9 to 11 whereas aluminum corrodes rapidly under these conditions. Aluminum has good corrosion resistance at a pH of 6 or 7 but mild steel corrodes under such conditions. In any case it is desirable to keep the pH around 7 since this tends to reduce radiolytic decomposition of the heavy water. Therefore mild steel would not be a suitable material in the system.

Moderator Storage, Drainage and Level Control

There is no reason why the calandria cannot remain full of heavy water, as in Fig. 1, 2 and 3. The reactor would then have to be regulated and shutdown by neutron absorber rods which are inserted into or withdrawn out of the reactor. There are, however, a number of reasons for supplying a second tank into which the moderator can be drained:

(a) It may be necessary to drain the reactor, in order to carry out maintenance or repairs.

(b) A very large decrease in reactivity is obtained when the moderator is drained from the calandria. It is, therefore, very convenient to be able to shut down the reactor by partially or completely draining the moderator from the calandria.

(c) A reactor trip or rapid shutdown is required if certain variables, such as reactor neutron power, exceed some predetermined values. Fast moderator drain provides a convenient method of quickly reducing the reactivity following a reactor trip. The tank into which the moderator is "dumped" is known as the dump tank. Moderator dump may be used as the only method of shutdown or as a backup to some other method, such as the insertion of absorber rods.

(d) Small variations in moderator level may be used as a very versatile method of reactor regulation. Such a means of reactor regulation requires a stable, accurate level control with a very fast response.

If a dump tank is to be incorporated into the system, additional equipment will be required to permit fast draining into the dump tank, transfer of water back into the calandria and variation in calandria moderator level. One of two methods can be used. Fig. 10 shows how the moderator system, shown in Fig. 1, is changed to accommodate the first of the two possibilities. Water is constantly pumped from the dump tank, through the heat exchanger, to the calandria. The large valve B is completely closed. A is a smaller control valve and, if it is also closed the level continues to rise in the reactor until it is full. If A is partially opened, the calandria level rises until the amount of leakage through A is equal to the flow into the calandria from the pump. To raise the calandria level, A is closed further. The level then rises until the flows again balance. The line CDE allows for free gas exchange between the two tanks as the water flows from one to the other. On a reactor trip, the large valve B opens to allow fast draining or dumping of the moderator into the dump tank.

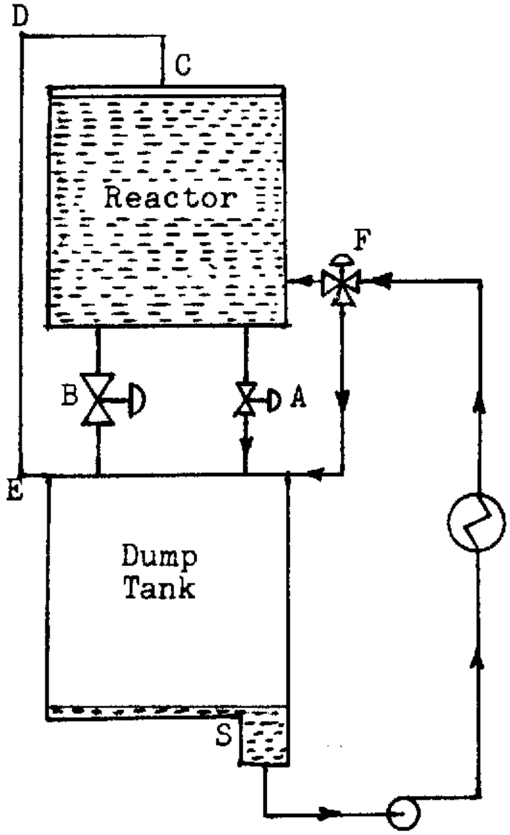


Fig. 10

To simplify the figure no pump or heat exchanger duplication is shown and the valves in Fig. 2 and 3 have been left out. There may also be more than one valve A and there will undoubtedly be several of type B in large drain lines.

In Fig. 11, there is, again, constant addition of water from the dump tank to the calandria. The water returns to the dump tank through the device W. This device, normally in the reactor vessel, is a weir arrangement over which the water spills. Such a weir arrangement is shown, diagrammatically, in Fig. 12. Because the water spills over the weir, a liquid-gas interface is formed at XY. The moderator is then kept in the calandria by establishing a Pressure differential between this liquid-gas interface and the top of the calandria. A pressure differential must, therefore, be established between the dump tank and the top of the calandria. The higher this pressure differential, the higher the moderator level in the calandria.

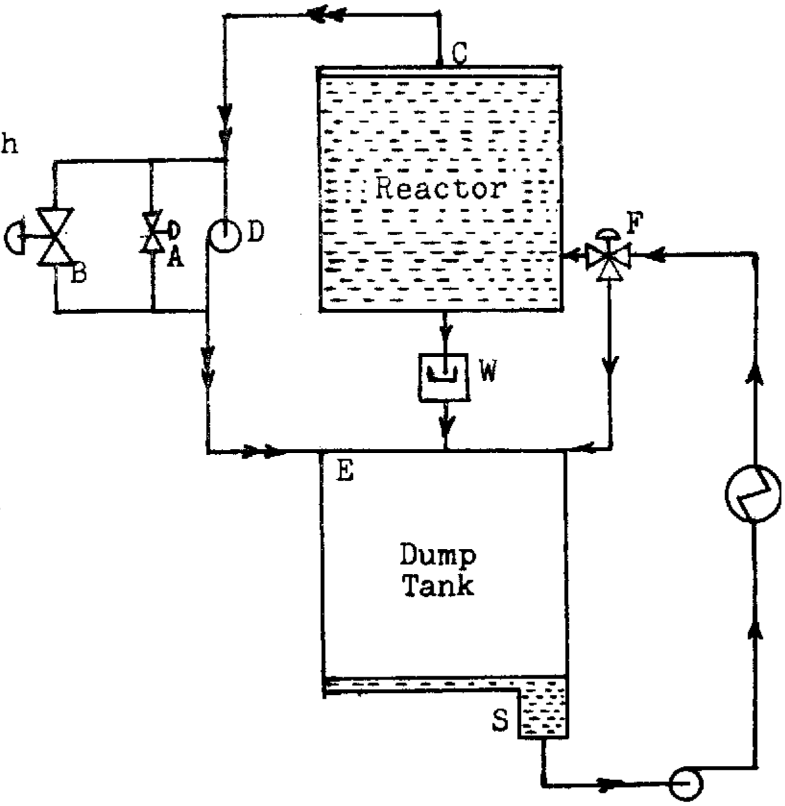


Fig. 11

The pressure differential, between the dump tank and calandria, is established by the cover gas system designated ↠ . Gas blowers or water-jet exhausters, D, maintain the pressure at C below that at E. If valve A and valve B were both closed, the pressure at C would become very low and the calandria would fill completely. However, a controlled gas leak rate can be established through the control valve A to give the pressure differential, between E and C, necessary to establish a specific moderator level in the calandria. Variation in the control valve leak rate can then be used to vary the cover gas pressure differential and, hence, the moderator level. A reactor trip will cause the large "dump" valve B to open. This equalizes the pressure at C and E allowing the moderator to empty rapidly into the dump tank.

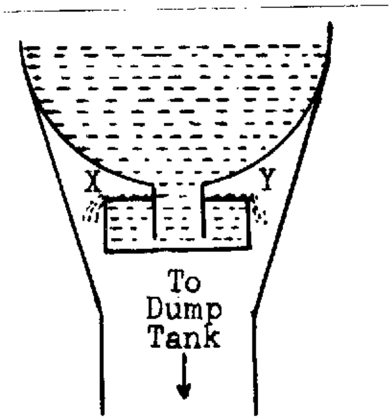


Fig. 12

The same considerations apply, to the valves in Fig. 10, as for the valves previously discussed. The following equipment requirements also apply:

(a) The dump tank must be large enough to contain the moderator and mechanically strong enough to withstand the shock of a moderator dump.

(b) The dump tank must be so shaped that it minimizes heavy water holdup while still providing adequate net pump suction head. An arrangement such as the sump S, (Fig. 10 and 11), could be used.

(c) The dump tank and piping materials will generally be the same as that of the calandria.

(d) Blowers, control valves and dump valves can be duplicated or triplicated to increase reliability.

(e) The maximum **moderator** pump-up rate must be such that the **rate of reactivity increase can not exceed some set value** The number and capacity of the blowers or water-jet exhauster, in Fig. 11, are based on this requirement. In Fig. 10, the pump capacity would be limited by this requirement and would result in more cooling capacity being required.

(f) The fast drain and gas pressure equalization lines, in Fig. 10, or the weir arrangement and gas pressure equalization lines, in Fig. 11, must be large enough to empty the reactor fast enough on a trip. This tends to favour the arrangement in Fig. 11, since the water lines must be bigger than the gas lines i.e. dump valves B, in Fig. 10, would have to be substantially larger than dump valves B, in Fig. 11.

(g) Flow diversion valves, F, may be incorporated in the moderator system. On a reactor trip, such a valve would divert the flow of water to the dump tank instead of the reactor. This would help to increase the dumping rate.

(h) All dump valves must be quick acting to ensure a rapid dump. This will likely mean that they are butterfly valves of the "air to close-spring to open" types.

(i) Interconnections between calandria and dump tank must have expansion joints in them to allow for changes in temperature in the two tanks.

(j) A gas, called the cover gas, is required, particularly in the system in Fig. 11. The requirements of the cover gas system are considered separately later in the lesson.

Moderator Purification System

A moderator purification system may be provided for several reasons:

(a) To remove corrosion products from the moderator. This corrosion product removal:

(a) (i) Reduces the corrosion product radioactive, in the system due to neutron absorption.

(ii) Reduces neutron capture in the reactor and so maintains the desirable neutron properties of the moderator.

(iii) Reduces radiolytic decomposition of the water since impurities encourage radiolytic decomposition.

(b) To control the pD of the water at a value which will reduce corrosion of calandria, dump tank, equipment and piping. The desirable pD in an aluminum system, for instance is about 6.

(c) To remove liquid poisons, such as boric acid, which have been added to the moderator to counterbalance excess reactivity in the reactor. Such poisons are added to maintain a full calandria with new fuel or when the Xenon poison has decayed during a shutdown.

Purification, or demineralizer, systems may be full flow or by-pass systems. In the full flow system the total moderator pump output would pass through the purification system, which would be in series with the pump. However, the moderator flow is too high, (about 6,000 Igpm at Douglas Point), to permit a full flow system to be used. Consequently, a by-pass purification system is used. The purification system is connected across the moderator pumps, the discharge head of the pumps providing the pressure necessary for the flow. An orifice, in the line, restricts the maximum flow through the purification system to 40 Igpm or 50 Igpm. There may also be a flow control valve, H, provided. Fig. 13 and 14 show two possible arrangements.

In Fig. 13, the inlet to the purification system is taken from the outlet of the heat exchangers. This takes advantage of the cooling provided by the heat exchangers but, since the moderator temperature is always below 180°F, this is not an important advantage. As will be seen later some types of interconnections with the heat transport system may make it necessary to adopt the arrangement in Fig. 14, with the system connected directly across the moderator pumps.

The low particulate materials concentration, or crud level, in the moderator system makes it unnecessary to have more than one filter, G, in the purification circuit and it may be considered unnecessary to have a filter at all, once the station is in normal operation. If a filter is used, it may be a disposable, cartridge type filter or a screen type filter which can be scraped and backwashed.

Soluble corrosion products are removed with ion exchange columns IX1 and IX2. There will be at least two of these in parallel so that one can be on line while the exhausted one is being changed.

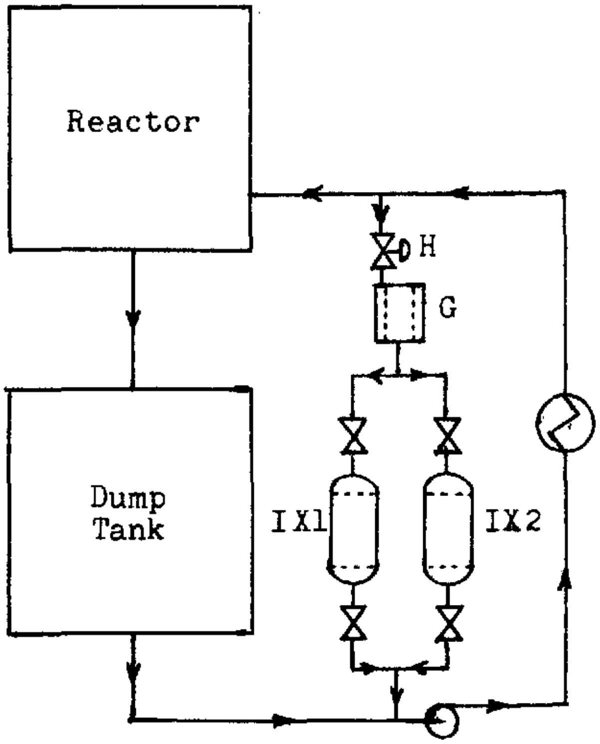


Fig. 13

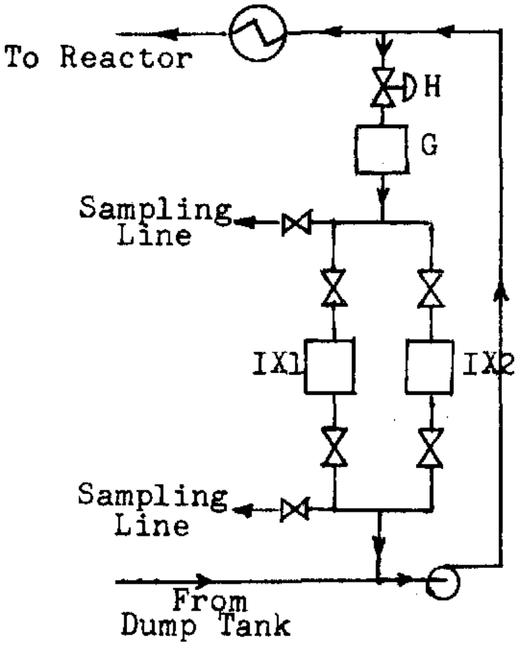


Fig. 14

Valves are then required for isolation. The ion exchange units themselves must be of stainless steel or must be lined with some compatible material such as PVC. The resin in normal use will be a mixed bed resin. If there is oil leakage into the moderator system or air inleakage into the cover gas, carbonates and nitrates will be formed. It may then be necessary to use a predominently anion bed in parallel. A predominently anion bed is also used to remove boron poison in the form of boric acid. The units may be entirely disposable, i.e. both container and resin being thrown away when the resin is exhausted. Alternatively provisions can be made to remove the resin only and to fill the unit with fresh resin. Heavy water conductivity measurements, at the inlet and outlet of the columns, indicate when the resin is exhausted. Facilities are also provided for taking samples at the inlet and outlet points, as shown in Fig. 14.

Before the resins are used they must be deuterated. Heavy water is passed through the resin beds until the H and OH ions, normally in the resin, are entirely replaced by D and OD ions. This prevents the hydrogen and OH ions from entering the moderator system and downgrading the moderator. When the resins are exhausted, the deuterium and OD ions must be recovered, by dedeuteration, to avoid loss of D_2O.

Aluminum systems are normally operated at a pD of 5.5 to 7 whereas the pD in stainless steel systems should be 7 or a little higher. However, more efficient boron poison removal occurs with a pD of 5 to 6 and, under these conditions, evidence of corrosion in stainless system should be carefully checked. The chloride ion must be eliminated from stainless steel systems because severe stress corrosion occurs with oxygen in combination with a chloride ion concentration of only 0.1 ppm.

Since activated corrosion products accumulate on the filters and columns and since radioactive N-16 and O-19 normally circulate through the columns, the filter and columns will be contained in a shielded cavity or pit. Removable shielding blocks allow access to valves and enable filters, columns or resins to be changed. A moisture detector must be located in the pit to indicate heavy water leakage.

Connections to Auxiliary Systems

(a) Spray Cooling - When the reactor is shut down, with the moderator drained from the calandria, the calandria tubes and reactor structure still receive some heat from the fuel. Heat is also being produced in the tubes and walls because of absorption of radiation emitted by decaying fission products. When the calandria is empty, or when all the fuel channels are not covered with moderater, auxiliary cooling is supplied from spray headers located at the top of the calandria, in the dump ports and elsewhere. The cooling water is supplied from the moderator system. Some advantage is gained by taking this water supply from the point J, at the heat exchanger outlet, as shown in Fig. 15. However, it could well be taken from the inlet to the heat exchanger, K, as shown in Fig. 16.

(b) Booster and Absorber Rod and Reactor Support Cooling - Heat is produced in booster rods and absorber rods both during reactor operation and at shutdown. The required cooling water can again be taken from the points J or K. The same arrangement is suitable to provide heavy water for calandria supporting rods. Any return flow to the moderator system will be to the pump suction, as shown in Fig. 16.

(c) Dousing Sprays - Dousing sprays are located above areas where pressure suppression is required in the event of line rupture. Heavy water dousing may be used initially in such areas, to avoid downgrading and light water dousing may be used only if the heavy water dousing is insufficient. Again the dousing supply may be taken from points J or K, but some advantage is gained if the supply water has been cooled in the heat exchanger

(d) Heavy Water Exchange Between the Moderator and Heat Transport Systems - Heavy water exchange between the moderator and heat transport systems may be required for one of two reasons. The moderator water may be used for emergency supply or injection to the heat transport system in the event that the heat transport system becomes depressurized as a result of a rupture. This ensures that heat is removed from the fuel during emergency conditions. If this is the only requirement, then the necessary connection can be made at the points J or K, as shown in Fig. 15.

The moderator system may, alternatively or in addition, be required to receive the swell from and make up the shrinkage in the heat transport system as it expands or contracts with changes in temperature. Since the heat transport system is at a higher pH than the moderator system, the flow of water to the moderator system, during a swell, must be fed through the moderator purification system. The interconnection is then made at the point L in Fig. 16. A check valve prevents the swell passing directly into the main moderator system. The second check valve prevents drainage of the purification system due to the fact that the ion exchange column are at a high point in the system.

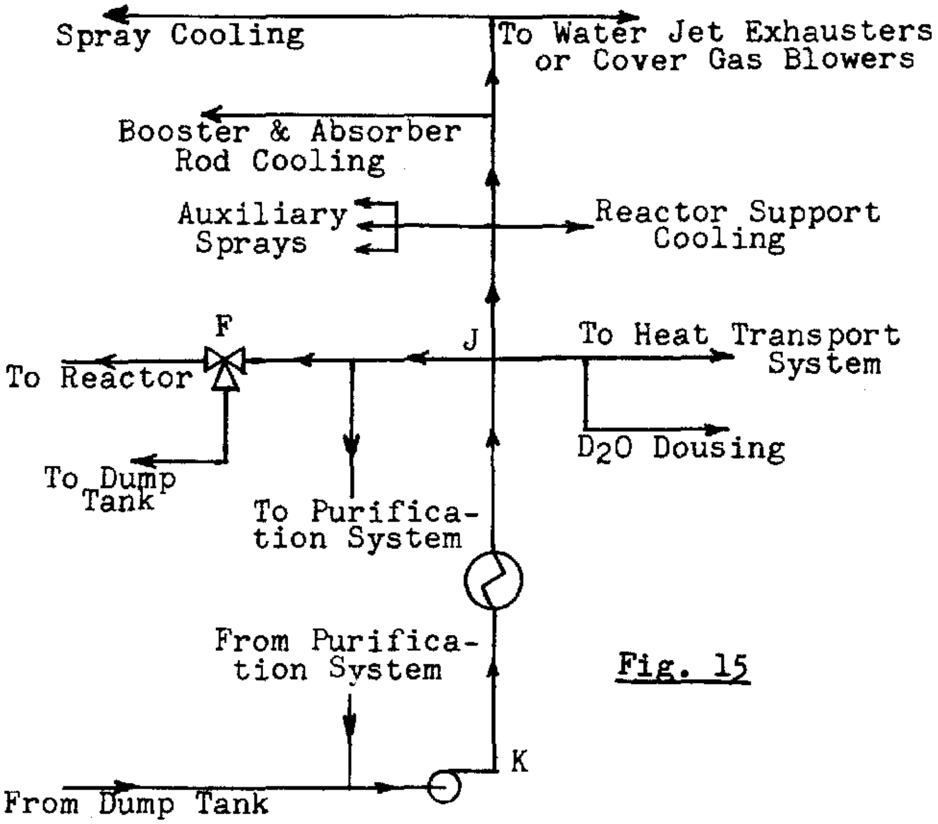


Fig. 15

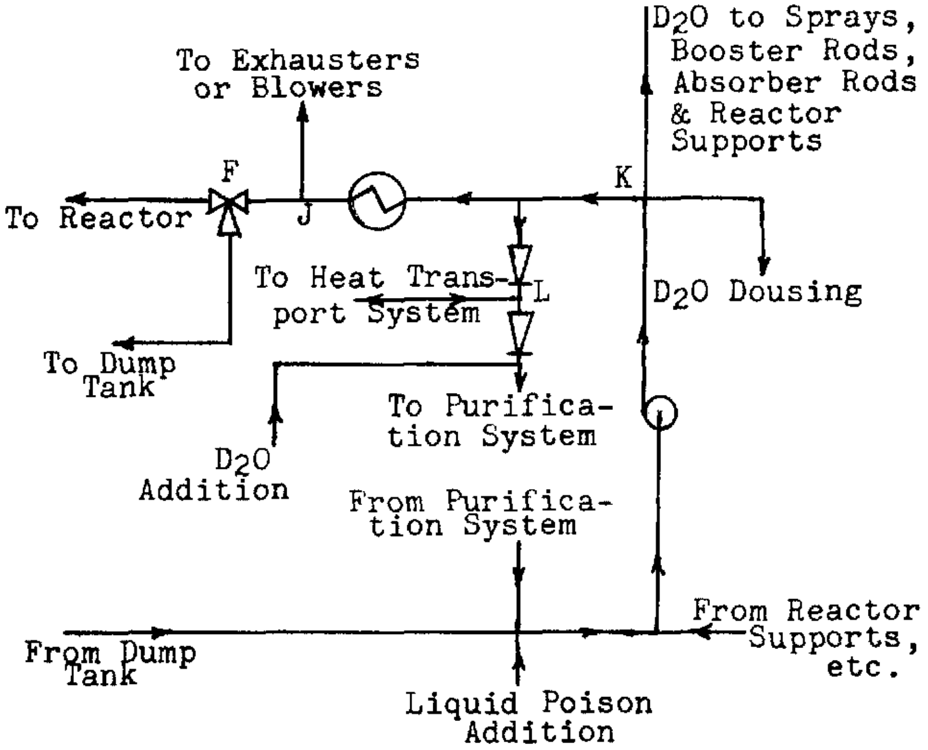


Fig. 16

The connection to the heat transport system is made at a point, such as at the outlet of the heat transport purification system, where the pressure is low.

(e) Heavy Water for Water-Jet Exhausters or Cover Gas Blowers - Water-jet exhausters or cover gas blowers maintain the pressure differential between the dump tank and calandria. The exhausters require a continuous water supply and so do the blowers if they are of the water seal type. To avoid downgrading, this water is supplied directly from the moderator system. The water must be as cool as possible and so it is always supplied from the outlet of the heat exchangers at J.

(f) Additions to the System - Provisions must be made for the addition of heavy water to the moderator system to make up for losses from the system by leakage or sampling. If the heavy water being added can be guaranteed to be free of impurities then the most convenient addition point is at the moderator pump suction. If there is any possibility of impurities being present, which would encourage radiolytic decomposition of the heavy water, heavy water should be added to the system through the ion exchange columns, as shown in Fig. 16. The second check valve, referred to above, now prevents the heavy water, being added, from entering the heat transport system and lowering the pH value.

Liquid poison addition to the system is made at the moderator pump suction so that it is dispersed through the system as rapidly as possible.

In all the above discussions, it is assumed that all the lines considered will have the appropriate valves fitted, even though these have not been shown. The connecting lines to the heat transport or dousing systems will have valves which open on a signal indicating an emergency condition. The D_2O addition line will have a manual isolating valve whereas the poison addition line is more likely to be fitted with a remotely operated isolating valve. All other lines will have provisions for isolation and may possibly have flow control valves. Note that in all cases, connections are made upstream, of any diversion valve, F, that may be in the system, since these auxiliary supplies are required during reactor shutdown.

The Cover Gas System

It has already been mentioned that water can be held in the calandria by establishing a pressure differential between the dump tank and calandria using water-jet exhausters or gas blowers. The level of water in the calandria can be varied by varying the cover gas leak rate through the control valve A (Fig. 11 and 17) and, thereby, varying the pressure differential between dump tank and calandria. When a reactor trip occurs, the dump valve B opens to quickly equalize the pressure between the dump tank and calandria. Further requirements of this cover gas system will now be discussed.

(a) Choice of Cover Gas

The cover gas is required primarily to provide a medium for establishing the pressure differential required to hold the water in the calandria. The essential properties of a cover gas may be stated as follows:

1. It should be chemically inert even in high radiation fields. This avoids the formation of compounds which would enter the moderator as impurities and increase the radiolytic decomposition of the water.
2. Its cost must not be excessive.
3. It should have a low neutron capture cross section. This is not so important from the point of view of neutron economy since the density is low. It is, however, important because of the resulting radioactivity induced. If neutron absorption does occur, the resulting nucleus should either not be radioactive or not be a gamma ray emitter.
4. Its chemical purity should be readily maintained.

There are many gases from which the choice could be made but all of them would be rejected on the basis of one or more of the above considerations. Air would be ideal because of its low cost and availability. However, it contains nitrogen and argon. These constituents capture neutrons to form radioactive carbon-14 and argon-41 respectively. Nitrogen, in a radiation field, combines with oxygen to form oxides which dissolve in the moderator to form nitric acid. This is most undesirable and is the main objection to using nitrogen as the cover gas. Hydrogen is unsuitable because of the explosion hazard associated with it.

All the gases, including the noble gases, can be eliminated in this way, except helium. Helium is chemically inert and does not acquire induced radioactivity. It is fairly expensive initially but once in the system its purity is easily maintained because it is inert. It is more expensive than nitrogen but the cost of ion exchange columns to remove nitric acid, when nitrogen is used, greatly offsets the difference in the cost of the two gases.

The main problem with helium as a cover gas is air inleakage in systems at pressures below atmospheric as it increases the nitrogen content. To avoid nitric acid formation and its consequences the system has to be purged and fresh helium added. It is imperative, therefore, that air inleakage be reduced to a minimum by using welded joints or flanged connections with reliable gaskets. It is an advantage, as in the Douglas Point system, to have the whole system at or above atmospheric pressure. Joints and connections must still, of course, be reliable to avoid loss of helium.

(b) Recombination Circuit

As was explained in a previous lesson, heavy water dissociates into deuterium and oxygen on being irradiated. This represents a loss of heavy water unless it is reclaimed. Furthermore an explosive concentration of deuterium would soon result. The deuterium and oxygen are combined by passing the helium, containing them, through a recombination unit. This recombination unit contains a catalyst such as palladium. The helium, therefore, also serves as a "carrier" or "sweep" gas, providing the means of transporting the dissociation products through the recombination units.

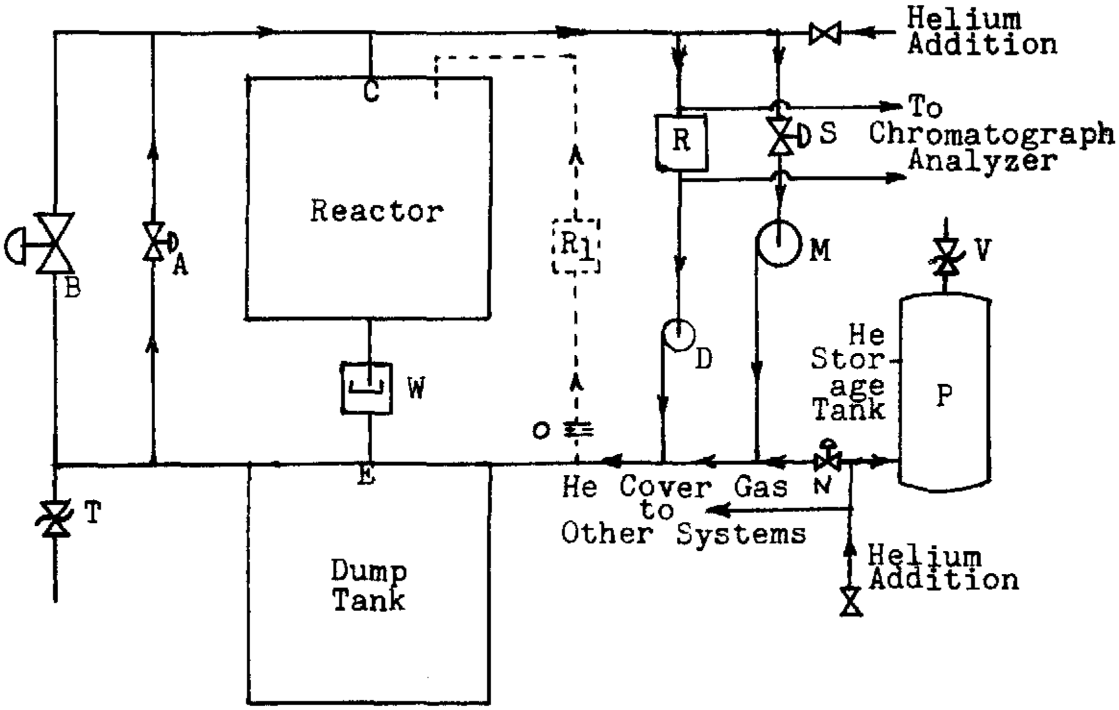


Fig. 17

The recombination units, of which there may be several in parallel, may be in-line units, as shown at R in Fig. 17. They are then in series with the blowers or exhausters, D, and process the total helium flow. Alternatively, they may be, as at R_1 in a by-pass circuit shown in dotted line. The flow through the by-pass circuit is limited by the flow orifice O. Such a by-pass circuit can be used to pass the clean helium into the top of the calandria to keep it continually purged of deuterium. Flame arrestors must be placed before and after the units to prevent the propagation of the flame resulting from the deuterium-oxygen recombination. The helium gas at the inlet and outlet of the recombination unit can be periodically sampled or continuously sampled in a chromatograph.

(c) Fast Pump Up

Provisions may be made for raising the moderator level more rapidly, immediately following a trip, to avoid a poison out. Such a facility would be in the form of a larger blower or additional exhausters, M, which would normally be isolated. The valve, S, would be automatically opened only when fast pump up was required and permitted.

(d) Helium Storage Tank

A helium storage tank or gas holder, P, is required to provide a helium feed or bleed to the system when necessary. Such a tank may be a gas holder riding on the system or it may be a storage tank connected to the system when a dump tank pressure signal opens the valve N.

(e) Pressure Relief

Valves, such as T and V, must be provided to prevent overpressure of either the system or the storage tank. These valves would vent into the reactor vault or some other area where the heavy water vapour in the helium would be recoverable.

(f) Gas Addition

Provision must be made so that helium gas can be added from pressurized bottles to either the system or the storage tank.

The system might also become depleted in oxygen i.e. the deuterium:oxygen ratio in D_2O might not be maintained because of oxygen combining with other substances, e.g. corrosion. This will result in a deuterium build-up despite the recombination units. There is, therefore, an oxygen addition line, (not shown), upstream of the recombination units.

(g) Miscellaneous Connections

Helium may be used as a cover gas for other systems such as heavy water collection system. It may also be used to prevent air inleakage through the dump valve stems or the pump seals. Connections must be provided, from the storage tanks for these purposes.

ASSIGNMENT

1. Briefly summarize the basic requirement of the moderator system.

2. (a) Why is it necessary to cool the moderator?
   (b) Why can the heat removed not be put to some practical use?
   (c) Why are the pumps and, sometimes, the heat exchanger duplicated?

3. (a) What type of pumps are used in the main moderator cooling system and how are heavy water losses from these pumps minimized?
   (b) What reasons are there for minimizing such losses?

4. Explain how the design and method of operation of a moderator heat exchanger may differ from a conventional heat exchanger and explain why these differences are necessary.

5. (a) What considerations govern the choice of valves in the moderator system?
   (b) What types of valves are used and for what purposes?

6. (a) What two major considerations govern the piping connections and layout and in what principles do these considerations result?
   (b) What other provisions may have to be considered?
   (c) Explain how the pipe material is determined.

7. (a) What reasons exist for supplying a dump tank in the moderator system?
   (b) Enumerate the basic requirements of the dump tank and dump pipes.

8. (a) Explain why a moderator purification system is required and why a by-pass system is used.
   (b) What types of resin beds are used for what purpose and how is it known that the beds are exhausted?

9. (a) Why are spray cooling and booster and absorber rod cooling connections required?
   (b) Give two reasons why heavy water exchange should be possible between the moderator and heat transport systems.

10. (a) What is the primary reason for having a cover gas system and how is it fulfilled?
    (b) What are the other possible uses of the cover gas?

11. List the essential properties of the cover gas and explain why the only choice is helium.

12. Explain why a recombination circuit is necessary.

A. Williams

Reactor Boiler and Auxiliaries - Course 133

HEAT TRANSPORT FLUID REQUIREMENTS

Thermal energy is produced in a reactor as a result of the fissioning of a fissile material. The heat transport fluid is used to transport this heat either to a boiler, where it is used to produce steam, or it is used as a thermodynamic heat engine fluid which passes directly to the turbine. There are two objectives of reactor design which are significant when the heat transport fluid is being considered. These objectives are: -

(a) That the heat be transported out of the reactor in a useful form and, therefore, at as high a temperature as possible.

(b) That the maximum amount of energy is obtained from the fuel (i.e. as high a fuel burn-up as possible).

The higher the heat transport temperature the higher the thermal cycle efficiency. In the boiler of a coal or oil fired station the source of heat (i.e. the combustion gases) is at the highest temperature of the system. The structural material, is at a lower temperature, which is essentially the steam temperature. In a nuclear station, the fuel elements are the heat sources and they must operate at 300°F to 500°F higher than the steam temperatures, in order to obtain reasonable heat transfer. The materials in the reactor must also withstand these higher temperatures. However, if materials were used, which could withstand these higher temperatures the capital cost of the station would be increased. Such materials might also capture more neutrons and, thus, lower the fuel burnup. Because of these conflicting factors a lower thermal efficiency is usually acceptable but this is offset by other compensating factors.

There is no theoretical upper limit to the power that can be produced by a reactor through fission. The maximum power is frequently determined only by the rate at which the heat can be transported from the reactor. The transfer of heat from the fuel elements can be improved either by further subdivision of the fuel, so as to increase the contact area, or by increasing the volume of fluid in the fuel channel. Both result in a decrease of the neutron multiplication factor, k, which means an increase in reactor critical size.

The discussion above shows that the heat transfer and nuclear requirements may conflict. It also illustrates that the heat transfer and heat transport characteristics of a particular fluid may substantially affect both reactor core and fuel element design. This lesson will consider both nuclear and non-nuclear requirements of heat transport fluids.

Nuclear Considerations

(a) Neutron Absorption Cross-Section

The heat transport fluid passes through the reactor core. One important requirement, therefore, is that this fluid have a small neutron absorption (or caputre) cross-section. The neutron economy of a reactor system has an important influence on the size and on the over-all cost of a reactor. There is a substantial volume of heat transport fluid in the reactor and so it is imperitive that the absorption cross-section per atom or molecule be small. Table 1 lists the thermal neutron absorption cross-section, σ_a, for some possible heat transfer fluids.

TABLE 1

Material	σ_a (barns)	Material	σ_a (barns)
Lithium - 7	0.033	Light Water	0.66
Bismuth	0.032	Heavy Water	0.0003
Sodium	0.5	Polyphenyls	0.33
56% Sodium-44% Potassium	1.1		
Mercury	380		

The cross-sections for gases are not listed since the absorption cross-section per atom is not the important factor. The low density of a gas, (i.e. the small member of atoms or molecules per unit volume), is of greater significance when neutron absorption is being considered.

The cross-sections listed are those for thermal neutrons and are pertinent only for thermal reactors. The absorption cross-sections for fast neutrons are substantially smaller and other factors are more important when deciding on a heat transport fluid for fast reactors.

It may be seen, from Table 1, that the most attractive fluid, from the point of view of neutron economy, are heavy water, lithium, bismuth and the gases. These can be used in thermal reactors using natural uranium fuel. The cross-section for the polyphenyls is also low enough to allow it to be used as a heat transport fluid in a natural uranium fuelled, heavy water moderated reactor. The use of any other heat transport fluid would necessitate enrichment of the fuel. However, if the light water is allowed to boil, with boiling occurring, say, half way along the fuel channels, then it can be used without fuel enrichment in a heavy water moderated reactor.

(b) Moderating Ratio

A good moderating ratio is desirable but not essential for a heat transport fluid in a thermal reactor. If the heat transport fluid has a high moderating ratio the amount of moderator used can be reduced, the fuel channel separation can be smaller and, hence, the reactor size is smaller. Furthermore, if the reactor is not overmoderated, (i.e. the fuel channel separation is only just sufficient to thermalize the average neutron), then loss of heat transport fluid leads to a reduction in reactivity. The resulting decrease in neutron absorption, however, causes an increase in reactivity. If the two effects cancel or the overall void coefficient is zero or slightly negative, then a high moderating ratio is desirable. It is not desirable if the void coefficient is positive (as in an overmoderated reactor such as NPD) or if it has a large negative value. Both conditions would result in power transients or power swings if voids occur due to boiling.

Good slowing down materials (i.e. of low atomic weights) are to be avoided in a fast reactor where the neutron energy must be as high as possible.

(c) Induced Radioactivity

If the neutron absorption cross-section of the heat transport fluid was zero radioactive nuclei would not be produced in the fluid by neutron capture, i.e. there would be no induced radioactivity. However, since this is rarely the case, the heat transport fluid activity becomes a factor requiring the following considerations:-

1. The radioactive nuclei produced should not emit gama rays or, if they are gamma emitters, the gamma ray energies must be as low as possible. If gamma ray emitters are produced, the system, external to the reactor, is not accessible during reactor operation and requires shielding to avoid exposure of station personnel to the radiation. This precludes maintenance to the equipment during operation at power. The higher the gamma ray energy the thicker the shield required.

An example of such induced activity is the production of N-16 in water by absorption of fast neutrons in oxygen.

$$O^{16} + n^{1} = N^{16} + P^{1}$$

The Nitrogen-16 nuclei emit 6 Mev and 7 Mev gamma rays. At NPD this resulted in a 4ft thick concrete shield being placed around the heat transport system.

The induced activity in organic fluids, (containing only hydrogen and carbon), on the other hand, only occurs as a result of neutron absorption in impurities.

2. The half-lives of the radioactive nuclei produced should be as short as possible. This allows the equipment in the heat transport system to be approached by station personnel soom after reactor shutdown. The half-life of the N-16 above is only 7.35 secs. However, the half-life of Sodium-24, produced in sodium, is 14.5 hours. This would delay maintenance on heat transport equipment until the activity decayed to an acceptible level.

3. The neutron activation should not result in the formation of a substance which could be an internal hazard if it leaked out of the system. There are two excellent, but entirely different, examples of the production of such a substance.

   Tritium (H^3), a beta emitter, is formed by neutron capture in deuterium ($H^2 + n^1 = H^3 + \gamma$). Being a hydrogen isotope it is absorbed by the body in tritiated water. This together with its long half-life, makes it an internal hazard.

   Bismuth-209 captures a neutron to form Bi-210, which is a beta emitter with a 5 day half-life. Bi-210 decays to Polonium-210, a high energy alpha emitter with a half-life of 138 days. Polonium is highly toxic and is one of the most powerful physiological poisons known.

(d) Radiation Stability

Radiation should not cause dissociation or damage to fluids used in the heat transport system. There must be a minimum of cross-linking, tar or coke formation and a minimum change in desirable physical properties resulting from such effects.

Heat Transfer Considerations

The temperature in a conventional fossil fired station is limited to that resulting from the combustion of the fuel. In nuclear fuel heat energy is being continuously produced by fission. If the rate of heat removal was less than the rate of heat generation, the temperature in the reactor would steadily rise until the reactor was destroyed. Therefore, the rates of heat generation and heat removal must be properly balanced. For any reactor the maximum operating power is limited by some temperature in the system. For instance the temperature at the centre of the fuel may be limited to reduce fission product gas diffusion to the sheath so that the fission product gas pressure does not cause sheath failure. Alternatively the fuel temperature may be limited in order to avoid a phase change which would cause distortion. The fuel surface temperature may be limited to avoid the sheath melting or to avoid local boiling and the resulting change in reactivity. The heat transport temperature may be limited to minimize corrosion or to avoid pump cavitation. During the design stage these maximum permissible temperatures must be established, the magnitude and distribution of the heat sources estimated and the temperature differences along the various paths of heat flow determined. The analysis involves considerations of heat conduction through solid components such as the fuel element and sheath and transfer of heat by convection from the solid surface to the heat transport fluid. The heat is then transported, by flow of fluid, along the reactor channel to a heat exchanger or some other heat sink. The heat transport fluid will therefore be chosen on the basis of its conduction, convection, heat transfer and heat transport characteristics.

For a heat path of length L, and constant cross-sectional area A, (e.g. slab), shown in Figure 1, the flow of heat by conduction, q, is given by: -

$$q = kA\frac{\Delta t}{L} = \frac{\Delta t}{L/kA} \quad ------- (1)$$

where k is the thermal conductivity in Btu/hr-ft-°F and Δt is the temperature decrease along the heat path. Because of the analogy to Ohm's Law, the quantity L/kA is often called the termal resistance of the heat path.

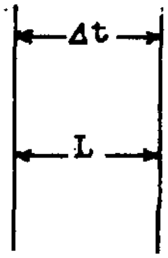


Fig. 1

Convective heat transfer between a solid and a fluid is governed by Newton's Law of cooling: -

$$q = hA\Delta t = \frac{\Delta t}{1/hA} \text{ - - - - - - - - - } (2)$$

where q is the rate of heat transfer from a surface of area A which is at a temperature Δt above its surroundings, h is the heat-transfer coefficient in Btu/hr-ft^2-°F. The quantity 1/hA is the thermal resistance to the heat transfer. These thermal resistances are additive.

Suppose, for instance, that heat is transferred, by convection, to face A of the slab, in Fig. 2. The heat is then conducted through the slab and transferred, by convection, from face B. Let the temperatures on each side of the slab be t_a and t_b, as shown, and the heat transfer coefficients are h_a and h_b.

The thermal resistance, R_1, at interface A is $1/h_aA$, that in the slab, R_2, is L/kA and that at interface B, R_3, is $1/h_bA$. These three can be considered as three resistances in series, as shown in Fig. 3. The total thermal resistance, R, is then given by: -

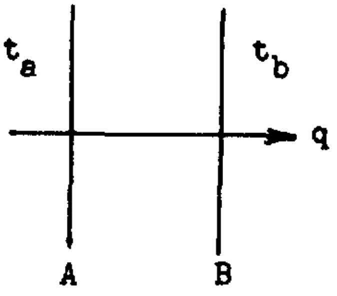


Fig. 2

$$R = \frac{1}{h_aA} + \frac{L}{kA} + \frac{1}{h_bA} = \frac{1}{UA} \text{ - - } (3)$$

$$q = \frac{t_a - t_b}{R} = UA(t_a - t_b) \text{ - - - - - } (4)$$

U is known as the overall coefficient of heat transfer.

The values of such quantities as h and U determine the suitability of a fluid as a heat transport medium. The value of h depends not only on the physical properties of the fluid but also on the shape and dimensions of the interface and the nature, direction and velocity of the fluid flow.

The value of h is larger, for instance, with turbulent flow than with laminar flow and in most cases, where forced convection is used for heat removal, the flow is turbulent. The wire wraps on the NPD fuel elements encourage turbulence. Heat transfer coefficients for turbulent flow of ordinary fluids can be expressed in terms of the Nusselt number (Nu) and the Prandtl number (Pr) for the fluid.

$$R_1 = \frac{1}{h_a A} \quad R_2 = \frac{L}{kA} \quad R_3 = \frac{1}{h_b A}$$

Fig. 3

$$Nu = \frac{hD}{k} \qquad \text{and} \qquad Pr = \frac{c_p \mu}{k}$$

where D is the pipe diameter c_p the specific heat at constant pressure and μ is the coefficient of viscosity.

For fluid flowing in a long straight channel

$$Nu = 0.023 \quad Re^{0.8} \quad Pr^{0.33} \quad - - - - - - - (5)$$

where $Re = \frac{Dv\rho}{\mu}$ is Reynolds number, v is the average velocity of the fluid and ρ is its density.

$$\text{Therefore:-} \quad \frac{hD}{k} = 0.023 \left(\frac{Dv\rho}{\mu}\right)^{0.8} \left(\frac{c_p \mu}{k}\right)^{0.33} \quad - - - -(6)$$

If the physical properties of the fluid, such as density, viscosity specific heat and thermal conductivity, are known the value of the heat transfer coefficient, for turbulent flow, can be calculated. Equation (6), however, does not hold for liquid metals. The values of the physical quantities in equation(6), particularly the viscosity, vary with temperature.

With liquid metals the most important heat transport mechanism is molecular conduction rather than by fluid movement as in gases and non-metallic liquids. For liquid metals an equation of the type: -

$$\frac{hD}{k} = 7 + 0.025 \left\{\frac{D\ v\ \rho\ c_p}{k}\right\}^{0.8} \text{ - - - - - - - - -(7)}$$

is used. Equation (7) gives values of h which are too high and, so, some modification of (7) is used in practice.

The following table gives the values of ρ μ, k and c_p at 400°F for (a) carbon dioxide at 10 atm. pressure and a velocity of 100 ft/sec, (b) water at 20 ft/sec and (c) liquid sodium at 20 ft/sec, flowing through a long straight tube of 0.1 ft diameter.

TABLE 2

	CO_2	Water	Liquid Na
ρ (lb/ft^3)	0.656	53.7	53.2
μ (lb/hr-ft)	0.0504	0.148	
k (Btu/hr-ft-°F)	0.0184	0.380	47.1
c_p (Btu/lb-°F)	0.265	1.0	0.320

Using these values in equation (6), for CO_2 and water, and in equation (7) for liquid sodium: -

h (for CO_2) = 124 Btu/hr-ft^2-°F

h (for water) = 8150 Btu/hr-ft^2-°F

h (for Na) = 9650 Btu/hr-ft^2-°F

The value of μ for water in the above table appears to be too low. However, the calculated values of h still serve to compare the heat transfer coefficients of the three fluids.

Table 3, below, lists the values of ρ, μ, k and c_p at the temperatures indicated. The values for the gases are all at atmospheric pressure.

TABLE 3

Fluid	Temp. (°F)	Density (lb/ft^3)	Specific Heat (BTU/lb) (°F)	Thermal Conductivity	Viscosity (lb/(hr)(ft)
Water (H_2O)					
m.p. 32°F	212	60	1.006	0.395	0.70
b.p.212°F	482	50	1.21	0.35	0.45
Water (D_2O)					
m.p. 39°F	215	66	0.993	0.372	0.79
b.p.215°F	504	54	1.18	0.305	0.28
Terphenyl					
m.p.250°F	600	53	0.60	0.066	0.77
b.p.750°F					
Sodium					
m.p.208°F	752	53	0.306	41.1	0.65
b.p.1621°F	1022	51	0.301	37.4	0.54
Sodium-Potassium (56 Na-44K)					
m.p. 66°F	752	51	0.252	16.0	0.56
b.p.1518°F	1022	49	0.248	16.4	0.47
Molten Salts(50NaF-46ZrF_4-4UF_4)					
m.p.970°F	1300	240	0.47	2.6	20.5
Air (dry)	1160	0.0246	0.268	0.0359	0.0946
	1700	0.0184	0.280	0.0434	0.1123
Carbon Dioxide	440	0.0670	0.241	0.0194	0.0463
	1160	0.0372	0.287	0.0359	0.0875
Helium	1160	0.00339	1.248	0.1570	0.1003
Hydrogen	1160	0.00171	3.54	0.2560	0.0451
Nitrogen	1160	0.0237	0.274	0.0350	0.0925
	1700	0.0178	0.288	0.0415	0.1115
Steam	440	0.027	0.48	0.018	0.044
	1160	0.017	0.52	0.038	0.072

Pumping Power Consideration

The power utilized in pumping the heat transport fluid through a reactor is given by: -

$$\text{Pumping power} = \text{Pressure drop} \times \text{Volume flow rate}$$

$$= (\Delta p)(A_f\, v) \quad \text{--------- (8)}$$

where Δp is the total pressure drop due to all causes, A_f is the total cross-section, or flow, area of all fuel channels and v is the flow velocity. For turbulent flow, the pressure drop due to fluid friction is given by: -

$$\Delta p_f = 4\, f\, \frac{L}{D}\, \frac{\rho v^2}{2g} \quad \text{------------- (9)}$$

where f is the friction factor, which depends on the Reynold's number, L is the length and D the diameter of the pipe.

The entrance and exit effects and the effects of fittings must be added to Δp_f to give Δp. However if these are ignored or assumed to be proportional to $\rho v^2/2g$ (which they are for abrupt contractions or expansions), equation (8) now becomes.

$$\text{Pumping power} = k\, A_f\, \frac{\rho v^3}{2g}$$

where k is a constant which depend on pipe length and diameter. The mass flow rate $M = \rho A_f v$

$$\text{Hence pumping power} = \frac{k\, M^3}{2g\, \rho^2 A_f{}^2}$$

It may, therefore, be seen that the pumping power required is, roughly, proportional to the cube of the mass flow rate and inversely proportional to the square of the fluid density.

For a given fluid of constant mean density, the mass flow rate required depends on the volumetric heat capacity ρc_p. The higher the volumetric heat capacity the lower the mass flow rate required to remove a specific quantity of heat. Thus, the higher the density and specific heat, the lower the pumping power required. Specific heats do not differ by more than a factor of around 15, eg, c_p for Sodium-potassium = 0.25, c_p for hydrogen = 3.54. However, densities of gases are generally about 100 times or more lower than those of liquids, and therefore gases require much larger pumping powers. Water has one of the most favourable pumping power characteristics, as may be seen from Table 3, followed by terphenyl and then by the liquid metals.

Other Non-nuclear Considerations

In general, a heat transfer fluid should possess the following additional characteristics: -

(a) Low Melting Point - It is desirable that the fluid have a melting point below room temperature to avoid the necessity of traceheating or preheating the system and to avoid freezing of the fluid during shutdown. Where the heat transport fluid passes through a heat exchanger cooled by lake or river water, as it would before passing through ion-exchange columns, it is desirable that its melting point be below 32°F.

(b) High Boiling Point - The most efficient conversion of thermal into electrical energy occurs with high heat transport temperatures. The high boiling point of a liquid permits operation of the heat transport system with little or no pressurization. This means that a strong pressurized system is not necessary which is an advantage from the operational safety point of view and also helps to reduce the capital cost of the station.

(c) Thermal Stability - The heat transport fluid must be stable at high temperatures.

(d) Non-corrosive Properties - It is necessary that the fluid should not attack materials, inside or outside the reactor with which it comes into contact, even at elevated temperatures. Corrosion causes damage to components which may mean that they have to be replaced. Their replacement after prolonged reactor operation may be difficult and expensive and is likely to lead to operating losses due to a lengthy unit down time. Corrosion products may also cause flow reductions in fuel channels and cause system contamination when they become radioactive.

Because of the fluid flow in the heat transport system mass transfer due to temperature gradients can be significant. For example, carbon from steel will dissolve in hot molten sodium and the carbon is then deposited at a cooler spot in the system.

(e) Non-toxic - Since there may be some leakage out of the heat transport system, it is essential that the heat transport fluid not be toxic nor result in a toxic substance when exposed to radiation.

(f) Low Cost - The volume of the heat transport system is second only to that of the moderator system. High cost of the heat transport fluid would, therefore, substantially increase the capital cost of the unit. Operational cost, due to fluid losses, would also be higher.

ASSIGNMENT

1. (a) Why are neutron capture cross-sections of heat transport fluids important in thermal but not in fast reactors?

   (b) When is a good moderating ratio desirable and when is it not desirable, for a heat transport fluid?

2. What induced radioactivity considerations are important when choosing a heat transport fluid?

3. (a) What physical properties of a fluid determine its heat transfer coefficient?

   (b) Briefly compare the heat transfer coefficients of carbon dioxide, water and liquid sodium indicating the primary reasons for the different values.

4. (a) What factors determine the pumping power required to circulate a fluid?

   (b) Why are the pumping powers required for gases so much greater than for liquids?

A. Williams

Reactor Boiler and Auxiliaries - Course 133

HEAT TRANSPORT FLUID COMPARISONS

No single substance, or mixture, can be found which satisfies all the heat transport fluid requirements which have been discussed. It is for this reason that several possible fluids have been and are being tried in reactor systems. The general characteristics of the most likely fluids will now be reviewed.

Light Water

Ordinary or light water is attractive, as a heat transport fluid, because it is readily available at low cost. It has a high specific heat and fair thermal conductivity. Its thermal conductivity is about 100 times lower than that of liquid metals but its value is well above those of organic liquids or gases. Hence the heat transfer characteristics of water are far superior to those of gases and organic liquids and are not much poorer than those of liquid metals and molten salts. Because of its good heat transfer coefficient, it is possible to decrease the flow rate of water and, since its density is relatively high, the pumping power required is low. The pumping power with water, for equivalent heat removal, is roughly in the order of one-tenth of that required with a gas at 10 atmospheres pressure.

Another basis for the comparison of heat transport fluids is the volumetric heat capacity, which is defined as the product ρc_p (in Btu/ft^3 - $^\circ$F). Water has one of the highest volumetric heat capacity and this enables a high rate of heat removal to be attained without an excessive rise in the heat transport fluid temperature. This helps to reduce thermal stresses in the system.

A further advantage of using light water as a heat transport fluid is that it can be used simultaneously as a moderator, which results in a fairly compact core. It melts at 32°F and, so, will not solidify during reactor shutdown.

The use of light water, however, has many drawbacks. The thermal neutron capture cross-section is too large to allow light water to be used as a heat transport fluid with any moderator unless some fuel enrichment is used. Water boils at 212°F under normal atmospheric pressure. Therefore, in order to extract the heat from the reactor at a high enough temperature, the heat transport system requires pressurization to prevent the water boiling. If the water is used simultaneously as moderator and heat transport fluid, the reactor can be enclosed in a pressure vessel. This does not involve additional material in the reactor core which increase neutron absorption but it does introduce the disadvantages, discussed previously, which are inherent in a pressure vessel design. If the moderator and heat transport fluids are to be kept separate, as in the pressure tube reactor concept, the additional strength required, because of pressurization, may result in more neutron absorption in the core. Either the pressure tubes must be made of zirconium alloy which has a low neutron capture cross-section but is expensive, or low alloy steel must be used for the pressure tubes thereby increasing neutron capture and necessitating the enrichment of the fuel.

Water at high temperature, even when pure and free from oxygen, becomes quite corrosive. With a pH of 5.5 to 7 the corrosion rates with zircalloy and stainless steel are quite acceptable but the corrosion rates with carbon steel or low alloy steel are not acceptable unless the pH is raised to between 9 and 11. This corrosion problem is more severe because of the presence of oxygen due to radiolytic decomposition of the water. This radiolytic decomposition can be reduced by removing ionic impurities. The decomposition of water is also discouraged by pressurizing and is not, therefore, as severe a problem in the heat transport system as it is in a low pressure moderator system. The cover gas space above the moderator allows the gas to escape from the water and encourages more decomposition. Experience at NPD G.S. has shown that it is better to allow the hydrogen concentration to reach an equilibrium value rather than use a degassifier to remove the decomposition products. The disadvantages with this is that it makes hydrogen available for absorption in zircalloy and this may result in hydrogen embrittlement of the zircalloy.

Radioactive O-19 and N-16 are produced in water as a result of neutron capture. Both these nuclei are gamma ray emitters and the N-16 gamma rays have very high energies. Extensive shielding is, therefore, required around the heat transport system equipment and piping external to the reactor. It also means that the heat transport system is not accessible for maintenance during reactor operation.

Although light water is relatively inexpensive, some cost is involved in maintaining it free of impurities. It is also undesirable to have the containing room and other equipment contaminated by radioactive fluid which could contain fission products from failed fuel elements. Leakage of water out of the system should, therefore, be avoided but the equipment would probably be more conventional than would be possible with heavy water.

Boiling Light Water

There are many advantages to be gained by permitting the heat transport system fluid to boil. If boiling water is used as a heat transfer fluid the large latent heat of vapourization can be used to increase the heat transport coefficient by a substantial amount. Fig. 1 shows how the heat flux, transferred to a fluid, varies with the temperature difference between the heated surface and the fluid. The shape of the graph should be regarded as being generally representative although actual values differ in different cases.

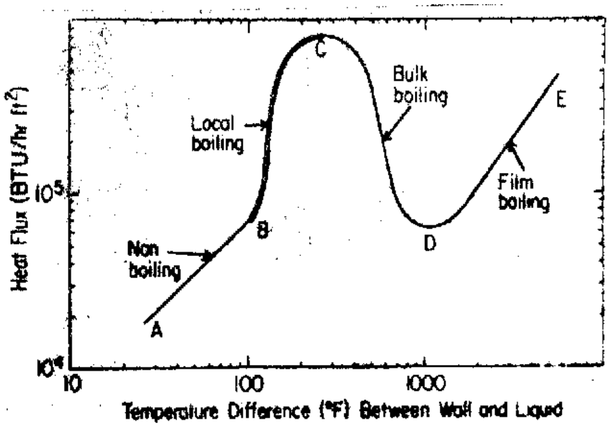


Fig. 1

The graph can be divided into a number of regions, in each of which the mechanism of heat transfer is different. While the heated surface temperature is below the water saturation temperature, (shown as the non-boiling section, AB), heat is transferred by single phase convection, i.e. ordinary convection in water. This is the method of heat transfer already considered.

In the local or nucleate boiling region, BC, the heated surface temperature exceeds the saturation temperature by a few degrees. Bubbles are formed on nuclei, such as solid particles or gas absorbed on the surface. As the bubbles break away, they collapse or continue to grow, depending on whether the bulk of the fluid is below or at the saturation temperature. The large increase in the heat flux in this region occurs as a result of the mixing of the liquid caused by the agitation of the bubbles. This greatly improves the heat transfer characteristics.

The heat flux continues to increase until the heated surface becomes blanketed by an unstable, irregular film in violent motion, with little direct contact between the water and the heated surface. Heat transfer is then by conduction and radiation through the film. The heat flux, consequently, decreases appreciably along CD. At D the film is stable and the heat transfer then improves as the surface gets hotter. However, such high temperatures are required, in the region DE, to attain comparable heat flux to those along BC, that they result in the destruction of the fuel or sheath. This is known as BURNOUT and must, of course, be avoided.

Care must be taken, when operating in the nucleate boiling region, that the operating point is not too near the maximum of the curve. A slight increase in heat flux would then cause a sudden change to film boiling, which usually results in burnout.

It would appear, therefore, that high heat fluxes can be obtained in the nucleate boiling region. This would enable the pumping power to be reduced. Nucleate boiling with the bubbles growing would lead to a net generation of steam in the reactor if the bulk fluid temperature is a little above the saturation temperature. This would allow a direct cycle to be used with the steam being fed directly into the turbine. This, in turn, eliminates the heat loss in the boiler of an indirect cycle. Fig. 2 illustrates two further advantages resulting from the direct cycle. The fuel element temperature, with the direct cycle, is lower for the same steam conditions. Alternatively the steam temperature is higher for the same fuel temperature. Secondly, the direct cycle can be operated at a much lower pressure than that required to prevent boiling in the indirect cycle using water as a heat transport fluid. This reduces thickness of pressure vessels or tubes and results in lower capital costs.

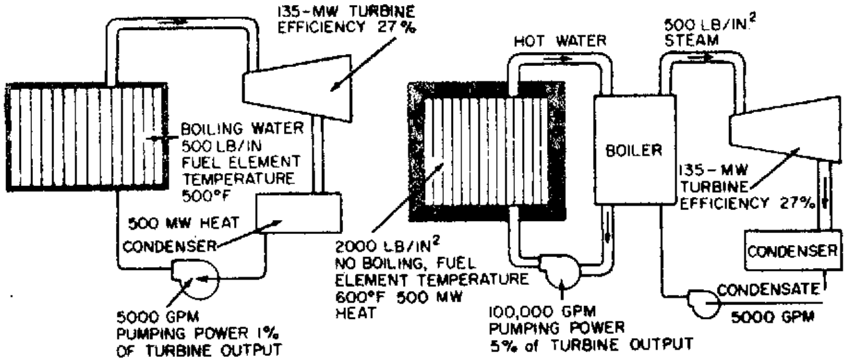


Fig. 2

The overall effect of void formation through boiling in a reactor using light water moderation and cooling is a decrease in reactivity. A reactor using boiling light water would, therefore be self regulating. The reduction in the high enthalpy water content *of the heat transport system would also ease the containment problem* should a rupture occur in the heat transport system.

There is one final advantage of using boiling light water which is of particular significance to the CANDU concept. If boiling occurs about half way along a fuel channel in a pressure tube reactor, the decrease in neutron capture, by the light water, is sufficient to allow boiling light water to be used in a natural uranium-heavy water moderated system. This would result in reduction in heavy water capital cost and in operational costs resulting from heavy water losses from the heat transport system.

The following are some disadvantages of using boiling light water as the heat transport fluid: -

(1) Because of carry over of radioactive nuclei with the steam, the turbine and associated equipment may have to be shielded to avoid personnel exposure. This also restricts access to the equipment during operation.

(2) The carry over of radioactive nuclei also necessitates the reduction of leakage from the turbine and auxiliaries to avoid general contamination of the building. This is particularly necessary because of the possibility of fission products being present. This leak tightness requirement would lead to higher capital costs for the turbine and auxiliary equipment.

(3) If boiling takes place half way along a fuel channel the latter portion of the fuel channel would be cooled by steam. Further subdivision of the fuel would then be required, to increase the heat transfer area, or the thermal rating of the fuel would have to be raised and this would probably require thicker fuel sheaths. The channel diameter would also have to be increased. The net result would inevitably be poorer neutron economy.

(4) In reactors using a heavy water moderator and light water as the heat transport fluid, there would probably be an overall positive void coefficient especially if the reactor is over-moderated. Formation and collapse of steam bubbles would, therefore cause the regulating system to "hunt" in trying to keep the power steady and a more sophisticated regulating system might be required.

During start up, a large increase in reactivity occurs as more and more fluid boils. Measures, such as poison injection or use of control rods, would be required to balance this reactivity increase and enable the reactor to operate with a full calandria and, therefore, at full power.

(5) If a high pH is required in the heat transport system, pH control may be by ammonia addition rather than by ion exchange resins. This will result in more induced radioactivity.

(6) There is a likelyhood of non-symmetric flux and power distribution axially because of the density variation along the fuel channels.

Fog and Steam

Steam, as a heat transport fluid, suffers from all the disadvantages of a gas. It has low density, low thermal conductivity and poor heat transfer and heat transport properties generally. The fuel elements would therefore require high surface-to-volume ratios such as could be obtained with fins. However, stainless steel would seem to be the only structural and cladding material that could be used with steam since it has adequate corrosion resistance at 1000°F Stainless steel has a low thermal conductivity and little advantage would be gained from using stainless steel fines. Thus, extensive subdivision of the fuel elements would be required. There seems no advantage to be gained, therefore, in producing steam outside the reactor just so that it could be used, in the reactor, as the heat transport fluid.

However, if steam produced in a boiling water reactor could be superheated in a superheating section of the same reactor or in a separate superheat reactor, then the advantages are numerous. Boiling water reactor, even under ideal conditions, produce saturated steam at about 500°F. Only 30% of the heat added to produce this steam can be recovered in the turbine. The poor quality of the steam results in large turbines and low thermodynamics efficiency. Erosion occurs in the turbine unless the blades and diaphragms are made of a material that can withstand this erosion. Superheating of the steam would provide a higher utilization of reactor heat because 60% of the superheat energy added is recoverable. It would also permit the use of standard modern steam turbines.

However, many problems remain unsolved, particularly those arising from material and fuel performances. The use of stainless steel as structural and cladding material and the extensive fuel subdivision that would be required lessen the possibility of superheating in a natural uranium-fuelled reactor. Recent developments indicate that zirconium alloys could be used as sheath material but not as fuel channel material at the higher temperatures that would be possible.

The use of H_2O fog as a heat transport fluid is being seriously considered. Fog is somewhat different from saturated steam. It could be considered to be very wet steam or steam containing suspended water particles. The main advantages of using light water fog are:-

(1) The density of the fog is much lower than that of liquid water. The resulting improvement in neutron economy permits the use of fog, as a heat transport fluid, in a natural uranium - heavy water moderated reactor.

(2) The mean density along a fuel channel, with fog, is less than with boiling water. A smaller reactor core is, therefore possible if fog is used instead of boiling water.

(3) The heat transfer coefficients, with fog, are very high, being of the same order of magnitude as for liquid metals.

(4) The heat transport properties are excellent since evaporation of the droplets takes place and advantage is taken of the energy absorbed in the form of latent heat.

(5) The direct cycle is used as with boiling water.

(6) The fog, behaves as a homogenous medium compared with boiling water, which is in two distinct phases. This would probably eliminate some of the nuclear instabilities associated with the boiling water reactor. The power distribution will also be symmetrical.

(7) Zircalloy appears likely to be acceptable as a structural and fuel sheath material which may mean that UO_2 could be used as fuel in a CANDU type reactor.

The radiation problems, associated with boiling water reactors, will also occur when fog is used as a heat transport fluid.

Heavy Water

The significant differences between ordinary water and heavy water are: -

(1) The appreciably smaller thermal neutron absorption cross-section of heavy water.

(2) The production of tritium (H^3) by neutron capture in deuterium.

(3) The high cost of heavy water.

The low thermal neutron cross-section would permit the use of heavy water, boiling heavy water or heavy water fog as the heat transport fluid in thermal reactors using heavy water as moderator and UO_2 as fuel. High burnups are attainable with low fuel costs.

The tritium produced in heavy water is a serious internal radiation hazard. Leakage must be kept to a minimum to avoid personnel exposure. The leakage problem is particularly severe with the high pressures and temperatures used in the heat transport system and even more so in the direct boiling D_2O or D_2O fog cycles.

The high cost of heavy water increases the capital cost of the station and the operating costs due to leakage. This offsets to some degree the low fuel costs. There is, therefore, an economic reason for reducing leakages as well. Heavy water which has leaked from the system must be recovered and the recovery systems, collection systems and closed ventillation systems required, further add to the capital cost.

One other problem can arise with heavy water. It freezes at around 39°F and it is likely to freeze, in winter, in heat exchangers cooled with river water.

In most other respects heavy water and light water have very similar characteristics with similar advantages and disadvantages.

Organic Liquids

Many of the advantages and disadvantages of organic substances as moderators apply equally well, or more so, to their uses as heat transport fluids. The advantages may be summarized as follows: -

(1) High boiling points and low vapour pressures. Terphenyl boils at 750°F and its vapour pressure is low even at temperatures well above this. Therefore very little pressurization of the

heat transport system is required, to prevent boiling, even when reasonably high temperatures are attained to increase the thermodynamic efficiency.

(2) Low corrosion rates permit the use of standard materials for fuel channel and sheath materials, provided they are otherwise suitable.

(3) Little or no induced activity through neutron capture, since carbon and hydrogen are the only components of the pure material. Radioactive nuclei are only produced if impurities are present.

(4) Fair high temperature stability up to 800°F.

(5) Low cost (15¢ - 20¢ per lb).

The organic material under consideration is a mixture of terphenyls. The probable operating condition are 800°F and 290 psia. This relatively high operating temperature permits the use of a steam cycle with an estimated efficiency of 37.9%. Turbine erosion, due to moisture in the steam, is also greatly reduced. Both the low heat transport pressure and high station efficiency lead to lower capital and fuel costs.

There are, however, some limitations on the use of organics in the heat transport system and there are problems which require solutions. The main disadvantages with organics are: -

(1) High melting point. The melting point of terphenyl is about 250°F and it is, therefore, solid at room temperature. Trace-heating of the system is therefore required.

(2) Low heat transfer coefficients, primarily because of the poor thermal conductivity. Lateral fins may be required, on the fuel elements, to increase the heat transfer area.

(3) Organic compounds suffer radiolytic damage which increases at elevated temperatures. The effect of the radiation is to cause polymerization to longer-chain compounds. This results in the formation of tars, coke or varnish which increase the viscosity. Gases, such as hydrogen, are produced. Continual purification, by distillation under reduced pressure, is required.

It has been found that radiolytic damage is somewhat reduced if the concentration of the higher polymers is kept at around 30%. However, this high concentration tends to cause fouling on fuel element surfaces, but it drops the melting point to such an extent that freezing does not take place even at room temperature.

The major problem associated with organic materials is the development of fuel channel and sheath materials that will enable full advantage to be taken of the higher operating temperatures. The thermal conductivity of stainless steel is too low for it to be used with the poor heat transfer properties of organics. In any case the use of stainless steel also requires fuel enrichment. Zirconium is unsuitable because the hydriding effect of organics leads to hydrogen embrittlement.

A sintered aluminium product (S.A.P.) has been under development. This material consists of 6% to 10% aluminum oxide in an aluminum matrix. A zirconium-Niobium alloy has also been developed which looks promising.

Uranium carbide is being developed as a fuel material since higher temperatures are acceptable with this material than with the oxide.

Liquid Metals

Liquid metals have excellent thermal properties. They have high boiling points, low vapour pressures, high thermal conductivities. Metals of low atomic weights, such as sodium and lithium, also have fairly high specific heats and volumetric heat capacities. In addition, liquid metals are stable at high temperatures and in high radiation fields.

The heat transfer characteristics of several liquid metals are superior to those of water and the same rate of heat removal would, therefore, be possible with smaller heat transfer areas. However, liquid metal volumetric heat capacities are lower than that of water and so the liquid metals require a higher flow rate to remove the same amount of heat. The pumping power required is not too different to that required for water.

These thermal properties of liquid metals make them an excellent choice as heat transport fluids in reactors operating at high temperatures or with high power densities. No pressurization of the system is required. The main disadvantage with liquid metals is their high chemical reactivity at high temperatures. They must be protected from oxidation and must be kept oxygen free to minimize corrosion.

Table 1 lists a number of liquid metals which might be considered as heat transport fluids. Their absorption cross-sections for thermal neutrons are given.

The cross-section for Lithium-7 is very low but Li-7 is only 92.5% of lithium metal. The balance is made up with Li-6 which has a much higher cross-section. Thus, lithium would not be suitable in a thermal reactor unless the lithium-6 were removed by physical separation. The cost of such a seaparation would be excessive. Natural lithium could, however, be considered for fast reactors although it is not attractive because of serious corrosion problems.

TABLE 1

Material	σ_a (barns)
Lithium - 7	0.033
Bismuth	0.032
Sodium	0.5
56% Sodium - 44% Potassium	1.1
Mercury	380

Corrosion problems are not serious with Bismuth and it has an attractively low thermal neutron capture cross-section. Its melting point of 520°F is much too high but alloying with lead produces an eutectic whose melting point is 257°F. Lead also has a low thermal neutron absorption cross-section. A significant draw-back in the use of bismuth and its alloys is the fact that it captures neutrons to form Bismuth - 210. This is a beta emitter, with a 5 day half-life, which decays to Polonium - 210. Po - 210 is an exceptionally insiduous toxic substance which is difficult to contain. The fact that bismuth expands on freezing is an additional complication so that precautions have to be taken to prevent the liquid metal from solidifying at any time.

On the basis of neutron cross-section and heat transfer characterisitcs, sodium appears the most suitable liquid metal heat transfer fluid. It is particularly suitable in fast reactors where its lack of moderating qualities make it attractive. If free from oxygen , it does not attack stainless steels, nickel, many nickel alloys or beryllium, at temperatures below 1100°F. Mass transfer does occur at higher temperatures. It melts at 208°F and may, therefore, solidify during shutdown and requires traceheating of the system. This problem can be overcome by alloying the sodium with potassium to produce an alloy which is liquid at room temperature. However, this Na-K alloy is inferior to sodium as a heat transfer medium.

The major disadvantages with liquid sodium are as follows:-

(1) The formation of sodium - 24 by neutron capture. This is a gamma emitter with a half-life of nearly 15 hours. Consequently shielding is necessary around pipes, pumps and heat exchangers and maintenance problems are increased.

(2) The chemically reactive nature of sodium. It is easily oxidized by oxygen in air and reacts violently with water. Care

must therefore be exercised, particularly at elevated temperatures, to avoid contact of the sodium with air or water. Consideration must be given to providing a secondary sodium or mercury system between the primary sodium loop and the water system. The intermediate mercury loop prevents leakage of sodium into the water system. A secondary mercury or sodium loop prevents transfer of Na - 24 to the steam cycle in the event of a leak occuring and reduces shielding requirements.

The necessity for avoiding water also imposes problems in cooling control or booster rods. Organic coolants may well have to be used.

(3) The problem of sodium leaks. Because of the combination of viscosity, specific gravity, surface tension and wetting characteristics, sodium can leak through very small openings. A system known to be leak- tight at low temperatures may not be so at elevated temperatures. It is very difficult to provide the leak-tightness which is so necessary because of the chemical reactivity of sodium.

It is thought that sodium can percolate through the interface between carbide inclusions in stainless steel and this makes the design of a containing system difficult.

Gases

There is much to recommend the use of a gas as a heat transport fluid. They are generally easy to handle and they have good radiation and thermal stability. However their heat-transfer characteristics are inferior to those of water and liquid metals. With gases the heat transfer area has to be increased by the use of fins and this introduces more neutron absorbing material into the core. The pumping power requirement with gases, are much higher than with other fluid, mainly because of the small densities.

In order to be able to compare gases as to their suitability as heat transport fluids,a merit index is calculated based on the ratio of the heat removed to the pump work expended. The results at 1 atmosphere pressure and a temperature of 212°F are shown in Table 2.

Air is an obvious choice as a heat transport fluid, because of its availability. However, as may be seen from the table, a large proportion of the power produced would be required to pump the air around the system. Furthermore, at high temperatures, both the oxygen and nitrogen in the air will react chemically with core material.

TABLE 2

Gas	Index	Gas	Index
Hydrogen	100	Helium	18
Methane	29	Carbon Dioxide	11
Ammonia	22	Air	7.5

Radioactive Argon-41 is formed by neutron capture in Argon-40 which is found in nitrogen. This is undesirable in a closed system. The thermal properties of nitrogen are similar to those of air and it is, therefore, equally unsuitable as a heat transport fluid.

Hydrogen is the most attractive gas on the basis of the merit index and low neutron absorption cross-section. Unfortunately, it would constitute a serious explosion hazard if it escaped into the air. The containment of hydrogen at elevated temperature and pressure is a difficult problem which requires the use of special material not subject to hydrogen embrittlement.

Helium is less attractive from the heat-transfer standpoint. However, it has a negligible neutron capture cross-section and is chemically inert. It is also stable at high temperatures and in intense radiation fields. It, therefore,has much to recommend it especially in high temperature applications and as the fluid in a closed cycle gas-turbine system. It is in use in the EGCR at Oak Ridge, in Peach Bottom and in the DRAGON reactor in the U.K. High gas pressures are required to reduce pumping power (e.g. 294 psig in DRAGON) and to reduce core size. A closed system would have to be used with special precautions taken to reduce losses, because of the high cost of helium.

Carbon dioxide is inferior to helium as regards heat transfer, pumping power and neutron absorption cross-sections. It is, however, readily available and is relatively inexpensive. The absorption cross-section is small and the gas is free from the dangers of toxicity and explosion. It does not readily attack metals and the reaction rate with graphite appears insignificant below 930°F if an inhibitor is added. It is probably the most practical gaseous heat transport fluid below about 1100°F. It is used, with methane as an inhibitor, up to 1067°F in the Advanced Gas Cooled Reactor in the U.K., with enriched UO_2 fuel sheathed in stainless steel.

Comparison of D_2O Moderated Plants Using Various Heat Transport Fluids

It is not intended that the figures quoted here should be accepted as an indication or a predication of what reactor systems are likely to be adopted in the future. The figures are presented to give some idea of relative costs as they have been estimated by various groups or individuals.

Unit energy costs for D_2O moderated reactors, with various heat transfer fluids, were compared in the 1964 Geneva Paper P-10 (Pon Lewis, Haywood, Primeau, Phillips, Merlo). Their estimated costs are given in Table 3 below:

TABLE 3

	Liquid D_2O	Boiling D_2O (Indirect Cycle)	Boiling H_2O (Direct Cycle)	Organic
Unit Energy Cost (mills/Kwh)	3.829	3.714	3.689	3.580

It is assumed that the water cooled reactors use rod-bundle UO_2 fuel while the organic cooled reactor use a nested annular uranium carbide fuel.

The water cooled reactors have since been reassessed using Tube-in-shell (T1S) UO_2 fuel. The results of this assessment for station outputs of 250, 500 and 750 Mwe are shown in Table 4, which is taken from 1965 CNA Conference Paper 65-CNA-203 (Brooks & Pon) entitled "Conceptual Design of a Nautral Uranium Boiling Light Water Reactor".

Another AECL Report, AECL 1730 published in May 1963, compares unit energy costs with liquid D_2O, boiling H_2O, H_2O fog and organic liquids as heat transport fluids. This comparison is shown in Table 5.

The tables seem to indicate the possibilities of lower unit energy costs with organic liquids or boiling light water .

TABLE 4

		250 MW(e)			500 MW(e)			750 MW(e)		
		Liq. D_2O	Boil. D_2O	Boil. H_2O	Liq. D_2O	Boil. D_2O	Boil. H_2O	Liq. D_2O	Boil. D_2O	Boil. H_2O
D_2O Cost = \$20/lb.										
Specific Capital Cost	\$/kWe	361	344	333	252	236	225	216	200	190
Burnup (Idealized Core)	MWd/teU	9967	10382	9497	10914	11465	10343	11365	11997	10829
Fuelling Cost (@ \$40.84/kgU)	mill/kWhr	0.571	0.503	0.545	0.521	0.454	0.495	0.499	0.433	0.472
Capital Cost	mill/kWhr	3.539	3.375	3.270	2.472	2.315	2.212	2.117	1.965	1.859
Operation & Maint. Cost	mill/kWhr	1.039	1.031	0.943	0.697	0.690	0.643	0.572	0.564	0.531
Unit Energy Cost (UEC)	mill/kWhr	5.149	4.909	4.758	3.689	3.459	3.350	3.188	2.962	2.862
Difference in UEC from Liquid D_2O	mill/kWhr	0	-0.240	-0.391	0	-0.230	-0.339	0	-0.226	-0.326
D_2O Cost = \$10/lb.										
Difference in UEC from Liquid D_2O	mill/kWhr			-0.297			-0.264			-0.258

TABLE 5

Coolant	Liquid D_2O	Boiling H_2O	H_2O Fog	Organic
Net Elect. Output (MWe)	457	457	454	457
Plant Direct Costs ($)	59,020,000	62,080,000	60,921,000	54,953,000
D_2O Inventory (@$20/lb)	14,300,000	12,408,000	10,790,000	10,034,000
1/2 Fuel Charge ($)	2,320,000	3,320,000	3,230,000	3,400,000
Indirect Costs ($)	16,100,000	16,100,000	16,100,000	16,100,000
Contingency ($)	12,000,000	12,000,000	12,000,000	12,000,000
Interest during Constn. ($)	11,400,000	11,600,000	11,300,000	10,600,000
Total Plant Cost ($)	115,140,000	117,508,000	114,341,000	107,087,000
Specific Capital Cost ($/Kwe)	252	257	251.9	234.3
Burnup (Mwd/Tonne U)	8680	8985	7600	11,360
Unit Fuel Cost ($/KgU)	52.00	49.64	52.37	38.50
Fuelling Cost (mills/Kwh)	0.80	0.69	0.86	0.42
Unit Power Cost (mills/Kwh)	3.86	3.77	3.89	3.45
Steam Cycle Eff'cy %	34.6	36.7	36.7	38.0
Net Station Eff'cy %	31.3	33.5	33.5	34.2

ASSIGNMENT

1. List the advantages and disadvantages of using light water as the heat transport fluid.
2. What additional advantages are to be gained by allowing the water to boil?
3. Under what circumstances would steam be used as a heat transport fluid and what advantages would result?
4. Explain why fog is superior to boiling water as a heat transport fluid.
5. What are the significant differences between using ordinary and heavy water as a heat transport fluid?
6. (a) Summarize the advantages and disadvantages of terphenyl as a heat transport fluid.

   (b) Why is it not in use in natural uranium - heavy water moderated reactors?
7. (a) Why would sodium appear to be more suitable than lithium or bismuth as a liquid metal heat transport fluid?

   (b) What are the major disadvantages with liquid sodium?
8. Briefly compare air, hydrogen, helium and carbon dioxide as heat transport fluids.

A. Williams

Reactor Boiler and Auxiliaries - Course 133

MAIN CIRCUIT CONSIDERATIONS

The properties of many possible heat transport fluids have been considered and there are an even larger number of heat transport systems that can be used with them. However, there are many principles that apply equally well to several systems. All heat transport systems can be classified as either indirect or direct cycles. In the indirect cycles, heat from the reactor is used to produce steam in a boiler or heat exchanger outside the reactor. The heat may be transported from the reactor by water, organic liquids, gases or liquid metals but the heat transport systems have many similarities.

In the direct cycle steam is produced in the reactor and boiling water, steam or fog would be the heat transport fluid.

Indirect Cycle Systems

The basic heat transport system used with an indirect cycle is shown in Fig. 1. The primary loop is a closed loop consisting of a steam generator, or boiler, a circulating pump and associated pipes and valves. The heat transport fluid is circulated through the reactor to the steam generator and back to the reactor. The fluid transports the heat from the reactor to the steam generator where it is used to boil ordinary water to produce steam. The suction of the circulating pump is at the outlet of the steam generator because the heat transport fluid is coolest at this point. This decreases the probability of cavitation occurring in the pump due to the lower suction pressure. Although this is the basic heat transport system used with an indirect cycle, additions or modifications may be required depending on the heat transport fluid used. These will be discussed under the following headings.

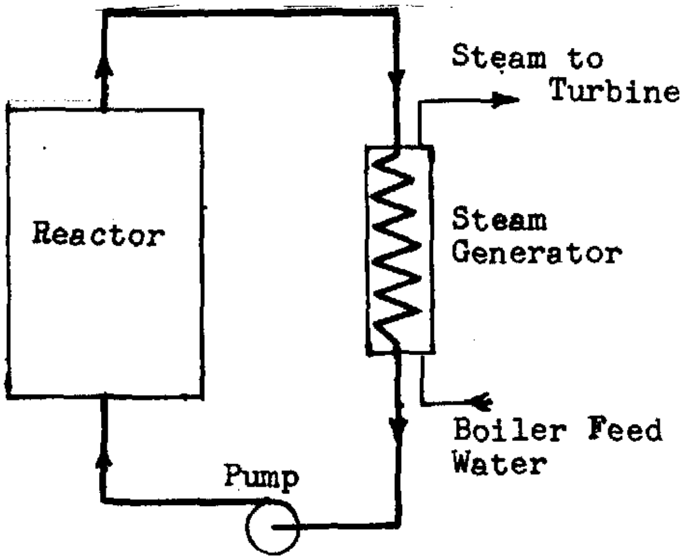


Fig. 1

(a) Pressurized Water Systems

Pressurized light water systems generally use the pressure vessel concept, in which the water, as both moderator and heat transport fluid, is in one common system. Many use the simple type of loop, shown in Fig. 1, with perhaps two or more such loops associated with each reactor as shown in Fig. 2. Two or more steam lines from separate steam drums feed into one turbine. The number of loops will be determined by the reactor power and the possible capacity of pumps and steam generators.

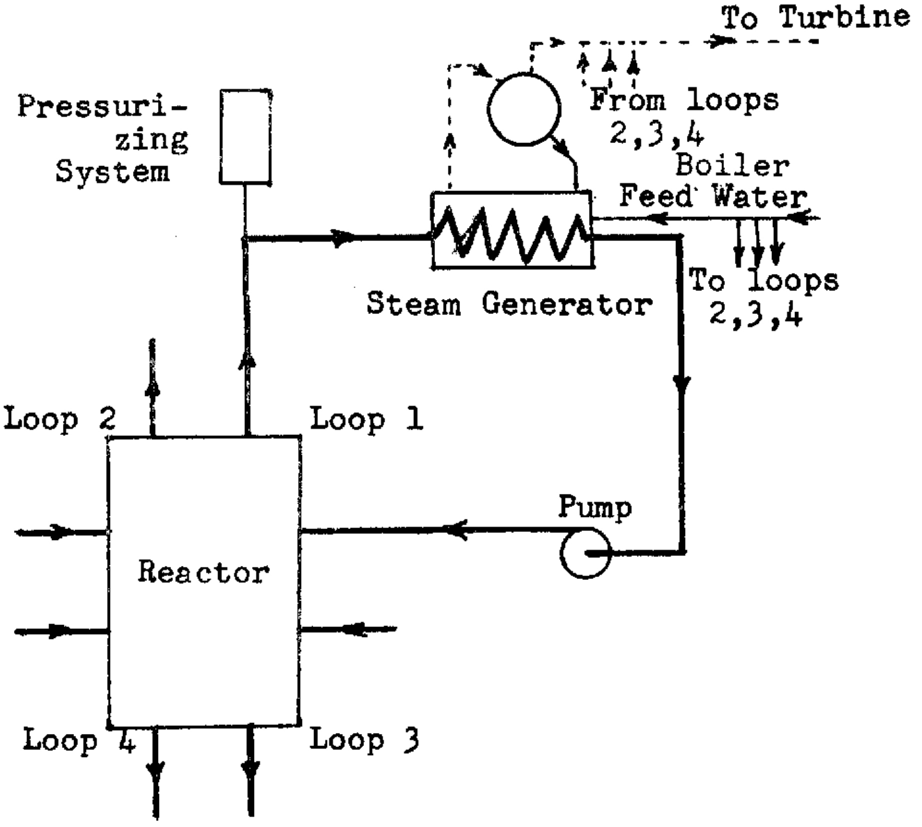


Fig. 2

A pressurizing system is required to maintain the loop pressures at 1500 to 2000 psig. Normally only one such system is required for all the loops.

In a pressurized heavy water reactor, such as the Swedish R-3/ADAM reactor, using the pressure vessel concept, the heat transport - moderator system arrangement is the same as that described above.

This type of multiloop system permits the station to operate with one loop out of service. It is likely, therefore, that valves will be installed in each loop so that the steam generator and the circulating pump can be isolated for maintenance. Such valves, if remotely controlled, could also be used to isolate one loop if leaks developed in it. If each loop is separately shielded, as in the Shippingport reactor, maintenance can be carried out on the isolated loop while the reactor is operating.

There are two types of pressure tube reactors. In both types the moderator used is heavy water, which is kept separate from the heavy water heat transport fluid. The pressure tubes or fuel channels can be vertical, as in the Carolinas Virginia Tube Reactor, or they may be horizontal as in the CANDU type. The vertical pressure tube system would use the primary loop system discussed above because it is a simple system and nothing more complicated is required.

In the CANDU system, advantage is taken of the fact that the fuel channels are horizontal, to use bi-directional fuelling. Since fuel moving is carried out against the direction of flow, the heavy water heat transport fluid is arranged to flow in opposite directions in adjacent channels. One of two arrangements are now possible as shown in Figs. 3 and 4. In the system shown in Fig. 3, the fuel channels, in which the flow is from right to left, feed through feeder pipes to a common header. The flow from left to right, through the reactor channels, passes into another outlet header. The flow from both headers pass through a common line, to the steam generator and thence, to the circulating pumps. The outlet of the pumps is fed to two inlet headers. One inlet header feeds the fuel channels in which flow is from right to left, and the other inlet header feeds the channels in which flow is from left to right. In effect the left-to-right and right-to-left reactor flows are in parallel and one loop only is required if it can cope with the required heat removal rate. If the heat removal rate is too high, then several steam generators can be placed in parallel and several pumps used in parallel. However, if the flow rates require prohibitively large pipe lines, two or more such loops would be necessary.

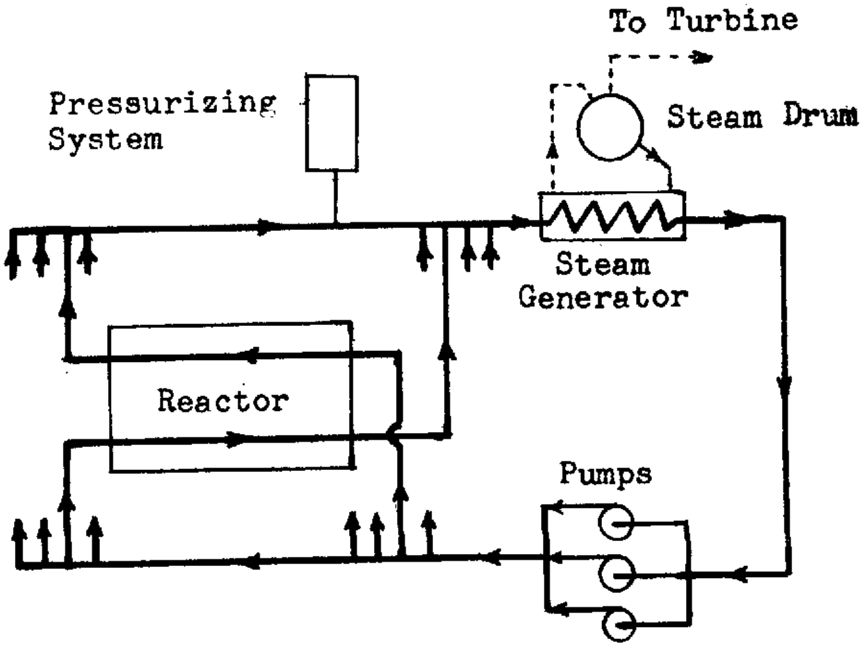


Fig. 3

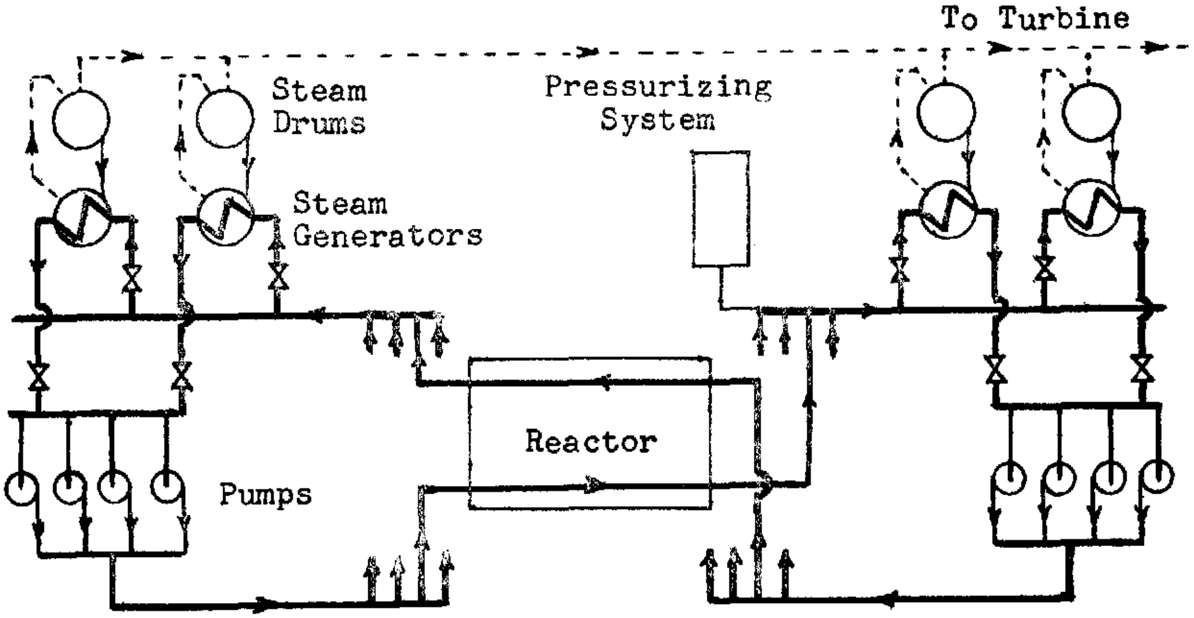


Fig. 4

Figure 4 shows an alternative arrangement. Here the right-to-left heavy water flow passes through feeders to outlet headers to the left of the reactor. These outlet headers feed one or more steam generators in parallel. The water from the outlet of these steam generators returns through pumps to the left side inlet headers of the reactor. Feeder pipes then carry the water to the fuel channels in which the flow is from left-to-right. The water is reheated as it passes through the reactor and feeds through the right side outlet headers, through one or more steam generators in parallel, through the pumps and back to the right side inlet headers. The cycle is then repeated. In effect the left-to-right and right-to-left reactor flows are in series in the one loop. There are, essentially, two heat sources and two heat sinks in the loop. Additional heat removal capacity can be installed by multiplication of steam generators and circulating pumps in the same loop. However if the heat removal rate required results in flow rates which require prohibitively large pipes, more loops would be necessary.

If multiple steam generators, pumps or loops are used, valves would probably be installed for isolation of steam generators, pumps or loops.

There are several principles and equipment considerations which apply to all pressurized water systems :-

1. The most important principle that must be applied is that IT MUST BE POSSIBLE TO REMOVE HEAT FROM THE FUEL AT ALL TIMES.

   Heat is produced in the fuel by fission of fissile material and by decay of fission products. Over 6% of the total heat production is by decay of fission product, as shown on the extreme left of the graph in Figure 5. Figure 5 also shows how the total reactor thermal power decreases, following a reactor trip. There is an initial rapid decrease to about 6% or 7% of full power in 10 or 20 seconds. Thereafter the decrease is slow because most or all of the heat is produced by fission product decay. If this heat is not removed, even during reactor shutdown, the temperature rise in the fuel would cause fuel sheath rupture or, possibly, melting of the fuel, sheath and fuel channels.

   There are many consequences of this heat removal requirement, some of which apply directly to the main heat transport circuit. The reactor is placed at a lower point in the system than the steam generator so that in the event of loss of circulating pumps, heat will be transported from the fuel to the heat sink by natural convection or thermosyphoning. This method of heat removal should be sufficient to remove a maximum of about 6% of full reactor power. To ensure that thermosyphoning is available at all times when an alternative heat removal system is not in use, an interlock would prevent isolation of all the steam generators or interruption of main circuit flow, unless an alternative system was first made available. It must be realized that thermosyphoning can only lower the heat transport fluid temperature to about 250°F because the water in the steam generators require a temperature of 212°F or higher to boil and, thus, provide a heat sink. An alternative method of heat removal is required for temperatures below this value.

   If there is insufficient flow through the heat transport system due to loss of one or more pumps, a reactor trip normally occurs. However thermosyphoning cannot cope with the required heat removal for perhaps 20 seconds or more. It is, therefore, necessary to fit the pumps with flywheels so that a sufficiently long pump run-down time is ensured to provide the forced convection required to remove the heat generated during this period of time.

   Further consequences of the heat removal requirement will be considered as they apply to the auxiliary systems.

2. The use of multiple loops, steam generators or pumps increases the reliability of the system. The heat removal rate or flow requirements may dictate the use of multiple loops or multiple

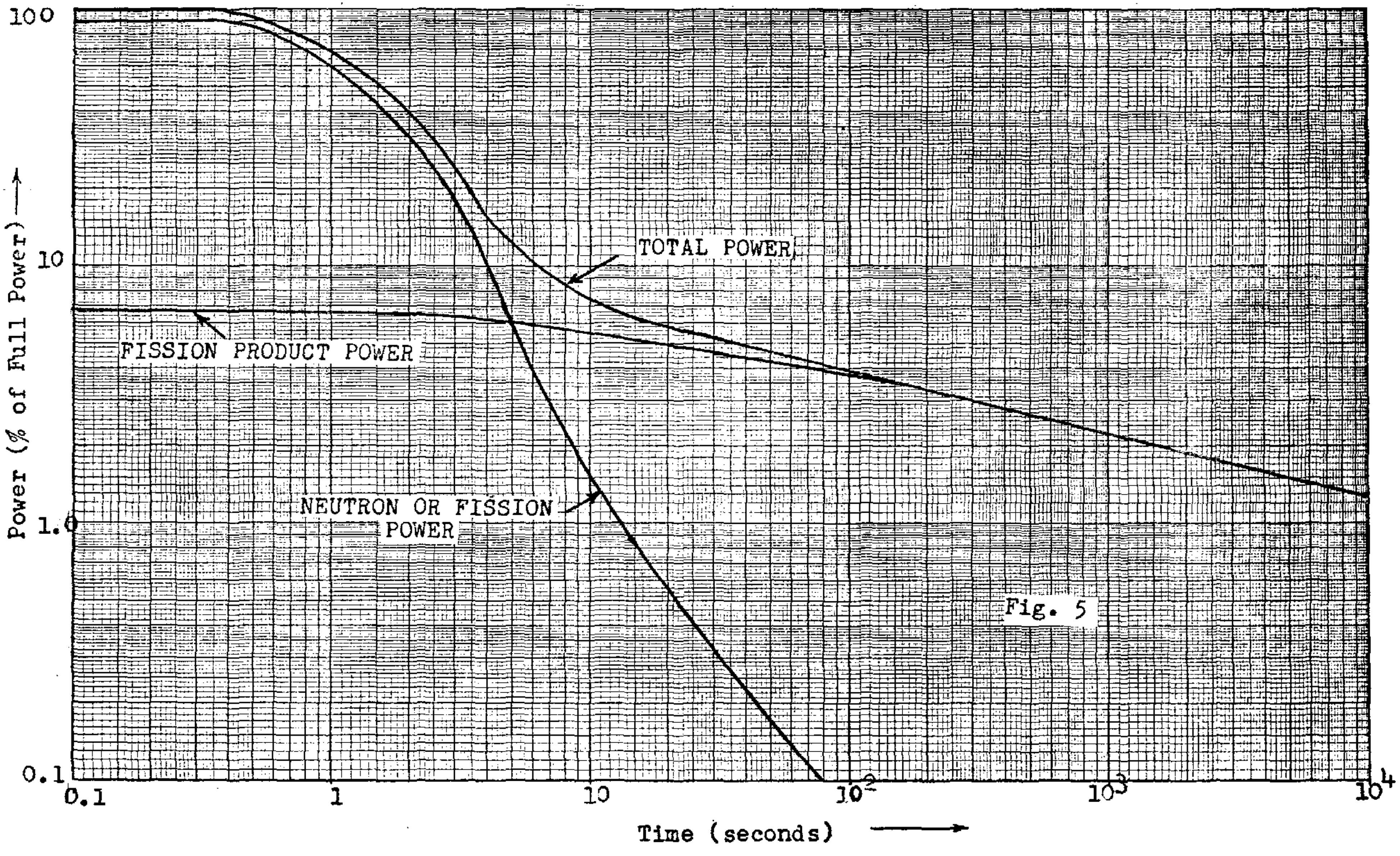

100
10
1.0
0.1
0.1
1
10
10^2
10^3
10^4
Power (% of Full Power) ⟶
Time (seconds) ⟶
TOTAL POWER
FISSION PRODUCT POWER
NEUTRON OR FISSION POWER

Fig. 5

components because large enough steam generators and pumps are either unavailable or only available at a prohibitive price. Nevertheless the system reliability is increased by using more than one loop, more than one steam generator or more than one pump, in parallel and this may well be a major consideration. If one loop or one steam generator becomes defective it can be isolated and sufficient excess capacity built into the remaining loops or steam generators to allow operation at full or slightly reduced power.

One or more pumps would be installed as standby pumps so that they can be used to replace defective pumps. Such a changeover of pumps could possibly be made while the reactor continued to operate and it would have to be possible during the poison override time following a reactor trip. With such an arrangement, check valves are required in the pump outlet lines to prevent backward rotation of the stationary pumps. A limited flow of system water might be required, through the stationary pumps, to keep them at or near system temperature. This would avoid stresses in the pump casing on pump changeover.

3. The circulating pump suctions are connected to the outlet of the steam generators which will be at the lowest temperature in the system. This decreases the probability of the water flashing to steam, at the lower pressure, and causing vapour locking of the pumps.

4. Decomposition of the water is suppressed by addition of hydrogen (or deuterium). It is not then necessary to provide degassing and recombination facilities.

Equipment considerations, in pressurized water systems, are as follows:-

1. Pumps

Parallel pump operation is again required and pump requirements will, therefore be similar to those for the moderator system. Large flow centrifugal pumps with medium discharge pressures and non-drooping head-flow characteristic curves, would be used. Because of the high pressures and temperatures in the heat transport system the leakage problem is much more severe than in a separate low pressure--low temperature moderator system. Leakage in the pumps, in the U.S. pressurized light water reactors, is minimized by using the canned rotor pumps previously discussed. However, because of the high capital cost of canned rotor pumps, leakage in the CANDU heavy water systems, have been minimized by using shaft sealed pumps. When such pumps are used, double high-pressure shaft seals must be used, the space between the seals being bled off to a collection system. The alternative is to seal the shaft with throttle bushings backed up by low-pressure mechanical seals to retrict

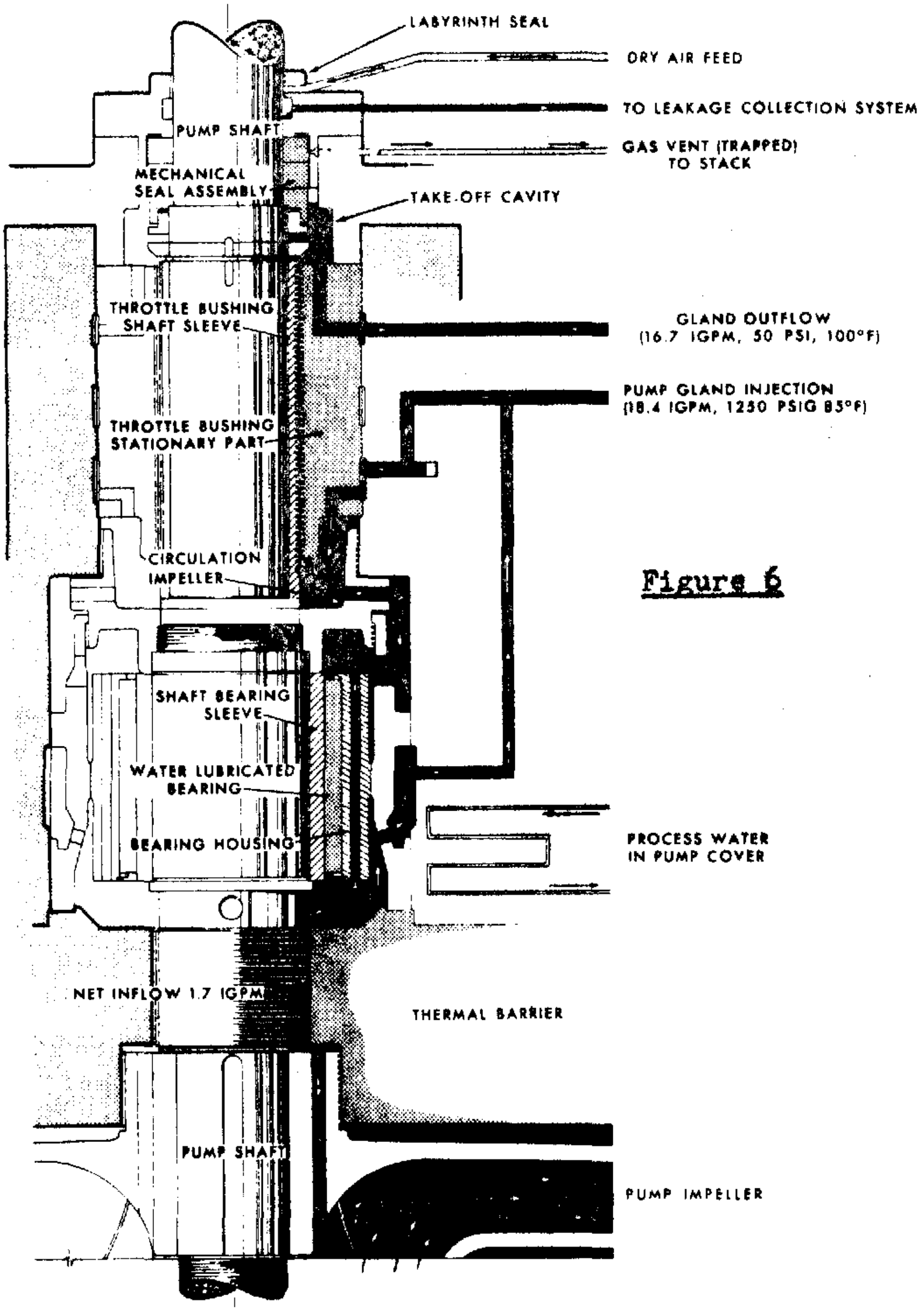


Figure 6

leakage. Such an arrangement is used in the pump shown in Fig. 6, which is the type of pump used at Douglas Point G.S.

It may be seen, from Figure 6, that cooling of the seal chamber is required. Because of the lower pressure at the pump suction, it is possible that dissolved gas may come out of solution inside the pump casing. Provisions are, therefore, required for effective venting of the pumps particularly prior to starting them. Such a venting system would feed into the collection system.

The pump bowl and casing would probably be made of cast stainless or chrome steel to reduce erosion problems.

2. Steam Generators

In the steam generators or boilers heat is transferred from the heat transport fluid to ordinary water, which is boiled to produce steam for the turbine. The most common type of steam generator is the U-type tube-and-shell heat exchanger with steam

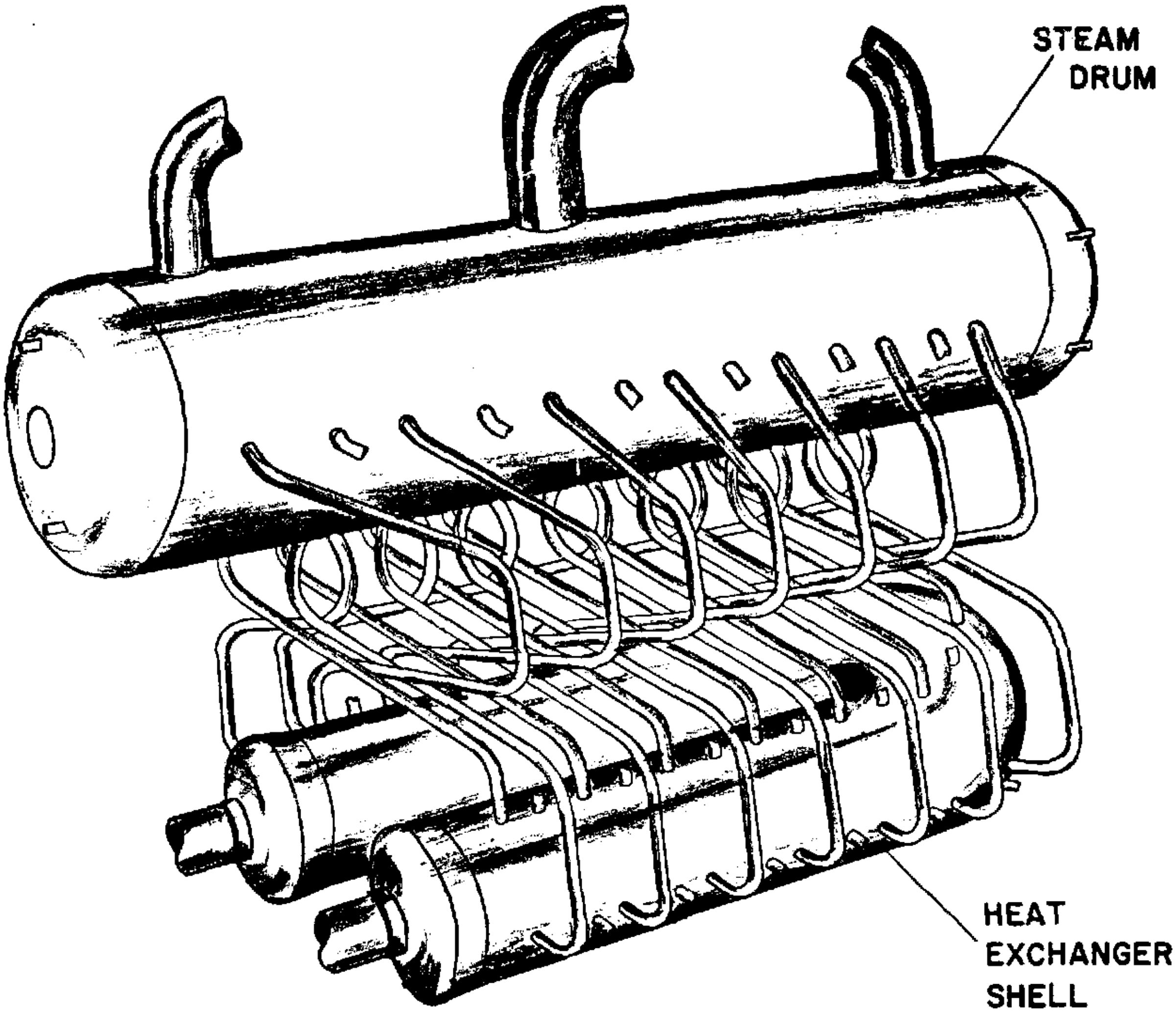


Figure 7

drum. The heat exchanger may be of the horizontal type shown in Fig. 7 or the vertical types in Figs. 8 & 9. The steam drum may be separate from the heat exchanger and there may be several heat exchangers associated with one steam drum. One steam generator in Douglas Point G.S., has ten heat exchangers, of the type shown in Fig. 8 with one steam drum.

Water at saturation temperature flows from the steam drum, through the downcomers and into the bottom of the heat exchanger. It boils in the heat exchanger and the steam enters the drum through the risers. The steam and water are separated in the steam separators and dry saturated steam passes to the turbine.

On the other hand an integral heat exchanger and steam drum may be used. Such a unit, 12 of which are used in Pickering G.S., is shown in Fig. 9. This integral arrangement is considered more economic and easier to transport and install.

Recirculation takes place inside the shell in an annulus formed by the outer shell and an inner shell which prevents the downcoming water from flowing directly onto the tubes. The lower part of the boiler is a preheater section and the baffles are used to prevent unacceptable thermal stresses in the tube sheet at the preheater.

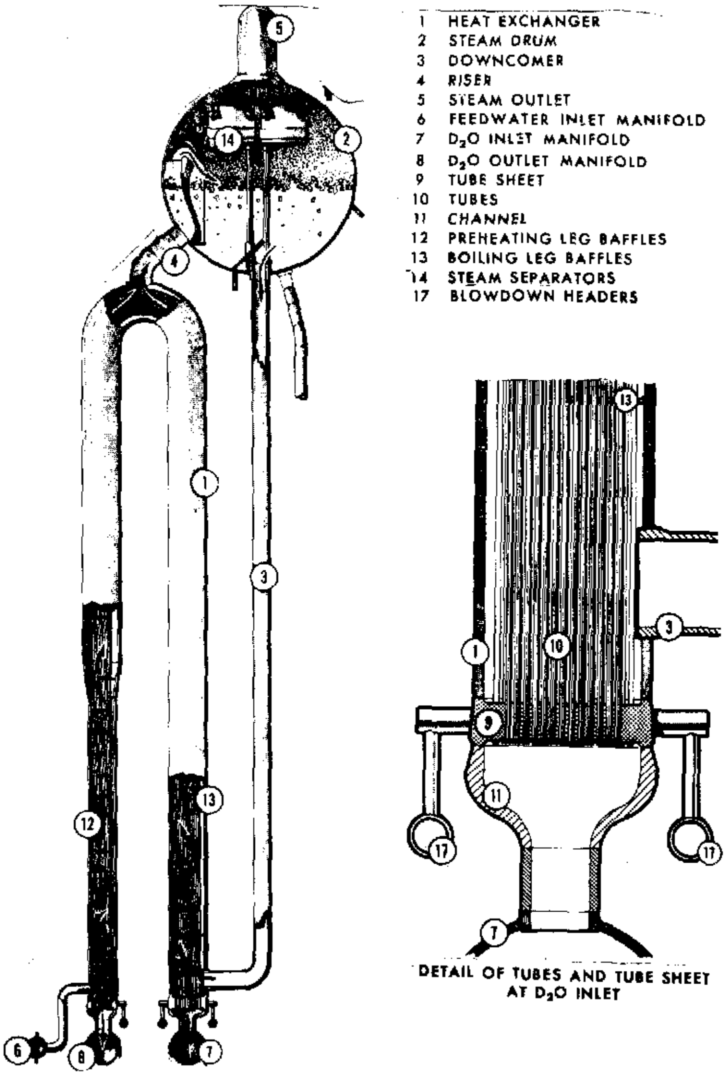


Fig. 8

In all three types the heat transport fluid, which is at a higher pressure than the light water, will pass through the tubes. The tubes will be made of a material, such as Monel or Inconnel, which is free of stress corrosion. The tube to tube-sheet joints are rolled or expanded and back-welded helium tight to reduce leakage to a minimum, particularly in a heavy water system. The tube sheets would also be of Monel or Inconnel or carbon steel clad with such metals. The shell side and the steam drum, with their associated risers and downcomers would be of carbon steel. The unit would be of welded construction in accordance with the ASME Boiler Construction Code, Section VIII for Unfired Pressure Vessels, and the requirements of the Tubular Exchanger Manufacturer's Association and the appropriate Provincial or State regulations.

The steam generator will have welded blowdown and blowoff connections and either the steam drum or the main steam line from the drum will have relief valves to avoid overpressurization.

3. Pressurizing System

One of three methods can be used to pressurize the heat transport system. The simplest method is that illustrated in Fig. 10. A pressurizing tank is connect to the system so that the tank is about half full of system water. The tank is also connected to a pressurized gas supply through a valve A. This valve, A, opens when the system pressure drops below a predetermined set point.

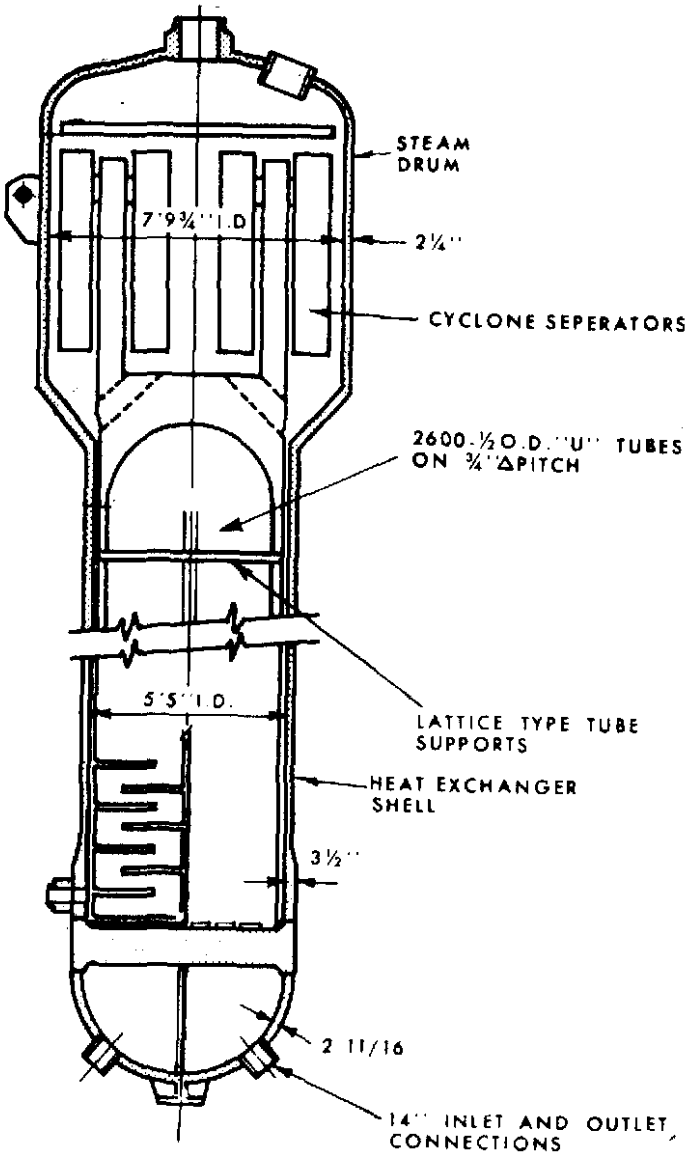


Fig. 9

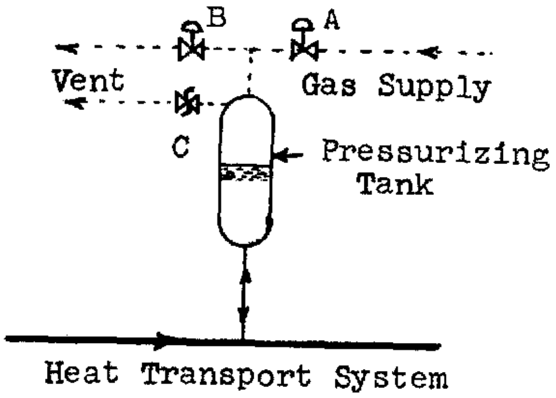


Fig. 10

If the system pressure increases above a preset value, the valve B opens to vent the tank. A relief valve, C, prevents overpressurization if B fails closed or if the pressure rise is too rapid.

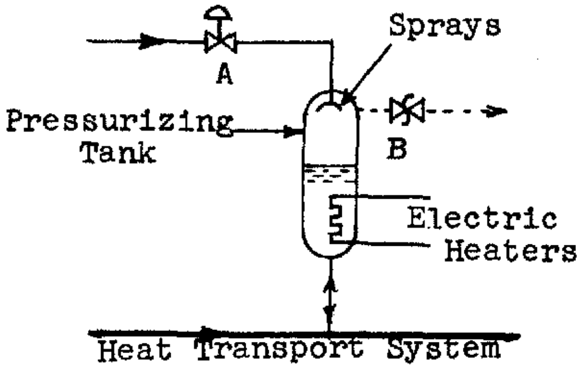


Fig. 11

A more commonly used method of pressurization is that shown in Fig. 11. Again a pressurizing tank is used and connected to the system so that it is about half full of water. Electric heaters are immersed in the water and are used to heat the water until the vapour pressure in the upper part of the tank is equal to the desired system pressure. The heaters are then controlled with heat transport pressure signals so that they maintain this pressure within narrow predetermined limits. It is possible that a spray system may be incorporated in such a tank for rapid depressurization of the system, if required, by condensation of the vapour. The water supply for the sprays would be taken from a point, in the heat transport system, at the highest pressure, such as at the discharge of the circulating pumps. The supply to the sprays would be controlled by valve A, which would be remotely operated. It is possible to continuously control the pressure in the tank, by using the sprays, if valve A is controlled by the system pressure. However, this tends to degas the water in the system which would lead to hydrogen removal from solution. The relief valve, B, prevents overpressurization of the system.

In both the above methods of pressurization, the pressurizing tank will be at the highest point in the system. The pressurizing tank would be of carbon steel and manufactured according to the appropriate ASME code for unfired pressure vessels.

In the third method of pressurization the system is "solid" ie, there is no gas or vapour space above the system fluid. Use is made, instead, of the natural compressibility of the water and a bleed and feed system is used as shown in Fig. 12.

Changes in pressure will occur in the system particularly during warming up (swell) or cooling down (shrinkage) of the system. If the pressure increases above a predetermined set point, valve A opens allowing bleed-off of water from the system, through the cooler, to the storage tank. The rise in level, in the storage tank displaces the cover gas into a suitable container. Valve

A closes as soon as the correct pressure is attained. If the pressure in the system falls below a predetermined set point, valve B opens, allowing inflow or feed of water from the storage tank into the system. The feed pump, which produces this inflow, must have a discharge pressure in excess of the system pressure. The compressibility of water, at about 1250 psig and 560°F, is such that the net addition of 1 cu. ft. of water to the heat transport system raises the pressure by about 50 psi.

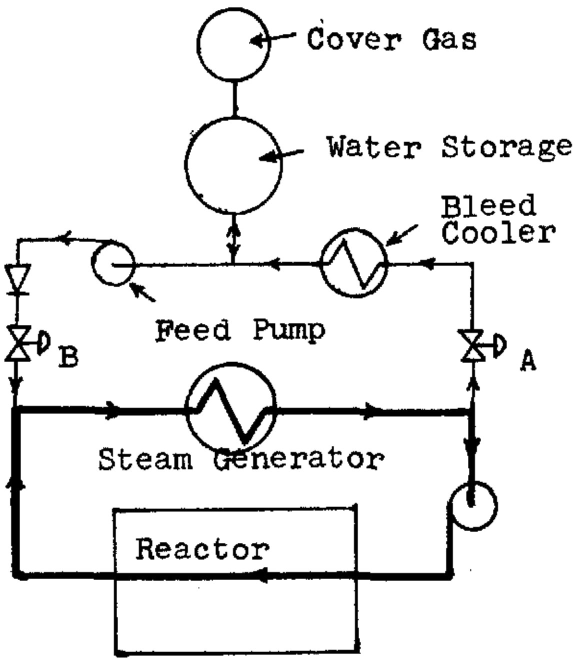


Fig. 12

It may be considered necessary to keep the feed pump operating continuously, in which case a small steady bleed from the system may be allowed which is balanced by an equal steady feed into the system. As may be seen later such an arrangement could be used as a purification circuit as well.

4. Piping and Valves

Since the heat transport system transfers heat from the fuel to the steam generator, it could contain fission products should a fuel sheath fail. The fuel sheath may be regarded as the primary fission product containment. It is imperative, therefore, that the heat transport system, which is the secondary containment, be reliable with a minimum of leakage out of it. With heavy water as the heat transport fluid, there is an additional economic reason for maintaining minimum leakage and a further reason is that of preventing the escape of tritium. Because of the high pressures and temperatures involved, reduction of leaks becomes a major problem.

Pipes would likely be manufactured from carbon steel such as ASTM A-106 grade B standard. The system must be of welded construction complying with either the requirements of the ASA B31.1, Code for Pressure Piping or, as in the case of large headers, with Section V111 of the ASME Boiler and Pressure Vessel Code for Unfired Pressure Vessels. Conditioning of the system will be required, by circulating water at 300°F through the system, in order to cover the pipe inner walls with a protective oxide and, thus, reduce corrosion.

Types of valves will be determined in the same way as for the moderator system. The valve bodies will, however, be made from carbon steel. Large valves will be provided with double stem seals and the inter-seal space drained to a closed collection system. A minimum number of small valves should be used and they should either be glandless types or double stem seals should be used.

Again the piping and equipment arrangement should be such as to minimize heavy water holdup but the congestion should not be such as to make maintenance so difficult that accidental spillage becomes more likely. Means must be provided for inserting isolating gas bags in the lines and for draining equipment or pipe sections.

(b) Gaseous System

Reactors using gas as the heat transport fluid can be divided into two groups:-

1. The "low" temperature systems which operate at temperatures below 1000°F because they generally use natural uranium metal and carbon dioxide gas. The alpha to beta transition in uranium restricts the fuel temperature to below 1220°F (660°C) and the gas temperature is kept below 1000°F because of the occurrence of a CO_2 - graphite reaction above this temperature.

   The gas pressure used in the heat transport system is generally around 100 psig to 400 psig. Pressurization of the system is not, therefore, a major problem except in so far as it affects the fabrication of the very large pressure vessels which are used. Multiple heat transport loops are used and the only significant difference between pressurized gas and pressurized water systems is in the design of the heat exchangers. The heat transport outlet temperature is around 650°F to 750°F and these temperatures permit some degree of superheating of the steam. The heat exchangers are usually of the tube and shell type with finned tubes to improve the heat transfer characteristics. Each heat exchanger is fitted with economizer, evaporator and superheater stages for both high pressure and low pressure steam. Typical heat exchanger arrangements are shown in Figures 13 and 14.

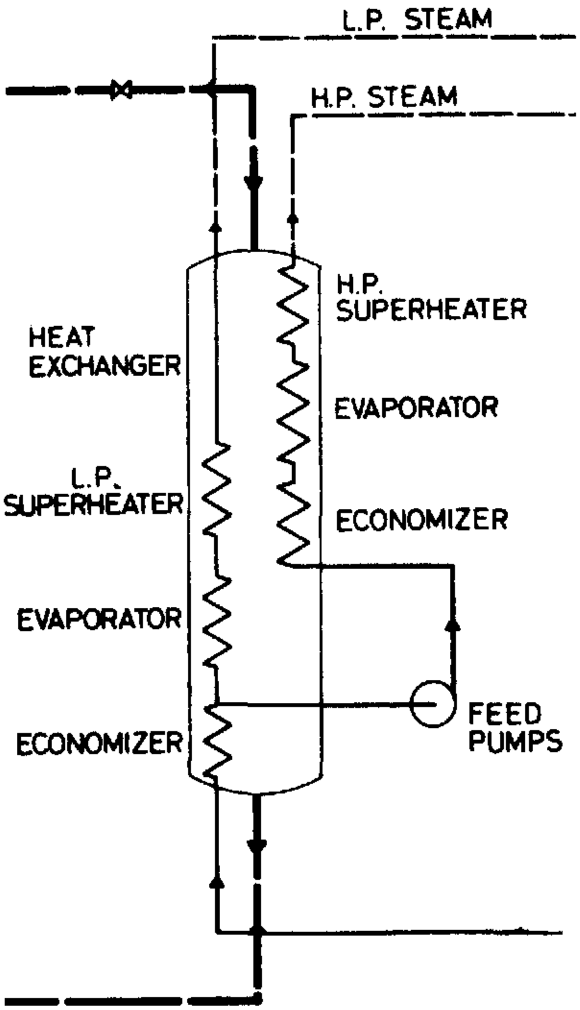


Fig. 13

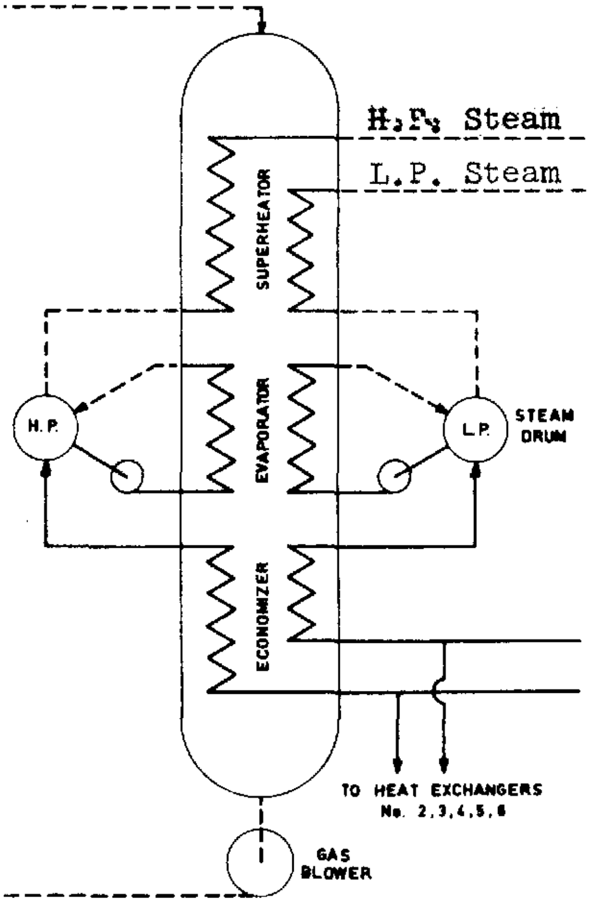


Fig. 14

2. The high temperature reactors which operate above 1000°F. These all use enriched uranium, generally as UO_2 and the fuel sheaths are made from stainless steel, zirconium, hastalloy, inconnel or graphite. The best heat transport fluid is helium since the CO_2 - graphite reaction still restricts the allowable CO_2 temperature. The higher temperatures permit a greater degree of steam superheating but the heat exchanger arrangement would probably be similar to those used with "low" temperature systems.

(c) Organic Liquid Systems

The major advantages of using organic liquids as heat transport fluids are the low pressures required in the system, even at temperatures up to 800°F, and the degree of steam superheating that can be obtained with these higher heat transport temperatures. Because of the low pressures involved (less than 200 psig) pressurization of the heat transport system may be achieved by simply applying gas pressure to the free surface of the liquid in the reactor itself or in an external surge tank. Alternatively pressurizing pumps can be used.

No hazardous chemical reaction occurs between organic liquids and water, and, therefore, the heat can be transferred directly from the primary organic circuit to boil the feedwater. An arrangement similar to that shown in Figure 15 would be used, with an evaporator, or boiler, and superheater. Both evaporator and superheater would be constructed of carbon steel and both would probably be of the shell-and-tube type.

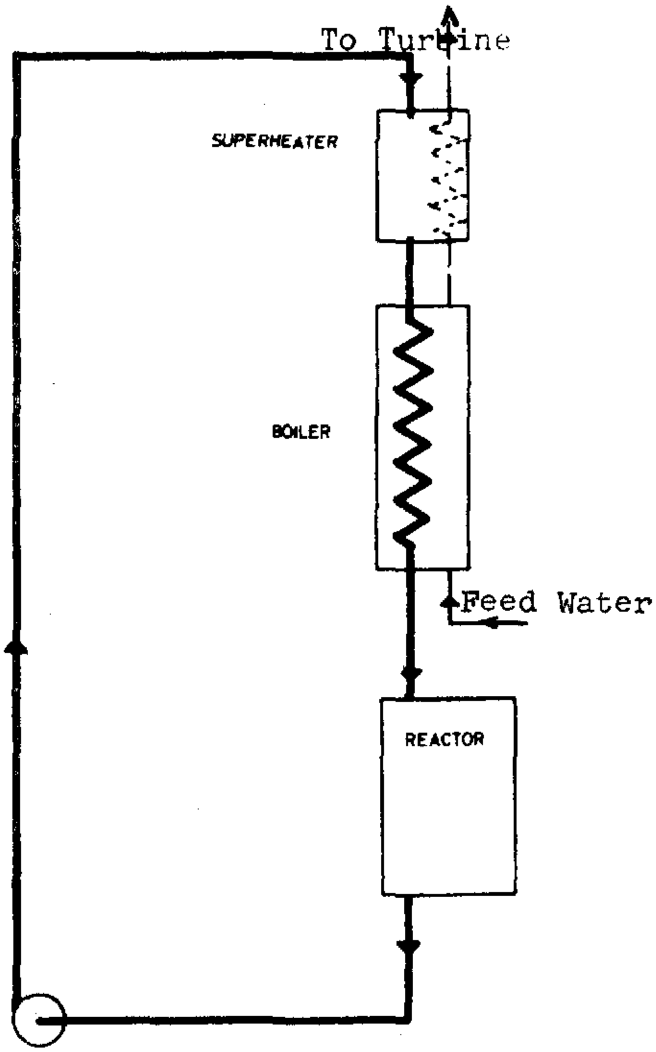


Figure 15

In the boiler the organic liquid would be on the tube side whereas, in the superheater, the steam would flow in the tubes and the organic liquid would be on the shell side. The boiler would have an internal steam separator to minimize water carry-over to the superheater.

Conventional "hot oil" pumps can be used to circulate the organic liquid round the heat transport system.

(d) Liquid Metal Systems

The extraordinary heat transfer properties of liquid metals make them attractive as heat transport fluids, particularly in fast reactors where the poor moderating properties of sodium, for example, make it particularly suitable. No pressurization of the system is required and the only pressurization which is likely to occur would be due to the circuilating pump discharge pressure or any static heads in the system. Liquid metals are capable of operating at the high temperatures required to generate superheated steam for modern turbines. The two most extensively used liquid metals are sodium and a sodium-potassium eutectic. The former is probably the better heat transport fluid but the latter is liquid at room temperature whereas traceheating would be required with sodium. There are two major problems when using either of these hot liquid metals to produce steam from water:-

1. The liquid metal systems are highly radioactive because of the formation of sodium - 24.

2. These liquid metals react violently with water.

Because of these problems, two circuits are used, as shown in, Figure 16. The primary liquid metal circuit, which is radioactive, transports heat from the fuel to an intermediate heat exchanger where it is used to heat a secondary fluid which is also a liquid metal.

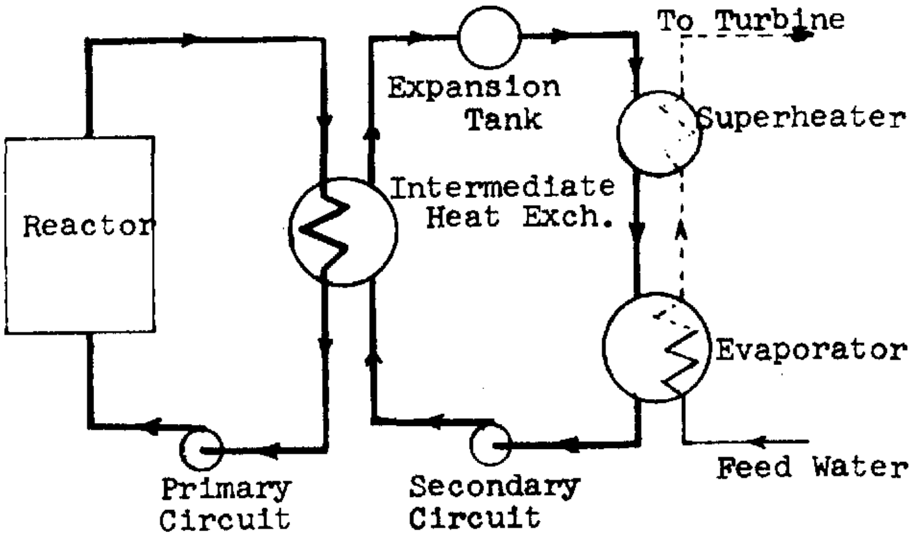


Fig. 16

The secondary circuit, normally not radioactive, has an evaporator and superheater to produce superheated steam. Both evaportator and superheater may be incorporated in the same steam generator. If mercury is used as the secondary fluid, there will be no reaction between it and the water in the steam generator. The evaporator and superheater could then be conventional tube-and-shell type exchangers with water or steam on the tube side.

If liquid sodium or sodium-potassium eutectic is used in the secondary loop, reaction between the secondary fluid and the water must still be prevented in the event of a tube failure. This is accomplished, as shown in Figure 17, by using double-walled tubes and double tube sheets. The liquid metal will then be on the shell side with water and steam in the inner tubes. The annular space between the double walls and the space between the tube sheets is filled with mercury or helium. This intermediate fluid prevents contact between the liquid metal and the water in the event of a tube failure and also acts as a monitoring fluid for detecting tube failures.

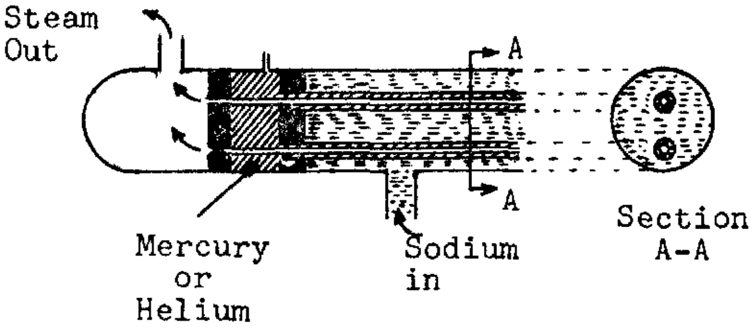


Figure 17

The intermediate heat exchanger, the evaporator and the superheater would probably be made of stainless steel.

The circulating pumps used with liquid metals may be centrifugal mechanical pumps similar to conventional pumps of this type. Leakages along the shaft can be reduced by having a liquid free surface inside the pump with a cover gas. Leakage of the cover gas is prevented with oil-filled, double-face shaft seals. Alternatively a freeze-seal is formed on the shaft by freezing the sodium leaking along the shaft.

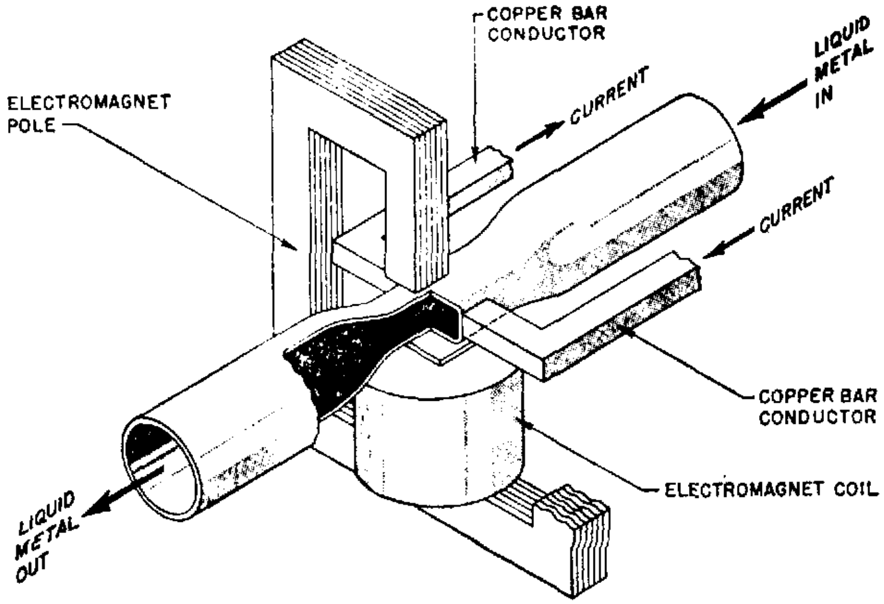


Figure 18

Advantage can be taken of the electrical conductivity of liquid metals to force it to flow along a pipe under the influenze of a magnetic field. The "electromagnetic" pump, shown schematically in Figure 18, results. A current is passed, through the liquid metal, in a direction perpendicular to a magnetic field. The liquid metal, being a current-carrying conductor, is exerted upon by a force which moves it along the pipe. The advantage of such an arrangement is that there are no moving parts except the fluid being circulated. However such pumps have low efficiencies and are, usually, only used on small lines where flow is occasional.

Direct Cycle Systems

In direct cycle systems the heat transport fluid is also the thermodynamic heat engine fluid which passes into the turbine. This fluid can be steam produced in the reactor and used to drive a steam turbine or it can be a gas which is heated up in the reactor and then passes to a gas turbine. Such heat engine fluids can be produced by a number of different heat transport systems:-

(a) Boiling Water Systems

Boiling water systems can, of course, be used in an indirect cycle in the same way as the pressurized water systems. The direct cycle however offers some advantages, which have already been discussed, and most boiling water reactors use the direct cycle. The simplest boiling water system is the "single" cycle system shown in Figure 19. Steam, produced in the reactor, passes to a steam drum which contains separators to remove entrained water from the steam. Dry saturated steam then passes to the turbine and the separated water is circulated back to the reactor. The feed water return is passed directly to the reactor, or flows to the steam drum where it mixes with the water in the drum and is returned to the reactor.

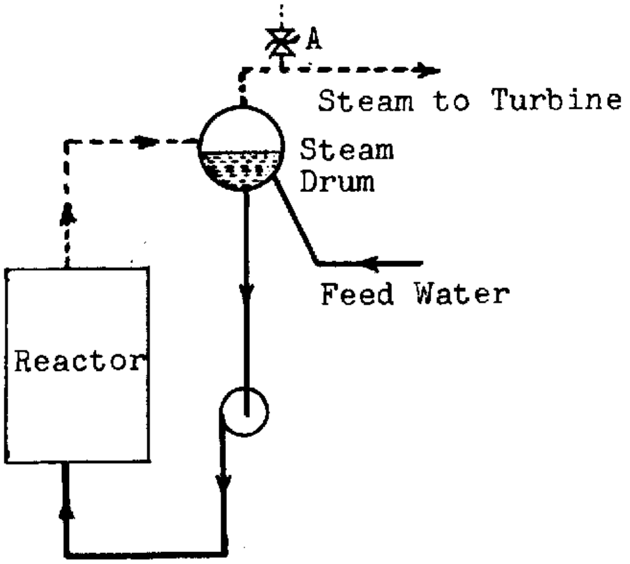


Figure 19

Several such loops may be required depending on the reactor power. In a pressure tube boiling water reactor, the steam from several fuel channels would feed into one steam drum and again several such steam drums may be required depending on the rating of the reactor.

No pressurizing equipment is required. The steam pressure produced depends on the rate at which steam is produced and the rate at which it is passed into the turbine. The steam pressure will rise until the two balance. Steam pressure can be increased further by increasing the rate of production, (i.e. increasing reactor power), or decreasing rate of flow into the turbine. The normal operating pressures are about 500 psia to 1000 psia and are, generally lower than in the pressurized water systems.

Piping and equipment requirements will be similar to those in pressurized water systems. Over-pressurization is prevented by means of safety valves such as A.

In the "dual" cycle system, shown in Figure 20, the reactor heat is carried off partly as steam and partly as hot water. The hot water is used to produce steam in a secondary steam generator either by flashing or by boiling feed water. The steam produced in the secondary steam generator is at a lower pressure than the primary steam and goes to intermediate stages of the turbine. This cycle improves the control characteristics of the reactor and results in a more stable system.

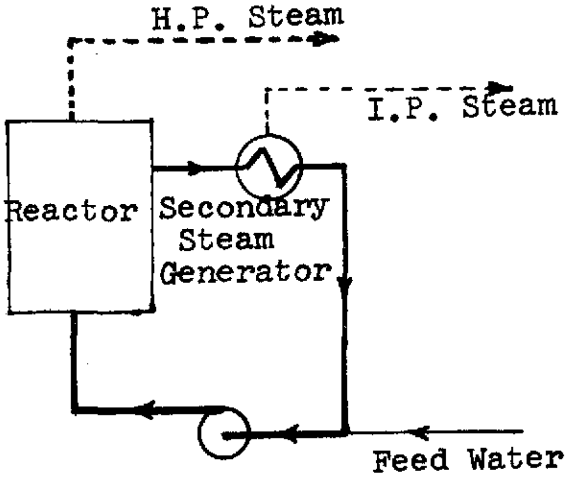


Figure 20

Figure 21 shows a modification of this "dual" cycle system which is used in the Dresden Station. Secondary steam, at a lower pressure than the primary steam generator. The heat required to boil the water is obtained from the water separated from the primary steam in the primary steam drum. The turbine governor is set to control the output of this inter-

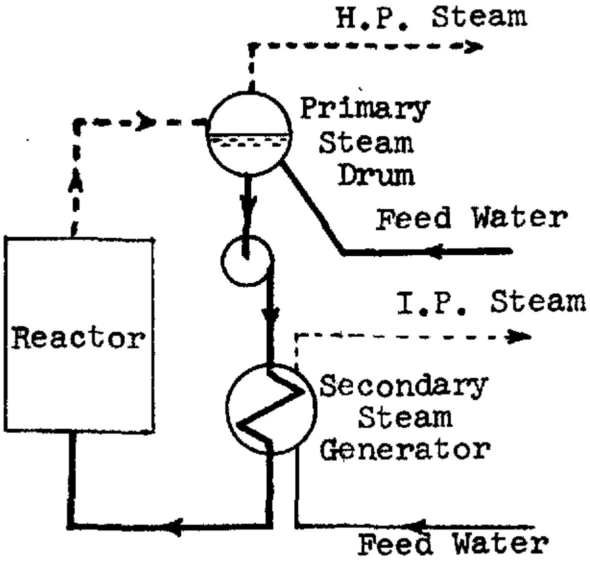


Figure 21

mediate pressure steam so that primary steam pressure does not vary with change in load.

The boiling superheat reactors use a "single" cycle system which is even simpler than that shown in Fig. 19. Superheated steam is obtained by passing steam, generated in the outer section of the reactor, through a central superheating region of the same reactor. No separators or steam drums are, therefore, required and the steam is passed directly into the turbine. The feedwater return from the turbine is pumped directly into the reactor.

(b) Fog Systems

The heat transport system, using fog as a heat transport fluid, is very similar to that using the boiling water system. Figure 22 illustrates such a system. The fog is produced by mixing some of the steam from the separator outlet with water, which was separated from this steam. The fog, as it passes through the reactor, is converted into steam. Entrained water is removed in the separator and saturated steam passed to the turbine. The steam content at the inlet to the reactor is about 15% to 25%. As heat is added to it, in the reactor, the steam content increase to 50% to 60% steam.

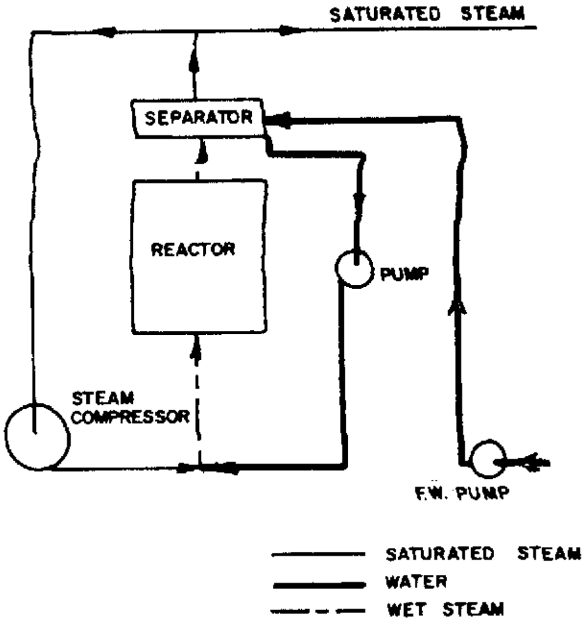


Figure 22

(c) Gas Turbine Systems

If high temperatures can be attained in a gaseous heat transport fluid, the gas can be passed directly into a gas turbine. This gives a similar advantage to that obtained with superheated steam. A direct cycle is used which does not require steam separators or steam generators.

ASSIGNMENT

1. Explain (a) Why, in a pressurized water system, the reactor is placed at a lower point in the system than the steam generator.

(b) Why the circulating pumps in such a system are fitted with flywheels.

2. What are the advantages of using multiple loop systems or multiple steam generators and pumps in a single loop?

3. Explain two methods of pressurizing water systems that are in common use.

4. What is the basic difference between a pressurized water and a pressurized gas heat transport system?

5. Explain the similarities and the differences between heat transport systems using organic liquids and liquid metals.

6. Describe one special design feature of steam generators or superheaters used with liquid metals.

7. Explain the essential difference between the "single" and "dual" direct cycles and the advantages of using the latter.

A. Williams

Reactor Boiler and Auxiliaries - Course 133

AUXILIARY CIRCUITS

The main types of heat transport systems have been discussed but considerations were only given to the circuit necessary to transport the heat from the fuel to the steam generator or turbine. Several auxiliary circuits are also associated with the main heat transport system. These may differ, in detail, according to what heat transport fluid is used but the function they serve is usually the same in most cases.

Standby Cooling

The main heat transport circuit is regarded as a highly reliable system for removing heat from the fuel under most circumstances. In a pressurized water system, for instance, there are spare pumps to improve reliability and, if all the pumps failed, natural convection or thermosyphoning would be adequate to remove fission product decay heat, provided that a heat sink in the form of a steam generator is available. Such a heat sink must, of course, be supplied with feedwater. However, even if the feedwater flow ceases, there is usually enough water in the steam generator to provide the required heat sink for 3 or 4 hours.

Since it is imperative that heat be removed from the fuel at all times, there are possible circumstances which make it necessary to provide an alternative heat removal system. These possible circumstances are:-

(1) The necessity of interrupting flow in the main heat transport circuit in order to isolate equipment for maintenance or because of leakage. Such isolation prevents forced or natural convection from occurring in an indirect cycle and an auxiliary heat removal system would be required.

(2) The loss of feedwater for a period long enough to exhaust the water already in the steam generator of an indirect cycle. A heat sink is no longer provided and natural convection can no longer continue.

(3) The necessity of having to cool the heat transport fluid, of an indirect cycle, below about 250°F. Since the water in the steam generator boils at 212°F, it cannot be used to cool the heat transport fluid below about 250°F. If lower temperatures are required (eg, for maintenance), then an alternative system is necessary.

(4) On turbine shutdown in a direct cycle. Steam flow and return feedwater flow are then interrupted and an alternative method of heat removal, from the fuel, is required.

Such alternative heat removal is possible by means of the purification system, which is considered later. Cooling of the heat transport fluid is usually necessary, in any case, before it is passed through ion-exchange columns and, so, the purification system could be designed as a standby cooling system as well. This arrangement, however, cannot be used with a direct cycle with an in-line purification system. In any case a separate standby or shutdown cooling system is simpler and provides greater reliability.

Figure 1 shows how reactor power decreases with time, following a reactor trip. It may be seen that, after about 80 seconds, power is produced almost entirely by fission product decay. This fission product power has decreased to about 1% of normal full power after about 3 hours. Further decrease of power is very slow and, therefore, the standby system should be capable of removing 1% of full reactor power over an indefinite period of time.

In the event that rapid isolation of the main circuit is required because of leakage, such isolation would probably be carried out immediately following the circulating pump rundown time. The reactor thermal power is still likely to be about 5% of full power, or higher, at this time and the standby system must be able to cope with this emergency condition. When the standby system is normally put into operation, the heat transport fluid temperatures will be between 200°F and 300°F but, under the emergency conditions outlined above, these temperatures are likely to be in excess of 500°F. Stresses resulting from these high temperatures make it undesirable to use the system, in this manner, too frequently.

A typical standby circuit is shown in Figure 2. It consists simply of a circulating pump, a heat exchanger, isolating valves and associated piping. The requirements of equipment and piping are similar to those for the main system. The heat exchanger would probably be of the U-tube, tube-and-shell type with monel or inconnel tubes and carbon steel shell. If heavy water is used in the heat transport system, double tube sheets would likely be used and the tubes welded to the tube sheets.

When the main system is in operation the standby circuit must be isolated. Conversely, when the standby system is in operation flow must be prevented through the main circuit, otherwise flow through the reactor is by-passed. The power supply to the pumps and the cooling water supply to the heat exchanger must be the most reliable available. The pumps would therefore, be supplied from Class III power (ie, Diesel generator

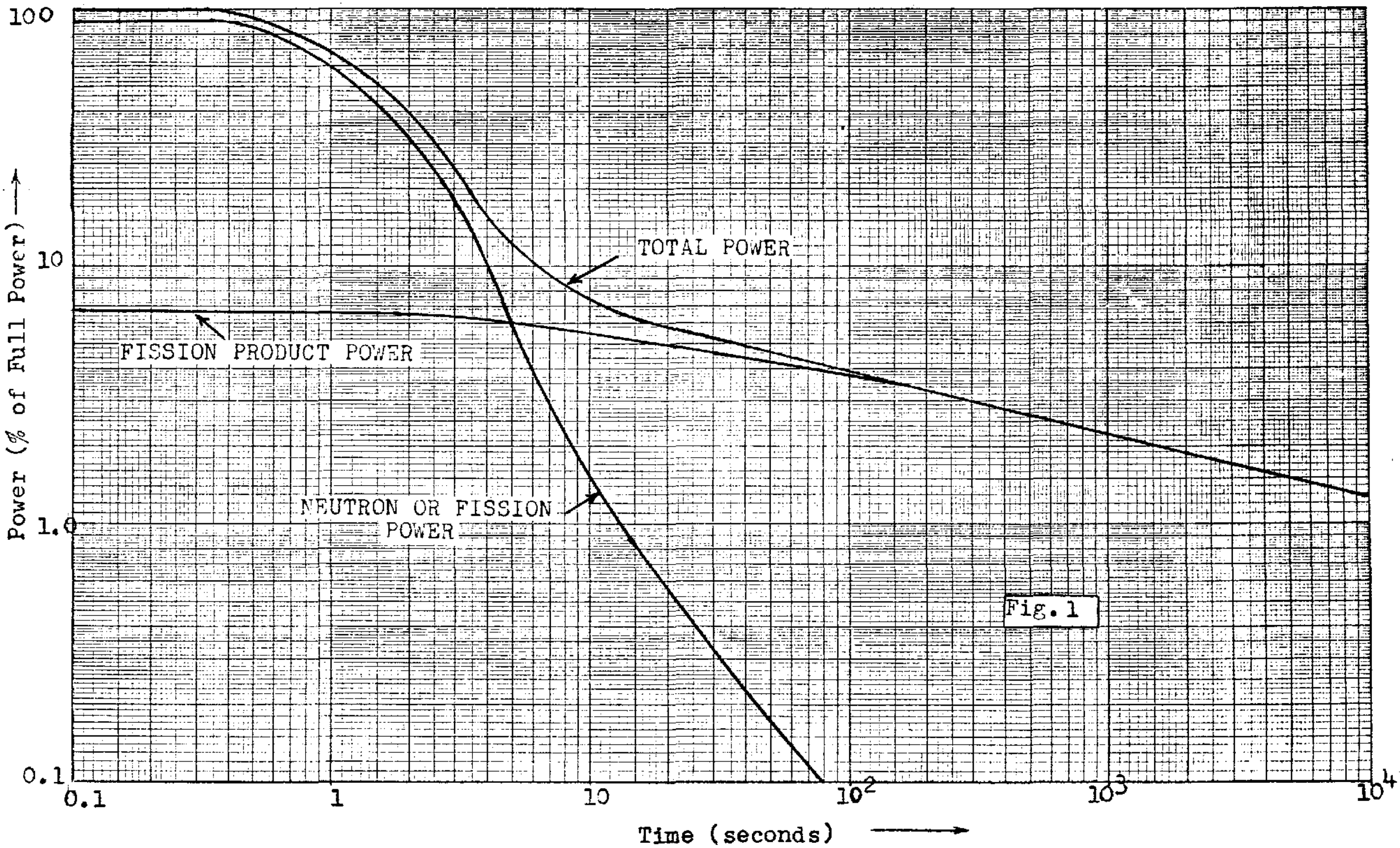

100
10
1.0
0.1
Power (% of Full Power)
0.1
1
10
10^2
10^3
10^4
Time (seconds)
TOTAL POWER
FISSION PRODUCT POWER
NEUTRON OR FISSION POWER

Fig. 1

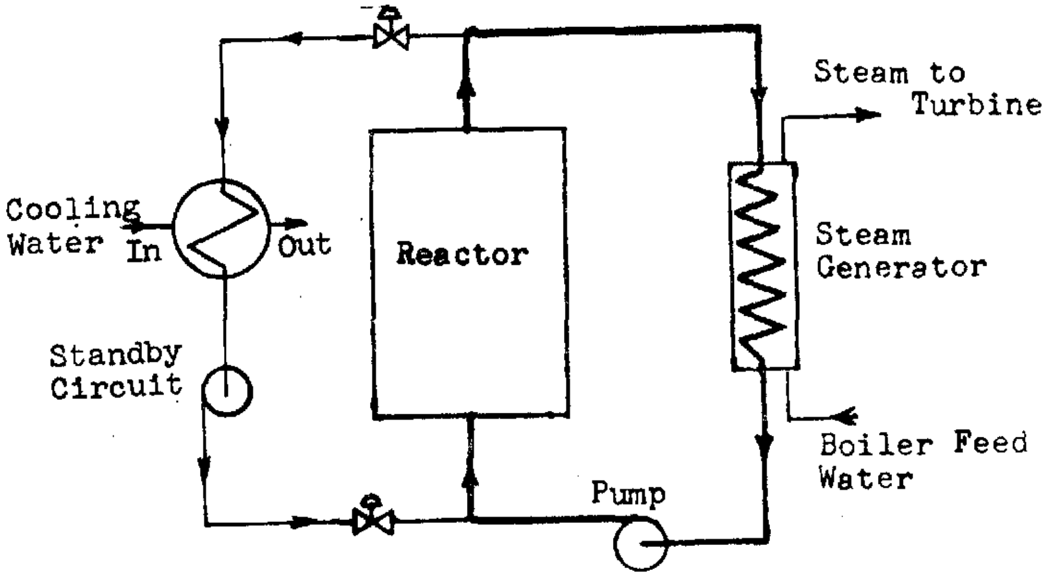


Fig. 2

back-up). The cooling water would normally be supplied from the process water system, but in the event of failure of this supply, cooling water would be provided from a gasoline pump.

Figure 3 shows a dual arrangement of standby circuits that could be used with a horizontal pressure tube reactor of the CANDU type.

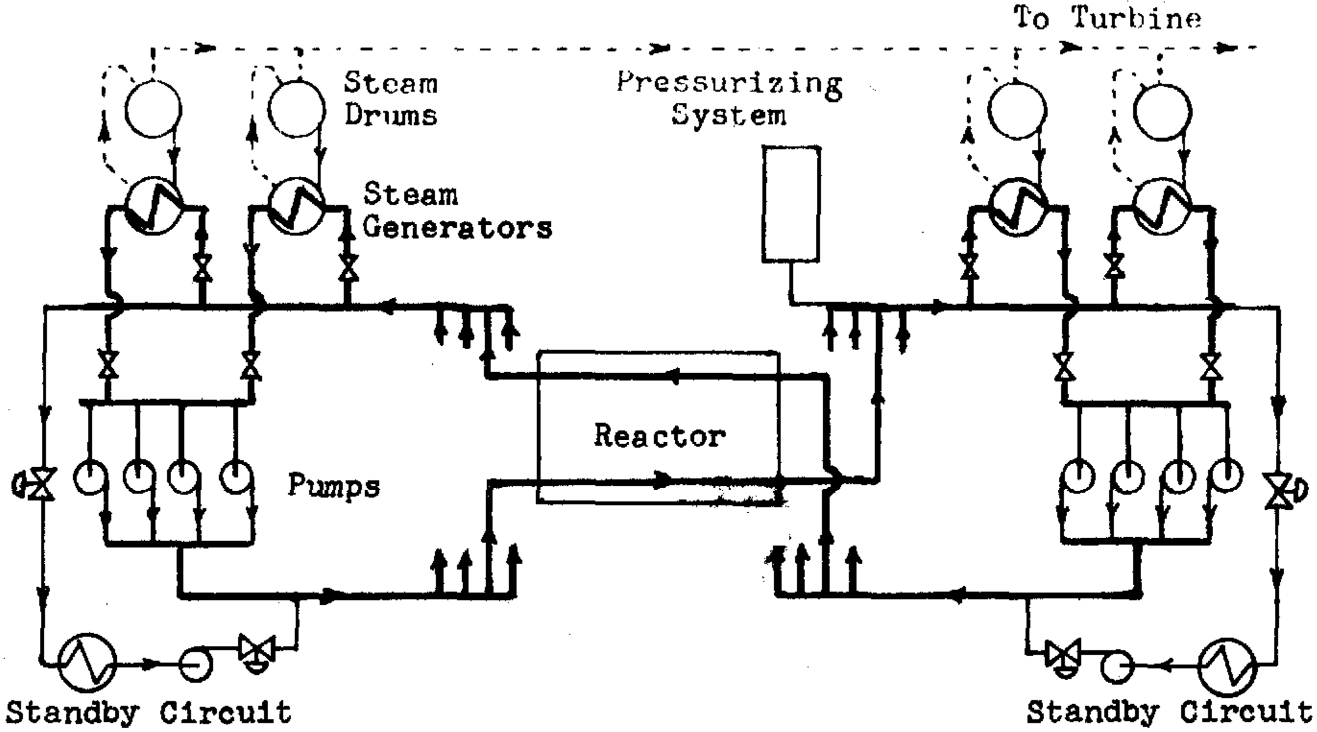


Fig. 3

Again flow through the main system must be prevented when the standby circuits are in operation.

Purification and Degassing Systems

These systems are being considered together because one common system is occasionally used for both purposes. As in the case of moderator purification, the heat transport purification is used to remove corrosion products and to control the pH or pD of the heat transport fluid.

In pressurized water systems the purification system is frequently a separate by-pass system, as shown in Figure 4. The requirements of the system and the design features which meet these requirements are as follows:-

(1) The heat transport system is at a temperature of approximately 500°F whereas the temperature at the ion exchange units should be 120°F or lower. Also the purified water leaving the ion exchange columns should have its temperature raised before being returned to the heat transport system. Both of these requirements are, at least partially, met by using the regenerative heat exchanger in the circuit. Water, from the heat transport system, enters the purification system through the tube side of this heat exchanger. Water returning to the heat transport system, from the purification system, passes through the shell side of this regenerator. Some of the heat from the heat transport fluid is, thereby, transferred to the returning fluid, the temperature of which is, therefore raised before it is returned to the main system. Further cooling of the system water may well be necessary, in a second heat exchanger, C, before it can be passed through the ion exchange columns, IX. The specifications for both heat exchangers would be similar to those for the heat exchanger in the standby circuit.

(2) The heat transport fluid pressure will be in excess of 1000 psig., whereas the ion exchange columns can withstand around 100 psig if standard units are used. A pressure reducing valve, V, is, therefore, required in the purification system. Excessive flashing would occur at this valve if the regenerative heat exchanger was not used to cool the water. The valve V may also be used to regulate the flow.

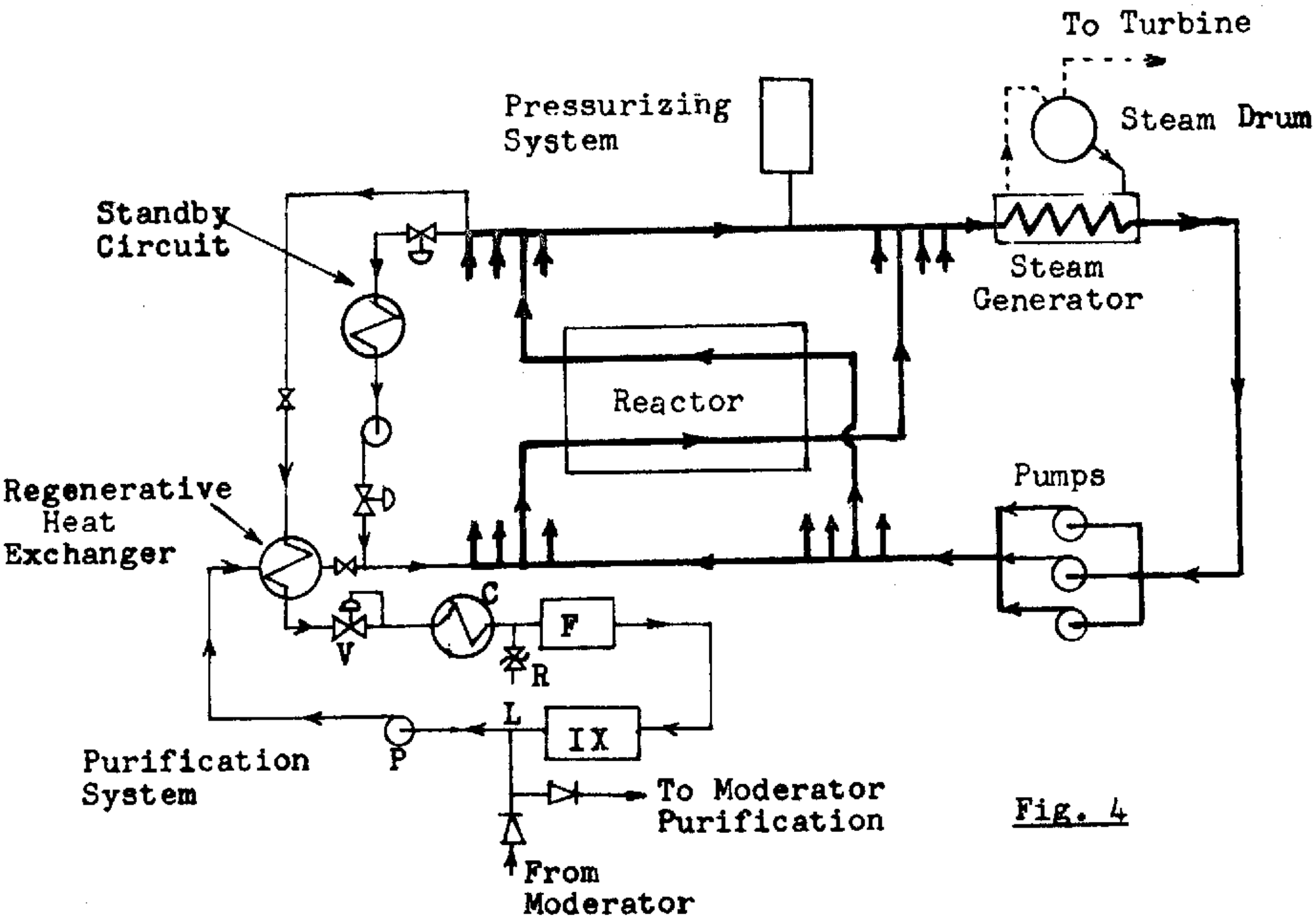


Fig. 4

(3) The crud level or particulate material concentration is likely to be higher in the heat transport system than in the moderator system, described previously, particularly when the system is initially operated. A filter, F, would therefore, be required either as a temporary arrangement, during initial operation of the heat transport system, or as a permanent fixture if the particulate concentration causes blocking of the ion exchange columns. A permanent filter could be of the backwashable or disposable type.

(4) A mixed bed resin would be used in the ion exchange column. Since the heat transport system would be predominently of carbon steel construction, a pH of 9.5 to 11 is required in the system. The cation resin would, therefore, be in the lithium form, the lithium exchange maintaining the desired pH. Supplementary addition of lithium hydroxide to the main system may also be required. Two such ion exchange columns, installed in parallel, would facilitate resin changing.

(5) The circulating pump, P, is sized to provide a flow of 40 Igpm to 50 Igpm. The pump discharge pressure must be greater than the main system pressure.

(6) Overpressurization of the filter and resin beds is prevented by means of the relief valve R.

When a system, such as this, is used for purification, some gas is likely to come out of solution because of the large drop in pressure that occurs. Some arrangement must be provided for venting at the highest points in the system. These gasses could, normally, be vented into the ventillation system exhaust. However with heavy water heat transport fluid, the gasses would be vented into the collection system so that the heavy water vapour can be recovered. It is also possible that fission product gasses may be present because of fuel sheath failures. Therefore, consideration must be given to venting into a delay tank, where the fission product gas activity can be allowed to decay, even with a light water system.

Water, under irradiation, dissocates into hydrogen and oxygen. However, as was mentioned earlier in the course, this dissociation is inhibited if there is an excess of hydrogen dissolved in the water. Such a condition can be achieved by removing the oxygen from solution or by adding hydrogen. Some oxygen will be removed through corrosion but a degassing system may be considered to remove the remainder. The same degassing system may be required to remove fission product gasses released during fuel sheath failures or to remove helium which may find its way into a heavy water heat transport system from the moderator. It is also possible that the hydrogen may be absorbed in zircalloy and cause embrittlement and that the hydrogen concentration may have to be controlled. These gasses can be removed in a degassing tank located in the purification system after the ion exchange columns. Such a tank would be vented to a delay tank and, if a heavy water heat transport fluid is involved, provisions would be made to recover the heavy water vapour (see Relief System consideration). An alternative method of degassing is available when pressurization of the main circuit is by means of a pressurizing tank containing electric heaters, as shown in Figure 5. The sprays in the tank can be used for continuous pressure control and degassing. As gas accumulates in the tank it can be released to the delay tank through the valve C.

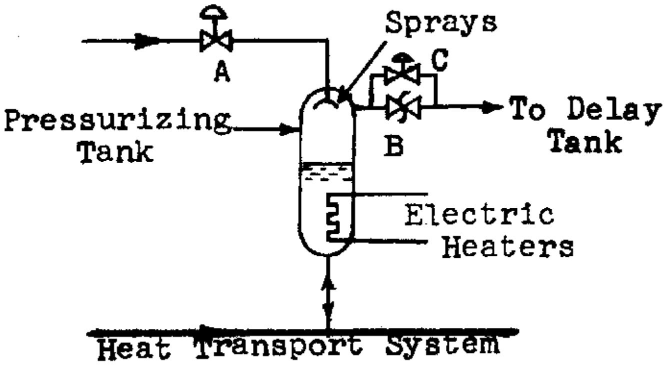


Fig. 5

Any degassing arrangement will, inevitably, take the deuterium out of solution, if the heat transport fluid is heavy water. This not only, encourages further radiolytic decomposition of the water, but also represents a loss of heavy water. A recombination unit might, therefore, be required to recover the deuterium and more deuterium or hydrogen may have to be injected into the heat transport fluid to inhibit further decomposition. Practical experience at NPD G.S. has indicated that a minimum of degassing has been required because little or no fission product gasses have been released and the oxygen has been removed without degassing. The deuterium has also been removed from solution and more deuterium has had to be added from time to time.

The feed and bleed system of pressurization can be design to incorporate both a purification and a degassing system, as shown in Figure 6. A continuous bleed and feed flow is passed through the ion exchange column and the degasser. If a feed flow, in excess of the normal flow, is required because of contraction in the main circuit, due to cooling, the feed pump obtains additional suction supply from the water storage tank. If faster bleed flow than the normal flow is required, because of swell, in the main heat transport circuit, the relief valve opens, at a predetermined pressure, to allow direct flow to the storage tank. Facilities are provided to valve out and by-pass either the ion exchange column or the degasser.

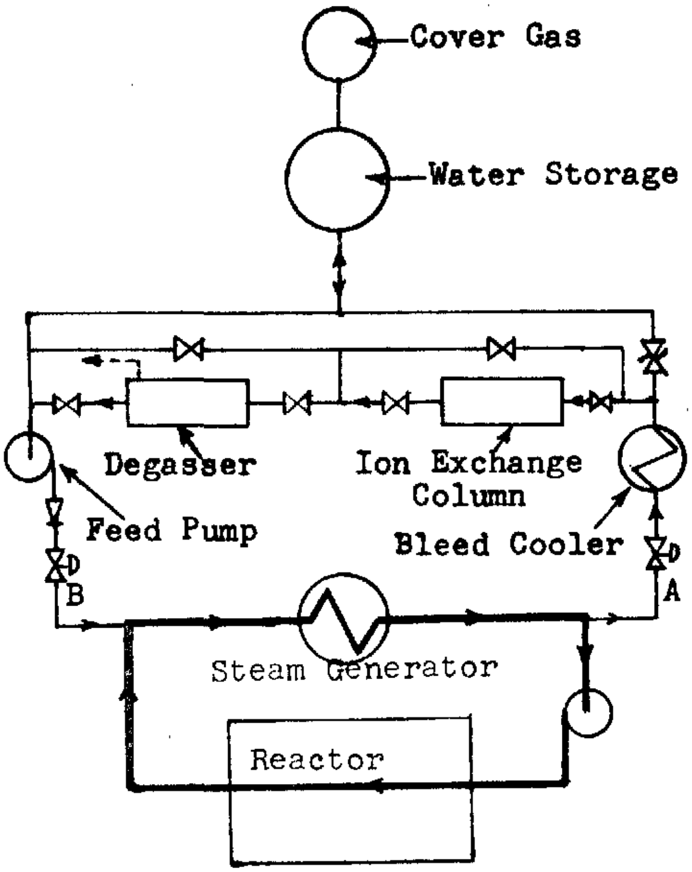


Fig. 6

The degasser is usually a packed column in which the pressure is kept as low as possible. The water to be degassed enters the top of the column and is stripped of gas by a small flow of heavy water vapour which enters the column at the bottom. The vapour and off-gasses, from the top of the column pass into the off-gasses system. Here the off-gasses are separated from the vapour and are passed through a recombination unit, (if deuterium has to be recovered), and into a delay tank, before being bled off into the ventillation exhaust system.

Most of the U.S. boiling water reactors, using the direct cycle, have the same type of purification and degassing system as that shown in Figure 4. The water entering the system is taken from the lowest part of the reactor vessel where the water is coolest. The purified water is then fed into the feedwater inlet to the reactor. Alternatively, the purified water can be returned to the turbine condenser or to the suction of the condensate pumps. The direct cycle system permits an in-line purification system to be used, whereby the ion exchange columns are placed in the condensate lines from the turbine condenser. It is possible that the pH may be controlled by ammonia injection into the feedwater rather than by lithium ion exchange.

When gas is used as the heat transport fluid, it is purified in a simply by-pass circuit which takes about 2% of the main system flow. The most troublesome impurities are dust and moisture. The dust is removed with filters and the moisture removed with driers. If gaseous impurities are present in carbon dioxide they can be removed by distillation at reduced temperatures. Gaseous impurities in helium are removed by molecular sieve CO_2 adsorbers, and catalytic $CO - H_2$ converters.

When organic liquids are used as the heat transport fluid, radiolytic damage causes formation of gasses and high molecular weight compounds, which are called HIGH BOILERS. The high boiler concentration is allowed to reach about 30% in order to reduce radiolytic damage. However a purification system is required to maintain this high boiler concentration. A degassing system is also required to remove the gasses. Figure 7 shows a typical degassing and purification arrangement. As seen from the figure both degassing and purification systems are by-pass systems.

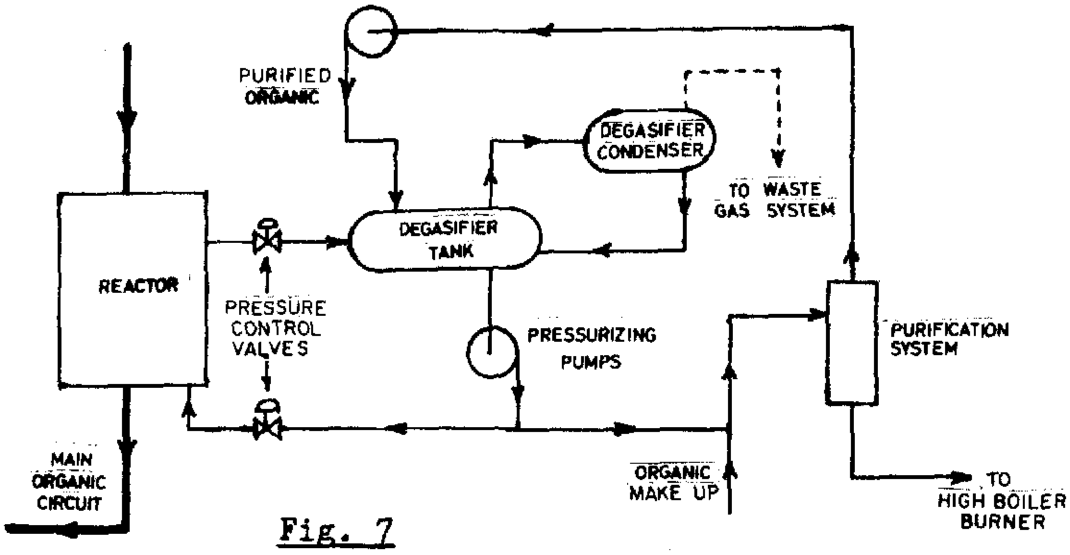


Fig. 7

The gasses and water

vapour are removed in the degassifier tank, into which the hot organic fluid is sprayed under vacuum. Organic vapours are condensed in the condenser and returned to the system. The gasses are vented to the ventillation exhaust system through the waste gas system. The organic is returned to the main system through the pressurizing pumps, which also produce whatever pressure is required in the system. This pressure is controlled by the pressure control valves.

A small fraction, (about 0.5 gpm), of the pressurizing pump discharge is fed to the purification system. This is a continuous distillation column in which the organic fluid is distilled under reduced pressure. The purified coolant is taken from the top of the column and returned to the degassified tank, and thence, to the main system. The residue from the bottom is mainly high boilers and activated products. These are stored until the radioactivity has decayed sufficiently and then burned in a hydrocarbon burner.

The organic fluid make-up is also processed through the purification system, to remove impurities, before being added to the main system.

In order to prevent corrosion and the plugging of pipes in a liquid sodium system, due to deposition of sodium oxide, the oxygen content in sodium must be kept below 10 ppm. The control of the oxygen content is accomplished by cold trapping or by high temperature "gettering" processes. The cold or hot traps are arranged in bypass circuits, as shown in Figure 8.

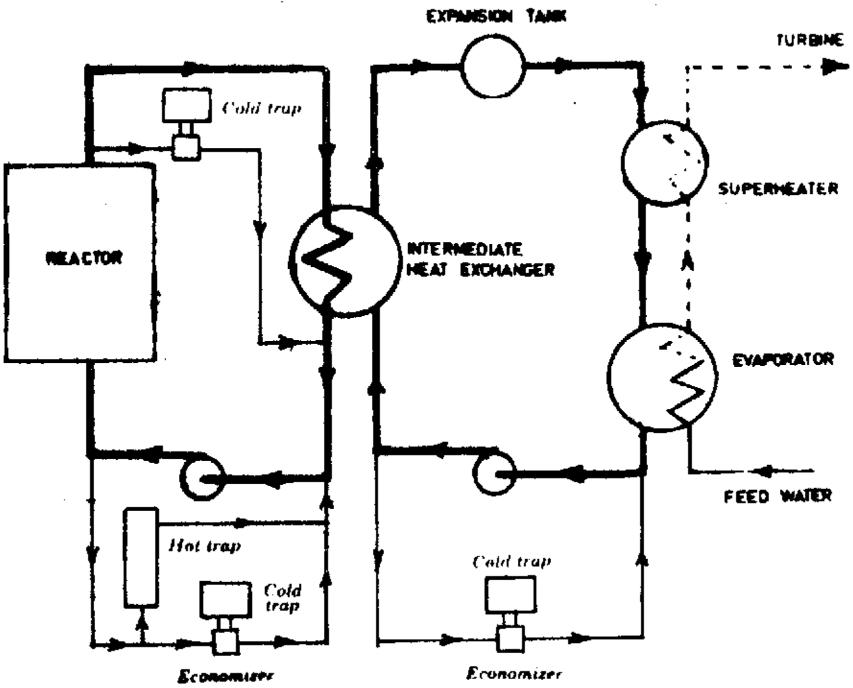


Fig. 8

Cold-trapping involves a reduction in the temperature of the sodium to the point where sodium oxide precipitates and can be filtered out. Figure 9 shows design details of a cold trap in which boiling toluene is used to lower the temperature of the sodium. The sodium oxide precipitates out on the steel mesh screens. Hot-trapping involves the use of a material, such as zirconium, which has a marked affinity for oxygen at very high temperatures. Such a hot trap is shown in Figure 10.

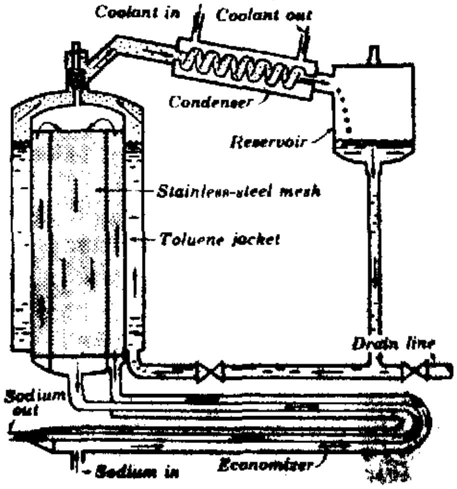


Fig. 9

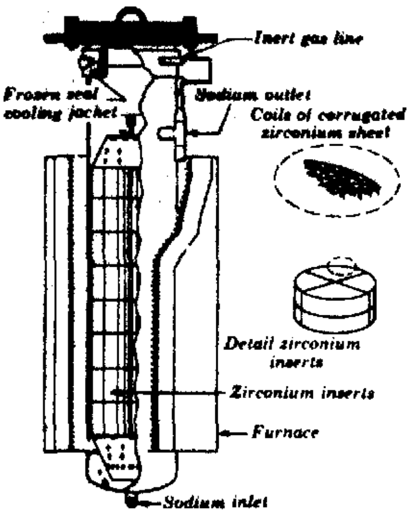


Fig. 10

Constant Volume Control and Emergency Injection

Since it is imperitive that heat must be removed from the fuel at all times, under no circumstances can the fuel be allowed to become uncovered. There are three possible conditions which could result in the fuel becoming uncovered and thereby prevent effective heat removal from the fuel:-

(a) Contraction of the main heat transport system due to a decrease in the system temperature on reactor shutdown.

(b) Small losses of fluid from the system through sampling, purification or monitoring systems or through leakage.

(c) Gross loss of fluid from the system because of a rupture in the heat transport system.

The first two sets of circumstances occur as a result of normal operation of the system, whereas the third may be considered an emergency situation.

In reactors using the pressure vessel concept enough fluid is present, above the core in the reactor vessel, to accomodate changes in volume that occur as a result of changes in temperature. With organic or liquid metal heat transport fluid a big enough pool of the fluid exists above the core to fill the core in an emergency.

When light water is used as a heat transport fluid small losses of liquid are easily made up through the valve, A, in Figure 11. The valve A would open on change in level either in the pressurizing tank or in the reactor. The supply to valve A must, of course be pressurized.

If a rupture in the heat transport system results in large losses of system water, emergency injection of water would occur through check valves into either or both inlet and outlet headers or into the steam drum and the inlet header of a direct cycle system.

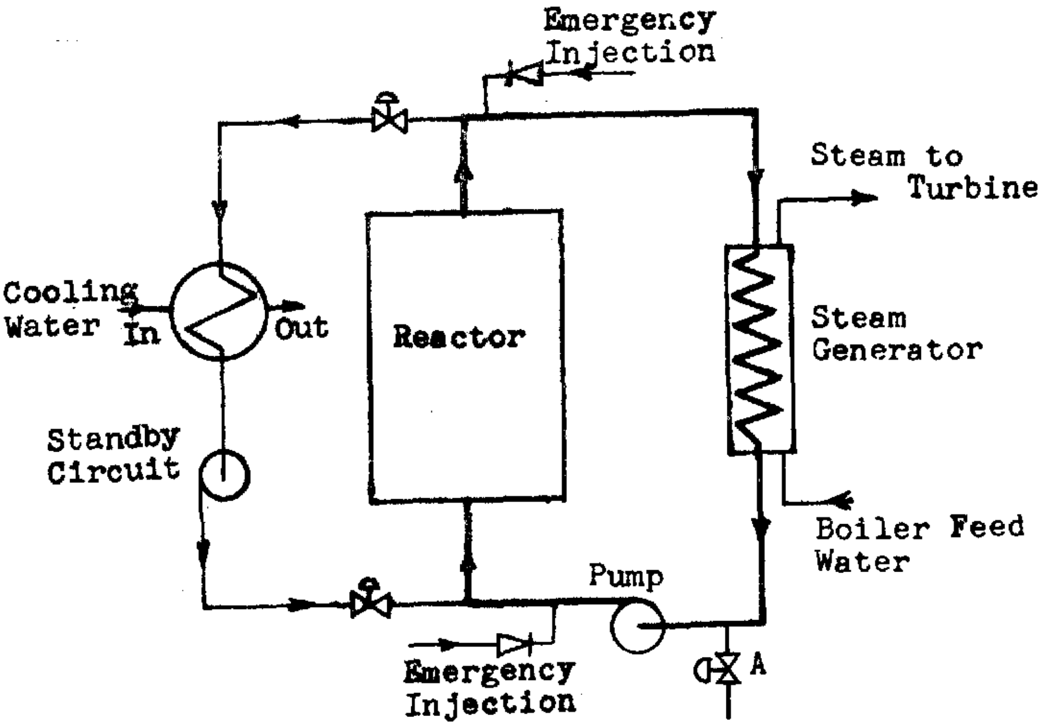


Fig. 11

The emergency water supply would come, initially from a storage tank. This supply would provide time for the system to be connected to a more continuous source of water, such as the process water system.

In a heavy water pressure tube system, pressurized on the feed and bleed principle, the system is solid. Swell or contraction in the system cause changes in pressure before changes in volume occur. The feed and bleed pressurizing system correct these pressure changes by addition or subtraction of water and no other separate system is required. Small losses are made up from the heavy water storage tank through the feed pumps.

When a pressurizing tank is used to pressurize the system, the most sensitive indication of a volume change in the system is the water level in the tank. A drop in level in the tank could cause a valve, such as valve A in Figure 11, to open and allow make-up water into the system. However,for such an arrangement to work at all, a supply of pressurized heavy water has to be available. The most convenient source of heavy water is the moderator system, and valve A could be supplied from the moderator system through a high discharge pressure pump.

Swell, in such a system would cause the pressurizing tank level to rise and level signals could be used to open a second valve and allow water into the moderator system. However, there are disadvantages to using such valve arrangements:-

(a) The moderator is at a lower temperature than the heat transport system and would cool it down if admitted directly into the system through valve A.

(b) The heat transport system is hotter than the moderator system and the quantity of water transferred during a swell would substantially increase the required cooling capacity of the moderator heat exchanger.

(c) The heat transport system is at a higher pD than the moderator and so water transferred to the moderator system has to pass through the moderator purification system. This last requirement however, is inevitable whatever transfer method is used, but it does mean that the water pressure must be reduced before it enters the moderator purification system.

All the above disadvantages can be overcome by transferring water in either direction through the heat transport purification system, as shown in Figure 12. Flow of water out of the heat transport system is controlled by valve V_1 and return flow to the heat transport system is controlled with V_2. During normal operation of the purification system V_1 allows 40 Igpm flow through the ion exchange column. The pump P_1 returns to the system 40 Igpm plus whatever water is required to make up the losses from the system and valve V_2 permits this flow into the system. The additional water is obtained from the moderator system, through a check valve, at the point L.

If the level in the pressurizing tank falls, due to contraction or small leaks, the valve V_1 closes and V_2 opens to permit make up water to be pumped from the moderator system. This water will be heated in the regenerative heat exchanger before entering the heat transport system. If the drop in level is sharp or it continues to fall a second pump, P_2, may be used, with a higher capacity than P_1, to provide additional make-up volume. If the level in the tank rises due to swell in the heat transport system, V_2 closes to prevent return of water to the system and water flows to the moderator system through the moderator ion exchange columns. This water has already been depressurized, by V_1, and cooled in the heat exchangers.

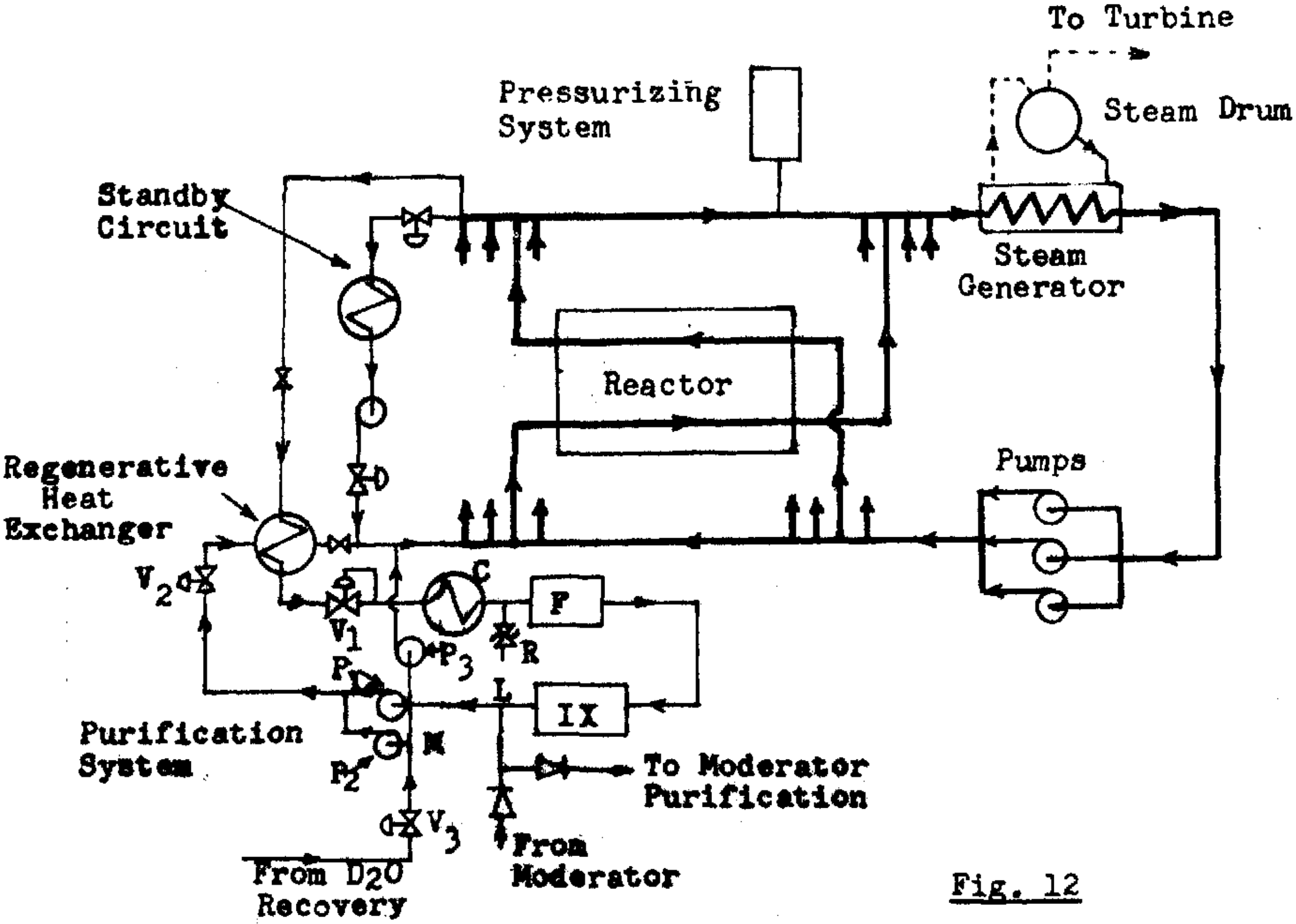


Fig. 12

Should a serious rupture occur in a heavy water heat transport system, it is preferrable to provide emergency injection of heavy water from the moderator system, to avoid serious downgrading of the system. The arrangement in Figure 12 can be used for such a purpose. A third high capacity pump, P_3, (e.g. 160 Igpm) starts on abnormally low pressurizing tank level, to provide the additional make-up volume required. This pump feeds directly into the inlet headers to avoid flow restricition in V_2. The heavy water leaking out of the system is directed, through drain lines, to a recovery system. It can then be pumped back into the moderator system to complete the cycle. Alternatively it may be pumped directly from the recovery system to the suction of P_1, P_2, and P_3 at the point M in Figure 12. Once such a cycle has been established it can continue until the heavy water supply is exhausted because of losses through seepage etc.

If the rupture is too severe for the above system to cope with, a light water injection system, feeding directly into the inlet and outlet headers as in Figure 11, may be required. Light water flow would probably be initiated by low pressure in the heat transport system while the system was still at high temperature.

If the system is pressurized with a feed and bleed system, there are no facilities, such as those described above, for heavy water emergency injection. Connections would therefore be required between the moderator system and either the headers or the charging system for this purpose. Flow to the moderator system would be prevented by check valves. Loss of pressure in the heat transport system, due to a serious rupture, would allow flow of moderator water into the heat transport system. Again a complete cycle would be set up by collecting the water, leaking out of the system, in a recovery sump and pumping it back into the moderator system. Heavy water losses will still occur through seepage and evaporation and provisions would therefore be made to have an alternative light water supply available.

Pressure Relief

Overpressurization of the heat transport system can occur as a result of :-

(a) Failure of the pressurizing system e.g. the heaters are not turned off when the pressure set point is reached.

(b) Failure of the injection, (or charging), pumps described above to shut off.

(c) Overpressurization during pressure testing due to failure of a pressure reducing valve on the pressurizing gas cylinder.

(d) Abnormal rise in reactor power.

It is considered that triplicated reactor regulating and protective systems provide sufficient safeguards against the occurrence of (d). It may also be argued that, under such conditions, the reactor core, is safer without safety valves connected directly to a pressurized heat transport system, since rapid ejection of fluid through large capacity safety valves may result in lack of heat removal from and consequent melting of the fuel. There is, however, a link between the source of power and the atmosphere through the steam generator, across to the steam side and out through the steam safety valves. About 125% of normal full power can be removed in this manner and this would be considered sufficient since the protective system trips the reactor at 110% of normal full power.

A direct cycle system would of course, be protected from possible overpower by safety valves, but direct overpressure relief in a pressurized indirect cycle system would only be provided incase of overpressurization by causes (a), (b) or (c).

In a solid system, connections can be made from the headers, through pressure relief valves R and S, shown in Figure 13, to a surge cooler A. Flashing steam and water passes into the surge cooler and the steam is condensed. The liquid is collected in the surge receiver B, which is drained when this becomes necessary. Provisions would have to be made to dispose of non-condensible gasses through the ventillation exhaust system. These non-condensible gasses would pass through a delay tank incase they contained fission product gasses and through driers if it was necessary to recover heavy water vapour.

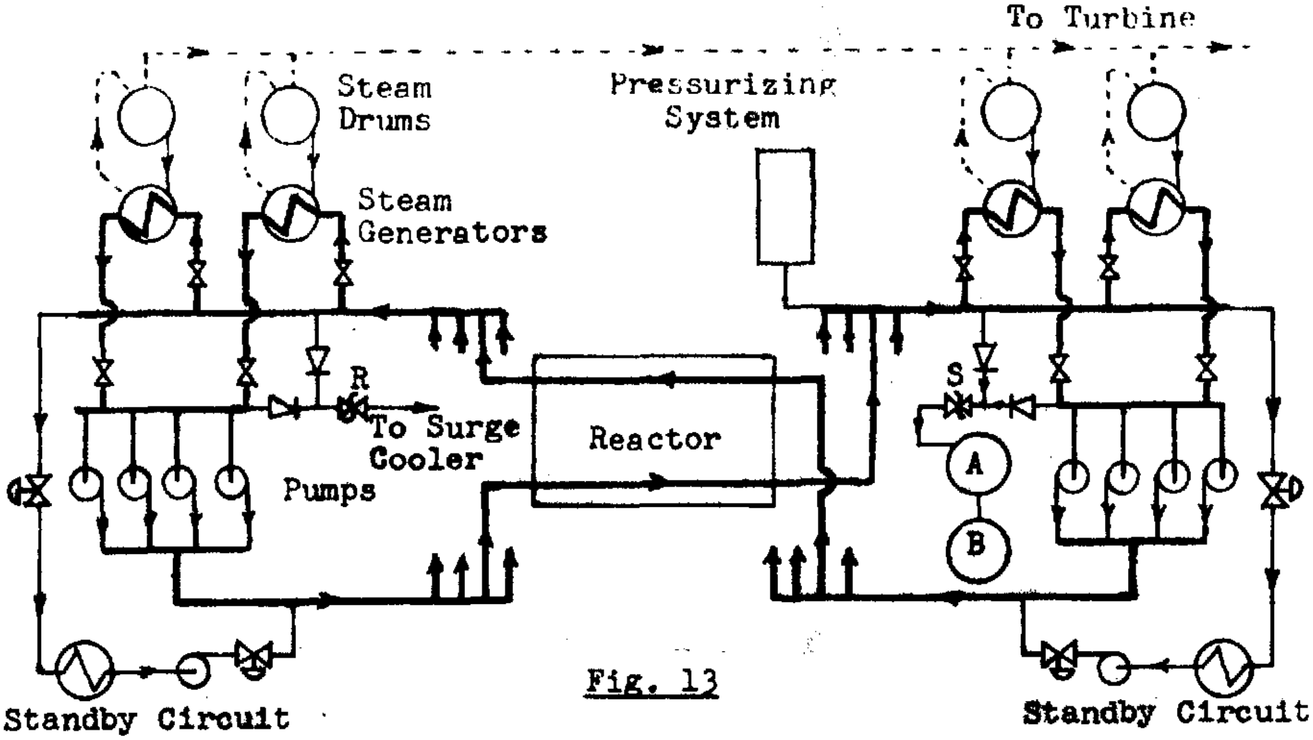


Fig. 13

If the heat transport system is pressurized by means of a vapour space produced in a pressurizing tank, the pressure relief valve is most conveniently connected directly to the tank. Vapour relieving through such a valve must be condensed in a cooler, particularly if it is heavy water vapour. It has been mentioned that such a pressurizing tank may also be used as a degassifier and vapour and non-condensible gasses are discharged from it periodically. This gaseous discharge should also go through the same cooler so that the vapours are condensed. A typical arrangement is that shown in Figure 14. The gas and vapour mixture enter the condenser where the vapour is condensed. The condensed liquid passes into the collection system through the steam traps. The non-condensible gasses are periodically released through the valve A, overpressurization of the condenser being prevented by the relief valve B. On being released, the non-condensible gasses pass through a drier, (probably containing silica gel), to the delay tank where the radioactive gasses decay. A continuous bleed from the delay tank to the ventillation exhaust can be arranged through the orifice O. The orifice O can be replaced with a remotely operated valve for periodic release of gasses from the tank. Overpressurization of the delay tank is prevented by means of the relief valve C. Condensation in the delay tank or the ventillation exhaust suction could cause a vacuum to be formed in the delay tank, which could cause the tank to collapse. This is prevented by the vacuum breaker D.

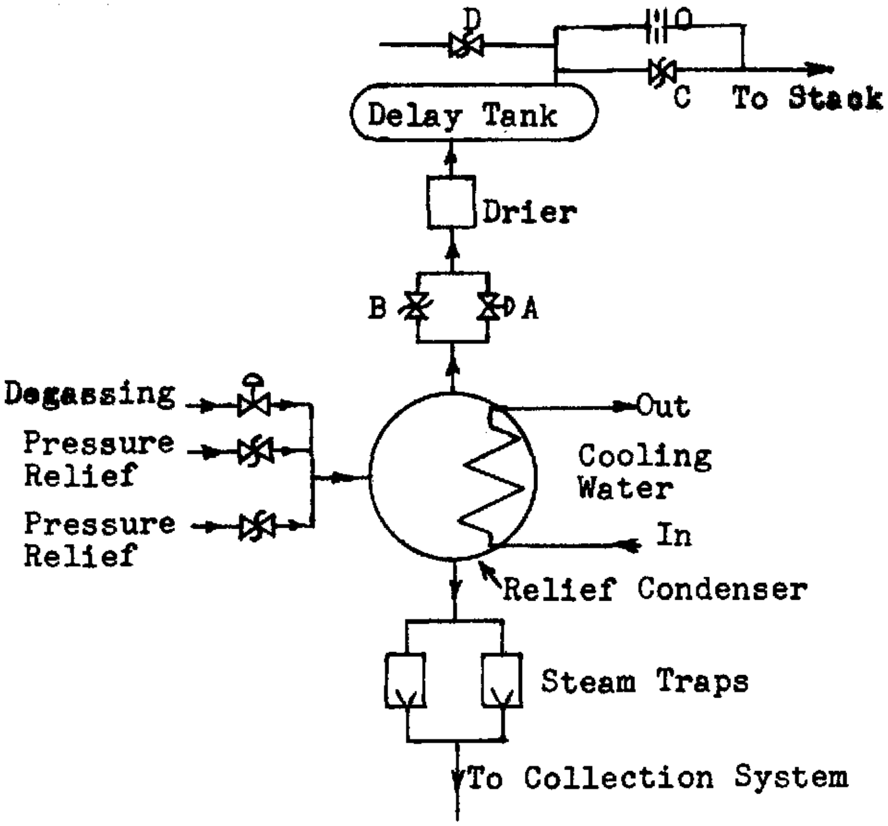


Fig. 14

Provisions can be made for relieving pressure at several locations into this system, e.g. pressure relief in the purification system, immediately ahead of the filters and ion exchange columns.

ASSIGNMENT

1. (a) Under what possible circumstances may standby cooling have to be provided?

   (b) Why is a standby cooling system not considered a back-up to the heat transport system?

2. What are the heat removal requirements of a standby system:

   (a) under normal conditions?

   (b) under emergency conditions?

   Explain your answers.

3. What conditions must exist in the main system before the standby system can be put into operation and why?

4. What are the principal design requirements of a heat transport purification system and how are these requirements met in a by-pass system.

5. Explain why degassing facilities may be required with a pressurized water heat transport fluid and explain what disadvantages such a degassing system may have.

6. How can a feed-and-bleed pressurizing system be also used for purification and degassing?

7. What purification or degassing is required and why:-

   (a) in an organic system?

   (b) in a liquid sodium system?

8. Under what circumstances could fuel in a reactor become uncovered?

9. (a) Explain what principles must be applied to a system designed to maintain a constant volume in a pressure tube reactor system and show how the purification system can be modified for this purpose.

9. (b) What further modification is required to provide emergency heavy water injection.

10. (a) Pressure relief in the heat transport system is provided to prevent overpressurization due to three possible causes. What are these possible causes?

    (b) Why is protection against overpressurization, due to abnormal rise in reactor power, not provided by direct relief? What alternative protection is provided?

11. What are the basic principles incorporated into the design of a pressure relief system for a D_2O system?

A. Williams

Reactor Boiler and Auxiliaries - Course 133

REFLECTOR SYSTEMS

Neutron leakage from a reactor can be decreased by surrounding the reactor core with a substance which scatters or reflects neutrons back into the core. Such a substance is called a reflector.

The Function of a Reflector

The function of a reflector is shown diagrammatically in Fig. 1. More neutrons escape from the "bare" core, shown in Fig. 1(a), than from the "reflected" core, shown in Fig. 1(b). Therefore, with a reflector, more neutrons are available for fission.

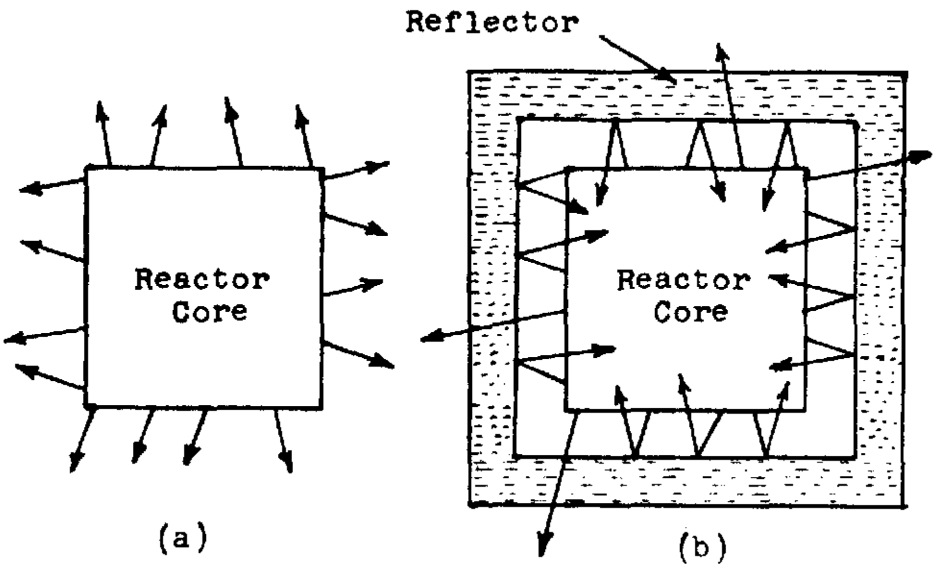


Fig. 1

The effects of placing a reflector around the core can be summarized as follows:

1. The neutron flux distribution is "flattened", ie, the ratio of the average flux to the maximum flux is increased. This effect is illustrated in Fig. 3 for a reactor using fast neutrons for fission, and in Fig. 2 for a reactor in which thermal neutrons are used for fission. The hump in the reflected curve in Fig. 3 is due to the fact that fast neutrons also escape into the reflector and become thermalized in the reflector.

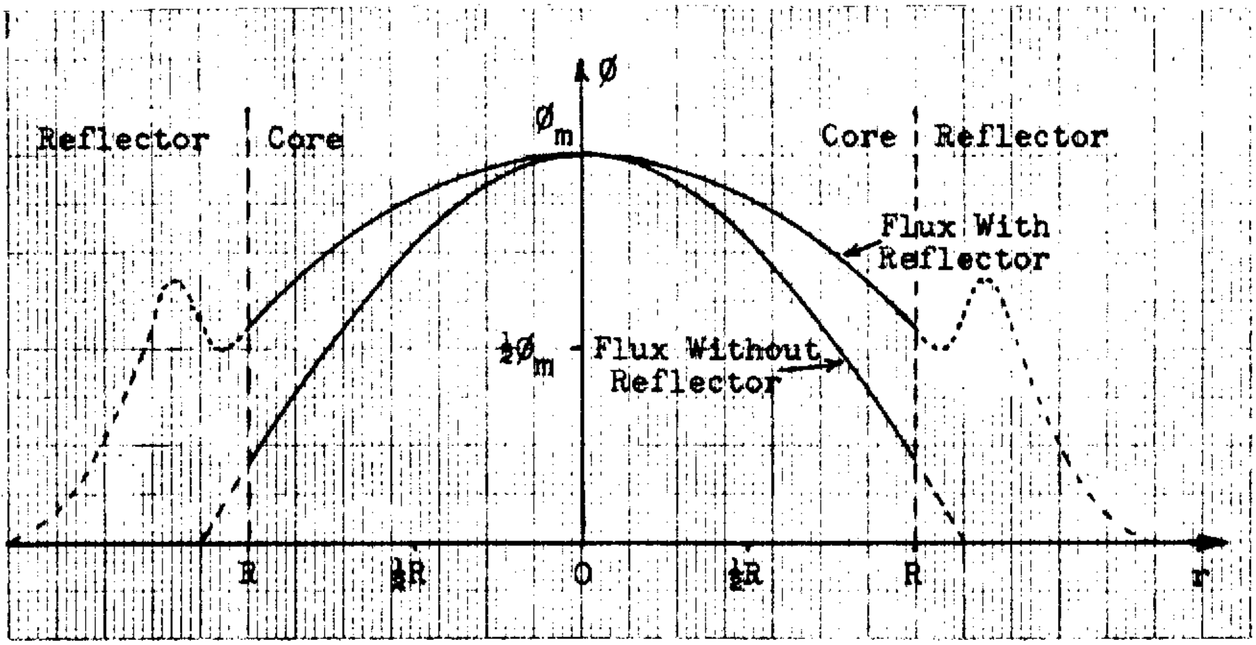


Fig. 2

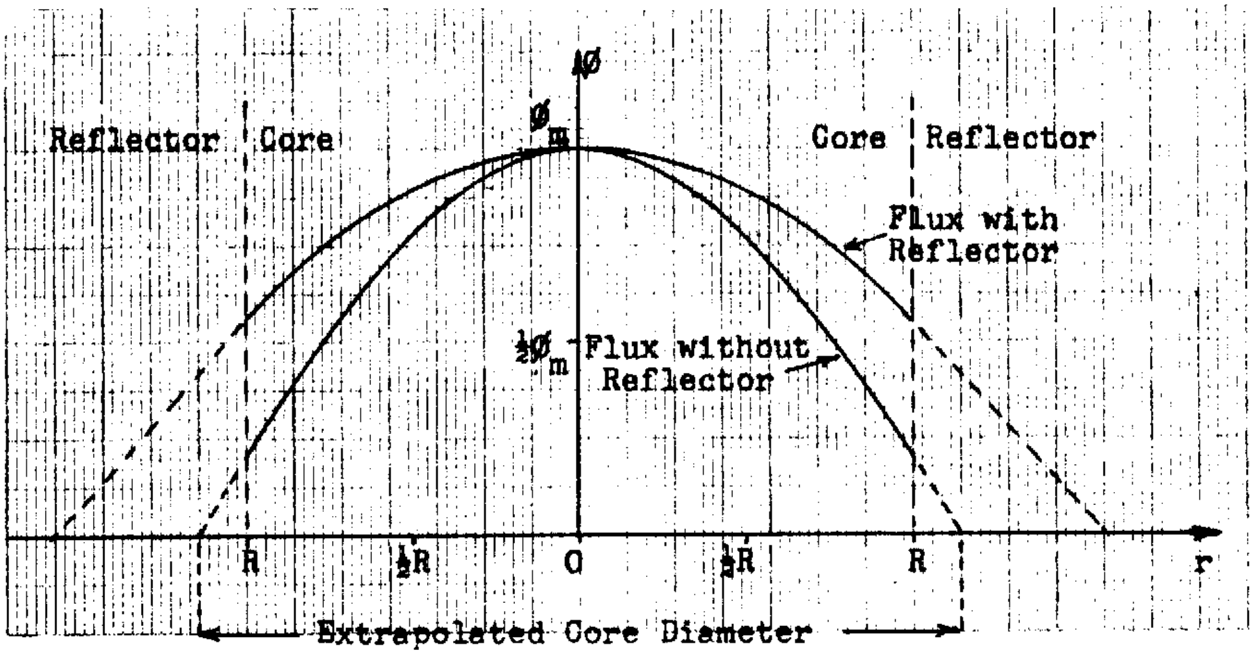


Fig. 3

The flux at the edge of the core, in the radial direction, is now over 50% of the maximum flux instead of only 20% in the bare core case. When both radial and axial flux distributions are considered in a reactor like NPD, the average flux is increased from 27.5% to 42% of the maximum flux.

2. Because of the higher flux at the edge of the core, there is much better utilization of fuel in the outer regions. This fuel, in the outer regions of the core, now contributes much more to the total power production.

3. The neutrons reflected back into the core are now available for fission. This means that the minimum critical size of the reactor is reduced. Alternatively, if the core size is maintained, the reflector makes additional reactivity available for higher fuel burnup.

4. The power, P, produced by a reactor (in megawatts) is given by:

$$P = \frac{\varnothing_a \times U}{3 \times 10^{12}} \qquad \text{- - - - - - - - - - (1)}$$

where $\varnothing_a$ is the average flux in neutrons/cm^2/sec and U is the total weight of uranium fuel in tonnes.

Therefore, if $\varnothing_a$ is increased the reactor power can be increased without changing the size of the reactor. The only way to increase $\varnothing_a$ in a bare reactor is by increasing the maximum flux. However, the maximum flux is usually limited by the maximum fuel heat rating or by the severity of the xenon transient. This increase in $\varnothing_a$, and therefore in power, is obtained, by using a reflector, without increasing the maximum flux, because of better fuel utilization in the outer core regions.

Reflector Properties and Comparisons

The neutrons are reflected back into the core by scattering collisions between the neutrons and the reflector nuclei. The efficiency of a substance as a reflector is measured by a quantity known as the REFLECTOR COEFFICIENT (β).

The reflector coefficient may be defined as the fraction of the neutrons entering the reflector which are reflected back into the core.

For a reflector in the form of a slab, the reflector coefficient for thermal neutrons is connected with the diffusion length, L, by the equation:

$$\beta = 1 - \frac{4D}{L} \qquad \text{- - - - - - - - - - (2)}$$

where D is a quantity known as the DIFFUSION COEFFICIENT which depends on the scattering and absorption cross sections of the material.

Although Equation (2) only applies to slab geometry, it can be used to illustrate that the greater the value of L and the smaller the value of D, the closer β becomes to unity, ie, the more efficient the material is as a reflector. The values of D and L and the maximum value of β that could be obtained with an infinitely thick reflector are tabulated below for H_2O, D_2O, graphite and beryllium.

TABLE 1

Material	L (cm)	D (cm)	β for an Infinite Slab	β for a Slab Thickness 2L
H_2O	2.76	0.17	0.780	0.772
D_2O	100	0.88	0.966	0.965
Beryllium	21	0.54	0.901	0.900
Graphite	64	0.94	0.944	0.940

The desirable properties of a thermal neutron reflector material may be summarized as follows:

1. Thermal neutrons are reflected by elastic scattering between them and reflector nuclei. Therefore, the macroscopic elastic scattering cross section, Σ_s, should be high. If this is the case, the value of D will be low. Thus the density and the value of σ_s must be high. Table 1 shows that H_2O has the lowest value of D.

2. The capture or absorption cross section, σ_a, should be low so that as few neutrons as possible are lost by capture. This requirement ensures a high value of L. However, since the escaping neutrons would have been lost, in any case, without the reflector, a low value of σ_a is not quite as important as it is in a moderator. Hence H_2O is quite acceptable as a reflector for a natural uranium-fuelled reactor, even though it is not as efficient as D_2O. A light water reflector is also an excellent fast neutron shield and its use might well help to avoid the use of a thermal shield.

These desirable properties are, of course, almost identical with those expected in a moderator. The best thermal neutron reflectors are those materials that make the best moderators because they have small σ_a/σ_s ratios.

In the case of fast neutrons, the most effective scattering mechanism is inelastic scattering. Therefore, the best fast reactor reflectors are the heavier materials such as uranium or thorium.

Selection of Reflector Material

The choice of reflector material is by no means based entirely on the above considerations. For example, the reflector used in a fast reactor is also used as a BLANKET, ie, a region, outside the core, in which fissile material is produced from fertile material. Therefore the heavy material used will inevitably be confined to Uranium or Thorium. Which one is used will depend on a number of factors such as:

1. The fissile material which is required to be produced, ie, Pu-239 or U-233.
2. The availability of the fertile material. Depleted fuel, containing mainly U-238, will accumulate as a result of power production with natural uranium-fuelled reactors. Thorium, on the other hand, is in shorter supply and will require mining and processing.
3. The fuel cycle which will give the best conversion factor or breeding gain.

The variables with thermal reflectors are even more numerous. From Table 1 it can be seen that D_2O is the most efficient reflector. However, this may not be the most important criterion. The following are some of the factors that would also have to be considered:

1. Desirable nonnuclear chemical and physical properties. Such factors as chemical inertness, high boiling or melting points, nontoxic properties, radiation resistance, availability and cost, apply to reflectors in much the same way as they do to moderators.
2. Moderator material. Since the desirable reflector properties are so similar to those of the moderator, it would be advantageous for the reflector to be merely an extension of the moderator. Thus a D_2O reflector would be used around a D_2O moderated reactor but not around a graphite or H_2O moderated reactor.

   If the reflector is merely an extension of the moderator, they can both be enclosed in the same reactor vessel and both can be cooled, purified, etc, by the same system.

3. The reflector thickness required. The reflector coefficient, β, increases, initially, as the reflector thickness increases. However, as shown in Fig. 4, very little is to be gained by increasing the thickness beyond a value equal to 2 L. Table 1 also shows that the value of β, for a reflector thickness of 2 L, is within 1% or less of the maximum possible value.

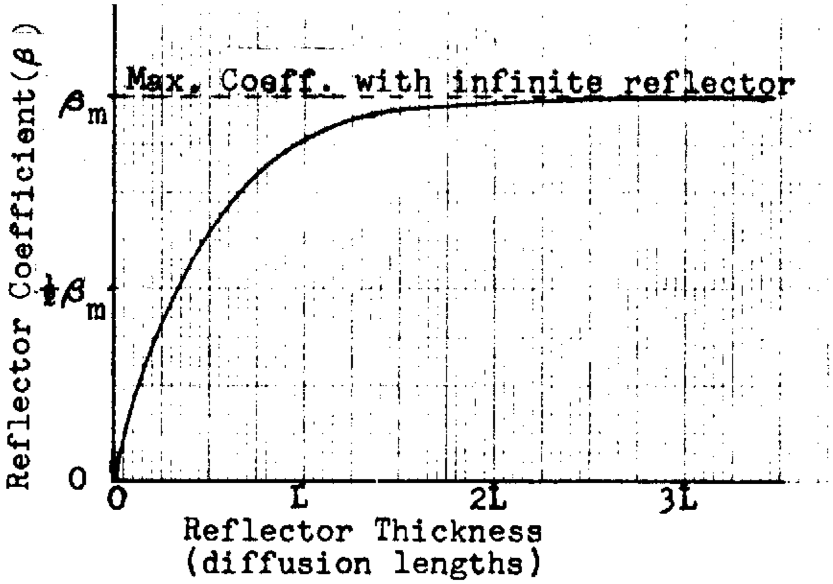


Fig. 4

The important consideration, however, is what thickness of each material would be required to equal 2 L.

For H_2O, such a thickness is only 5.5 cm, whereas for D_2O it is 200 cm. It would require a considerable amount of additional D_2O to extend the moderator by 200 cm especially as this additional D_2O is being placed on the outside of the core.

Factors (2) and (3) are conflicting factors only with D_2O because of the high cost. They could be conflicting factors with graphite because of the increase in size of a pressure vessel which is already large. A compromise solution may be required with D_2O depending on factor (4), which follows.

4. The size of the reactor which, in turn, determines the relative neutron leakage. The larger the reactor the smaller the neutron leakage. Thus the neutron leakage from the Douglas Point reactor is smaller than from the NPD reactor. The more leakage there is out of the reactor, the more important it is to have as high a value of β as possible.

In Douglas Point, a large reactor with D_2O moderator, the leakage is relatively small. A value of β within about 3% of the theoretical maximum was, therefore, considered adequate. A D_2O reflector 74-cm thick will give such a value of β and, so, the moderator was extended by this amount.

In NPD, a small reactor with a high leakage, a 200-cm thickness of D_2O reflector is justified on the grounds of neutron economy. However, the cost of such a reflector would be out of all proportion to the cost of the moderator in the 167-cm radius core. Hence, a 55-cm thick D_2O reflector was used for a value of β within 5% or so of the maximum. A further increase in β was then still required and such an increase could be obtained by placing an H_2O or graphite reflector outside the D_2O reflector. The selection of H_2O or graphite was determined by technical and economic factors.

H_2O requires a containment vessel and if the H_2O is contained in the outer annulus of a double-walled vessel there is always a possibility of H_2O leakage into the D_2O, with consequent downgrading of the D_2O. However, it was felt that the vessel wall could be fabricated leak-tight and H_2O was then chosen on the basis of the following factors:

(a) H_2O is such an excellent fast neutron shield that the use of a thermal shield was avoided.

(b) A graphite reflector would have required a separate CO_2 cooling system. The H_2O can be cooled by merely circulating it through a heat exchanger.

(c) Technology of water handling is much better known in Canada than that of graphite, which gives the design group more confidence in its selection.

(d) The use of H_2O in a double-walled vessel is more economic than the use of graphite, which has to be fabricated, stacked and supported around the reactor vessel.

Reflector Systems

In fast reactors the blanket, which acts as a reflector, is constructed in a similar manner to the core itself. It will consist of elements, similar to fuel elements made from natural uranium, depleted uranium or thorium. Such an arrangement will require cooling and this cooling is most conveniently provided by the heat transport fluid which removes heat from the core. Hence, no separate system is required specifically for the reflector.

In thermal reactors, where the reflector is an extension of the moderator, the moderator and reflector systems will, generally, be common systems. However, when the reflector used differs from the moderator material, a separate reflector system is required. Such a system would have the following general requirements:

1. If the reflector is a liquid, a steady flow must be maintained to prevent stagnation corrosion of the containing walls.

2. Heat is generated in the reflector by absorption of radiation and through slowing down of fast neutrons. Therefore, circulation of a liquid reflector through a heat exchanger is required to remove this heat.

   With a solid reflector, such as graphite, CO_2 cooling would likely be used and a heat exchanger used to remove heat from the CO_2.

3. A purification system will be required. In a water system, such a purification system must keep the pH in the system at a value that will minimize corrosion. It must also remove corrosion products, particularly nitric acid, which is formed, under irradiation, from the nitrogen and oxygen dissolved in the water.

   A CO_2 cooling system would require the same purification as a CO_2 heat transport fluid.

4. Make-up facilities are required.

5. A head tank, in a liquid system, or a gasholder in a CO_2 cooling system would ensure that the system is full and allow for thermal expansion and contraction in the system.

If the reflector is a liquid, as at NPD, enclosed in the outer annulus of the reactor vessel, there are additional requirements of the system. Any excessive temperature differentials between the reflector and the moderator, inside the inner separating wall, will cause excessive thermal stressing of the separating wall or of the calandria tubes. Thus, the temperature differential is maintained at zero or between narrow defined limits. These limits must be maintained both under normal steady temperature conditions and when the moderator temperature is rapidly changing during startup or following a trip.

A typical system for such an H_2O relector is shown in Fig. 5.

The main circuit consists of circulating pumps, P_1, P_2 and P_3, the associated isolating and check valves, a heat exchanger, H, and a head tank, T. Two out of the three pumps would be in operation with the third on standby. This increases the reliability of the system. Centrifugal pumps would be used to obtain the necessary flow. The heat exchanger should be oversized in order to ensure that the reflector is cooled or heated at the same rate as the moderator under all conditions. It will, therefore, be sized on the basis of the maximum rate of cooling that would be required in the reflector to maintain the correct temperature differential between it and the moderator. A valve in the cooling water line, such as V_1, would be modulated by this temperature differential. A steam or hot water line into the heat exchanger may also be required to maintain the correct temperature differential during reactor startup. Such a heating arrangement has not been found to be necessary at NPD. Some flow of cold water may be required continuously through the heat exchanger to prevent stagnant corrosion on the shell side. Under certain conditions, steam or hot water may have to be mixed with this minimum cold water flow. Thus, the valve V_2 is also controlled by the moderator-reflector temperature differential.

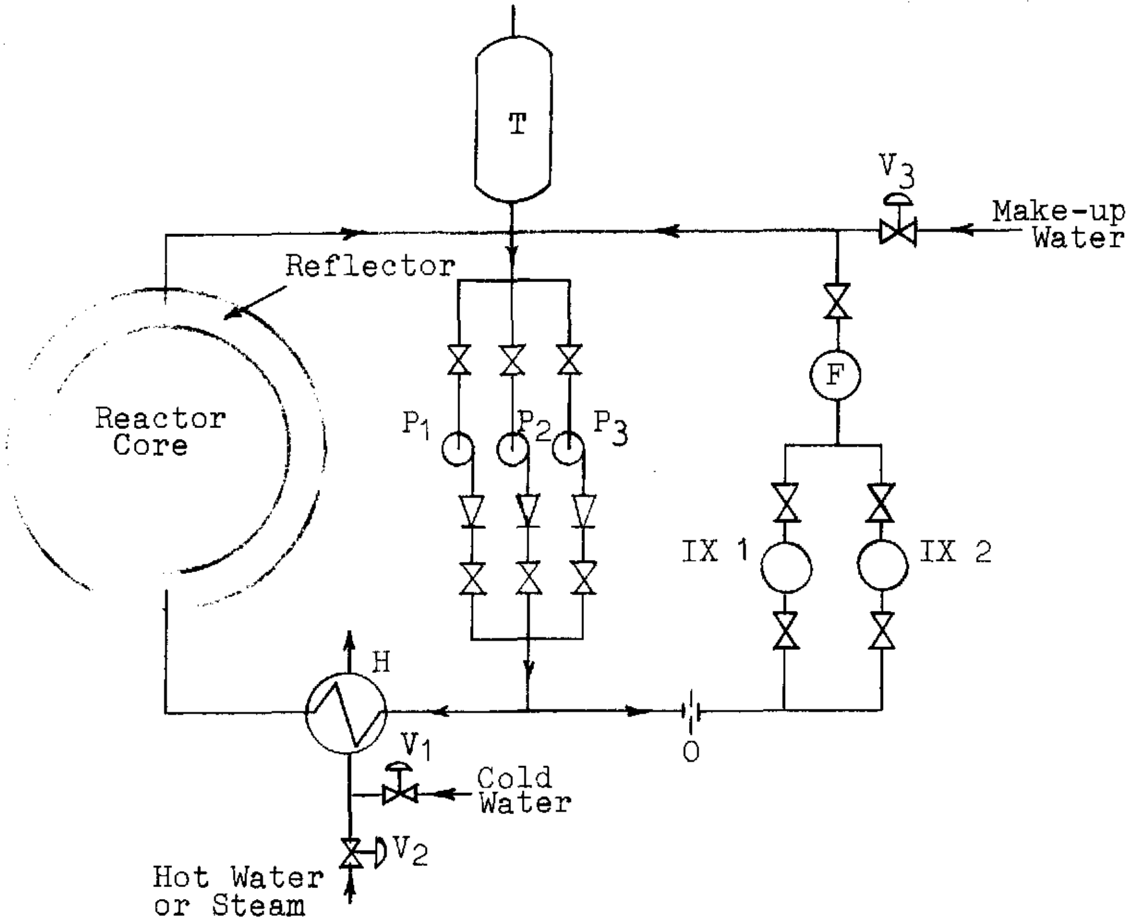


Fig. 5

The head tank, T, provides a static head for the pump suction, allows for thermal expansion and contraction, and ensures that the system is full at all times.

The purification system, which is a bypass system, consists of a filter, F, and two ion-exchange columns, IX 1 and IX 2, connected in parallel. The filter could be a disc type which can be cleaned by scraping or a disposable type. One ion-exchange column would be "on-line" and the other on standby to facilitate column replacement. The column will contain a mixed bed resin for reasons outlined below. The material of construction is also considered below. The flow rate through the circuit is determined by the size of the orifice, O.

Make-up water is delivered, on demand, into the system through the control valve V_3. This make-up water will have been treated in the same way as that used in the boiler-feedwater system.

Water leakage out of the system is not, economically, as important as it is from a heavy water system, although it does represent a loss of water which has been treated. However, the system will contain radioactive N-16, O-19, and corrosion product nuclei so that leakage out of the system should be kept to a minimum. The system would, therefore, be of welded construction wherever possible and leakage along pump shafts and valve stems prevented by suitable seals or packings.

Since the reflector is in the neutron conservation region of the reactor, the wall or walls separating the light water from the heavy water reflector would be made of a small capture cross section material such as aluminum. The system piping would, then, also be of aluminum. Pumps and valves would be of stainless steel whereas the other components would be of aluminum. The pH of the system must, therefore, be maintained between 6 and 7.

Cooling of the reflector must be maintained even when the reactor is tripped. If a unit outage resulted in a Class 4 power supply failure, circulation in the system must still be maintained. The required reliability is obtained by placing one of the operating pumps on Class 3 power.

A gas cooling system, for a solid reflector, would look very similar to that in Fig. 5 with the pumps replaced by blowers and the head tank replaced by a gas holder. The purification system would, of course, be designed to meet the same requirements as for a gas heat transport fluid.

ASSIGNMENT

1. (a) What is the basic function of a reflector?

   (b) Define the quantity used to determine the reflector efficiency.

2. (a) What are the essential properties of a material which is to be used as a thermal neutron reflector?

   (b) What materials make the best fast neutron reflectors and why?

3. (a) Explain the other factors which have to be considered when a thermal neutron reflector is being selected.

   (b) Discuss the choice of reflectors in the NPD and Douglas Point reactors.

4. Briefly describe the general requirements of a reflector system.

5. What are the factors which determine the pump selection, the pump arrangement and the pump power supplies in a liquid reflector system?

6. What supplies may be required on the shell side of the reflector heat exchanger and why?

7. What are the purposes of the head tank?

8. On what basis would the reflector system piping and component materials be selected?

A. Williams

Reactor Boiler and Auxiliaries - Course 133

SHIELD COOLING SYSTEMS

Shielding in a nuclear-electric generating station is required for gamma rays only, for fast neutrons only, or for gamma rays and fast neutrons together.

For gamma rays alone, the main requirement is for high density material, the most effective gamma ray shields being those made of dense material containing atoms of high atomic number. The most common gamma ray shielding materials, in decreasing order of effectiveness, are lead, steel and concrete.

The shielding requirements for fast neutrons are:

(a) The shield must contain material rich in hydrogen to slow down the neutrons by elastic scattering (eg, water, polyethylene or masonite).

(b) The shield must contain other elements to slow down the neutrons by inelastic scattering (eg, iron or titanium).

(c) The shield must contain materials that will readily capture thermal neutrons.

(d) Since thermal neutron capture usually results in the production of gamma rays, sufficient high density material must be present to reduce the capture gamma ray intensity after the neutrons have been thermalized. The elements required for (b) would probably meet this requirement.

When both fast neutrons and gamma rays are being emitted, as from the reactor core, both the above requirements must be simultaneously satisfied. There must, therefore, be an adequate amount of dense material and material rich in hydrogen. If the ratio of heavier element to hydrogen is too high, the low energy neutrons are not effectively slowed down and stopped, resulting in a buildup of low energy neutrons near the outside of the shield. This effect will tend to increase the formation of capture gamma rays towards the outside of the shield.

On the other hand, an excess of hydrogenous material will tend to cause a buildup of high energy neutrons further into the shield, due to the relative lack of inelastic scattering. Since hydrogen is an effective gamma-scattering material, but a poor gamma-removing material, excessive hydrogen will cause a buildup of gamma ray intensity further into the shield.

The single most important source of radiation in a station is the reactor itself and the shield around the reactor is, therefore, known as the primary shield. There are two types of primary shields:

(a) The operational shield which is the shield between the reactor and areas that are continuously accessible during full power operation. This shield must meet the requirements for both gamma rays and fast neutrons. The operational shield is usually made from ilmenite concrete, for economic and structural reasons, since it contains a proper balance of hydrogenous and dense material. This shield is sometimes known as the BIOLOGICAL shield, since its main purpose is to protect station personnel against the damaging effects of nuclear radiation.

(b) The shutdown shield which is the shield between the reactor and areas accessible during shutdown only. This shield is required for gamma ray attenuation only. This shield can also be made of ilmenite concrete and is usually made mainly of this material if the reactor is contained entirely within the primary shield, such as is the case at NPD. However, at Douglas Point and Pickering the reactors are refuelled from outside the end shields and the fuel channels have to penetrate these shields. Problems of fabrication and alignment are, then, encountered which result in the central portions of these end shields being made mainly from steel. In fact, in Pickering these central end shields are integral with the calandria and are suspended with it.

Cooling of Concrete Shields

The interaction between nuclear radiation and the nuclei of materials results in a transfer of energy which heats the material. The lower energy gamma rays and neutrons contribute more to the heat generation than the more penetrating radiation. Consequently a high proportion of the total heat generation occurs in the first few inches of the shield. Since the inner face of the shield also receives radiant heat from the reactor, the problem is compounded. Fig. 1 shows how the heat generated varies with the distance into the concrete from the inner face, in the case of the NPD primary shield.

Because of the poor conductivity of the concrete and the large thicknesses involved, this heat is not easily removed. If no cooling was provided there would be a substantial increase in the shield temperature. It was estimated that, without cooling on the NPD shield, a maximum temperature of about 280°F would be obtained.

The effects of high concrete temperatures are twofold:

(a) Spalling and cracking may occur in the concrete, resulting in radiation streaming through the shield. Such cracking may occur as a result of chemical changes in the material or as a result of thermal stresses set up by nonuniform temperature distributions in a thick wall.

(b) Water is driven out of the shield by the high temperatures and the retained water content in the shield decreases. This will make the shield less effective as a neutron shield.

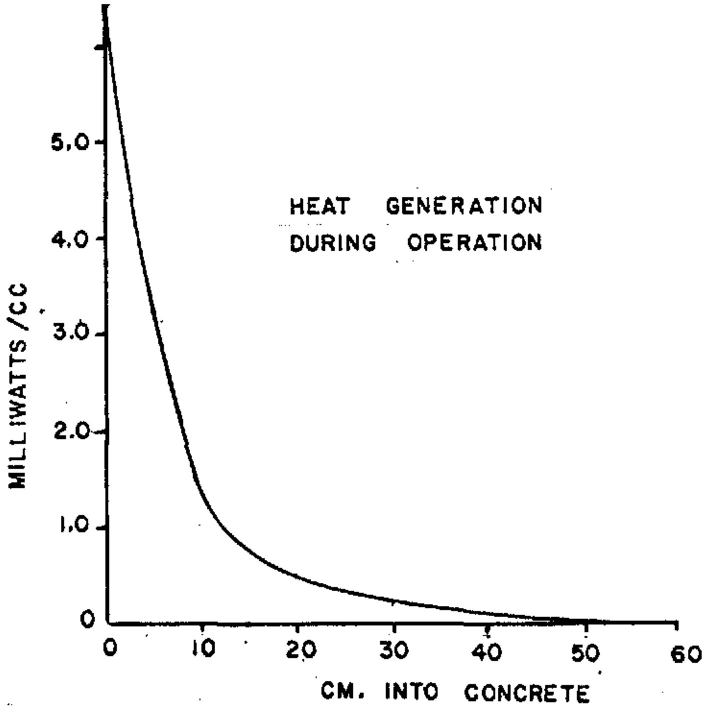


Fig. 1

The factor that limits the shield temperatures is usually the latter. The retained water content is a more critical factor than thermal stressing.

In view of these possible consequences of high shield temperatures, steps must be taken to keep the temperatures from becoming excessive. The maximum temperature specified for the NPD and Douglas Point shields, for instance, is 130°F.

Heat generation in the shield on reactor shutdown must also be considered. The fission product gamma rays, emitted when the reactor is shut down, are of much lower energy than some of the gamma rays produced during operation. This means, of course, that the shutdown shield thickness required is substantially less than the operational shield thickness. However, the persistance of low energy gamma rays in the shield requires that cooling be provided for some time after shutdown. Fig. 2 shows the heat generation, at various distances into the NPD shield, immediately after shutdown and 5 minutes later. Fig. 3 shows the variation of the heat generation with time.

If Figs. 1 and 2 are compared, it may be seen that the heat generated, immediately on shutdown, is nearly three times higher than during full power operation. It decreases to the operational value about 1 minute after shutdown. This transient increase in heat generation occurs because the moderator is dumped from the calandria, on shutdown, which results in an initial increase in the lower energy gamma ray intensity. The cooling capacity of the shield cooling system must allow for this transient.

It may be seen, from Fig. 3, that, after an initial rapid drop in heat generation, the heat generated, thereafter, decreases fairly slowly with time. Cooling must, therefore, be provided for some time after shutdown and until the maximum shield temperature remains below 130°F without cooling.

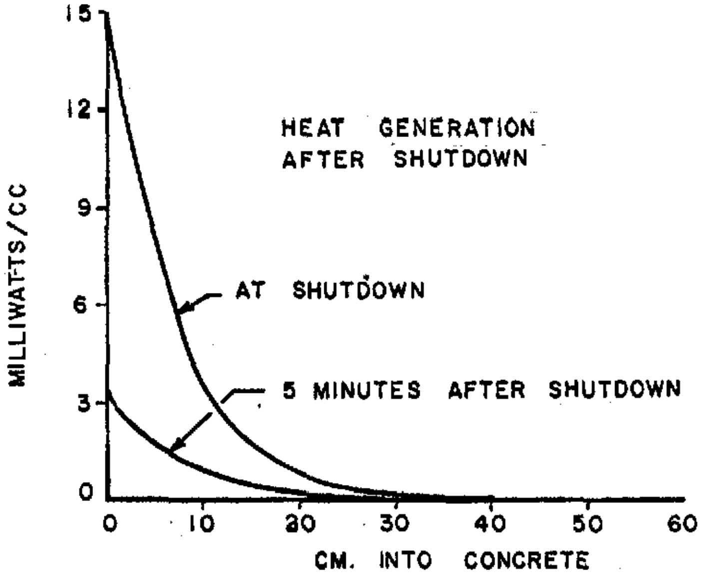


Fig. 2

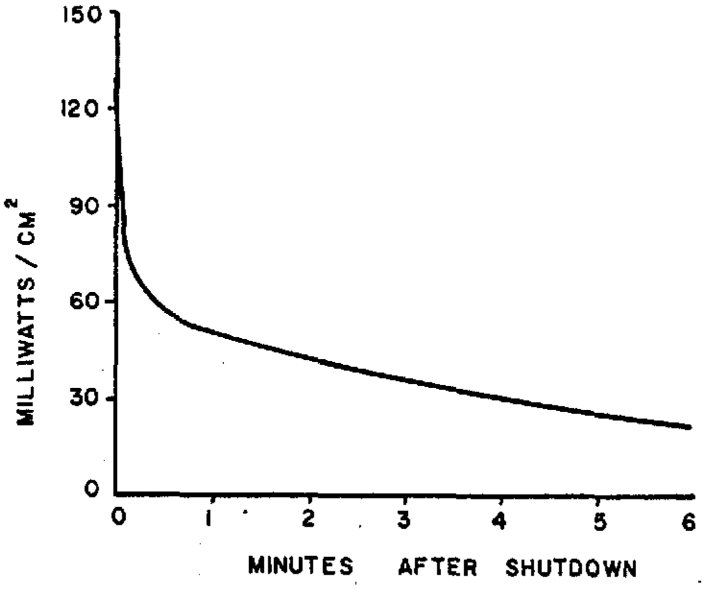


Fig. 3

Fig. 4 shows a typical concrete cooling system. The concrete is cooled by pipes embedded about 3" or 4" below the inner concrete face. The pipes, made of carbon steel, would be about 1" in diameter. Where the heat generation is low, the pipes would be widely separated, perhaps being as much as 12" or more apart. In the areas opposite the centreline of the reactor, the pipe spacing would be reduced to about 4". This represents about the minimum spacing possible without serious weakening of the concrete in this area which would probably result in the concrete spalling and breaking away. Should more cooling be required, a second row of cooling pipes would be installed deeper into the concrete.

The cooling pipes are arranged in panels. All the panels in one wall or in one layer of pipes are supplied from a common header but each panel has a separate return flow through a flow regulating valve and a flow indicator (not shown). The flow can thus be balanced to obtain a uniform temperature throughout any one area. The top of each loop in the vertical panels and each high point in the system must be vented through bleed-off valves to prevent air-locking of the tubes. A closed system is used in preference to a continuous system to avoid admission of air which, unless continuously vented, would cause air-locking and corrosion.

The return flow from the panels pass through a tube-in-shell type heat exchanger, E_1. Greater reliability is obtained by having a second, 100% duty, heat exchanger, E_2, on standby. The heat exchanger outlet temperature would be controlled by a modulating valve in the cooling water line. It may be necessary to avoid thermal shock to large heat exchanger by preventing the system water from dropping below some predetermined temperature. A temperature controller would, then, be installed in the return flow line, which would close the modulating valve as the temperature dropped. If the temperature continues to fall, after the valve has closed completely, a heater, H, would be switched on to prevent further drop in temperature.

Two 100% duty centrifugal pumps would also be provided for reliability, one pump operating and one on "auto" standby. A flow meter in the main return line will cause the standby pump to start on low flow. The pumps are provided with isolating valves and with check valves (not shown) to prevent backward rotation of the standby pump.

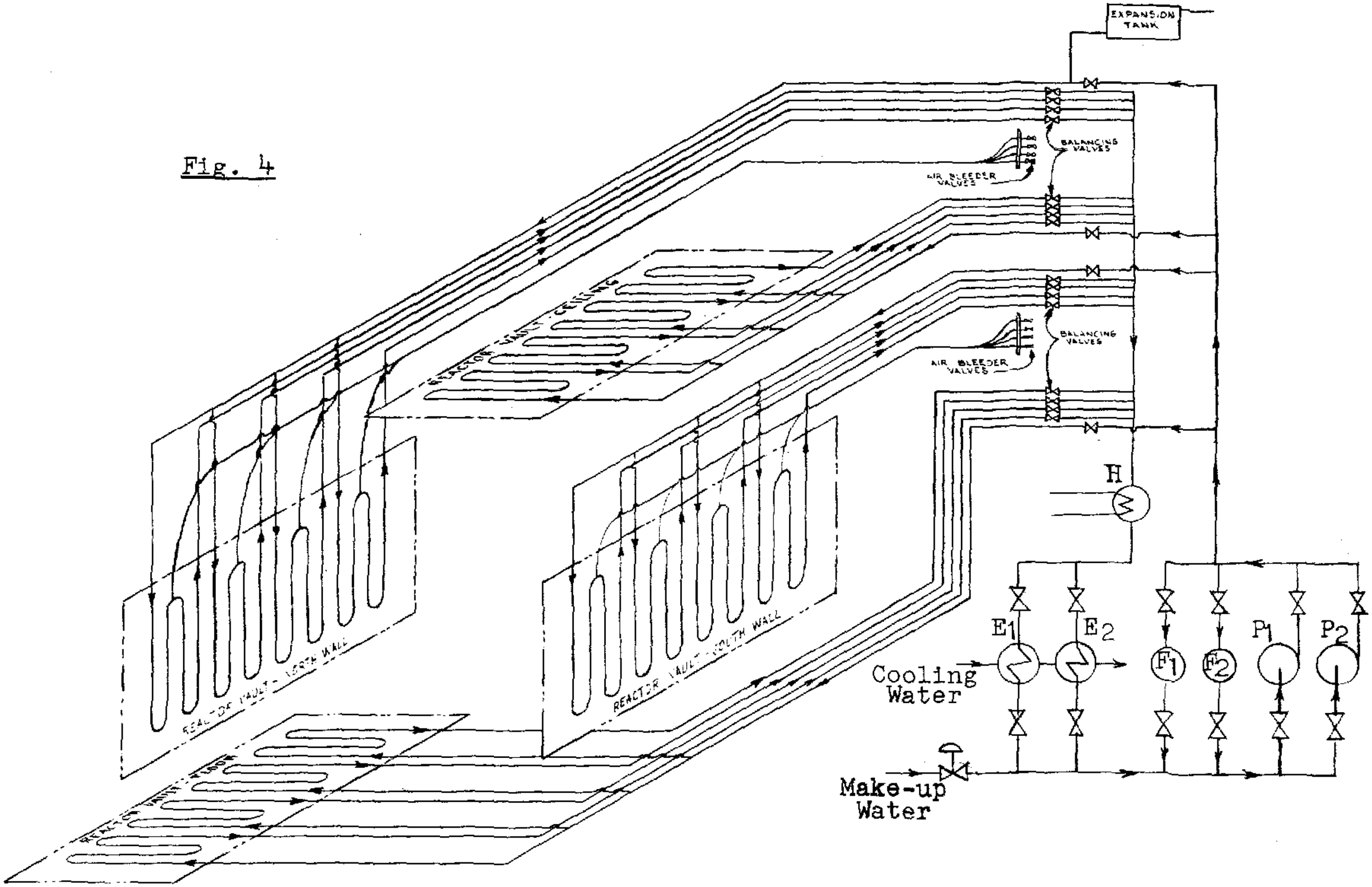
The system would, preferably, be filled with demineralized water, although process water is used at NPD. No ion-exchange columns are considered necessary but part of the pump discharge may be bypassed through either of two filters, F_1 and F_2. These filters may well be of the disposable paper cartridge type.

A vented expansion tank is connected to the high point in the system. This tank maintains a constant pressure on the system and allows for expansion and contraction in the system. Any air in the piping tends to find its way into this tank and is then released through the vent. Level switches in the tank initiate make-up water addition to the system, if the water level drops. A further drop in level is annunciated.

Cooling of Steel Shields

As was mentioned earlier in this lesson, if the reactor pressure tubes penetrate the end shields, steel end shields would be used to minimize fabrication and alignment problems.

Fig. 4


EXPANSION TANK
BALANCING VALVES
AIR BLEEDER VALVES
BALANCING VALVES
AIR BLEEDER VALVES
REACTOR VAULT - CEILING
REACTOR VAULT - NORTH WALL
REACTOR VAULT - SOUTH WALL
REACTOR VAULT - FLOOR
H
E1
E2
F1
F2
P1
P2
Cooling Water
Make-up Water

Such shields may form an integral extension of the reactor vessel as at Pickering. A typical end shield arrangement is shown in Fig. 5.

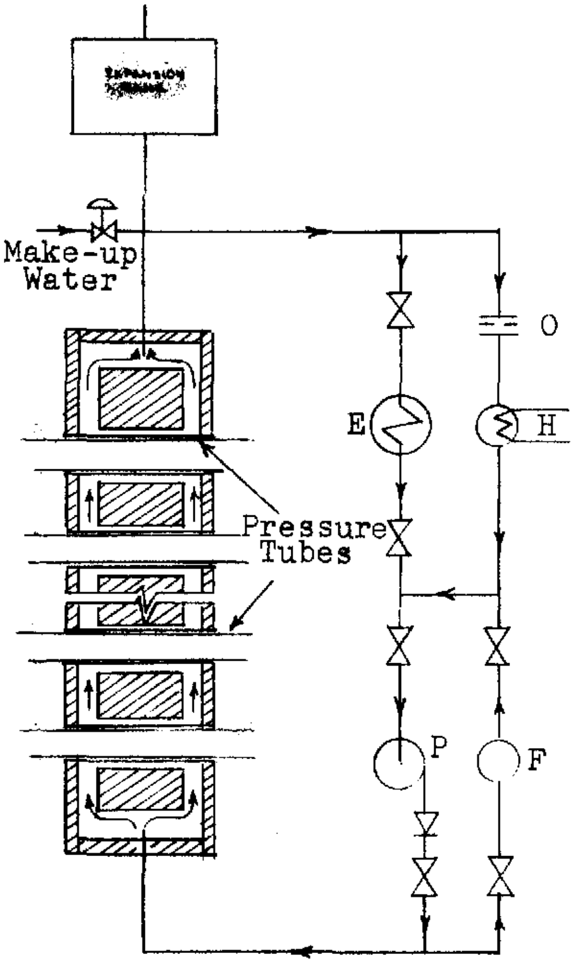


Fig. 5

Each shield consists of a tube and shell type structure containing a steel slab core about 3' thick. A 2" or 3" space separates this core from two steel sheets, each 1-1/2" or 2" thick. These steel sheets act as tube sheets for the lattice tubes through which the reactor pressure tubes pass. The holes in the core are oversized to provide annular channels between the core and the lattice tubes. These annuli form horizontal cooling channels which interconnect the two vertical spaces between the core and tube sheets.

Such end shields can be gas or liquid cooled. The main disadvantage with gas cooling would be the high pumping powers required. There are also additional advantages to using a water cooling circuit:

(a) Photoneutrons may be produced during shutdown in a heavy water moderated reactor. Some hydrogenous material may, therefore, be required in the end shields.

(b) Water is readily available for make-up and is relatively cheap.

The cooling system, shown in Fig. 5, is very similar to the one used for the concrete cooling. Only one heat exchanger, E, and one circulating pump, P, is shown for simplicity. There would be duplication of these and the filter, F, in practice.

The main difference between this system and that in Fig. 4 is that the electric heater, H, is in parallel with the heat exchanger instead of in series with it. This arrangement enables the temperature of the end shields to be kept above the predicted notch ductility temperature of the steel (150°F) even during shutdown periods. A portion of the return flow from the shield bypasses the heat exchangers and passes through the heater. The amount of bypass flow is determined by the orifice, O. Temperature signals from sensing elements in the pump suction line modulate the cooling water supply to the heat exchangers. If the return water temperature drops, the cooling water supply is reduced. If the temperature continues to drop the cooling water supply ceases and a further drop in temperature then causes electrical power to be supplied to the heaters.

Loss of cooling to this system cannot be tolerated so that the operating pumps would be on Class 3 power.

The Use of Thermal and Water Shields

It is possible that the heat generation in a concrete shield may be so severe that too many cooling pipes would be required too close together. The concrete strength and effectiveness would then be seriously affected. In such cases, one of two approaches can be adopted:

(a) The space between the reactor vessel and the reactor vault wall can be filled with light water. Such a light water shield is to be used by C.G.E. in the Karachi Nuclear Power Project. It provides excellent fast neutron shielding and some gamma ray attenuation and substantially reduces the heat generation in the biological concrete shield. The light water shield is easily cooled by a conventional cooling circuit such as those described for reflector cooling.

(b) A steel shield, known as a thermal shield, can be placed between the reactor and the concrete shield. A high percentage of the heat generation then occurs in the steel but, because of its superior thermal conductivity, such a steel shield is not difficult to cool with water or with gas flow. The steel is also, structurally, able to withstand higher temperatures.

The thickness of a steel calandria vessel wall can be increased, beyond that required structurally, in order to provide, at least some, of the thermal shielding required. This has been done with the Douglas Point reactor vessel. An alternative, which has been adopted in the Pickering reactor, is to provide a peripheral thermal shield around the core, inside the calandria vessel. The Pickering internal thermal shield is 4-1/2" thick and eliminates the need for a separate thermal shield inside the vault. With both these approaches, cooling is provided by the moderator.

If a steel thermal shield is used inside the vault wall, the arrangement would be similar to that shown in Fig. 6. Where only a thin shield is required, a single steel sheet about 2-1/2" thick would be used. If a greater thickness is necessary, two sheets would be used to facilitate cooling. The shield can be cooled by gas or water flow but, in either case, a completely welded vault liner would prevent leakage of the cooling fluid into the vault. The cooling circuit would be similar to those already described.

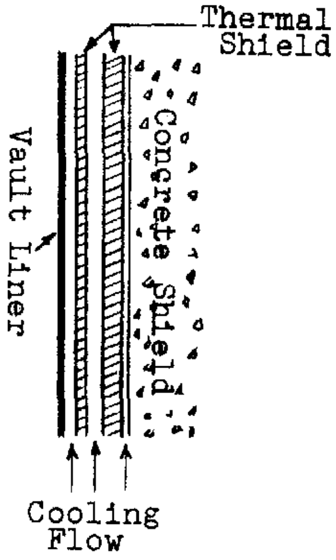


Fig. 6

ASSIGNMENT

1. What are the possible consequences of high temperatures in a concrete shield? Which is the factor that limits the shield temperature?

2. Explain why the heat generation immediately on shutdown may be of greater importance than that during reactor operation.

3. What limits the heat removal capacity of a concrete cooling system?

4. What advantage is to be gained by arranging the concrete cooling pipes in panels with separate return flows?

5. Explain how the reliability of the system is increased and how thermal shock to the heat exchangers is prevented.

6. Under what conditions would steel end shields be used and in what way would they be constructed?

7. (a) What are the advantages of using water cooling rather than gas cooling on the end shields?

7. (b) How and why would the end shield cooling system differ from the concrete cooling system?

8. What possible approaches can be adopted if the heat generation in a concrete shield is too severe for a cooling system to remove?

A. Williams

Reactor Boiler and Auxiliaries - Course 133

THE FUNCTIONS OF REACTIVITY MECHANISMS

The reactivity mechanisms, in a reactor, represent the final control element which causes a change in the neutron multiplication factor, k, or the reactivity δk. There are three general requirements of a control system:

1. It must keep $k = 1$ and $\delta k = 0$ during steady power operation, so that it must compensate for changes in reactivity that occur due to fuel burnup, poison buildup and temperature changes.
2. It must allow δk to become positive or negative in order to change reactor power at the required rate.
3. It must decrease k sufficiently, or provide a large negative δk value, for rapid shutdown of the reactor when this is required.

It is neither desirable nor convenient for these three functions to be executed by the same system. There are, therefore, two systems which are, preferably, entirely independent:

(a) The regulating system which regulates the value of k as required in (1) and (2) above.

(b) The protective system which provides the large negative reactivity for a rapid reactor shutdown as in (3) above.

The reactivity mechanisms used as the final control element in the two systems may also be entirely separate and independent. However, some reactivity mechanisms may serve both purposes.

A reactivity mechanism, used for regulation, must always be capable of providing a reactivity change larger than the following:

1. The random and short-term reactivity variations due to refuelling or temperature changes in the moderator, heat transport fluid and fuel.
2. The long-term variation of reactivity due to fuel burnup.
3. The large excess reactivity in the reactor because of the low fission-product poison level with fresh fuel or at startup following a long shutdown.

4. The reactivity decrease during the first 30 to 45 minutes of the xenon transient which follows a reactor shutdown or power reduction.

The regulating system may also be required to control the neutron flux distribution in the reactor. Heat transport fluid flow through each fuel channel is regulated to give the same exit temperature for all channels with the expected flux and power distribution in the reactor. If the temperature is the maximum achievable without overheating the fuel, such a condition represents the maximum efficiency of energy transportation to the turbine. Under these conditions any distortion of the flux and, therefore, of the power will cause some of the fuel to be overrated unless the total power is reduced. Such total power reductions can be avoided if reactivity mechanisms are used to decrease the reactivity locally where excessive power is being produced and, thus, reduce the power in that region. Such flux and power distortions may well be caused, or at least amplified, by xenon oscillations. Regional flux control by independent reactivity mechanisms is essential if xenon oscillations are likely to occur.

Protective reactivity mechanisms must provide a large reactivity load (ie, negative reactivity) for rapid shutdown. The rate at which this negative reactivity is introduced, as well as its total value, must compensate for the total reactivity increases possible as a result of malfunction of a process system such as the regulating or heat transport systems.

Before the reactivity mechanisms are compared, further consideration should be given the reactivity variations which have to be controlled.

Possible Reactivity Variations

The random and short-term reactivity variations are most frequently caused or connected with on-power refuelling. Burnt or depleted fuel must be replaced so that the fissile atom concentration is sufficient to maintain a chain reaction. Liquid fuel can be continuously replenished so that the correct fissile atom concentration is maintained. Such liquid fuel systems are not yet practical. Solid fuel must be changed discontinuously. After each refuelling operation the fissile atom concentration is excessive and must be compensated for by increasing the reactivity load, ie, by removing excess neutrons not required to maintain the chain reaction. The fissile atom concentration then returns to the required value just before the next refuelling and the reactivity load must be correspondingly reduced during this period.

With on-power fuelling the normal reactivity change during refuelling is small (less than 0.15 mk in Pickering) and the rate of change of reactivity is also small (less than 0.1 mk per second in Pickering). The replacement of all the fuel bundles with new fuel in a central channel would result in a reactivity increase of 0.5 mk in Douglas Point and 0.22 mk in Pickering. The rate of reduction of reactivity, in either reactor, with no refuelling is about 0.4 mk per day. These reactivity changes and rates of change are compensated for without difficulty.

In reactors which cannot be refuelled on power, the refuelling frequency will not be greater than once every six months. Each refuelling then results in a large increase in reactivity. The compensating reactivity load must be decreased over a short period of time to allow for poison buildup and it must be decreased over an extended period of time to allow for fuel burnup.

Little change in moderator temperature should normally be expected unless there is a change in the efficiency of the heat removal system. However, substantial changes occur in the fuel and heat transport fluid temperatures due to changes in reactor power. The power coefficient of reactivity is usually negative for stability and because of safety requirements. Thus, an increase in power causes a decrease in reactivity. In Douglas Point an increase from zero to full power causes a 5.3 mk decrease in reactivity. A compensating reactivity load must, therefore, be inserted, at low power, which can be removed as reactor power increases.

With initial fuel in Douglas Point, at zero power, an increase of heat transport fluid temperature to 504°F causes a decrease of 7 mk in reactivity.

The long-term variations of reactivity with burnup are generally much greater than those discussed above. For instance, in Douglas Point there is a decrease of 55 mk from the initial fuel charge to the equilibrium fuel condition, assuming equilibrium xenon in both cases. The equilibrium xenon load is also large compared to the values above being about 28 mk in both Pickering and Douglas Point. Thus a reactor with fresh clean fuel would have an excess reactivity of 82 mk compared to an equilibrium core with equilibrium zenon load.

Since the maximum xenon transient load is more than double the equilibrium load, it is not practical to have enough excess reactivity in the reactor to avoid a poison-out altogether. Normally, enough excess reactivity is available to provide a poison override time of about 45 minutes. The rate of growth of the xenon load in Douglas Point and Pickering, following a reactor shutdown from full power, is about 23 mk per hour. Thus, in 45 minutes the xenon load will have increased by about 16 mk.

It may be seen from the above considerations that there are three classes of reactivity variations which have to be considered:

(a) Small reactivity variations as a result of refuelling or temperature changes.

(b) The xenon transient load increase during the poison override time.

(c) The large decreases in reactivity due to such things as equilibrium poison buildup and long-term fuel burnup.

There is one overriding consideration and that is the necessity of regional control to prevent flux and power distortions due to xenon oscillations.

Types of Reactivity Mechanisms

All reactivity mechanisms either increase neutron production or decrease neutron losses in order to change the value of k. Neutron production can be increased by addition of fissile material. Neutron losses can be changed by:

(a) Changing neutron absorption in materials other than the fissile material. This can be done in a variety of ways:

  1. By inserting or removing neutron absorbers into or out of the reactor core.

  2. By changing the moderator density so that a change occurs in the absorption by nonfissile fuel nuclei.

(b) Changing neutron leakage out of the reactor either by:

  1. Changing core size or

  2. Changing reflector thickness.

It is clear that one type of reactivity mechanism, relying on one of the above methods of reactivity variations, could not meet all the regulation requirements discussed previously. It would be even less likely to meet the protective system requirement as well. Thus, several types of reactivity mechanisms would be used in the same reactor and these could be classified as to types and functions as follows:

(a) BOOSTER MECHANISMS which would be used to increase reactivity, during xenon transients, to provide the required poison override time. Such booster mechanisms could be:

1. Absorbers, such as Cobalt rods, which are in the reactor during normal operation and are removed to reduce neutron losses during a xenon transient.
2. Additional fuel rods which are inserted into the reactor only when required for additional reactivity during a xenon transient.

(b) Variable REACTIVITY LOADS to compensate for small variations in reactivity. Since such reactivity loads must be continuously variable during reactor operation, they must depend on a continuously variable neutron removal system such as:

1. Variable neutron leakage achieved by variable moderator-reflector level which varies the core size or the reflector thickness above the core.
2. Variable moderator temperature to vary neutron leakage and relative neutron absorption in nonfissile fuel material.
3. Variable moderator density, such as obtained by bubbling a gas through it, to achieve the same results as in (2).
4. Variable neutron absorbers in the core.

(c) POISON SYSTEMS to simulate equilibrium fuel conditions or equilibrium xenon loads in a reactor in which sufficient excess reactivity has been provided to compensate for fuel burnup and xenon buildup to equilibrium. This is known as SHIM control.

(d) SHUTDOWN MECHANISMS to provide the large negative reactivities required following a reactor trip. Such large negative reactivities can be obtained by:

1. Dumping the moderator, if it is a liquid, and thus greatly increase neutron leakage, or
2. Inserting absorber, known as SAFETY or SHUT-OFF RODS.

ASSIGNMENT

1. Distinguish between the functions of the regulating and protective systems.

2. What five requirements are demanded from reactivity mechanisms used for regulation?

3. Why is it not possible for one type of reactivity mechanism to meet all the requirements in (2)?

4. Explain the four types of reactivity mechanisms that can be used to meet the regulating and protective system requirements.

A. Williams

Reactor Boiler and Auxiliaries - Course 133

COMPARISON OF REACTIVITY MECHANISMS

When a CANDU type reactor has reached the equilibrium core condition, the most important method of controlling reactivity is by on-power refuelling. However, in solid fuel systems continuous variations in reactivity cannot be obtained by such a method. Other means of compensating for small variations in reactivity must, therefore, be used. This is required even more in reactors where on-power refuelling is not used. Reactivity mechanisms must also be used for controlling other types of reactivity variations, as outlined in the previous lesson.

These reactivity mechanisms will now be discussed further and the various types of mechanisms available compared.

VARIABLE REACTIVITY LOADS

These are the continuously variable reactivity loads which are required to compensate for small variations in reactivity caused by on-power refuelling, temperature changes. The characteristics required for such a mechanism must satisfy the following limitations:

1. The maximum reactivity worth must be larger than the maximum reactivity increase for which it must compensate (eg, the refuelling of one complete channel with fresh fuel).
2. The rate of reactivity load insertion must be greater than the maximum rate of reactivity increase for which it must compensate (eg, caused by refuelling).
3. The rate of reactivity load removal must be greater than that of the fastest reactivity decrease for which it must compensate (eg, due to moderator or other temperature change).
4. Where such a mechanism is also used to counterbalance flux distortions, such as those caused by xenon oscillations, the distribution of reactivity load between the various reactor zones or regions must be such that the system be capable of counterbalancing the largest flux distortion that can occur.
5. The rate of load removal must never be larger than the rate of insertion of the protective reactivity mechanisms.

As was stated in the previous lesson, such variable reactivity mechanisms must depend on a continuously variable neutron removal system. Such systems will now be considered further and their advantages and disadvantages discussed.

Moderator Level Variation

Variation of moderator level can result in one of two further variations. If all the fuel channels are not covered, it results in variations in core size. Once all the fuel channels are covered, it results in variation in reflector thickness, assuming that the reflector is merely an extension of the moderator beyond the edge of the core. In both cases, the end result is a variation in neutron leakage.

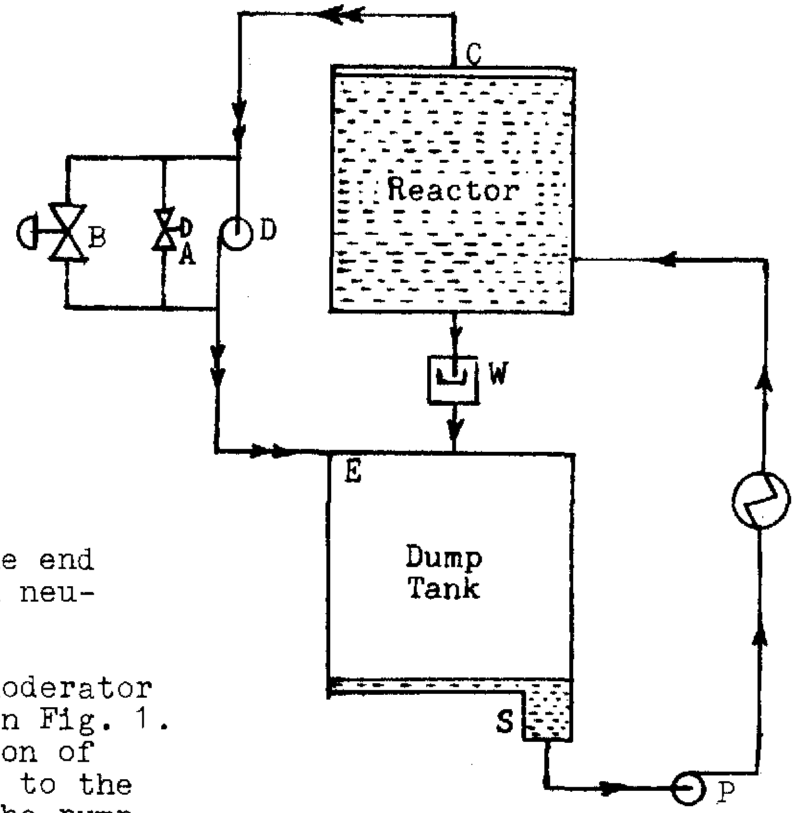


Fig. 1

The mechanism for moderator level control is shown in Fig. 1. There is constant addition of water from the dump tank to the reactor vessel through the pump P. The water returns to the dump tank through the dump port W, which is, in effect, a weir over which the water spills. A liquid-gas interface is established at the weir and the pressure differential between this interface and the point C, at the top of the calandria, determines the height of moderator in the reactor vessel. The necessary pressure differential is established by the gas blowers or water-jet exhausters at D. The regulating valves, A, permit the controlled gas leak rate between E and C which is required to establish a specific moderator level.

The main advantages with such a system are:

1. A simple arrangement of valves and blower only are required which can be located in accessible or partially accessible areas.

2. The use of valves for regulation permits the use of triplicated control systems with virtually complete independence of the three control channels.

3. The same system can also be used as the shutdown mechanism by simply dumping the water out of the reactor into the dump tank. Large dump valves, B, open, on a reactor trip, to rapidly equalize the pressure between E and C. This aspect will be considered further in the lesson.

One major disadvantage of moderator level control is that the neutron leakage variations, which it initiates, cause a distortion of the flux distribution through the reactor core. The low moderator level that results with fresh fuel or because of the absence of xenon poison would cause such a distortion in flux that it forces a reduction in power. The low moderator level could also result in some calandria tubes not being immersed. Spray cooling of these tubes would then be required to prevent stresses due to differential expansion.

The second disadvantage is that moderator level control applies to the reactor as a whole and is not suitable for regional or zonal control of flux distribution.

Moderator Density Variation

The density of any moderator can be changed by varying its temperature. The resulting variations in both density and neutron temperature cause a change in neutron leakage, a change in neutron capture in nonfissile nuclei and a change in the ratio of fissions to absorptions in the fissile nuclei. However, the moderator temperature coefficient changes with fuel burnup because of the different effect of neutron temperature changes on Pu-239 and U-235. This is not, therefore, considered a useful control mechanism.

The density of a liquid moderator can also be varied by bubbling a gas through it. Such a change in density changes the neutron leakage out of the reactor. The main problem with such a method is to maintain uniformity of bubble sizes, so that no practical system has been developed although the method has been considered. It has the advantage of not causing flux distortion but it is not suitable for regional control.

Variable Neutron Absorbers

Neutron absorption in a reactor can be increased by introducing neutron absorbers into the core. Such absorbers must be easily removed to increase reactivity when necessary. In a liquid-moderated reactor the simplest method of introducing such an absorber is in the form of a soluble poison. The soluble poison can be injected into the suction of the moderator circulating pump, P, in Fig. 1, ie, outside the core. It has several advantages:

1. No complex driving mechanisms are required in high radiation fields.
2. No permanent structures inside the core, such as guide tubes, are required.
3. There is conservation of neutrons when the absorber is not in use.
4. Because the absorber is uniformly distributed throughout the moderator, there is negligible flux distortion.

Such a method is, however, unsuitable for this type of fine control because the removal of the poison from the moderator is a slow and expensive operation, involving the use of ion-exchange columns. In addition, such a poison system is unsuitable for regional control of flux distribution. However, it is suitable to simulate the equilibrium xenon load and its application, in this manner, will be discussed later.

The alternative is to introduce local neutron absorbers in various regions or zones. The absorber can be of any shape and either solid, liquid or gaseous, provided it has a large neutron absorption cross section. A solid absorber is usually in the form of a rod which is inserted into the reactor along a guide tube or thimble. The rod is driven by a mechanism which allows for its insertion or removal at a definite rate.

The Douglas Point reactor has two such thimbles, like the one shown in Fig. 2, in the reactor axial centre plane. Each tube contains two absorber rods of stainless steel which move in and out in opposite directions. They are moved in and out by lead-screw drives. One advantage of the solid rod type of absorber, over the liquid or gaseous ones, is that, in liquid moderated reactors, it can be cooled by circulating the moderator through the guide tube. Thus, in Fig. 2, moderator fluid enters at A and eventually spills into the reactor at B.

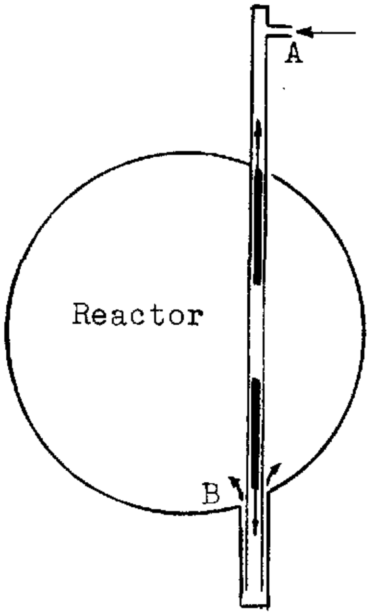


Fig. 2

The following are the desirable properties of neutron absorbers:

1. High neutron absorption cross section at the right neutron energies so that less absorber is required. This simplifies the design problems associated with the introduction of the absorber into the core and reduces the permanent reactivity reduction caused by the guide tube.

   The absorption cross section should be highest for neutrons of prevailing energy in the core. However, a cross section which is fairly large over a wide energy range may be preferred to a larger cross section at the prevailing energy which becomes very small at other energies.

2. Low rate of depletion of the absorbing isotopes.

3. Low neutron scattering cross section by absorber or its cladding. This condition is necessary to avoid reducing the efficiency of the absorber by scattering neutrons out of the absorber before they can be absorbed.

4. Low neutron activation which would increase the problem of radiation protection for personnel and equipment.

5. Adequate mechanical strength for solid absorbers.

6. Small weight to make it easier to move.

7. Negligible corrosion of solid absorber by coolant or of container by fluid absorber.

8. Good chemical and physical stability under high temperature and irradiation conditions.

9. Reasonable cost and availability and ease of fabrication if a solid.

0. Good heat transfer properties for ease of cooling.

1. No deposition, by plating out, etc, of dissolved absorbers on container walls and connecting lines to avoid a permanent reactivity load.

The final choice of absorbing material will depend on a compromise between these properties, some of which are conflicting. Possibly the foremost consideration will be the economic feasibility of a particular absorber for the reactor system under consideration.

The low rate of depletion requirement contradicts the high absorption cross section requirement, because the rate of depletion is proportional to the cross section. It would appear, then, that the low depletion rate sets an upper limit on the value of the cross section. However, the depletion rate can be reduced by one of two methods:

(a) The absorber can be made thick enough to cause the absorption on the surface of the absorber to depress the neutron flux inside the absorber. The inner absorber nuclei then become very inefficient, in terms of neutron absorption, and they initially become depleted very slowly. They act as spares which are used after the surface nuclei have been used up.

(b) Elements can be used, as neutron absorbers, which are transformed into another absorbing isotope when they absorb a neutron. The depletion of the original isotope is, therefore, compensated by the production of another absorber. In some cases a chain of transformation into new absorbers is possible by successive neutron absorption.

Comparison of Possible Solid Absorbers

Table I shows the absorption cross sections, of the most likely absorber materials, for thermal neutrons and for resonance energy neutrons.

As may be seen from the table, Hafnium has a relatively low thermal cross section but is still considered a very efficient absorber because of its high cross section in the energy region immediately above the thermal energies (the so-called EPITHERMAL region). It is readily fabricated and has very good corrosion resistance in high-temperature water. One major advantage of using Hafnium is that each isotope on capturing a neutron changes into a series of neutron capturing isotopes according to:

$$Hf^{177} + n^1 = Hf^{178}; \quad Hf^{178} + n^1 = Hf^{179}; \quad Hf^{179} + n^1 = Hf^{180}$$

Hafnium is available as a by-product in the production of Zirconium and its chemical properties are similar to those of Zirconium. It has adequate mechanical strength for use in rods.

Boron has a higher thermal cross section than Hafnium and a reasonable epithermal cross section, since its cross section varies inversely with the neutron velocity. It is, therefore, more efficient than Hafnium for thermal reactors in which the epithermal flux is low. It is usually dispersed in other materials, such as steel, to give it better corrosion and other physical properties. On neutron capture it transforms into He-4 and Li-7. This transformation may cause swelling and cracking of the containing material. Because of this, Boron would be better used as a liquid absorber.

TABLE I

Material	Abundance (per cent)	Thermal σ_a (barns)	Thermal Σ_a (cm^{-1})	Major Resonances	
				Energy (ev)	σ_a (barns)
Boron		755	107	—	—
Boron-10	20	3800	—	None	
Silver		62	3.64	—	—
Silver-107	51.3	31	—	16.6	630
Silver-109	48.7	87	—	5.1	12,500
Cadmium		2450	113	—	—
Cadmium-113	12.3	20,000		0.18	7200
Indium		190	7.3	—	—
Indium-113	4.2	58	—		
Indium-115	95.8	197	—	1.46	30,000
Samarium		5600	155		
Samarium-149	13.8	41,000	—	0.096	16,000
Samarium-152	26.6	225	—	8.2	15,000
Europium		4300	90		
Europium-151	47.8	7700	—	0.46	11,000
Europium-153	52.2	450	—	2.46	3000
Gadolinium		46,000	1400		
Gadolinium-155	14.7	61,000	—	2.6	1400*
Gadolinium-157	15.7	240,000	—	17	1000*
Hafnium		105	4.71		
Hafnium-177	18.4	380	—	2.36	6000†
Hafnium-178	27.1	75	—	7.8	10,000
Hafnium-179	13.8	65	—	5.69	1100†
Hafnium-180	35.4	14	—	74	130

* Gd^{155} and Gd^{157} have several important resonances in the energy range from 2 to 17 ev.

† Hf^{177} and Hf^{179} have several important resonances in the energy range from 1.1 to 50 ev (and smaller ones up to about 100 ev).

Cadmium has a very high thermal cross section that drops fast for higher energies. Thus, it is not efficient in reactors with high epithermal fluxes. Moreover, its high cross section increases its burnup rate. Finally, it has a low melting point and poor corrosion resistance. To overcome these disadvantages, it is usually alloyed with other absorbers (eg, Silver and Indium).

Silver has too low a thermal cross section to be used alone in thermal reactors but it can be used as a good base for alloys containing such high thermal cross section absorbers as Cadmium, since it contributes epithermal absorption.

The rare earths, Europium and Gadolinium, have very high thermal cross sections and good epithermal cross sections. Moreover, Europium under neutron irradiation goes through a series of transformations similar to Hafnium, from Eu-151 to

Eu-156. They are not widely used as absorber materials because their properties are not well established.

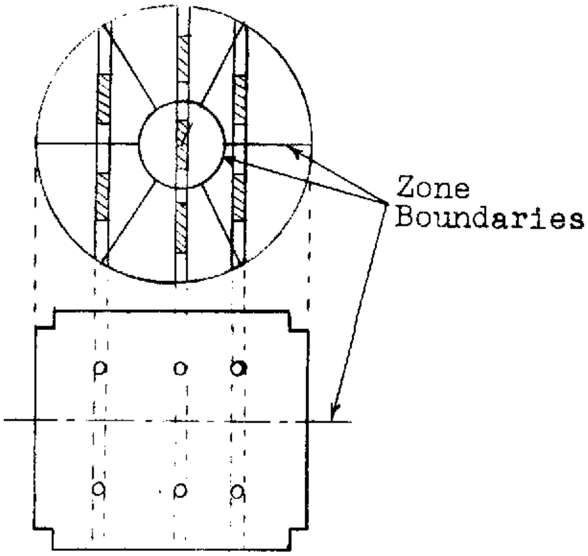
The introduction of any localized absorber into a reactor will cause some flux distortion. However, this factor may be used to shape the neutron flux to some desired optimum. This type of reactivity mechanism is also better suited to the regional control of flux disturbances, since the insertion of each absorber is regulated by an independent mechanism which can be controlled by independent local flux detectors.

Absorbers, particularly in the form of rods, are also suitable as shutdown mechanisms and this is discussed later.

The use of absorbers, in the form of rods, has one major disadvantage. They require complex driving mechanisms which must operate reliably in high radiation fields.

<u>The Use of Fluid Absorbers</u>

To avoid the disadvantage of complex driving mechanisms, the absorber can be introduced, as a gas or a liquid, into partitioned volumes or compartments in tube or thimble penetrating the core. Such an arrangement, shown in Fig. 3, is used at Pickering. Light water is introduced into each compartment through small diameter tubing. Small diameter tubing must be used to feed the compartments in order to ensure that most of the absorber is in the compartment and not in the tubing. Even so, the neutron absorption in the compartment walls and in the interconnecting lines adds to the permanent reactivity load in the core. Therefore, there should be as few separate compartments as possible. The use of liquid absorbers does not, however, prevent the flux distortions associated with localized absorbers, but again this system can be used to shape the neutron flux and to provide regional or zonal control. Its use for zonal control is clearly demonstrated in the Pickering system, shown in Fig. 3.



<u>Fig. 3</u>

The reactor is divided into 14 zones. First, the reactor is divided into two axial slices and each slice is then divided into one central and six outer zones. As may be seen, from Fig. 3, there is a fluid absorber compartment in each zone. The light water in each compartment is continuously circulated for cooling and chemical control. A typical flowsheet is shown in Fig. 4.

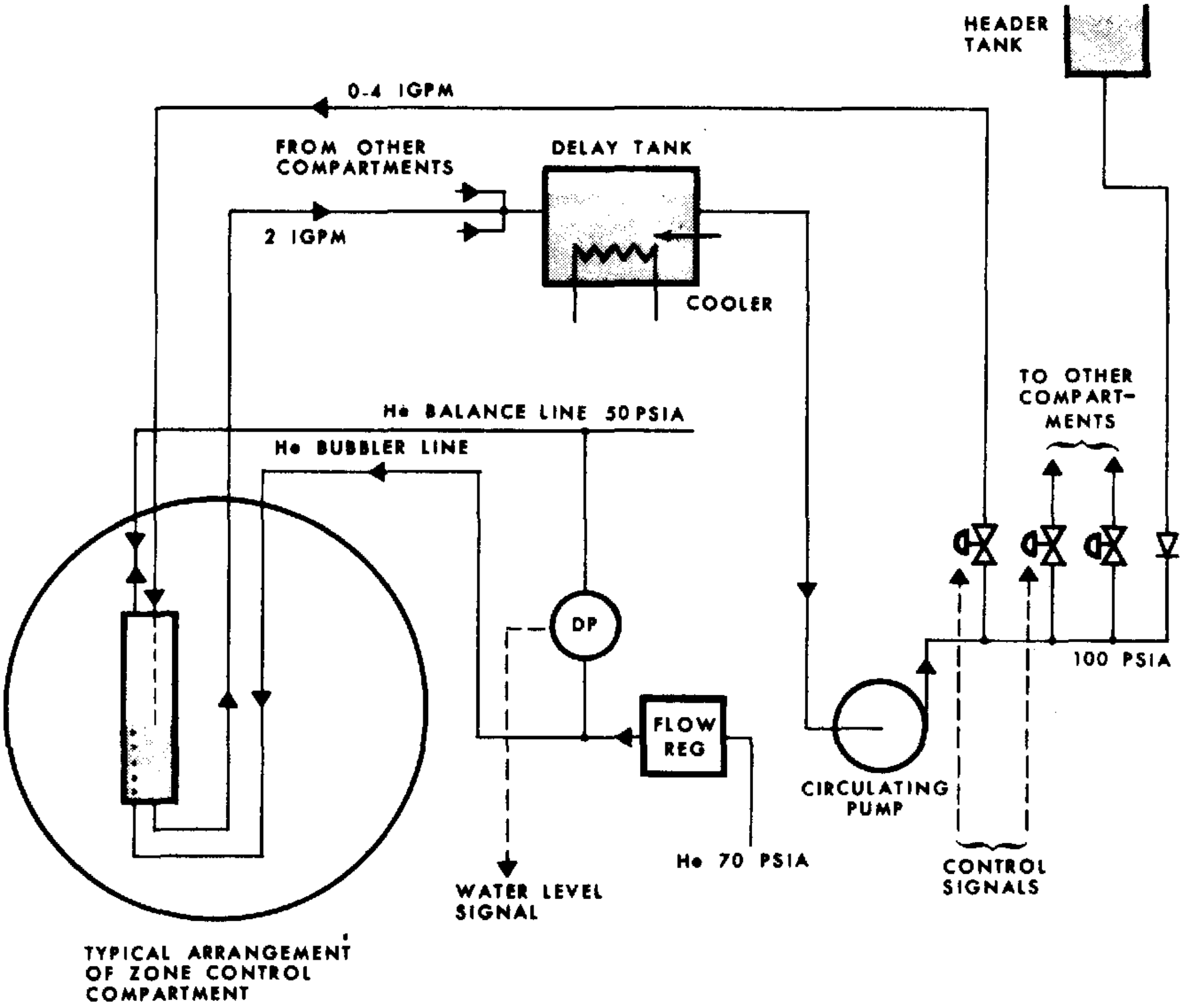


Fig. 4

The depth of water in each compartment is independently controlled through electropneumatically operated control valves which are accessible during operation. There is a constant outflow of 2 Igpm from the bottom of each compartment and a controlled inflow, in the form of a jet from the top, from 0 to 4 Igpm. This arrangement ensures that the light water is always circulated and cooled. If all compartments receive the same control signal, the reactivity rate would be about ±0.2 mk per second.

A delay tank permits the induced O-19 and N-16 activity to decay so that the valves, pumps and blowers are accessible during operation. The water level in each compartment is indicated by the gas bubbler method. The head tank permits full control to be retained during a failure of both water pumps for a few minutes. If the calandria pressure rises, all connecting tubes would be closed so that light water could not be ejected from the compartments even if the tube wall collapsed.

The estimated internal diameter of the compartments is 3.5" and the connecting tube will be about 0.3" internal diameter. About 450 lb of light water is needed to fill all the compartments.

Any suitable liquid could be used instead of water, such as a solution of any of the solid absorbers previously discussed. Some method would have to be used to ensure that the dissolved absorber did not become a permanent reactivity load in the reactor by plating out on the compartment wall or the tubes.

One advantage of using a gaseous absorber is the further regulation achievable by varying the pressure of the gas. However, this advantage might be offset by its large temperature coefficient, which is due to its large variation in density with temperature. Helium-3 suggests itself as a possible gaseous absorber since it has a thermal absorption cross section of 5.3×10^3 barns. However, its isotopic abundance is only 0.00013 and the only other naturally occurring Helium isotope, Helium-4 has a very low absorption cross section.

SHUTDOWN MECHANISMS

The considerations that apply to shutdown mechanisms are similar to those for variable reactivity loads. Such a protective mechanism must be capable of adding enough reactivity load rapidly enough to shut down the reactor safely under any condition. This may be accomplished either by rapidly absorbing neutrons with absorbers or by rapidly increasing neutron leakage. Because of the magnitude of the reactivity load required and the speed with which it must be inserted, only two alternatives can be considered:

(a) If the increased leakage method is to be chosen, it must be achieved by a large and rapid decrease in core volume. This is possible only in liquid-moderated reactors, by dumping the moderator out of the core.

(b) Rapid increase of neutron absorption is best achieved by the rapid insertion of absorber rods, known as safety or shutoff rods, into the core. The solid rod mechanism can be designed with greater simplicity to fail safe rapidly enough under any condition including the failure of the rod mechanism itself.

The choice between the two alternatives will depend on relative reliability, speed of response and cost. Moderator dump has the advantage of simplicity with no mechanisms located in inaccessible, high radiation areas. The only moving mechanisms required are the valves which open to equalize the moderator cover-gas pressure between the reactor and the dump tank, and such valves can be located in accessible areas. It is a very satisfactory mechanism for a small reactor, such as NPD, where it produces a 5 mk reactivity decrease in less than 1 second and 70 mk in 5 seconds.

However, in large reactors, such as those in Pickering, such a dump requires the rapid movement of several tons of water from the reactor vessel into the dump tank. This introduces engineering problems in the design of dump ports which will allow such a rapid dump and still support a calandria full of water. Such large dump ports, which are placed on the core boundary, represent a reactivity load, and they also imply more heavy water holdup in D_2O lattices. Moreover, after a complete dump, with the moderator pump-up rate limited by safety considerations, the time necessary to return the moderator to the calandria increases the probability of poison-out with its consequent power production loss. It is estimated that, in a Pickering unit, the pump-up time is 50 minutes, whereas the poison override time provided by booster mechanisms is only 45 minutes.

It could be argued that one big advantage of such a dumping facility is that it uses the same mechanisms as moderator level regulation. However, in large reactors, the moderator level mechanism can not be used as the only method of reactor regulation because it does not permit zonal control against flux distortions caused by xenon oscillations. It could, thus, be argued that it is not required and that variable absorbers offer more advantages.

In a reactor which does not have moderator dump facilities (eg, graphite-moderated reactors), safety rods must be used. The main disadvantage with such a system is that it requires mechanisms for rapid insertion which must operate reliably in high radiation fields where maintenance is difficult because of inaccessibility. The choice of material for safety rods is based on similar requirements to that of regulating rods. However, the high neutron absorption requirement is now much more important than a low rate of depletion since the rod is only inserted following a reactor trip. Since regulating rods are required for small reactivity variations and safety rods for large reactivity changes (24 mk in Pickering), the two mechanisms are likely to be completely independent. This arrangement is also preferable because of the desirability of having the protective system independent of the regulating system. Safety rods are usually permitted to fall into the core under gravity to obtain the required fast insertion without relying on moving mechanisms. Regulating rods, on the other hand, must be moved in and out of the core by some mechanism. However, the same rods have been used in some reactors for both purposes. The regulating mechanisms move the rods in and out as required and magnetic clutches allow the rods to fall into the core when a reactor trip occurs. This does mean, however, that the tip of the rod, which is being used for regulation, is being continuously depleted.

The reliability of a safety rod system can be improved by increasing the number of rods. However, the main disadvantage of the system is that it could be most unreliable at a time when it was most required. Any condition, such as an earthquake or a power excursion, which would initiate a reactor trip, could also distort the safety rod guide tubes so as to prevent insertion of the safety rods. Thus, in the CANDU concept, moderator dump is still used as a backup method for reactivity reduction. The initial dump requirements are not, then, as severe as when moderator dump only is used. Moreover, the dump can be stopped as soon as the correct functioning of the safety rods has been established. This reduces the time required to pump the moderator back into the calandria.

BOOSTER MECHANISMS

Booster mechanisms are required to increase the reactivity, during xenon transients following a power reduction, to provide the required poison override time. Such a xenon transient is illustrated in Fig. 5. A power reduction is assumed to occur at B and the xenon load builds up along BDC. If "a" represents the poison override time required, then the excess reactivity which must be available is that which is shown at D. Thus, to provide a 40-minute override time in a Pickering unit, the booster mechanism must provide 16 mk of available reactivity. It must be remembered that a poison-out can only be avoided if the reactor is returned to 65% or 70% of full power in order to burn out the xenon faster than it is being produced. This means that the decrease in reactivity due to the power coefficient must be allowed for in addition to the increase in xenon load during the poison override time. Therefore, the total increase in reactivity required is that equal to BD plus that due to the power coefficient. In order to assess the poison override time required, an evaluation must be made of the probable frequency and duration of power reductions and an estimate made of the savings, through extra power production, which can be obtained by permitting such an override time. The cost of the booster system must then be subtracted from these savings and the difference optimized with respect to the booster reactivity worth. The rate of insertion of this excess reactivity is made as fast as is permissible by safety considerations. Thus, its rate of insertion must never exceed the rate of reactivity decrease that can be achieved by the protective mechanisms.

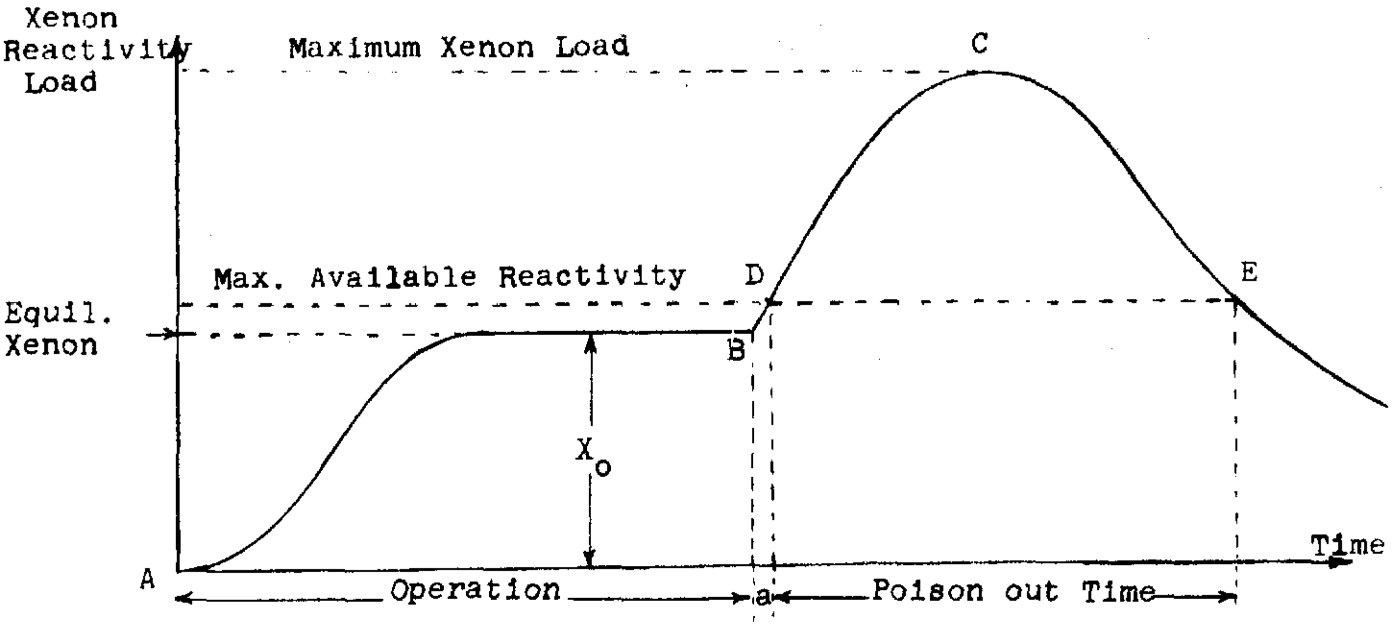


Fig. 5

A booster mechanism can increase reactivity by reducing neutron removal or by increasing neutron production. The former is achieved by designing the reactors to operate with absorbers in the core and removing these absorbers during the xenon transient. The alternative is to increase neutron production by inserting additional fuel into the core.

If removable absorber rods are used, the reactor must be made larger than required and the excess neutrons absorbed in the booster mechanism instead of being used to produce power. This is uneconomical unless there is some other justification for using such a system. In heavy water moderated reactors, neutron economy is of even greater importance, since this is the justification for using such an expensive moderator. The use of absorbers as booster mechanisms is, therefore, contradictory to the choice of moderator. The fissile material, on the other hand, is only inserted when required and it does not add to the fuel inventory in the reactor.

The additional fissile material is inserted in the form of a fuel rod, the mechanism being similar to that illustrated in Fig. 2. Smaller quantities of fissile material are required with solid fuel compared with liquid fuel because the latter has to be circulated outside the core. It is also convenient to cool a rod by means of the moderator, if this is a liquid. The disadvantage of using a rod is the need for complex driving mechanisms in high radiation fields. The booster fuel is usually highly enriched with fissile atoms in order to reduce the amount of fuel to be added. This, in turn, minimizes the change in reactor structure required and reduces the reactivity load of the guide tubes. The smaller volume of the booster fuel does, however, cause more flux distortion and increases the cooling problems. U-235 is preferred to Pu-239 as the fissile material because of its higher delayed neutron emission which gives a longer reactor period for the same increase in reactivity.

As was stated earlier, the use of removable absorber rods as booster mechanisms can only be justified on economical grounds. For instance, if the neutrons lost for power production could be used to produce some useful isotope then there may be some justification for using absorber rods. The absorber rod could, for example, be made from Cobalt-59 from which Co-60 could be obtained.

$$\mathrm{Co}^{59} + n^1 = \mathrm{Co}^{60} + \gamma$$

Alternatively, Thorium rods could be used to produce U-233.

$$\begin{tikzcd}[row sep=small, column sep=tiny] \mathrm{Th}^{232} + n^1 \; = & \mathrm{Th}^{233} + \gamma \arrow[d, "\beta"'] \\ & \mathrm{Pa}^{233} \arrow[d, "\beta"'] \\ & \mathrm{U}^{233} \end{tikzcd}$$

Such absorber rods, known as ADJUSTER rods at Pickering, could also be used to adjust the neutron flux to some optimum shape. However, the use of such rods represents a loss of production of useful power and, in a power reactor, their use can only be justified by an acute demand for the particular isotope which is produced.

POISON SYSTEMS

To maintain constant power production from a reactor, the production of neutrons must exactly balance their removal. Neutrons are produced from fission as follows:

$$U^{235} + n^1 = X + Y + \nu n$$

where X and Y are fission products and ν neutrons are produced per fission.

Neutrons are removed by capture in core material or by leakage from the core.

Each fission process represents a loss of one fissile nucleus and the production of two new neutron absorbing nuclei. Thus, unless the fuel in the reactor is continuously replenished, the neutron production will decrease because of the depletion of fissile nuclei and neutron removal will increase because of the production of fission product absorbers or poisons. If the reactor is designed to be critical with new fuel, it would be necessary to continuously add new fissile material as the fissile material is depleted and the fission product poisoning increases. If the fissile material in the fuel is to be properly utilized (ie, high fuel burnup achieved), new fuel must be added, initially at least, without removing old fuel, since this old fuel still contains a high proportion of the original fissile atoms. Some new fissile atom contribution is obtained from the conversion of fertile atoms (such as U-238) into fissile atoms. However, this compensates only to a very limited extent for the U-235 burnup.

The addition of new fuel without removal of old fuel poses a core design problem and it involves overrating the fuel initially or underrating it later. On the other hand, to remove fuel with low burnup is uneconomic. The only alternative is to provide an excess of fuel initially in the core so that it can be left in until its burnup value is economical. The excess reactivity with new fuel must then be temporarily compensated for by increased neutron removal.

The xenon poison load is initially zero in a reactor and increases to its equilibrium value after about 60 hours of reactor operation. When the reactor is shut down the xenon transient causes the xenon load to increase for 10 hours or so and it then decays, during a sufficiently long shutdown, to an essentially zero value. Thus, following a long shutdown, some neutron removal mechanism must be introduced to balance the lack of poison load.

With a fresh core, the excess reactivity for which compensation is required is the sum of the load due to fuel depletion and due to equilibrium xenon poison. Typical values for different types of reactors are shown in Table II.

The time necessary to bring the fuel charge to equilibrium burnup is of the order of months or years and the compensating load must be removed over this period of time. The compensating load for the equilibrium xenon must, however, vary much more rapidly. It may, therefore, be seen that large reactivity loads and relatively slow time constants are involved. The reactivity mechanisms used to compensate for these large reactivity variations are known as SHIM controls.

TABLE II

Reactor	Douglas Point	Organic-D_2O	Gas-Graphite	Light Water	
				Boiling	Pressurized
Burnup Load (mk)	49	10	-	60	70
Poison Load (mk)	38	31	25	40	33

In NPD this shim control, following a reactor shutdown, is obtained by operating with low moderator level. For initial startup, depleted fuel had to be used in addition. Such a low moderator level produces serious flux distortions which would be compounded, in a large reactor, by xenon oscillations. The consequent power distortion causes a forced power reduction which could not be tolerated in a large power reactor, since it results in a serious loss of revenue. Some other reactivity mechanism is, therefore, required which could be used, on initial startup, with depleted fuel, since such fuel is useful in shaping the initial flux distribution to as close to the equilibrium distribution as possible.

The Advantages of a Poison System

The most obvious method of neutron removal that can be used for shim control is by insertion of neutron absorbers into the core. In reactors using a solid moderator such absorbers would have to be in the form of rods, since the poison cannot be dissolved in the moderator. However, fast removal of the absorbers is no longer a requirement because of the long time constants involved. There are other factors that are much more important:

1. There must be as little interference with the flux distribution as possible. The size of the reactivity load involved increases the possibility of serious flux distortion and consequent overrating of fuel.

2. The absorber cannot inadvertently be removed. If it is in soluble form, it can only be lost by loss of moderator.

   These safety requirements preclude the use of liquid poison in the heat transport system or a gas poison in the annuli between the pressure tubes and calandria tubes. The loss of gas or the interruption of gas flow through malfunction of the control system could leave the reactor prompt critical because of the large reactivities involved. Similarly a loss of coolant accident would be worsened by the loss of poison.

3. It must require a minimum of mechanical control since it is difficult to develop sufficiently reliable mechanisms which must operate in high radiation regions with low accessibility.

4. When the absorber is removed, it must not leave a permanent reactivity load in the core.

Conditions (1), (3) and (4) cannot be met with absorber rods but they are satisfied with a liquid poison system. Condition (2) is also satisfied if the liquid poison is dissolved in the moderator.

Poison System Considerations

The following nuclear characteristics must be considered in evaluating the poison to be used:

1. The neutron absorption cross section. The cross section at the neutron energy corresponding to the moderator temperature to be expected must be such that the poison concentration required satisfies the chemical and metallurgical requirements of the moderator system. It must also satisfy the requirement of ease and cost of control of the poison concentration.

2. Rate of depletion by neutron capture. Again it is an advantage for one poison nucleus to be transformed, by neutron capture, to another poison nucleus. If this does not happen, the depleted nuclei must be removed and replaced continuously.

3. Type and intensity of activation due to neutron capture. This will determine whether or not additional shielding is required around the moderator system and, if so, how much. Maintenance of equipment and replacement of ion-exchange columns could be seriously hindered if there is high neutron activation of the poison which produces a penetrating gamma ray emitter.

The poison must also have the following nonnuclear characteristics:

1. Low cost and good availability.
2. Ease and economy of removal from the moderator. Since such a poison is normally removed by ion-exchange resins, this will be determined by the volume of resin required, the cost of the resin, the range of concentrations which can be processed with one bed and the rate of poison removal.
3. Negligible effect on the radiolysis of heavy water. Any excessive D_2 concentration caused by increased radiolysis results in an explosion hazard and excessive oxygen concentration causes increased corrosion.
4. Chemical stability under core conditions.
5. Negligible effect on moderator system corrosion problems so that there is no change in the pD or in the quantity of dissolved or suspended solids in the system.
6. Negligible deposition inside the calandria. This is not too serious a problem if the depletion rate is high.

The two most common poisons are Cadmium-113 and Boron-10. Natural Cadmium and natural Boron are compared in Table III on the basis of the criteria established above.

The comparison of Table III shows that the main advantages of Boron over Cadmium are the higher absorption rate per unit concentration which results in a lower concentration being required and in a lower rate of depletion.

The main advantage of using Cadmium is its greater ease of removal. The boric acid formed, when the Boron is introduced into the moderator, is a very weak acid. Consequently, only small concentrations of it can be held by the ion-exchange resins for a given flow through the resins. Therefore, the volume of resin required is much larger than for Cadmium, even though the amount of resin loaded for each startup is of the same order of magnitude.

One advantage that Boron has over any other poisons, except Lithium, is that it does not contribute any additional activity to the moderator system. The neutron is absorbed by an (n,α) reaction and the α-particle is easily absorbed.

Theoretically, any of the neutron absorbers listed in Table I can be used as a liquid poison if it has a water-soluble compound which meets the system requirements. If Lithium is used, because of its (n,α) reaction, it would still be inferior to Boron because of the much greater concentrations required. The only soluble inorganic compound of Hafnium listed is its oxychloride which is too corrosive for use in a poison system.

TABLE III

Property	Natural Boron	Natural Cadmium
σ_a (Thermal)	759 barns	2450 barns
Σ_a/Concentration	42 cm^2/gm	12.9 cm^2/gm
Initial Concentration in Reactor*	10 ppm	24 ppm
Rate of Depletion*	0.05 ppm/day	0.4 ppm/day
Activation	None	800 mr/ppm in moderator equipment room
Compound Used	B_2O_3	$Cd\ SO_4$
Method of Removal	Ion-exchange resins	Ion-exchange resins
Resin Necessary for Startup*	10 cu ft completely loaded. 60 cu ft used.	5 cu ft completely loaded
Mean Life in Core*	23 days	4 days

* Douglas Point reactor values.

There are no radiolytic, stability, corrosion or deposition problem with either material at these concentrations.

The absorption of neutrons in a poison represents a waste of neutrons unless the absorption produces a useful isotope. In a light water moderated system, the introduction of heavy water meets this requirement. The addition of D_2O decreases the slowing down power of the moderator. This results in a larger number of neutrons entering the fuel at resonance energies which, in turn, results in higher conversion of U-238 to Pu-239.

ASSIGNMENT

1. List the advantages and disadvantages of moderator level as a variable reactivity load.

2. (a) List the desirable <u>nuclear</u> properties of neutron absorbers and indicate how some of these conflict.

   (b) How can such a conflict between two of these properties, in particular, be resolved?

3. Briefly compare the suitability of some possible absorber rod materials.

4. What particular disadvantage is avoided by using a fluid absorber instead of a solid absorber?

5. Briefly compare the suitabilites, or otherwise, of moderator dump and safety rod insertion as shutdown mechanisms, explaining why moderator dump is still used as a backup to safety rods in the CANDU system.

6. Explain why booster fuel rods are highly enriched and indicate what problems might be caused by such enrichment.

7. Explain why booster absorber rods are considered uneconomical and indicate how their use can be justified.

8. Why are liquid poisons considered suitable for shim control but not for regulation?

9. Compare the use of Boron and Cadmium as soluble poisons.

A. Williams

Reactor Boiler and Auxiliaries - Course 133

UPGRADING REQUIREMENTS AND PROCESSES

The following terms are in common usage in the heavy water upgrading field:

Heavy Water	-	A mixture of ordinary water (H_2O) and heavy water (D_2O) in which the deuterium isotope predominates or is at least present in considerably greater amounts than found in ordinary water.
Isotopic Purity	-	The weight % of D_2O in a D_2O-H_2O mixture.
Upgrading	-	Any process which increases the fraction of D_2O in a quantity of heavy water. "Upgrading" and "reconcentration" are used interchangeably in the field.
Downgraded Heavy Water	-	Heavy water in which the proportion of D_2O is lower than "reactor grade".
Reactor Grade	-	A term applied to heavy water in which the fraction of D_2O is equal to or greater than an arbitrary value based on an economic study of a given reactor system.

Heavy water which has become downgraded by the addition of H_2O contains highly valuable D_2O. Restoring downgraded heavy water to "reactor grade" (usually $\approx$ 99.8 wt % D_2O) may cost less than the lost fuel burnup that would result if it were re-used in its downgraded condition. In the operation of a CANDU-PHW reactor, high purity D_2O is essential if good neutron economy is to be maintained. The addition of 0.25% H_2O to pure D_2O more than doubles the neutron capture cross section of the material. Consequently, the isotopic purity of the moderator should be kept at "reactor grade". In NPD, for example, a 1% decrease in D_2O isotopic purity would cost an estimated $175,000 annually in loss of fuel burnup.

Downgrading of Heavy Water

While high purity heavy water is most desirable in a heavy water moderated and cooled reactor system, it is very difficult to maintain this purity because of the ease with which heavy water downgrades when it contacts ordinary water. Although heavy water systems are designed to avoid mixing of heavy and light water, heavy water can become downgraded be several means, some of which are:

1. Sampling heavy water for chemical control.
2. Collecting heavy water in an open vessel when draining a piece of equipment, such as a pump, for maintenance work.
3. Opening a piece of equipment, allowing the light water moisture in the air to contact (and downgrade) residual heavy water on the exposed internal surfaces.
4. Spilling heavy water due to equipment failure. Downgrading results due to contact of the heavy water with light water moisture contained in the air or on surfaces.
5. Failure of a heavy water heat exchanger. Either outright loss of heavy water to the light water coolant or downgrading of the heavy water may result.
6. Deuteration and dedeuteration of new and spent ion exchange resins.

The formation of tritium (${}_1H^3$) by neutron capture in D_2O which has been irradiated complicates the problem of handling downgraded D_2O, and may be a factor in determining where the upgrading will be done - either on-site or at a distant upgrading unit. Ideally, on-site upgrading could eliminate the tritium problem by the use of a closed "in-line" upgrading unit. However, several factors must be considered, to determine whether on-site upgrading is more economical than some alternate scheme. Some of these factors are:

1. The cost of shipping and handling the downgraded D_2O.
2. The savings in rem of ${}_1H^3$ uptake by the elimination of handling, if an "in-line" unit is used.
3. The additional D_2O needed for station inventory.
4. The cost of modifications to an existing station to install an on-site unit.

Upgrading Processes

The processes that are used for upgrading heavy water are the same as those used for producing heavy water from natural water. However, the selection of an upgrading process is easier than a production process because the downgraded water normally has a much higher isotopic purity than ordinary water, the starting point for a production process. An upgrading unit is the final stage of a long, tapered production plant, and therefore the economic prospects of a process depend on the capital operating and maintenance costs, hydrogen-deuterium separation ability and the ease with which the isotopic purity of the feed may be changed. All these factors must be carefully weighed in each case.

The separation factor (γ) is the best single measure of the ease or difficulty of a separation, and is the relative volatility of the gas and liquid, as:

$$\gamma = \frac{\left(\frac{y}{1 - y}\right) \text{gas}}{\left(\frac{x}{1 - x}\right) \text{liquid}} = \frac{\left(\frac{\text{atom fraction } H_2}{\text{atom fraction } D_2}\right) \text{gas}}{\left(\frac{\text{atom fraction } H_2}{\text{atom fraction } D_2}\right) \text{liquid}}$$

at any point in the process. In gaseous diffusion separation, it is equal to $(M_1/M_2)^{\frac{1}{2}}$, where M_1 and M_2 are the masses of the isotopes undergoing separation.

The relative volatility (γ) increases with falling pressure and temperature; for constant pressure the size of the plant is proportional to $(\gamma - 1)^2$, and the steam consumption, if using distillation, proportional to $(\gamma - 1)$. It therefore pays to operate at as low a pressure as possible. On the other hand, the volume of steam to be handled increases with falling pressure, so that a compromise has to be made to obtain the optimum operating point.

Several methods have been used for the separation of hydrogen and deuterium, and some of these have been listed in Table I. The processes can be broadly put into three classifications:

(a) high energy,

(b) large volume,

(c) low energy coupled with low volume.

Although the value of γ is of great importance, since it determines plant size (approximately proportional to $1/(\gamma - 1)\ln\gamma$, flow rates, etc), there is no correlation between γ and the economic prospects of a process. The energy consumption, plant size and cost have to be carefully weighed in each case.

TABLE I

Hydrogen-Deuterium Separation Processes

	γ	$1/(\gamma - 1)\ln\gamma$
High Energy Requirement		
gaseous diffusion	1.20	26.3
thermal diffusion	1.05	400
electrolysis of water	10	0.0485
Large Volume Requirement		
water distillation	1.05	400
ammonia distillation	1.03	1110
freezing of water	1.02	2500
Low Volume, Low Energy		
hydrogen distillation	1.4	7.35
chemical exchange		
H_2O/H_2	3	0.454
H_2O/H_2S	2	1.43
NH_3/H_2	6	0.111

Of the chemical exchange processes, the H_2O/H_2 reaction appears to be the most attractive for H and D separation. It offers a relatively high γ, good opportunities for heat recovery and can utilize inexpensive equipment capable of high through-put rates, such as conventional sieve-plate towers for gas-liquid contacting. In comparison with distillation, the plant volume is smaller because γ is much larger. The reaction

$$H_2O + HD \rightleftharpoons HDO + H_2$$

however, requires a catalyst - Pt on colloidal charcoal, and because of the low solubility of H_2 in H_2O, practical exchange rates can only be obtained at high pressure. Efficient homogeneous catalysts for this exchange reaction have not yet been found. Moreover, while the mass transfer aspects of the H_2/H_2O process have not been extensively studied, it is likely that the catalyst problem will introduce complications, and more development work is required before the process could be used for upgrading.

Up to the present time, the GS (Girdler-Spevack) process (H_2O/H_2S exchange) is mostly used for initial concentration to about 10 - 20%. The H_2/NH_3 process also appears promising.

However, for D_2O upgrading, where the feed D_2O isotopic purity is relatively high, the economic aspects and H_2O-removal ability of what is in effect the final stage(s) of a production plant are of prime importance; water distillation or electrolysis generally proves to be best suited to this application, for the following reasons:

1. Distillation and electrolysis are less complicated, and more reliable than, for example, the GS process. This becomes particularly noticeable in the relatively small units used in final enrichment since the cost of final enrichment (capital and operating) accounts for only 2 to 4% of the total cost, if starting with natural water. Costs are no longer proportional to plant size in this application.
2. Processing can be performed under vacuum, preventing losses due to leaks, outleakage of $_1H^3$ and resulting in an improved γ at decreased pressure.
3. The water can be used directly, in the case of distillation, and an easily removeable electrolyte (KOH) can be added for electrolytic upgrading.
4. The required H_2O removal capability can be obtained most economically without an excessively large installation.

ASSIGNMENT

1. Suggest ways by which reactor grade D_2O may become downgraded during the day-to-day operation of a nuclear station.
2. Of the many hydrogen-deuterium separation processes available, which generally prove to be most suitable for D_2O upgrading and why?
3. The presence of tritium complicates the handling of irradiated, downgraded D_2O. Suggest ways in which this hazard may be minimized when filling drums of downgraded water for shipment, and how the hazard could be largely eliminated at a station.

R. Popple

Reactor Boiler and Auxiliaries - Course 133

PRINCIPLES OF ISOTOPE SEPARATION

The smallest element of an isotope-separation unit which effects some separation of the process material is called a separating unit. The simplest type of separating unit or stage is one which receives one feed stream and delivers two product streams, one partially enriched in the desired isotope, the other partially depleted. When the degree of separation by a single stage is less than the degree of separation desired between product and waste, it is necessary to connect several stages in series. Such a series-connected group of stages is called a cascade. An example of a cascade is a complete distillation column.

A cascade in which no attempt is made to reprocess the partially depleted product stream (ie, the tails) leaving each stage is called a simple cascade. In a simple cascade, the feed for one stage is the partially enriched product stream (ie, the heads) from the next lower stage in the cascade. (Note: the terms "enriched" and "depleted" in D_2O upgrading work refer to the D_2O content of the D_2O - H_2O mixture.) When partially depleted stage tails have sufficient value to warrant reprocessing, a countercurrent recycle cascade as shown in Fig. 1 may be used. This cascade flow scheme is by far the most common. Such a countercurrent cascade separates F l/hr of feed of composition x_F into P l/hr of product enriched in the desired isotope (D_2 in this case) of composition x_P, and W l/hr of waste, depleted in the desired isotope of composition x_W. The portion of the cascade between the feed point and product end is called the enriching section; the portion between the feed point

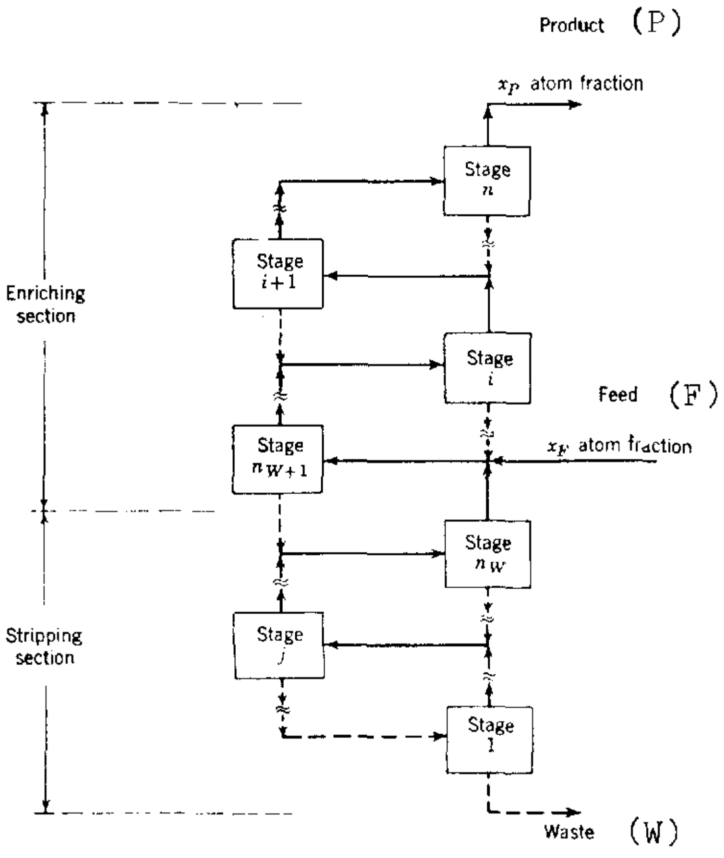


Fig. 1

and waste end is called the stripping section, used to increase the recovery of the desired isotope from the feed. Stages of the cascade are numbered consecutively from 1 at the waste end of the plant to n at the product end. The highest stage of the stripping section is numbered n_W.

General Cascade Equations

In a cascade such as in Fig. 1, feed, product and waste quantities and compositions (the external variables) must satisfy the material-balance equations

$$F = P + W$$

$$Fx_F = Px_P + Wx_W$$

Since there are two equations and six variables, it is possible to specify four external variables independently. For example, these might be product rate and product, feed and waste compositions. In such a case, the other two variables would be given by

$$\text{Feed rate:} \quad F = \frac{P(x_P - x_W)}{x_F - x_W}$$

$$\text{Waste rate:} \quad W = \frac{P(x_P - x_F)}{x_F - x_W}$$

The number of theoretical stages (n) required to effect a specified separation increases as the over-all separation increases and the separation factor (γ) approaches unity. This number of stages (or transfer units) is given by the equation

$$n = \frac{2 \ln \left(\frac{x_P (1 - x_F)}{x_F (1 - x_P)} \right)}{\ln \gamma} - 1$$

twice the theoretical minimum at total reflux.

If G and L are the quantities of gas (vapour) and liquid flowing up and down a column respectively, then the reflux ratio is given by

$$v = \frac{L}{G - L}$$

When the column is operating on total reflux, v approaches ∞, since the amount of vapour flowing up the column is equal to the amount of liquid flowing down. The quantity of liquid flowing down, which at the high reflux ratios usually used, is about equal to the quantity of gas, determines the cross-section of the column, while the number of stages (n) determines its length. The product nv is proportional to the volume of the column and has a minimum value for which the column will usually be designed.

If γ is assumed to be constant for each stage (ie, if the pressure drop through the column is assumed to be zero) and to be equal to the average value in a distillation column, the actual number of plates needed for a given enrichment may be calculated from

$$\text{Enrichment} = \gamma_{AVG}^{\;nE}$$

where E is the average plate efficiency, and the enrichment is the ratio of atom fractions of D_2O in bottoms to tops product.

Examples

1. Calculate the feed and waste rates (in l/hr) which would be obtained at equilibrium using a distillation column capable of processing 90% D_2O feed into 3.9 l/hr of 99.8% D_2O product and a waste composition of 25% D_2O.

Solution: Using the material-balance relations, the feed rate is given by:

$$F = \frac{P\,(x_P - x_W)}{x_F - x_W}$$

$$= \frac{3.9\,(0.998 - 0.25)}{0.90 - 0.25}$$

$$= \underline{\underline{4.5\ \text{l/hr}}}$$

Similarly, the waste rate is given by

$$W = \frac{P\,(x_P - x_F)}{x_F - x_W}$$

$$= \frac{3.9\ (0.998 - 0.90)}{0.90 - 0.25}$$

$$= \underline{\underline{0.6\ \text{l/hr}}}$$

2. Estimate the number of stages required in an electrolytic cascade to upgrade a 50 wt. % D_2O feed to 99.8 wt. % D_2O (Assume: $\gamma = 10$).

<u>Solution</u>: The number of stages required is given by

$$n = \frac{2 \ln \left(\frac{x_P\ (1 - x_F)}{x_F\ (1 - x_P)} \right)}{\ln \gamma} - 1$$

$$= \frac{2 \ln \left(\frac{0.998\ (1 - 0.50)}{0.50\ (1 - 0.998)} \right)}{\ln 10} - 1$$

$$= 4.4$$

$$\approx \underline{\underline{5\ \text{stages}}}$$

NOTE: An integral number of stages must be used. Since 4 stages would be inadequate, 5 stages would be required.

3. Estimate the theoretical equivalent number of stages in the Douglas Point Sulzer packed distillation column, assuming:

$\gamma = 1.05$

$x_F = 0.90$

$x_P = 0.998$

3. Why are more stages than this required in practice?

Solution:

$$n = \frac{2 \ln \left(\dfrac{x_P \ (1 - x_F)}{x_F \ (1 - x_P)} \right) - 1}{\ln \gamma}$$

$$= \frac{2 \ln \left(\dfrac{0.998 \ (1 - 0.90)}{0.90 \ (1 - 0.998)} \right) - 1}{\ln 1.05}$$

$$= \underline{\underline{164 \text{ stages}}}$$

The above calculates the equivalent theoretical number of stages, assuming each stage to be 100% efficient. In practice, stage efficiencies are less than 100% efficient, and in the above example, about 280 equivalent stages would be required, assuming typically 60% stage efficiency.

ASSIGNMENT

1. State what is meant by a stage, a simple cascade and a recycle cascade as applied to isotope separation.
2. Write down the general material-balance equations, and indicate the meaning of each symbol used.
3. What is the reflux ratio and what is meant by total reflux?
4. Estimate the number of plates required in a distillation column to effect an H_2O - D_2O separation into 1% D_2O tops and 95% D_2O bottoms, if the plates are 50% efficient and the average stage separation factor is 1.05.

R. Popple

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THE DISTILLATION UPGRADING PROCESS

Water distillation for D_2O upgrading is based on the difference in boiling points of D_2O and H_2O. Generally, the magnitude of this temperature difference is a measure of the difficulty of separating the two components, and in this case the difference (1.4°C) is small so that the difficulty of separation is appreciable.

Mechanism of Operation

A D_2O-H_2O mixture contains D_2O of boiling point 101.4°C and H_2O of boiling point 100°C at atmospheric pressure. If sufficient heat is added to such a mixture, the H_2O-component will boil first, and any vapor leaving the mixture will contain a slightly increased proportion of H_2O. The mixture left behind will then contain a slightly increased proportion of D_2O. In this case, the mixture left behind will ultimately become "reactor grade" D_2O (and will remain at the bottom of the column) and the mixture leaving as vapor will ultimately become waste product (and will be drawn off at the top of the column).

Several stages connected in series and assembled in a vertical cylinder form a cascade which is known as a distillation column. Heating of the D_2O-H_2O mixture takes place at the bottom of the column in a reboiler. Vapor rises through the column, is condensed at the top of the column, and descends through the column as liquid while other vapor rises and comes in contact with it. The descending liquid, already slightly depleted in D_2O, tends to pick up higher boiling point component (D_2O) from the vapour, while at the same time giving up lower boiling point component (H_2O) to the rising vapor. The internals of the column are designed to promote thorough and efficient mixing of the liquid and vapor phases.

The fluid obtained by condensing the overhead vapor (called distillate) is depleted in D_2O content from that of the original mixture. This distillate may be either partially or wholly drawn off the top of the column or it may be allowed to descend the column by gravity, to contact the vapor rising up the column. The portion descending is called the reflux (Fig. 1). When all the distillate is allowed to descend, the column is operating on total reflux.

The distillate produced during D_2O upgrading using a distillation column contains proportionately more light water than the original D_2O-H_2O mixture. Unless the D_2O content is 2% by weight or less, it is uneconomical to discard it; rather, it is cheaper to store the distillate and upgrade it at some later time. During operation of the column, the portion of the distillate not recycled as reflux is "tapped" or "drawn" off, and although of lower grade D_2O than the column feed, must be saved for later reprocessing.

Since γ increases as the pressure at which distillation is done is reduced, a vacuum distillation unit is used for upgrading heavy water. The vacuum, usually about 0.25 atmos, is limited due to the large vapor volumes that would result at pressures much below this point.

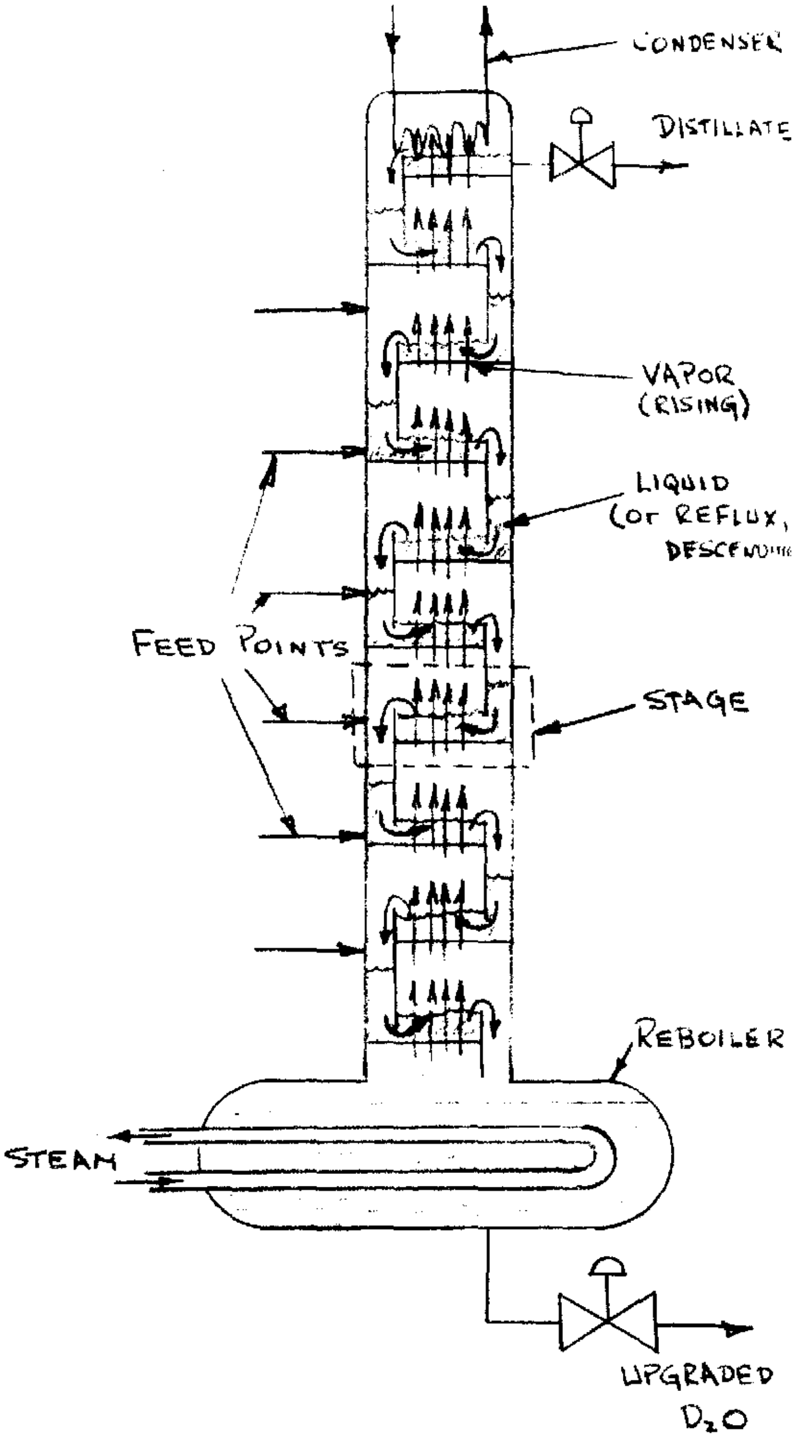


Fig. 1

Types of Columns Used

The columns presently in use for D_2O upgrading are of two types, classified according to the internals of the column:

1. Sieve-plate columns.
2. Packed columns.

Sieve-plate Columns

The internals of a sieve-plate column consist of a series of horizontal perforated plates through which rising vapor can pass, together with a weir arrangement to allow the liquid to descend in the column (Fig. 2). Repeated contacting of liquid and vapor is thus promoted as the vapor rises and the liquid descends in the column. The vapor leaving a given plate is slightly more enriched in H_2O; the liquid, slightly more enriched in D_2O. The column at NRU is a sieve-plate type, designed and built by Canadian Badger Ltd.

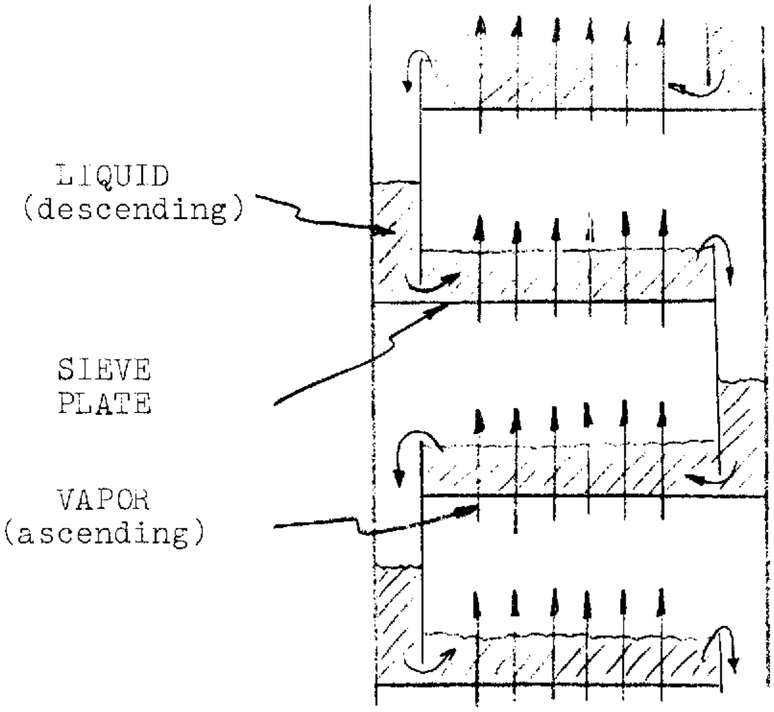


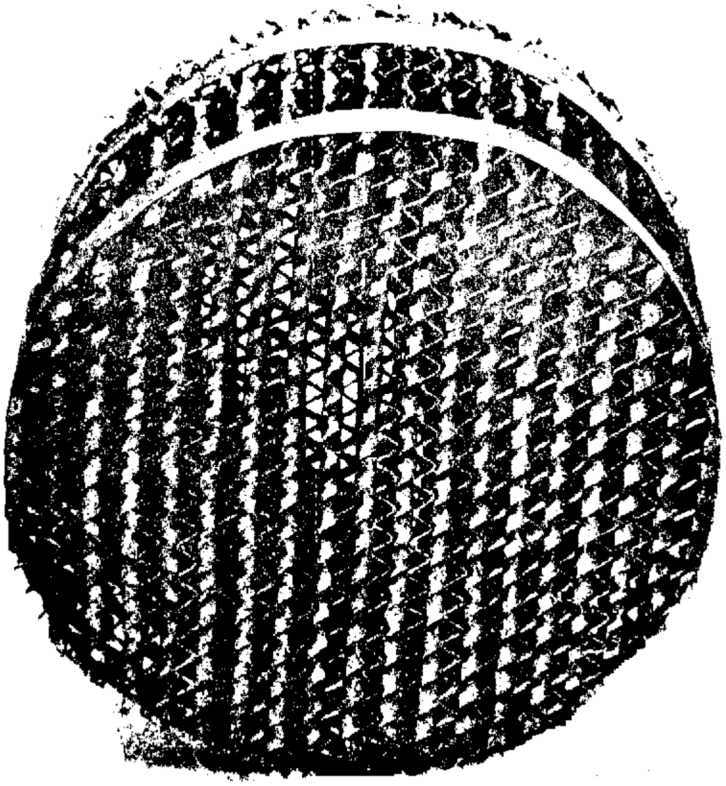
Fig. 2

Generally speaking, it is now usually possible to obtain more efficient vapor-liquid mixing with a packed column, resulting in increased separating ability for a given size of column. Sieve-plate columns on the other hand have a comparatively low first cost. However, to accommodate the upgrading needs of present-day nuclear installations, a sieve-plate column becomes impractical due to the large number of stages and hence size of column required.

Packed Columns

The internals of a packed column are not arranged in a series-connected cascade of stages like those of a sieve-plate column; rather the internals consist of a packing, usually in the form of a metallic gauze throughout the length of the column. An equivalent number of stages is then calculated, based on the separating ability of the column. Fig. 3 shows the type of gauze pellet used in the Douglas Point packed column. Provision must be made along the column for the addition of feed, and uniform spreading of the feed across the column at the particular elevation(s) chosen.

Depending on the particular application, the equivalent of between 150 and 500 stages are needed for an H_2O-D_2O separation. By comparison, columns employed in chemical plants seldom contain



Sulzer Packing

This "pellet" is a few centimetres in diameter. These gauze "pellets" are packed into thin tubes, which run the length of the column, with provision for the addition of feed. The tubes are assembled into a cluster of about the same inside diameter as the column. Each Douglas Point column is about 25 centimetres in diameter.

Fig. 3

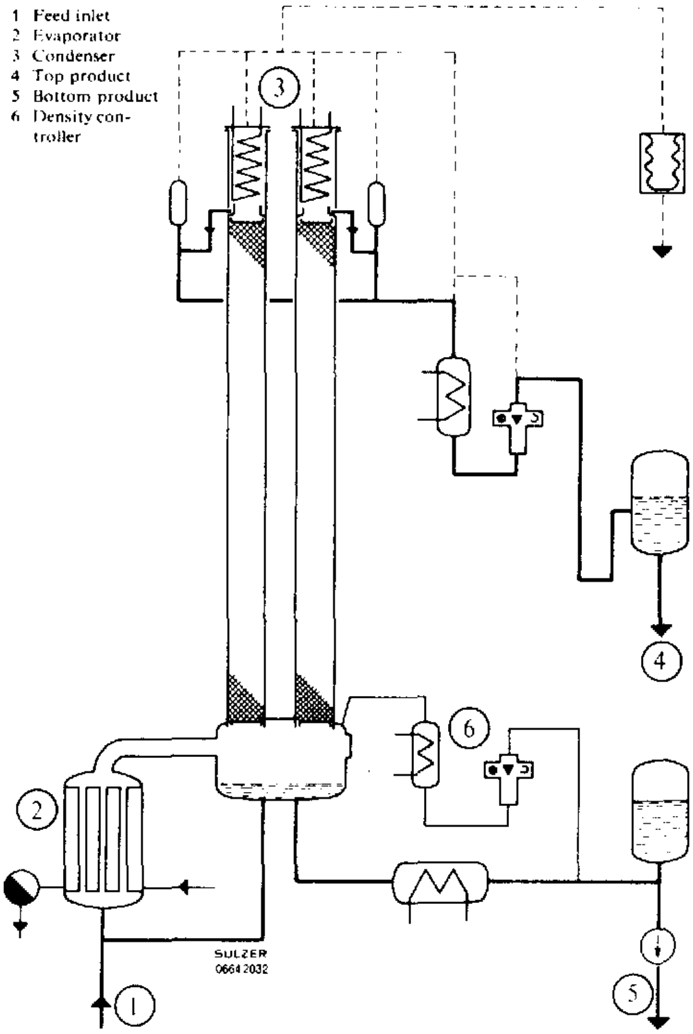
more than 100 stages. Working at reduced pressures proves advantageous as with the sieve-plate column, since the single stage α increases with falling pressure.

The following comparison serves to emphasize further advantages of the packed column:

	Packed Column	Plate Column
Stages/m	10 - 60	1 - 10
Pressure loss (mm Hg/stage)	0.2 - 0.5	2 - 10
Hold-up (ℓ/m^2 of stage)	3 - 8	10 - 50
Starting-up time (days)	3 - 4	7 - 8

In a packed column, the gauze "pellets" are arranged in long thin-walled tubes as described under Fig. 3. An important feature of most packed columns is that they usually consist of two or more towers, connected in parallel to the reboiler <u>at the bottom only</u>; each tower has its own condenser at the top. The reason for doing this is to prevent the individual towers from "competing" with one another, and can best be appreciated by considering the column during operation on total reflux (ie, when no product is being withdrawn). In this instance, the quantity of vapor ascending through any cross section is equal to the quantity of liquid descending. This applies to towers in parallel at the bottom only, even though the quantities may differ slightly from one tube to another, since phase reversal takes place at the top of the tube. However, if the columns are connected in parallel at the top also, the separation is quite adversely affected, since a net flow up one tower and down another can easily result, reducing the reflux ratio. Optimun operation for an H_2O-D_2O separation will result at very high reflux ratios.

A simplified flow diagram of a typical upgrading unit showing the main components which would be used on-site at a nuclear station is shown in Fig. 4. Most of the auxiliary equipment is labelled in the diagram; note the use of separate condensers for each tower, connected in parallel at the reboiler end only. Packing in the towers is indicated by double cross-hatching. Top and bottom product purities are measured by a density controller, operating on the principle of density difference between H_2O and D_2O.



<u>Fig. 4</u>

H_2O-D_2O distillation columns are now being built having a large diameter with as many as 40 - 50 towers connected to a single reboiler, each with its own condenser; the over-all diameter of the towers assembled into a single column may measure as much as 2 - 3 metres. The design and separating ability of a given installation will depend on the specified duties of the unit.

Method of Operation

There are two ways in which a heavy water upgrading distillation unit may be used. While a particular column may be designed primarily for one mode of operation it is desirable, for upgrading work, that it be flexible enough to work in both ways. The first mode of operation is called "stripping". This technique removes or strips light water from the incoming feed without appreciably raising the D_2O concentration of the feed. In this type of operation the column vapor flow or boiling rate is essentially the same as the feed rate. The second method of operation is called "enriching". In this case downgraded heavy water is fed to the distillation column at a much lower rate than the column vapor flow or boil-up rate. The heavy water flows downward with the column reflux while the light water is carried off in the vapor rising up the column. The heavy water thus becomes enriched or upgraded as it flows to the bottom of the column.

In both cases the high grade product is removed from the bottom of the column while the light water is removed in a portion of the condensed overhead vapor or distillate. In the stripping operation the product is reactor grade heavy water while in the enriching operation the product may either by reactor or may be a lower grade depending on D_2O content of the feed and on the characteristics of the column. In both cases, of course, the product flow rate is the feed flow rate less the quantity of water that is collected as distillate.

The distillation column must operate continuously in order to achieve the best possible performance. The best separation of components, in this case light and heavy water, is only obtained if the column operates "at equilibrium". This is a steady state condition where the values of D_2O concentrations, from the bottom of the column through the many stages to the overhead vapor, remain steady. A period of 2 to 8 days is required for a heavy water distillation column to reach equilibrium conditions after it has been started up. For the start-up period, the column is operated on total reflux; once equilibrium has been achieved, feed to the column is started. Most packed columns require "flooding" before operation at total reflux commences, to allow proper wetting of the gauze.

In order to increase the flexibility of the distillation column for upgrading work, facilities can be provided to permit the column to work on what may be called a "semi-batch" operation. This uses fairly large batches of feed of a wide range of D_2O concentrations and requires the column to come to a new equilibrium condition for each batch, although the operation is continuous while working on a batch. The column may be operated either as in stripping or as in enriching. The increased flexibility is, of course, obtained at the expense of some capacity since time is required to bring the column to equilibrium for each batch of feed.

ASSIGNMENT

1. Explain how an H_2O-D_2O separation takes place in a distillation column. Is it an easy separation? Why?

2. What is meant by total reflux and when is this mode of operation used in practice?

3. How are distillation columns classified and what types are currently in use for D_2O upgrading? Which is generally better suited to D_2O upgrading and why?

4. How does an "enriching" operation differ from a "stripping" operation?

R. Popple

Reactor Boiler and Auxiliaries - Course 133

THE ELECTROLYTIC PROCESS

When water is decomposed electrolytically into hydrogen and oxygen, the deuterium content of the hydrogen produced at the cathode is substantially lower than that of the water remaining in the cell. As electrolysis continues, the water becomes progressively enriched in deuterium. The gas produced in the latter stages of electrolysis is richer in deuterium than the water initially fed; this gas may be burned and the resulting "combustion water" condensed and returned to the cell. The high separation factor ($\gamma = 10$) attainable and the resulting small size of the plant are the main advantages of the electrolysis process when applied to D_2O upgrading.

Mechanism of Operation

Since the equilibrium constant for the reaction

$$H_2O + HD \rightleftharpoons HDO + H_2$$

is 3.88 at 25°C and 2.73 at 100°C, it is apparent that the separation factor of 10 observed in electrolysis must be due to some mechanism other than the establishment of equilibrium in this reaction at the cathode surface. One plausible explanation is that the hydrogen ion is discharged more readily at the cathode than the deuterium ion, thus accounting for the tendency of hydrogen to concentrate in the gas. Laboratory findings give some support to this theory. Another possible explanation is the decreased relative mobility of the D+ and HD+ ions in the liquid compared to the H+ ions, so that H+ ions reach the cathode more quickly than the D+ or HD+ ions.

In electrolysis process as applied to heavy water upgrading, a chemical (KOH) is added to the downgraded heavy water to form a current conducting electrolyte, through which a low voltage, high dc current (2000 amps or higher) is passed in an electrolytic cell. As the electrolysis proceeds, ordinary water in the H_2O - D_2O mixture is decomposed more readily than heavy water. The electrolyte remaining behind becomes richer in deuterium content as the process proceeds, until the electrolyte has the desired D_2O concentration. The process is then stopped and the chemical (KOH) extracted to leave reactor grade heavy water.

During the electrolysis, the electrolyte tends to heat up. If the temperature of the electrolyte is not controlled, valuable

D_2O vapour would be carried away from the cell, and thus adversely affect the separation factor.

Description of Electrolytic Cell

The present D_2O electrolytic reconcentration facilities in Canada operate using discarded cells previously used by Consolidated Mining and Smelting, Trail, B.C. in the final stage of their heavy water production plant. Each cell consists of a water-jacketed cylindrical iron body which serves as the cathode (Fig. 1). The anode in each cell is a heavily nickel-plated iron sleeve, insulated from the cover of the cathode, through which it passes, by a gas-tight rubber insulator. The anode is held in a position clear of the cathode by hard rubber studs placed symmetrically around its outer and inner surfaces near the bottom. An inverted iron cylindrical dome is raised from the bottom of the cathode forming an inner cathode which is water-cooled. This reduces the ineffective cell volume and provides a more constant cathode surface area (ie, current density). The piping to each cell is insulated electrically by the installation of sections of rubber hose.

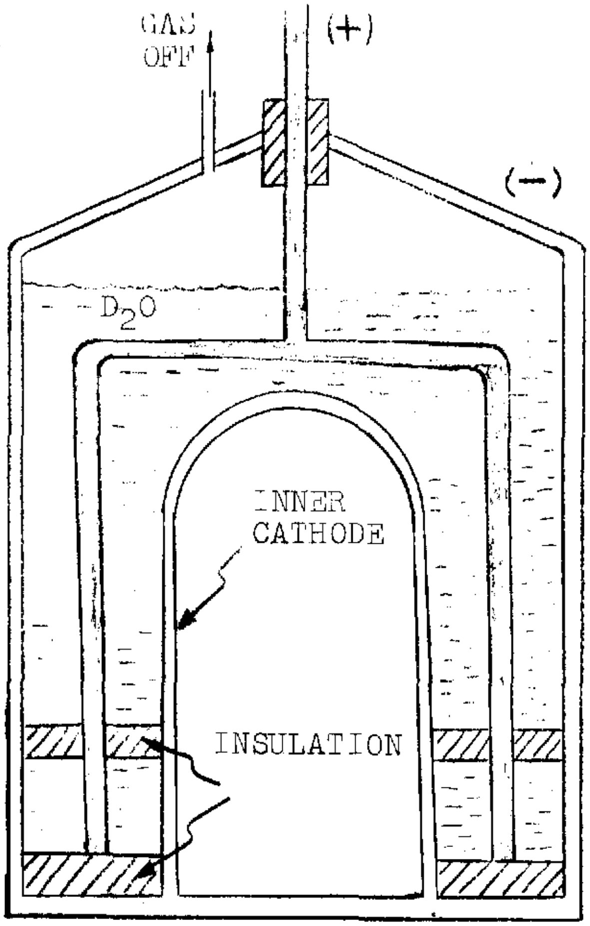


Fig. 1

Method of Operation

Electrolytic reconcentration appears to be best suited to the batch method of operation, although "semi-batch" operation appears feasible. Using this method, a given quantity of downgraded D_2O is upgraded to reactor grade, with possible further feed additions at higher isotopics only when the cell isotopic has reached the new feed isotopic. Fig. 2 shows a simplified flow scheme for a typical upgrading unit. If the unit is connected on-line to the reactor system, cleaning of the feed would be done using the usual

FLOW SHEET - ELECTROLYTIC UPGRADING PLANT

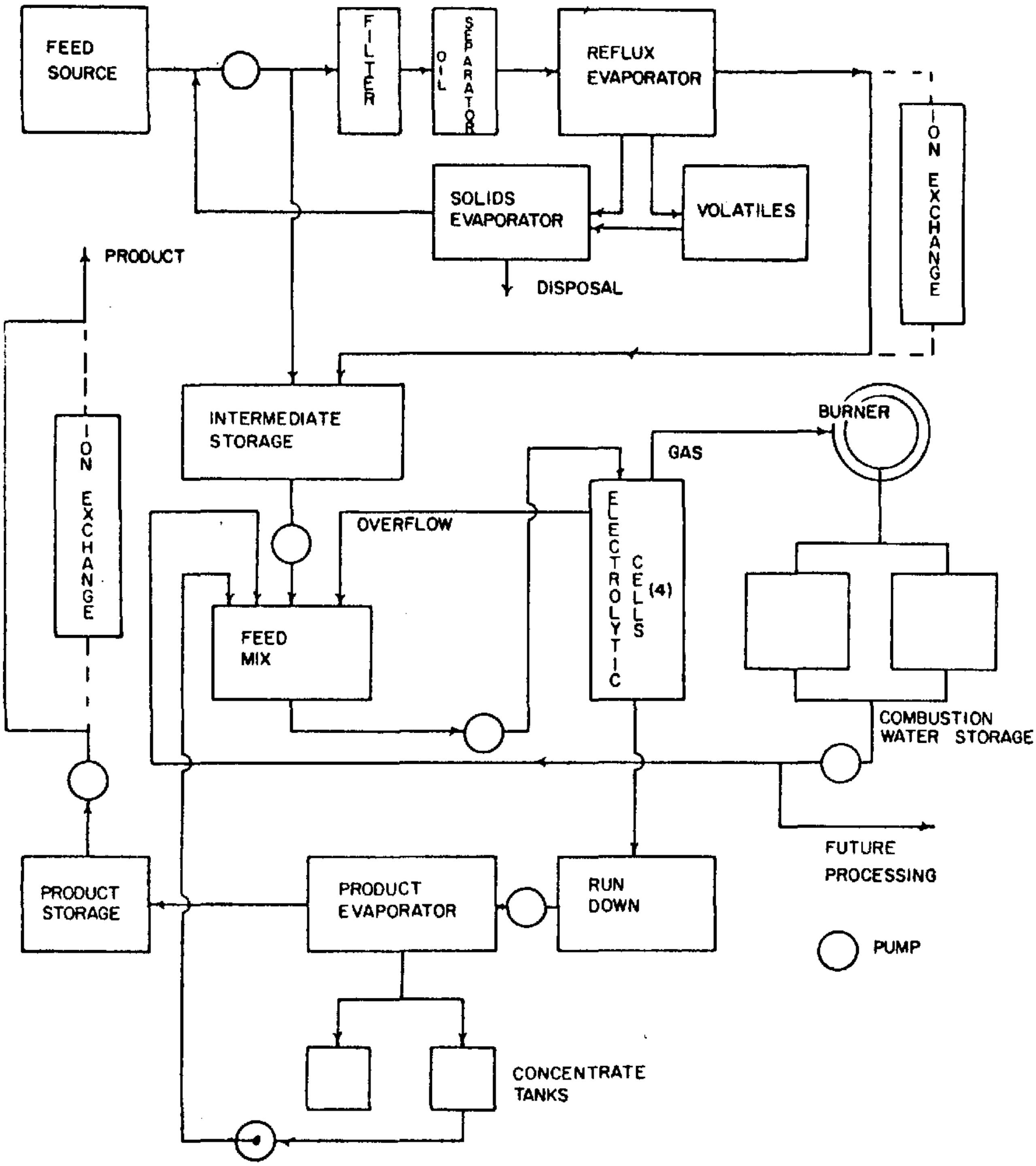


Fig. 2

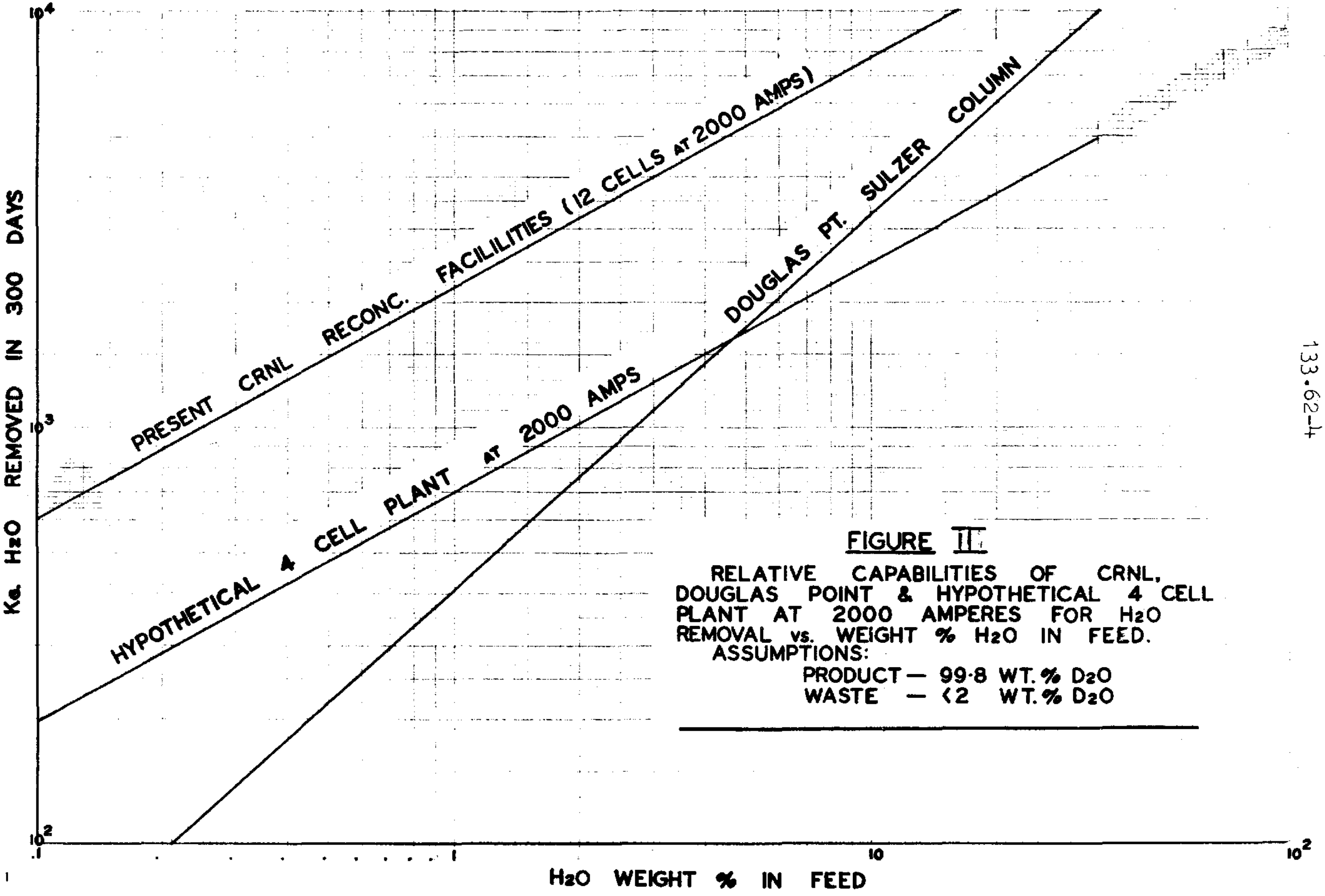


FIGURE II

RELATIVE CAPABILITIES OF CRNL, DOUGLAS POINT & HYPOTHETICAL 4 CELL PLANT AT 2000 AMPERES FOR H_2O REMOVAL vs. WEIGHT % H_2O IN FEED.

ASSUMPTIONS:

PRODUCT — 99·8 WT. % D_2O

WASTE — <2 WT. % D_2O

133.62-4

and CRNL 12-cell unit curves have the same slope, since they both operate at 2000 amps; the H_2O removal capacity of one is three times the other, since it has three times as many cells. Note also that the H_2O removal capacity of the 4-cell plant is greater than that of the Douglas Point column at isotopics above 95 wt. % D_2O (ie, 5% H_2O on graph); alternatively, at lower isotopics, the Douglas Point column has the edge. Thus, the choice of a facility will depend to a large extent on the feed isotopic content either based on past experience, predictions or both.

The electrical consumption and hence power costs are easily estimated for an electrolytic plant. For a 4-cell plant, operating at 2000 amps, utilizing 4 series-connected cells at 2.5 volts per cell and 85% efficiency of rectification, the required power would be

$$P = \frac{4 \times 2000 \times 2.5}{0.85}$$

$$= \underline{\underline{23.5 \text{ kw}}}$$

The operating costs are usually, however, about 90% labour, for an electrolytic reconcentration plant.

ASSIGNMENT

1. What are the main advantages and disadvantages of the electrolytic method of reconcentrating downgraded D_2O?
2. What are the main pieces of equipment required to utilize the electrolytic method for on-site D_2O upgrading?
3. On what quantity is the production of 'reactor grade' D_2O by electroylsis dependent?
4. How does the H_2O removal ability of a 4-cell electrolytic plant operating at 2000 amps compare with that of the Douglas Point columns? How does it compare if 6 cells are used at the same amperage?

R. Popple

Reactor Boiler and Auxiliaries - Course 133

COMMON PROBLEMS

Heavy water upgrading is the end stage of the process of heavy water production. There are some auxiliary processing steps which, if carried out, avoid some difficulties in upgrading. There are also some problems associated with the handling of heavy water, which must be considered no matter what upgrading process is used.

Heavy Water Recovery

Heavy water recovery starts with the loss, whether deliberate as in sampling or draining, or inadvertent, as in spills due to equipment failure, of reactor grade heavy water from a reactor system. The system may be a moderator, a heat transport or a purification circuit. Due to the cost of heavy water, an attempt is always made to recover water which has escaped from a reactor system so that it can be returned to the system.

To assist in achieving this, special facilities are provided for heavy water reactor units. First, there is a "closed" collection system with gathering pipes to pick up heavy water leakage from such obvious points as valve stems and pump glands. Secondly, if possible, the plant areas housing equipment containing heavy water are ventilated by a closed circuit system with condensing and drying facilities to permit recovery of heavy water carried in the air stream. Thirdly, the drainage from areas containing heavy water is collected in a sump with facilities provided to pump water from the sump into drums or other containers in the event that it turns out to be heavy water. In all cases the intent is to recover the heavy water with a minimum of downgrading. Inevitably, some of the heavy water is downgraded and, of course, some heavy water is normally deliberately downgraded during the processes of deuterizing and dedeuterizing ion exchange resins used for heavy water purification. As a result, the final stage of the recovery of heavy water is upgrading.

The upgrading unit, no matter what process is used, is itself subject to loss of heavy water through leakage, sampling, etc. Although it may not be possible to provide recovery systems for the upgrading unit, its design, construction and operation must always be such as to minimize heavy water loss. This means that, in a unit where the deuterium is extracted from the heavy water at some intermediate stage in the process, special precautions are also necessary to avoid loss of the deuterium. It is also important that the design and operation of the upgrading unit be such as to minimize the possibility of further downgrading by inadvertent contacting of the heavy water being processed with light water.

Inventory Control

Heavy water is an expensive material and is a prescribed substance under the Atomic Energy Control Act, which means that an organization requires an Atomic Energy Control Board license to possess large amounts of reactor grade heavy water. It is therefore necessary to account for losses and some form of inventory control is required. Normally it is preferable to record the inventory in terms of 100% D_2O. It is necessary to be able to measure the weight and isotopic purity or D_2O content of the water being processed, the water held up in the unit and the water removed from the process either as final product or as low grade waste.

Where batch processing is done, either weigh scales or previously calibrated tanks are used to determine the amount of D_2O being put in, collected from the process and remaining in the unit at the end of the batch run.

If continuous processing is done, in-line measurements may be possible so that a running inventory can be kept. Such a unit would also require facilities for draining the unit to permit a check on the contents by means of either a calibrated holding tank or weigh scales.

Finally where heavy water is transported in 45 gallon drums the normal procedure is to place a drum on a platform scale for either emptying or filling. For proper inventory control it is necessary that a consistant and careful weighing procedure be used. For example when emptying drums the normal emptying device may leave 2 or 3 lbs of heavy water, called a heel, in the drum. A variable procedure of removing the heel, sometimes, and leaving it, at other times, may introduce an error unless a very careful handling procedure is followed.

Freezing Problem

Heavy water freezes at 39°F. Like ordinary water, heavy water expands on freezing with the same danger of rupturing its container. Not only can this disable the upgrading unit but it can lead to direct loss of valuable heavy water when the ice melts or to indirect loss by downgrading if the ruptured container happens to be a light water cooled heat exchanger tube.

In either the distillation or electrolytic method of heavy water upgrading, there are temperature control requirements, resulting in heavy water being cooled with some form of cooling water usually obtained from a nearby river or lake. During the winter and early spring months, it is quite probable that natural surface water temperatures will be as low as 33°F to 36°F. It is therefore necessary to have sufficient control of cooling water flows so that heat exchangers and condensers do not freeze. Although the probability of freezing is perhaps greater during periods of reduced cooling requirements such as during startup or shut down of the upgrading unit, the danger of freezing during a period when the unit is shut down because a temperature control valve does not completely shut off the flow of cooling water can not be ignored. Considering the value of heavy water and the associated radioactivity hazard, it is not acceptable to protect equipment from damaging overpressures, due to heavy water freezing by relieving the pressure to atmosphere.

The freezing problem must be considered when transporting heavy water. If the upgrading unit is located remotely from the heavy water reactor being served, the heavy water must be transported in heated equipment during winter months. Also it is necessary to provide suitable heated storage facilities, for the incoming downgraded heavy water and outgoing reactor grade product, at the upgrading plant.

Radioactivity Problem

Heavy water that has been in or passed through the core of a nuclear reactor contains some radioactive material. There is tritium, a low energy beta emitting isotope of hydrogen, which is produced when a deuterium atom captures a neutron. In addition, there are radioactive corrosion products from the reactor's heavy water system and, if the reactor heat transport fluid is heavy water, there may be radioactive fission products due to fuel sheath imperfections. The radioactive corrosion and fission products can be removed by preliminary purification before upgrading to a considerable extent. However, these materials are often emitters of penetrating gamma rays and so heavy water containing them can be a serious radiological hazard for personnel working nearby. Also these materials, being normally non-volatile, tend to be concentrated with the reactor grade final product. Even though partially removed before upgrading, the concentration may become high enough to require shielding for the portions of the equipment containing the concentrated product.

Tritium is a serious radiological problem only if taken into the body. The low energy of the emitted beta radiation is such that tritium is not an external radiation hazard, but is hard to detect in the presence of other radioactive materials which might be present. Tritium, because it is a hydrogen isotope, occurs in heavy water as a form of "extra heavy" water and cannot be readily separated except by a process similar to the upgrading processes. This means that in the heavy water upgrading unit, tritium concentrates in the final product. Unfortunately tritium in the "extra heavy" water molecule is readily taken into the body either by breathing air containing the vapour or by direct contact with and absorption through the skin. Once absorbed, the tritium containing water readily exchanges with water in the body tissues and brings the tritium into intimate contact with the tissues so that the low energy beta radiation is quite capable of causing damage. To protect personnel against the tritium hazard, the following points should be considered:

1. Minimize loss of tritiated heavy water from containers and equipment by leakage and evaporation.
2. Provide adequate ventilation for the area to protect against the necessary minimum losses.
3. Provide protective clothing and clean, fresh, breathing air for people who may become exposed to tritiated heavy water during operation and maintenance of the equipment.
4. Provide washing and shower facilities so that a person, who has contacted tritium bearing heavy water, can promptly wash thoroughly to minimize the absorption.

Purification Requirements

The purification of heavy water is the removal of impurities other than light water. These consist of dirt, oil, etc., appearing in spilled heavy water after flowing on or over machinery, floors, etc., or which may be present in an unclean container used to collect the heavy water. Lubricating oil and concrete dust are two examples of such contaminants. Radioactive materials other than tritium, are another source of impurity. Since these impurities tend to concentrate in the product of the upgrading process and impair the efficiency of the process, it is preferable to remove them by a preliminary purification step.

Standard cleaning techniques are used including filtration, evaporation, centrifugal separation and ion exchange purification. Each of these steps increases to some extent the loss of the heavy water which is being recovered. It is difficult to recover all the heavy water contained in the filter medium carrying the impurities. Centrifugal separation may not give an acceptable purification without a small portion of the heavy water being discarded with the impurities. Ion exchange purification requires use of deuterized resin to avoid further downgrading of the water and so the spent resin containing impurities must be dedeuterized to recover as much of the contained heavy water as possible. In addition, the extra handling itself tends to increase losses through evaporation or spillage. Where radioactive impurities are present, all of these processes will require suitable shielding. Since tritiated heavy water is involved, facilities must be available to protect personnel from this hazard also.

R. Popple (R-0)

Reactor Boiler and Auxiliaries - Course 133

GENERAL CONSIDERATIONS

The subject of nuclear fuels is an extremely broad one, since the forms in which nuclear fuels can be produced are many and varied. This lesson will deal mainly with the effects of irradiation on metal fuel, U, and the ceramic fuel, UO_2, from a general viewpoint. No attempt will be made here to fully explain the phenomena described.

Metallic Uranium

The effects of irradiation of metallic uranium on its mechanical properties are a generally marked decrease in tensile strength even after short exposure, changes in hardness (cold-worked uranium irradiated to low exposure softens, whereas annealed uranium hardens), decreased impact strength of cast, heat-treated or hot-rolled uranium by a factor of 3 or 4. Embrittlement of the metal does occur, so that any bend tests done on slender specimens have been found to fail by brittle fracture. The changes in properties can be generally said to depend directly on the integrated exposure to neutrons, and the neutron flux level.

The physical properties of metallic uranium are markedly affected by irradiation. Limited data from in-reactor measurements indicate a decrease in thermal conductivity of 10 to 15%. This property is an important one because decreases in thermal conductivity on irradiation might result in an increase of uranium core temperature to an unsafe level. Since the predicted accuracy of the methods used to obtain these data is probably ± 20% at best, the validity of the results in an absolute sense is dubious. Qualitatively, however, the results are quite valuable, in that they indicate there is no gross change in thermal conductivity of uranium after irradiation. An apparent increase in the modulus of elasticity of about 15% has, however, been reported.

The most significant effects of exposure to neutron flux on natural uranium metal are the pronounced loss in ductility, the disintegration of the metal at nominal burn-ups, the increases in volume and the growth of preferably oriented specimens. The changes in tensile properties are much too great to be attributed to vacancy-interstitial or thermal spike effects. The decrease in room temperature ductility may be attributed to the presence of fission products. It is quite probable that the noble gases xenon and krypton have a pronounced effect on the mechanical properties, and they are believed to be primarily responsible for the disintegration of certain types of uranium after relatively short burn-ups. Unalloyed enriched uranium disintegrates after low burn-up and resembles coke, suggesting that the formation and expansion of gas bubbles may play a part in the disintegration.

The presence of fission products leads to certain highly undesirable effects both in the fuel and, if they escape, in the coolant. At the instant of fission, the two fission products smash through the uranium lattice, dislodging uranium atoms, generating vacancies and interstitials, finally lodging as foreign atoms in the lattice. At sufficiently high temperatures, the gaseous fission products (Xe, Kr) diffuse and agglomerate, causing voids, gross local decreases in uranium density and even pronounced swelling. Swelling and decreases in density are often accompanied by cracking and eventual disintegration of the uranium; owing to the radioactivity of fission products, gross contamination of the reactor and/or reactor site may occur if they are released into the coolant by a fuel element failure or penetration of the sheathing.

Uranium metal offers a maximum conversion ratio, ie, a greater yield of plutonium per gram of uranium. However, the sheath of a solid fuel element must be capable of retaining all the fission products even though the fuel is distorted by irradiation and even though the properties of the sheathing material change. Metallic uranium has an anisotropic crystal structure. The effect on it of many fissions is to produce often dramatic changes in the shape of the fuel, which could lead to blockage of coolant flow in a channel. While this problem can be overcome by suitable metallurgical treatment, and does not exist at all in isotropic uranium dioxide, the problem of the accumulation of gaseous fission products remains.

Ceramic Fuel

The ceramic fuel in most common use today is uranium dioxide (UO_2). UO_2 has good corrosion resistance at high temperatures; this has been demonstrated in NPD where fuel elements have operated with defected sheathing for periods of over 2 years and post-irradiation examination of the UO_2 showed little or no difference between defected and undefected elements. Its coefficient of thermal expansion is low and it is dimensionally stable under long irradiation. It is capable of withstanding high temperature operation (melting point: 2800°C) and many thermal cycles.

UO_2 has, however, a low thermal conductivity, which limits the power rating that can be attained without melting. Power reactor operation in Canada to date has however been well below the UO_2 melting point. Fission product gases are largely retained by the UO_2 at lower temperatures but can leak out at high temperatures. Extensive irradiation tests to burn-ups as high as 100,000 Mwd/te U showed that the UO_2 swelled by no more than 8% of the volume when at temperatures below about 700°C, and restrained by 80 to 170 atm. While many unanswered questions still exist, recent studies on the irradiation behaviour of UO_2 conclude that the average fission product atom does not travel far before being trapped in the lattice of the material. The fission product gases can agglomerate and form observable bubbles; the ceramic UO_2 cracks under thermal stresses and must be contained and supported by a sheath.

The build-up of pressure inside the sheathing due to release of fission products from the UO_2 does not appear to be a problem with the Canadian design. Although in-reactor measurements are extremely difficult, the pressure within a fuel element at a burnup of $\approx$ 10,000 Mwd/te operating at maximum Pickering heat rating is predicted to be $\approx$ 80 atm. While operating, this is balanced by the pressure in the coolant system and when removed or shut down, should be decreased by a factor of about 10 due to the decreased temperature of the UO_2, well below the burst strength of the tube. Less than 20% of the gases produced are released from the UO_2.

The thermal conductivity of UO_2 changes very little on irradiation; the factors affecting the conductivity of the UO_2 are the temperature, porosity, excess of oxygen and the impurities present.

The Meaning of $\int \lambda d\theta$

In conventional laboratory practice, one is accustomed to measuring the temperature distribution in a system and thus having knowledge of its condition. Alternatively, some other quantities, eg, heat generation, may be measured, and the temperatures deduced from established physical properties of the materials. However, the measurement of temperatures in operating fuel elements is extremely difficult, largely because of the high temperature gradients existing in them. Consequently, many of the physical properties of the constituent materials have not been determined under operating conditions. Specifically, the thermal conductivities of many fuels under irradiation have not been established. Some parameter which can be measured experimentally is required to define the condition of the fuel, in a manner comparable to temperature, and this parameter is

$\int_{T_s}^{T_o} \lambda \, d\theta$ where λ is the thermal conductivity of the fuel material, θ is temperature, T_s is the surface temperature, and T_o the central temperature. Usually, the symbol is simply written as

$$\int \lambda \, d\theta$$

The use of a definite integral as a single parameter, without reference to its components, will be familiar to those who have used "entropy". Although the parameter is most widely used in connection with UO_2 cylinders, the arguments leading to its association with measurable quantities are general and not confined to a single fuel. If the central fuel temperature were known, it would be a useful criterion for comparing the severity of different irradiations. In ignorance of it, $\int \lambda \, d\theta$, obtained from the equation

$$\int \lambda \, d\theta = \frac{1}{4\pi} q F$$

can be used for this purpose. In this equation, q is the heat generation per unit length of the pellet (W/cm) and F is a factor which accounts for the flux depression within the pellet. The flux depression in the Douglas Point or Pickering outer elements can be neglected, and F taken as unity. For larger diameter or enriched pellets, F must be calculated, and has a value less than unity.

As an example, the $\int \lambda \, d\theta$ for an average Douglas Point outer element, producing 500 W/cm would be given by

$$\int \lambda \, d\theta = \frac{1}{4\pi} (500)(1)$$

$$= 40 \text{ W/cm}$$

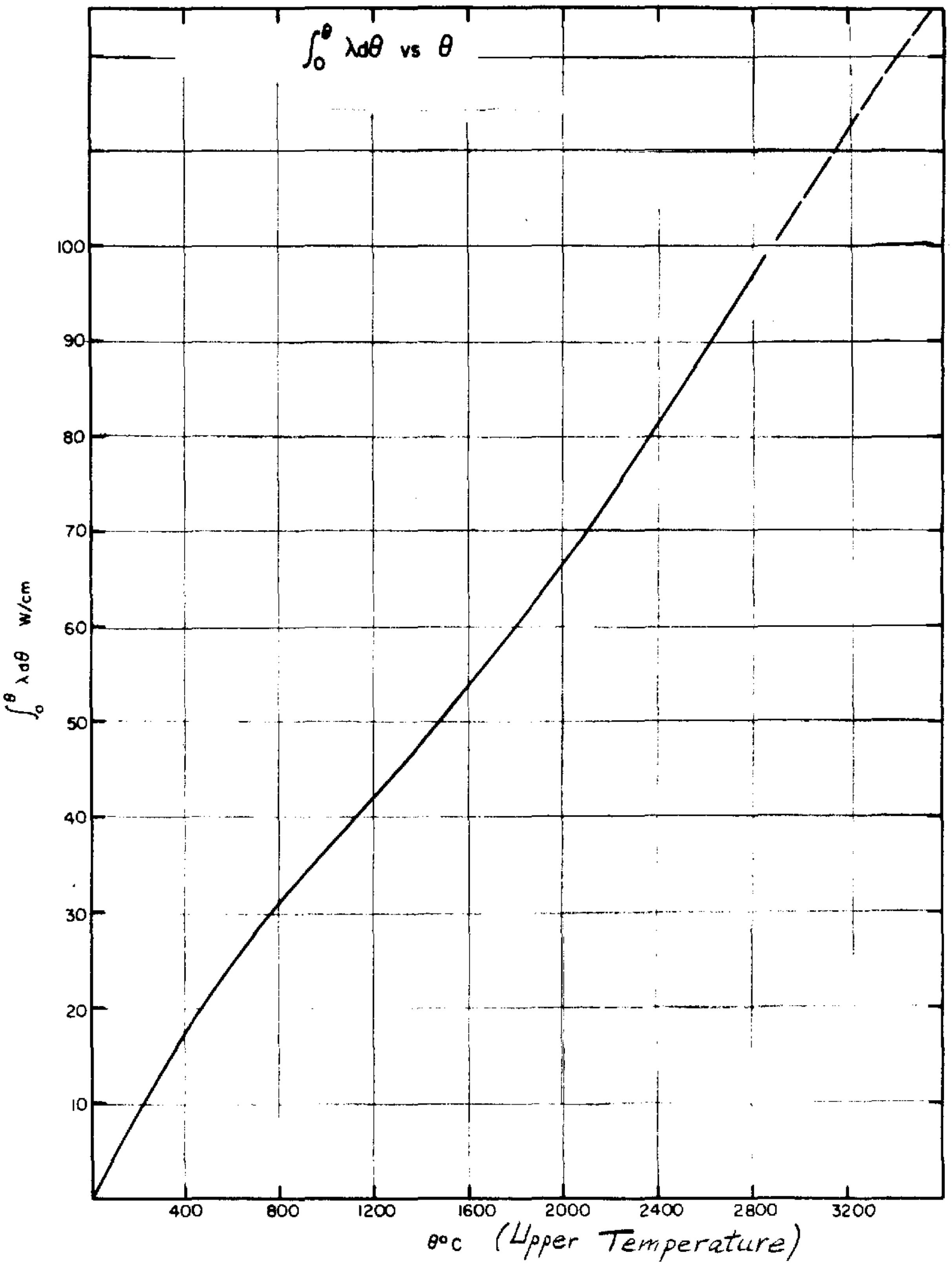
A graph, given as Fig. 1 has been included in this lesson to give some idea of the associated UO_2 temperatures. Reference to Fig. 1 will show that the upper temperature corresponding to

$\int \lambda \, d\theta = 40$ W/cm is about 1100°C; to

obtain the central UO_2 temperature, to this must be added the surface temperature of the UO_2, about 500°C in this case, giving the UO_2 central temperature at nominally 1600°C, well below the melting point of 2800°C.

ASSIGNMENT

1. How does UO_2 compare with metallic uranium as a high burn-up fuel, with regard to dimensional stability, fission product retention and corrosion resistance?


$\int_0^\theta \lambda d\theta$ vs θ
$\int_0^\theta \lambda d\theta$ W/cm
10
20
30
40
50
60
70
80
90
100
400
800
1200
1600
2000
2400
2800
3200
θ°C (Upper Temperature)

<u>Fig. 1</u>

2. Are gaseous fission product pressures inside operating UO_2 fuel elements of the Canadian design considered to be a problem? Why?

3. Why is $\int \lambda \, d\theta$ used to indicate the severity of a fuel irradiation, in preference to some other parameter, such as temperature?

R. Popple

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PREPARATION AND MANUFACTURE

The production of uranium, in the form and purity needed for fuel manufacture, entails a series of carefully designed refining operations. In addition to the pure metal, alloys and several chemical compounds of uranium are produced as basic fuel materials. The bulk of the work done in Canada in this line concerns natural uranium products, but enriched and depleted materials are also being processed in increasing quantities.

Uranium concentrate from the mines is shipped to the refinery as "yellowcake", a precipitate of di-uranates, and this is the starting material for all the refinery processes.

The impurities in yellowcake vary according to the source of the material, so that the preliminary refining operation is the blending of batches of yellowcake from several sources. Although the total concentration of impurities is unaffected by the blending process, it has the advantage of providing a means of control of individual impurities.

Chemical operations commence with the digestion of the yellowcake with nitric acid to produce an acid slurry of uranyl nitrate. This is purified in a series of extraction columns, where the slurry is contacted with an organic solvent which dissolves the uranyl nitrate, leaving most of the impurities in the aqueous phase. The organic solution is scrubbed with a small amount of water which removes most of the remaining impurities, and is then contacted with large amounts of pure water, causing the uranyl nitrate to transfer back to the aqueous phase and leaving impurities in the organic solvent. The final aqueous solution of uranyl nitrate has a purity which meets the high standard required for nuclear fuel production, and after concentration in a two-stage evaporator, contains about 1400 grams of uranium per litre.

Having produced a uranium compound of the required purity, it remains to convert this compound into a derivative which is suitable for nuclear fuel manufacture. In one series of operations, the uranyl nitrate is heated to give uranium trioxide by thermal decomposition. The trioxide, also known as "orange oxide", is cast into pellets which are treated by a moving bed process in two stages. The pellets are heated, and pass down a reaction vessel where they are met by a stream of hydrogen, causing reduction to the dioxide (brown oxide). In a second heated vessel, the pellets are changed, by a stream of hydrogen fluoride, into uranium tetrafluoride or "green salt". Excess acid is removed by a scrubber and the green salt is collected on filters.

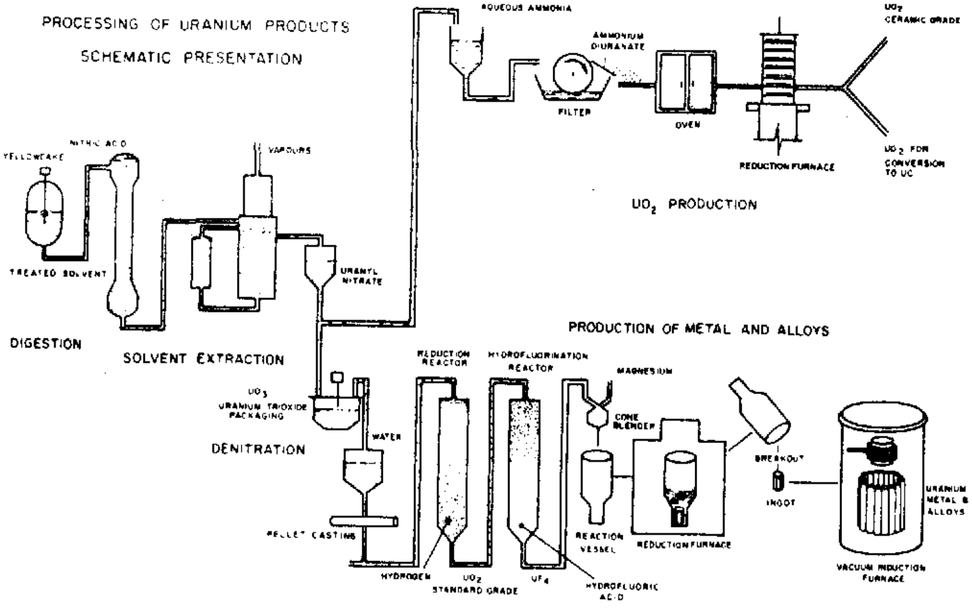


Fig. 1

Uranium Metal

Uranium metal is made from the tetrafluoride by a thermite-like process in which the salt is reduced by magnesium metal. The reaction takes place in a vessel with a refractory lining which is made from magnesium fluoride - a by-product slag recovered from previous production. Weighed quantities of powdered uranium tetrafluoride and granulated magnesium are blended and packed uniformly in the reaction vessel which is then capped with refractory slag. The vessel is placed in a furnace and heated slowly over a 12 - 16 hour period to 650°C; the thermite reaction starts spontaneously and is complete in two or three minutes during which time the temperature rises to 1900°C. The metal and slag are both molten when produced, and the uranium collects at the bottom of the vessel. After two days of cooling, the metal is removed from the vessel as a cylindrical ingot which is about 50 cm in both diameter and height, and weighs about 2 tonnes.

The uranium ingots may be directly fashioned to the required shape, by forging and rolling, or they may be remelted under vacuum and precision cast to shape.

Uranium Dioxide

Uranium dioxide has already been described as one of the stages in manufacturing green salt. The dioxide produced in this way, when pulverized or fused, can be fabricated into fuel elements by swaging, vibratory compacting and other methods.

The high-stability fuel for power reactors, however, uses a different quality of uranium dioxide (known as "ceramic grade" dioxide) which is produced by a somewhat different process resulting in a powder with a high specific surface area. This ceramic grade of dioxide can be sintered, without the addition of binding materials, to form high-density pellets. In this process, high purity uranyl nitrate solution is mixed with ammonia to precipitate ammonium di-uranate (ADU). The precipitate is dried and pulverized and then placed in trays stacked in a closed retort containing an atmosphere of inert gas. The retort is heated, causing the ADU to decompose into uranium trioxide; this is reduced to the dioxide by passing hydrogen into the retort to displace the inert gas. After the process is completed, the retort is allowed to cool and the dioxide is removed. The product is pulverized to remove agglomerates and batches are blended to insure consistency in the quality of the uranium dioxide powder.

Uranium Carbide

Uranium carbide has a higher density and thermal conductivity than uranium dioxide and is a fuel material with considerable promise, particularly for use in organic-cooled reactors. To produce the carbide, uranium dioxide and graphite are pulverized and thoroughly blended. Sintering the mixture in an induction furnace produces a low density uranium carbide, which is transferred to a skull-arc furnace. Here, the carbide is melted and is cast into high-density slugs of the required shape which, after surface grinding, are ready for incorporation in fuel rods.

Manufacturing Technique

The procedure for manufacturing a fuel bundle depends on the type of bundle to be produced. The description which follows relates to the production of Douglas Point fuel; although there are many differences of detail, the manufacture of NPD and/or Pickering fuel follows the same general pattern.

Fig. 2 shows details of a Douglas Point first charge fuel bundle. The zircaloy tube for making the sheaths is received in batches and samples from each batch are checked for straightness, mean internal diameter, wall thickness and variation of internal diameter. This inspection is followed by an ultrasonic test for flaws which is applied to every tube. Acceptable tubes are cut to length, degreased, and loaded into the magazine of a special type of welding machine for the attachment of end plugs. Twenty tubes are loaded at one time, nineteen of them being a batch of tubes which will be made into one fuel bundle, plus one tube which is a dummy and is used for checking the welding operation.

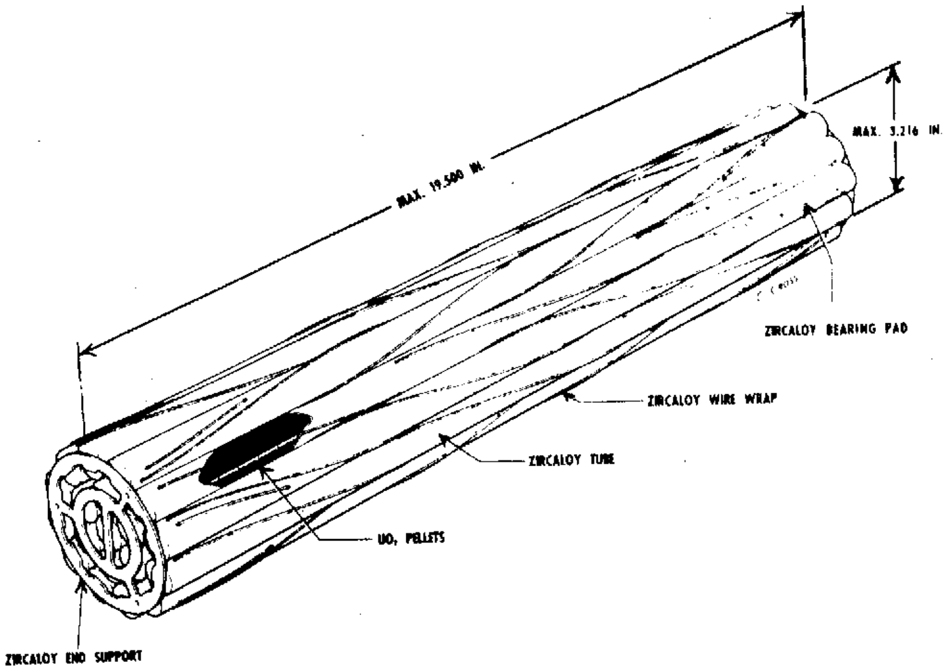


Fig. 2

End plugs, machined from zircaloy bar stock are also loaded into the welding machine, which is a resistance welder using an electro-magnetic follower to apply pressure to the end plug while the weld is produced. Each weld is performed individually. The welded sheaths are placed in a 19-element jig with open ends upward. The UO_2 pellets are now loaded into the tubes by hand; as each sheath is filled with 24 pellets, the space remaining is checked to ensure that there will be sufficient clearance for the second end plug.

The filled sheaths are returned to the welding-machine magazine, which is evacuated and then filled with inert gas consisting of argon with 10% helium. The second end-plugs are welded into place to seal the sheaths and the elements are completed by a machine in which the flask from the welding operations is removed and the end plugs are profiled to their final shape. The completed elements are tested for leakage by means of a sensitive helium leak-detector.

The elements are cleaned by pickling and the appendages required for maintaining inter-element spacing attached (wire-wrap spot-welded every inch in the case of the Douglas Point first charge). The welds on a test sample are given a shear test to determine the weld strength. The elements which will form the outer ring of the bundle now have bearing pads added. These are also zircaloy wire of thicker diameter than the wire wrap to ensure bundle-to-pressure tube spacing.

The end-plate to end-plug welds are now effected in a jig which rigidly holds the completed elements in their "bundle" positions. These welds and their structure are an improvement over the original NPD design, and will likely be used for Douglas Point replacement fuel, and Pickering. The end-plug has a shallow, conical protrusion on it with a small platform at the base of the cone. When pressed against the end plate, a current impulse is passed between the two. The high current density at the tip of the cone causes the zircaloy to melt, and the end plate settles down onto the platform, producing a strong, reliable weld.

Other Fuel Designs

There is an extremely broad variation of nuclear fuel in use at the different research institutions and power stations operating today. A few examples of these fuels are given in Fig. 3. These sketches are each drawn to scale, but not all on the same scale.

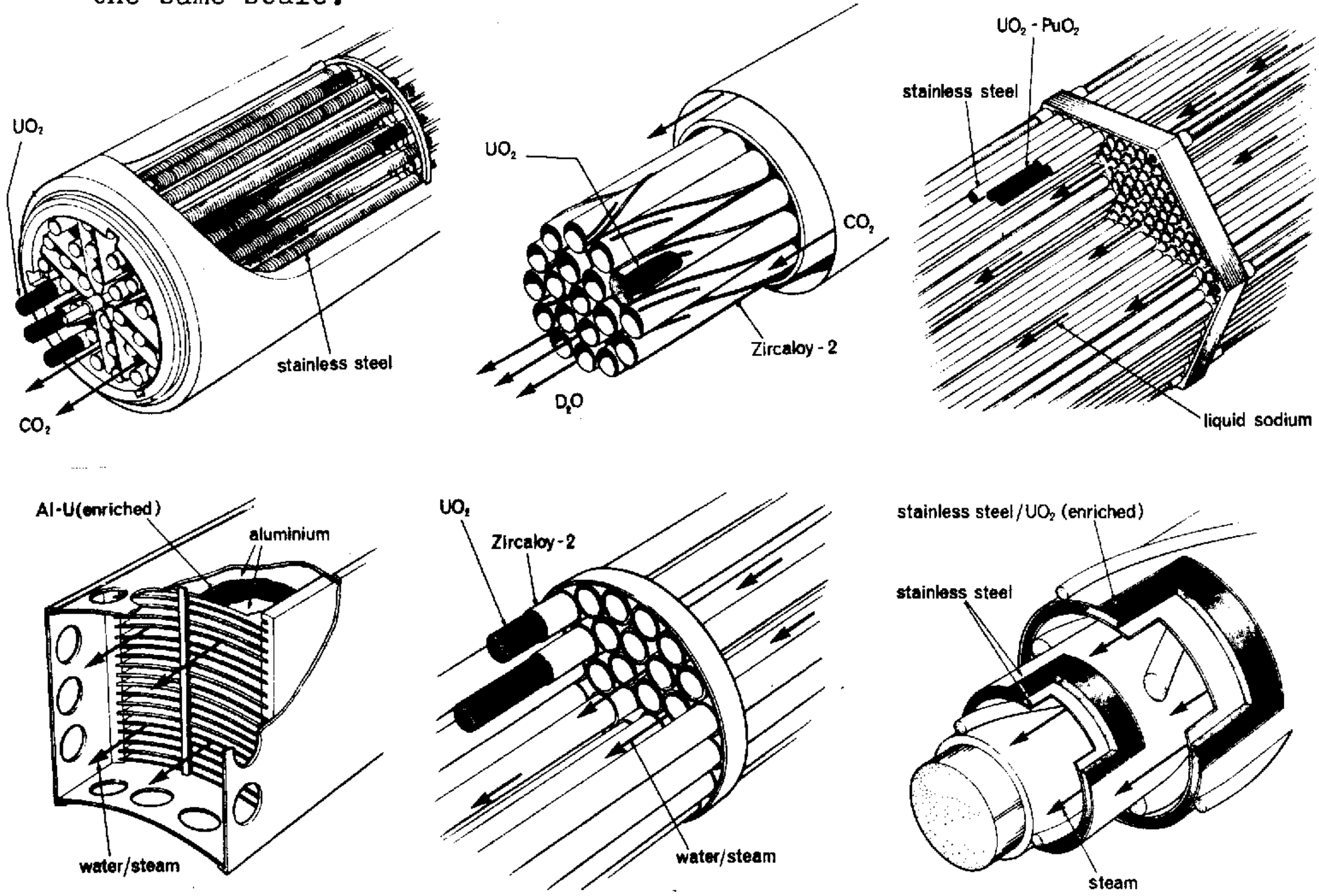


Fig. 3

In the upper row, left to right, the fuels are that of the Advanced Gas Cooled Reactor (AGR); the Douglas Point Reactor and the Prototype Fast Reactor (PFR) in the United Kingdom. In the lower row, left to right, the fuels shown are that of the Materials Test Reactor (MTR), similar in design to that at CRNL in the Pool Test Reactor (PTR); the British Steam Generating Heavy Water Reactor (SGHWR) and the Pathfinder Boiling Water Reactor (BWR), utilizing the superheat element shown.

Ease and cheapness of reprocessing spent fuel in order to recover any remaining fissile material must be kept in mind in designing fuel elements, but cheap initial fabrication is usually more important. In this context, because of the toxicity of plutonium and the radioactivity usually associated with contaminants of U-233, the cost of fabricating fuel elements containing these fissile materials is greater than the cost for fuels enriched in U-235. The standard of reliability of nuclear fuel elements is extremely high, but manufacturing techniques have been developed to meet this high standard at an acceptable cost.

Many of the requirements for solid fuel elements are conflicting - for example, greater reliability can be achieved by making the sheath thicker but only at the expense of increased parasitic neutron absorption - and the weight which must be attached to each factor depends on the type of reactor. The great majority of fuels for power reactors which have achieved competitive performance or which are currently in an advanced stage of development are in the form of metal, an oxide or a carbide, arranged in either a tubular or cylindrical shape.

R. Popple

Reactor Boiler and Auxiliaries - Course 133

THE CANADIAN DESIGNS AND THEIR PERFORMANCE

The success of the natural uranium heavy water reactor system, which has reached the point where it can compete in cost with conventional power stations, has demanded a high performance from the fuel elements and a considerable amount of development work has been concentrated on fuel element design. The fuel must be in a form which is stable under irradiation for long periods while immersed in heavy water at high temperature and pressure. Also, because natural uranium contains only a small proportion of fissile material, it is important to make the most efficient use of all available neutrons, and therefore, the fuel elements must contain a minimum of neutron-absorbing material.

The major step in satisfying these stringent requirements has been the development and manufacture of fuel elements consisting of sintered UO_2 pellets enclosed in zirconium-alloy sheaths. Processes for making the pellets, already described, have been developed which give a product of great stability. The problems of fabrication and quality control of zirconium-alloy sheaths with wall thicknesses as low as 0.38 mm have been solved. The complete fuel elements have been manufactured in large quantities by two manufacturers and are capable of burnups in excess of 10,000 Mwd/te U while operating at high heat ratings.

Performance of UO_2 Fuel Assemblies in Canada

Before summarizing the performance of the Canadian fuel design to date, it would seem appropriate to give a brief description of the different fuel designs produced by Canadian manufacturers. The vast majority of power reactor fuel under development and irradiation today in Canada consists of 50 cm long bundles of either 7- or 19-element bundles, which fit inside an 8 cm I.D. pressure tube. Pickering and Gentilly G.S.'s will utilize 28-element and 18-element bundles respectively, designed to fit within 10 cm I.D. pressure tubes. The 7-element bundles in NPD are no longer considered to be a potential future fuel design, but were an important intermediate step in the development of smaller diametered elements, now in use at Douglas Point and to be used at Pickering G.S. The 7-element bundles are held together by either rivetted or fusion welded end plates, with inter-element and bundle-pressure tube spacings maintained by spiralled, spot-welded wire wrap. The NPD 19-element bundles, initially conceived because early irradiation tests at CRNL indicated that a core fuelled entirely with 7-element fuel would lead to elements near the centre of the core being operated at ratings which at that time were believed to be too high, are of similar design, but the elements are of reduced diameter (1.5 cm vs 2.5 cm).

The Douglas Point fuel bundles, several of which are undergoing irradiation in NPD, evolved from the NPD 19-element bundle, and are remarkably similar except for different detailed design of wire-wrap and the addition of bearing pads to improve the wear properties between the bundle and the coolant tube. Other changes involved manufacturing methods (described previously) to reduce costs. The tungsten inert gas (TIG) end cap and element-to-end plate welds of NPD were replaced with resistance welding for Douglas Point. (See Fig. 1, lesson 133.71-2.)

The second generation of fuel for Douglas Point is still under development, but will have no wire wrap. Instead, element spacing is accomplished by the use of small spacers at the bundle mid-plane, and bundle-to-pressure tube spacing is accomplished by means of three planes of straight bearing pads. Fig. 1 shows the type of brazed spacer in plan view being developed; note that the spacers are at an angle of 30° to each other to prevent inter-locking of mating spacers. Fig. 2 shows an elevation and cross-sectional view of a typical straight bearing pad. The reason for the elimination of the wire-wrap from the Douglas Point design was that out-reactor flow endurance tests on wire-wrapped bundles indicated that at the high flow rates needed for Douglas Point fuel, (12.6 kg/s maximum) the elements would vibrate enough that in the life of the fuel, the wear on adjacent sheathing by the wire-wrap may be sufficient to penetrate the sheath (called "inter-element" fretting). The criss-crossed spacers allow one to tolerate a higher fretting rate, and fretting occurs on a raised surface.

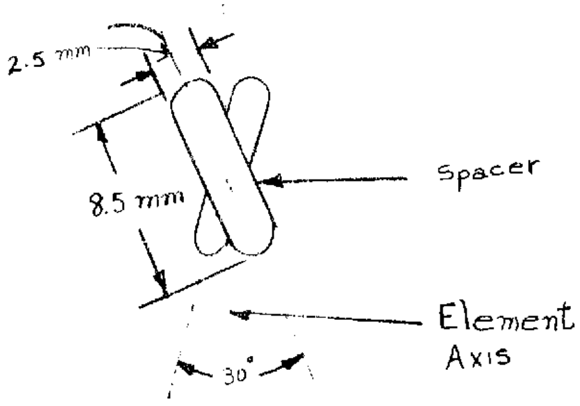


Fig. 1

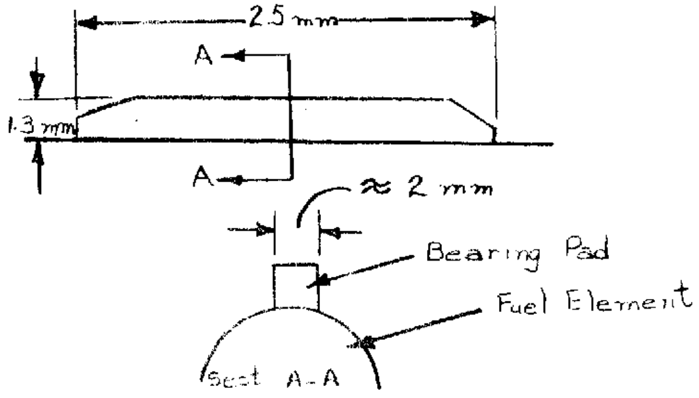


Fig. 2

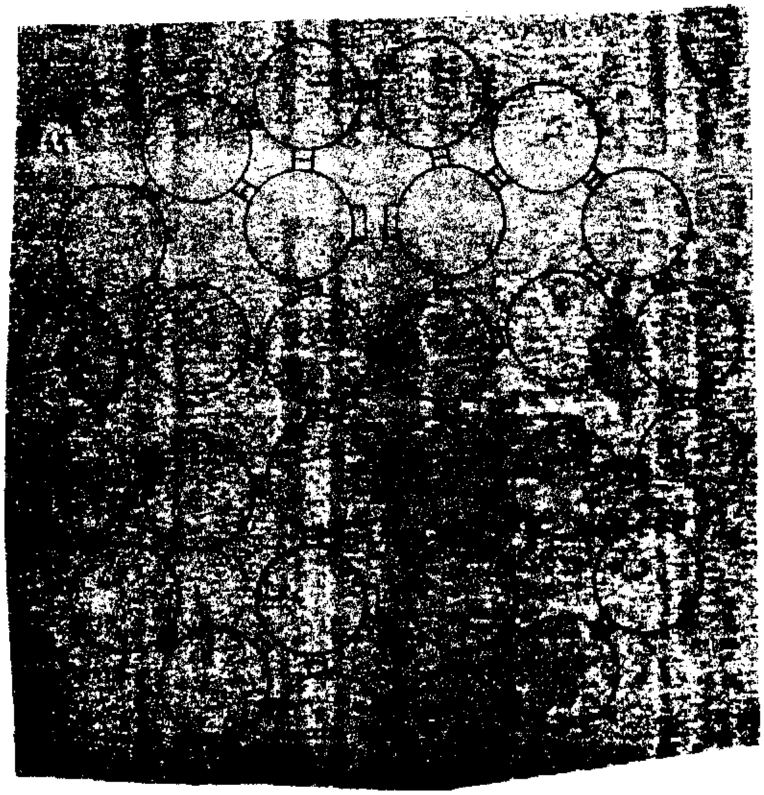


Fig. 3

The Pickering reactors will utilize 10 cm pressure tubes. In order to take advantage of the technology developed for the 1.5 cm elements in use at NPD and Douglas Point, the size of element will remain the same, but the number of elements increased from 19 to 28. Fig. 3 shows a transverse cross-section of such a bundle at the axial mid-plane; note the use of spacers the same as those for the Douglas Point replacement fuel. The bearing pads have been omitted from the sketch.

Extensive tests on the 19-element designs have been carried out in the loops of the NRU reactor at CRNL. In addition, the irradiation of statistically significant numbers of bundles of each of the 8 cm O.D. designs is continuing in NPD and commencing at Douglas Point. In NPD, the irradiation of the first charge of fuel has reached equilibrium burnup (6400 Mwd/te U); moreover, a large number of bundles have been replaced with fresh fuel while attaining burnups in excess of 10,000 Mwd/te U on certain development bundles, at Douglas Point ratings ($\int \lambda d\theta$ = 36 w/cm). Post-irradiation examination of high burnup elements has revealed them to be in excellent condition. While the UO_2 generally was cracked and very brittle, there has been no evidence of any inter-action between the fuel and sheathing.

Bundles of the 19-element CANDU-PHW design have been successfully operated in the U-2 loop at CRNL in steam-water mixtures for prolonged periods, ie, in a boiling environment. Post-irradiation examinations have revealed that the UO_2 - Zr-2 fuel design can be successfully operated in a two-phase coolant, and tests are continuing. Dimensional stability of the bundles tested has shown a maximum length increase of less than 1/4% at $\int \lambda d\theta$ = 55 w/cm, much higher than the maximum rated Pickering elements and high burnup bundles have been shown to be structurally sound enough to pass specified pre-irradiation strength tests.

The first test of Pickering fuel has been completed; the test consisted of the irradiation of six 28-element bundles, four of which were operating at average outer element ratings comparable to the highest rated Pickering elements ($\int \lambda d\theta =$ 42 w/cm). Some bundles operated in a boiling environment with up to 6 wt % exit steam quality. Examination of the bundles, irradiated to a maximum burnup of 3000 Mwd/te U, showed them to be in excellent condition. Further Pickering tests are continuing.

Hydrogen Pick-up in Zircaloy-2

The deuterium released in the dissociation of heavy water can be picked up by the sheathing, to form ZrD_2, and the oxygen forms an oxide coating on the sheathing, according to the equations:

$$Zr + 2\ D_2O = ZrO_2 + 2\ D_2\uparrow$$

$$D_2 + Zr = ZrD_2$$

Excessive ZrO_2 thickness could adversely affect the heat transfer properties of the sheathing. The ZrD_2 does not present a problem when the sheathing is at high temperatures; when cooled, however, the ZrD_2 platelets precipitate out, and appear as short dark lines when viewed at high magnification. As long as the platelets are oriented circumferentially, they do not present a problem; if they precipitate preferentially in the radial direction, the decrease in local density could lead to cracking of the sheathing. Fig. 4 shows a cross-section of sheathing, with ZrD_2 platelets predominately circumferential, magnified to 250X. The vertical distance is 0.38 mm in the figure. A criterion has been developed, called the F_n number, which is the proportion of platelets oriented within 30° of the radial direction, for comparison of samples.

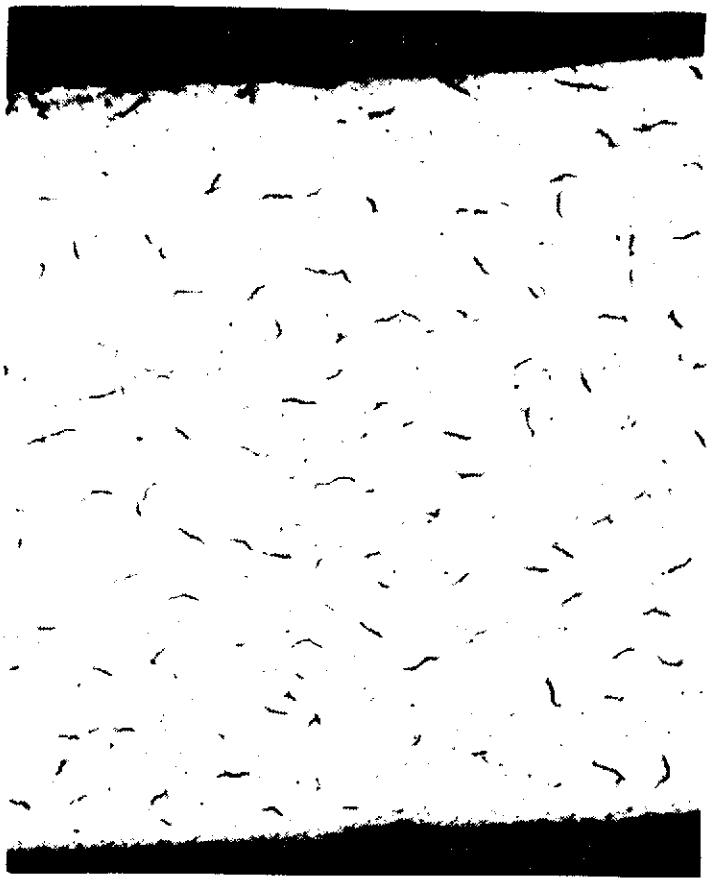


Fig. 4

Radial hydrides may develop in areas of stress concentration and they tend to migrate down temperature gradients to the coldest parts of the bundle (eg, end plates, end caps). Embrittlement of the sheathing results. In view of the hydrogen pick-up in Zircaloy-2, future Canadian fuels will be clad in Zircaloy-4 which is an alloy similar to Zircaloy-2, but has a slightly lower nickel content, which evidently leads to a reduced tendency to absorb hydrogen.

CANDU-BLW Fuel

The fuel for BLW is still under development, and is generally similar to that employed in the NPD and Douglas Point reactors. It consists of natural uranium dioxide pellets sheathed in Zircaloy tubes to form fuel elements. Eighteen of these elements are assembled into a fuel bundle configuration. As shown in Fig. 5, these eighteen elements are arranged in two concentric rings, the inner ring containing six elements and the outer ring twelve elements. The inner ring surrounds an unfuelled central position which is occupied by a tubular gas-filled Zircaloy tie-rod. This tie-rod is used to assemble ten bundles into a fuel assembly. Each fuel channel in the reactor contains one of these assemblies.

Zircaloy-4 is used to sheath the UO_2 pellets, which are larger in diameter than the Douglas Point or Pickering elements (2.0 cm vs 1.5 cm). The nominal maximum rating is $\int \lambda d\theta = 48$ w/cm; but the nominal maximum specific power output is lower than Douglas Point (24 w/gm U vs 34 w/gm U), due to the larger element diameter. Like the CANDU-PHW pellets, the BLW pellets have a shallow spherical dish at one end, the purpose of which is two-fold:

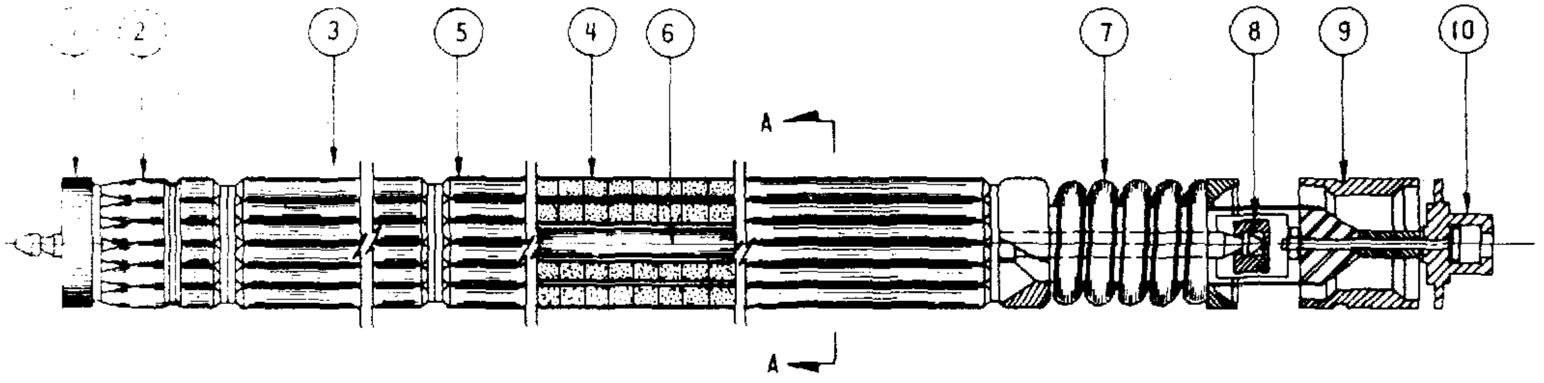
1. To increase the void volume in the element to accommodate the fission product gases, and reduce the internal pressure.

2. To allow some expansion of the pellets.

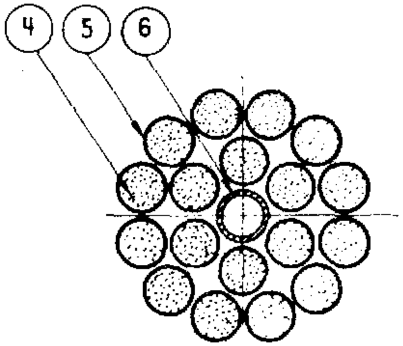
Zircaloy-4 was chosen as a sheathing material because of its lower rate of uptake of corrosion product hydrogen, as outlined previously. The table below gives a comparison of some of the relevent properties of these two materials; the figures are in weight %. The capture cross-sections of the two materials are about the same, with that of Zircaloy-2 slightly lower.

Element	Zircaloy-2	Zircaloy-4
Sn	1.20 - 1.70	1.20 - 1.70
Fe	0.07 - 0.20	0.18 - 0.24
Cr	0.05 - 0.15	0.07 - 0.13
Ni	0.03 - 0.08	0.007 maximum

Sum of Fe, Ni and Cr in Zr-2 must range between 0.18 and 0.38
Sum of Fe, Ni and Cr in Zr-4 must be 0.28 minimum



1. UPPER KEY~ VARIABLE THICKNESS
2. FLUX SUPPRESSOR
3. FUEL BUNDLE~ 10 PER ASSEMBLY
4. UO_2 FUEL PELLETS
5. SHEATH
6. TUBULAR TIE ROD
7. SPRING
8. LOWER KEY
9. FUELLING MACHINE LOCATING FINGER GROOVE
10. UPPER HALF OF T-SLOT CONNECTOR TO SHIELD PLUG



SECTION A-A

Fuel Assembly

Fig. 5

Other Advanced Fuel Designs

Other designs which are in the process of development involve fundamentals changes in the geometry of the fuel bundle. One of these is an annular fuel consisting of several concentric cylinders of fuel material with spaces between them to allow for the flow of coolant, surrounded by a tubular sheath. Another form of fuel rod, known as the tube-in-shell assembly (Fig. 6), consists of a large-diameter tubular sheath, filled with uranium dioxide, through which pass several small diameter coolant tubes. The ends of the sheath are closed by end-plates with holes into which the coolant tubes are sealed. The main advantage of the tube-in-shell bundle is the high fuel-to-coolant ratio offered, meaning that it could be used with a light water coolant. Testing to date has indicated that it does not offer any advantage when operated at high exit steam qualities; however, manufacture and production costs could be reduced with further development, since sintering, grinding, etc, of pellets would be eliminated. Vibratory compaction is used to pack the UO_2 in the shell. Two in-reactor tests have been tried to date, both ending with sheathing failures.

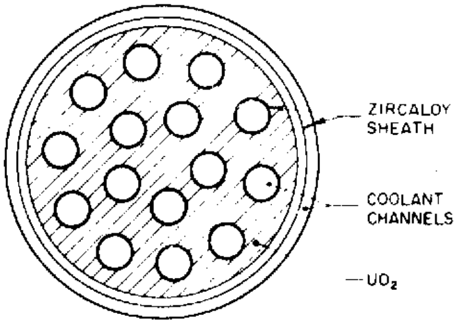


Fig. 6

ASSIGNMENT

1. Why must parasitic neutron absorption in the fuel be minimal with the CANDU-PHW and CANDU-BLW reactor designs?

2. Why was the wire-wrap eliminated from the design of the Douglas Point reference fuel design? What replaces it?

3. Summarize the operation to date of the Canadian power reactor fuel, to illustrate the compatibility of UO_2 and Zr-2.

4. How does corrosion hydrogen affect Zr-2 sheathing? What alloy offers reduced hydrogen pick-up rates?

5. What is the main advantage of the tube-in-shell fuel?

R. Popple

Reactor Boiler and Auxiliaries - Course 133

SAFETY STANDARDS AND SAFETY MEASURES

Two important factors make it imperative that safety measures be given serious consideration in nuclear power stations:-

1. The extensive casualties that could theoretically result from a nuclear accident and

2. The sensitivity of the public to any risks associated with the production of nuclear energy.

There was a tendency during the early development of nuclear reactors to demand more and more safety measures despite the extremely favourable safety record of the nuclear industry. To combat this tendency an attempt has been made, in Canada, to set a safety standard by which to judge the adequacy of safety measures.

This lesson discusses the development of this standard and the basic safety measures taken to meet this standard in our nuclear power stations.

Safety Standards in our Present Economy

All human activity involves some element of danger and the possibility that persons may be injured or killed. However, the probability of accidents are normally weighed against the benefits from the activity and if the risk appears small enough it is accepted. If the risk appears too high, social and ecomomic pressures result in safety measures being adopted to reduce the risk.

The accepted standard varies quite widely with different activities due to these economic pressures. In the transportation industry a much higher death risk is accepted for travel in private cars than for travel in airplanes or ships. One reason for this is that there is a much greater public reaction when there are a relatively large number of casualties regardless of the low frequency of airplane or ship accidents.

Whenever a new industry is established new risks are introduced. If greater risks are eliminated by the displacement of existing industry our overall safety is improved and the new risks are accepted, particularly if the new industry results in an improvement in our living standard and, hence, in life expectancy.

In the nuclear power industry it is recognized that no absolute guarantee can be given that accidents will not occur. However, a satisfactory safety standard is provided if the new risks introduced are less than the risks eliminated by the displacement of other industry.

Conventional Hazards at Nuclear Stations

Past experience with conventional hazards in other power stations and other industry has resulted in the adoption of various safety practices and safety measures. These measures imply an acceptable risk of conventional accidents and they are followed to provide the same protection against such conventional accidents in nuclear power stations.

Nuclear Hazards at Nuclear Stations

The new risks introduced by the operation of nuclear power stations result from the possibility that there may be a nuclear accident involving the escape of dangerous quantities of radioactive fission products from the fuel to populated areas where it could cause injuries or deaths. No nuclear accidents affecting the general public have occurred. Nevertheless, past experience with the behaviour of radioactive materials is limited. If pessimistic assumptions are made of the results of the escape and distribution of these materials it can be established that a nuclear accident could result in extensive deaths.

A satisfactory safety standard for a nuclear power station may be defined as one where the risk of loss of life due to nuclear accidents is no greater than the risks avoided by the elimination of coal fired power stations. However, since the potential loss of life due to a single accident is much higher, the potential frequency of nuclear accidents must be much lower.

During the early development of the CANDU type reactor statistics were gathered on the total accidental death frequency experienced in the preparation and transportation of coal, construction and operation of a coal fired power station. A comparison of this figure with that experienced in the preparation of fuel and the construction of a nuclear power station indicated that the total risks for each type of station would be similar if the annual fatality rate resulting from the operation of a 200 Mwe nuclear power station did not exceed 0.82 deaths per year. A safety factor of 5 was applied, to provide improved performance during early operation, resulting in a proposed allowable addident risk of 0.17 deaths per year.

Nuclear Safety Standards

This general philosophy for setting an allowable accident risk was used to calculate the required performance standard for equipment provided at NPD. The allowable annual fatality risk was reduced to 0.01 deaths per year since it was considered that the safety standard for the nuclear industry should be higher than that experienced in the replaced coal industry. The maximum potential death toll per accident was estimate to be 1,000, which results in a calculated maximum allowable nuclear accident frequency of once in 100,000 years. Obviously this standard can never be verified by actual accident experience. However, when the standard is expressed as an allowable annual nuclear accident probability of 10^{-5} it provides a basis for judging the effectiveness of the various measures which protect against the accidental escape of radioactive fission products.

Similar annual accident probabilities are being used to judge the effectiveness of safety measures provided at the Douglas Point and Pickering generating stations.

Protection Against Nuclear Accidents

Nuclear accidents which affect the public can only occur if the radioactive materials generated in the fuel escape and travel to a location where they may cause injuries or deaths. Obviously a single barrier cannot be depended upon to provide the required safety standard so the fuel and station equipment are constructed in a manner which provides many barriers against the escape of radioactive materials.

(a) Containment

The term containment, as used in these lessons, is intended to include all barriers which prevent movement of fission products from the fuel to occupied locations. The main containment barriers provided in the CANDU type reactors are:

1. Fuel - The fuel is designed to retain most fission products as long as the fuel temperature remains within its normal range.
2. Fuel Sheath - The fuel sheath provides a protective cover over the fuel and prevents escape of the small fraction of fission products which are released from the fuel in the form of gases.
3. Primary Envelope - This includes the piping and equipment containing the primary coolant. This envelope prevents escape into the building of any fission products which have escaped past the fuel and the fuel sheath.

4. Building - Since the reactor and primary envelope are contained in a normally unoccupied area of the station, the building enclosure provides shielding and prevents the release from the building of fission products, which have escaped past the fuel, fuel sheath and primary envelope. Equipment is provided to control any pressure surge so that the isolation provisions remain intact.

5. Exclusion Area - This is a designated area surrounding a nuclear station from which the public is normally excluded. Thus, all persons may be quickly evacuated from the area if there is any release of fission products from the station.

(b) Protective Systems

The provision of many containment barriers results in a negligible probability of significant quantities of radioactive materials escaping from the station except in circumstances where some fault causes damage to more than one of the containment barriers. Thus, if the fuel is extensively overheated, the fuel, the fuel sheath and the primary envelope may all be damaged and the resultant release of high energy coolant containing fission products may cause a pressure surge which could damage and penetrate the building enclosure.

Since fuel overheating can result only from increased power production or reduced heat removal the most important measure for preventing escape of fission products is to provide reliable process equipment which will control reactor power within a specified range and provide adequate cooling of the fuel.

Since all equipment is subject to failure the process equipment alone may not reduce the annual probability of a nuclear accident to the required standard and, therefore, independent protective systems are provided at each nuclear power station. These protective systems, which are listed below, are designed to prevent or reduce escape of fission products from the fuel into the building, if there is a failure of essential process equipment.

1. Reactor Boiler Protective System - This system independently shuts down the reactor thereby reducing fuel power production whenever various process variables exceed specified limits. The protective system prevents fuel failures, and thus nuclear accidents, following all process equipment failures except those involving loss of cooling. The design and operation of Reactor Boiler Protective Systems is discussed in another course.

2. Emergency Injection Water System - This system provides fuel cooling if the normal heat transport fluid is lost. The

2. injection water supply is obtained from an independent source and may be either light water or heavy water. Injection water equipment is discussed in the following lesson.

3. Emergency Power Supplies such as that obtained from a diesel generator, in the event that normal power supplies are interrupted by a simultaneous loss of line and loss of generator production. Power is, thus, maintained for essential process systems such as calandria spray cooling.

4. Standby Water Supplies such as that obtained from a gasoline engine, in order that essential cooling and fire protection be maintained when normal process water supplies are interrupted. Both (3) and (4) will be discussed in other courses.

ASSIGNMENT

1. On what basis is a safety standard established for a new industry such as the nuclear power industry?

2. Define a "satisfactory safety standard" for a nuclear power industry.

3. In what allowable nuclear accident frequency does such a standard result?

4. What stages of containment are provided in a nuclear station?

5. What is the primary purpose of the protective devices provided in a nuclear station?

R. Kelly

Reactor Boiler and Auxiliaries - Course 133

EMERGENCY INJECTION AND CONTAINMENT SYSTEMS

Protective devices, in a nuclear station, prevent or reduce the escape of fission products from the fuel into the building enclosure, if failure of process equipment has occurred. Containment provisions prevent or reduce escape of radioactive material from the building if both process equipment and protective devices fail.

One protective device, the emergency injection system, and all the containment devices will be discussed further in this lesson.

Emergency Injection Systems

If there is an appreciable loss of the normal heat transport fluid during reactor operations, there is a decrease in pressure and the remaining fluid boils. This reduces the heat transfer from the fuel. If the Reactor Boiler Protective System responds rapidly enough to shut down the reactor, heat generation in the fuel is reduced to that provided by fission product decay. However, if the fluid is not replaced fuel temperatures may rise sufficiently to cause the fuel to melt and allow the release of fission products into the heat transport system. The emergency injection system automatically provides a separate alternative supply of cooling water, in such an event, so that heat removal from the fuel is restored and the rise in temperature is restricted.

The most obvious separate source of heavy water for such an injection system is the moderator system. In NPD, Douglas Point and Pickering, there are interconnections between the moderator system and the heat transport system. The arrangement in NPD is shown in Fig. 1. In the event of a serious loss of heat transport fluid, heavy water, from the moderator system is pumped into the reactor inlet headers by the "charging" pump, P.

Floor drainage is arranged so that spilt heavy water is collected in a recovery sump. It is then possible to return the recovered water to the moderator system or directly into the charging pump suction, through remotely operated valves.

A similar system is used in Douglas Point and Pickering with minor differences. No charging pump is used so that D_2O

injection takes place under moderator pump pressure, through a check valve and an isolating valve. Therefore, D_2O injection will not occur until the heat transport system pressure drops below the moderator pressure. Recovered heavy water is returned to the moderator pump suction.

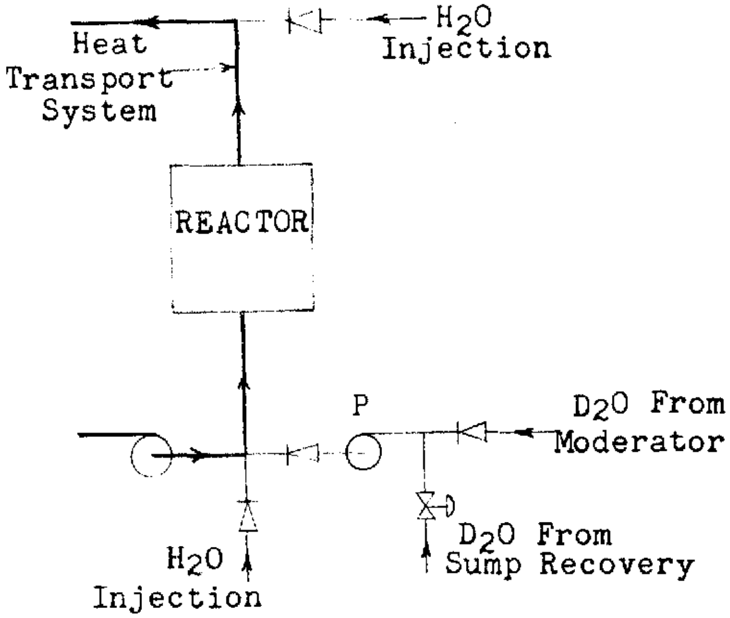


Fig. 1

An alternative supply of water may be required if the D_2O injection cannot cope with the situation or if the moderator water level drops too low to supply emergency injection. In NPD a light water injection system is provided as an alternative. Pipes from a 250,000 gallon storage tank are connected, through check valves, to the inlet and outlet headers. If the loss of heat transport fluid is extensive enough to drop the system pressure below the static head in the storage tank, light water will be admitted into the system. In Douglas Point and Pickering, provisions are made to connect a light water supply to the system if and when it is required.

Containment Provisions

It was stated in the previous lesson that there are four stages of containment of fission products, the fuel itself, the fuel sheath, the heat transport system and the building. The building and other containment provisions are required to prevent escape of radioactive material into the environment in the event that the first three stages of containment fail. All systems containing, or potentially containing, significant quantities of radioactive material and operating at elevated temperatures or pressures are placed within a containment structure designed to:

1. limit the release of radioactive material to a value set by the exposure criteria for a particular site,
2. withstand any calculated pressure increase, which may occur as a result of an accident, without exceeding the specified leak rate,
3. have adequate provisions for testing at predetermined intervals.

Since such a structure contains the reactor boiler plant and the associated shielding and material handling facilities, it would, of necessity, be very large. Such a structure is usually made of steel or concrete.

The potential energy release inside such an enclosure would cause a pressure increase beyond the capabilities of most practical structures. Severe thermal stresses would also result from such a release. Therefore, additional measures are required to absorb part of the released energy. These additional measures are:

(a) Use of Pressure Relief Systems

In this type of system the building itself may be used to contain the radioactive materials. However, to avoid the effect of the initial pressure increase on the structure, the initial blast resulting from the explosion is allowed to escape through a relief duct into the atmosphere.

Such a relief duct, as used in NPD, is shown in Fig. 2. This relief of pressure reduces the pressure build-up from the explosion and the structural thicknesses required to withstand such a pressure. It is assumed that the initial escape of steam and air only contains tritium, since fuel failure will only occur subsequent to the loss of heat transport fluid. Immediately following the initial blast, a door closes in the relief duct, preventing escape of fission products released after fuel failure.

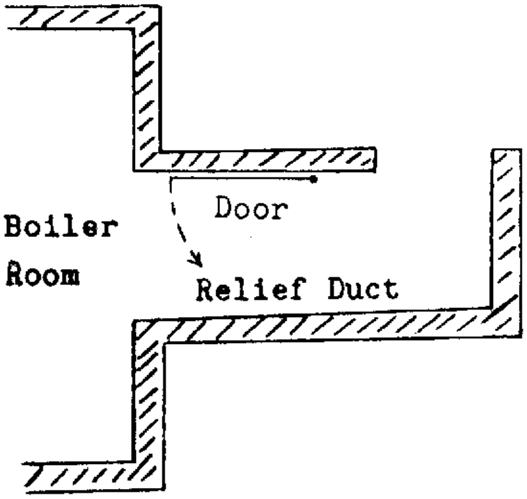


Fig. 2

Pressure relief may be used, as in Pickering G.S., as a means of reducing the pressure build-up even though a containment shell is used. Such a pressure relief would reduce the thickness of the containment shell required. The type of arrangement proposed is as shown in Fig. 3. Each reactor and associated heat transport system will be housed in a 4 ft thick cylindrical concrete containment building.

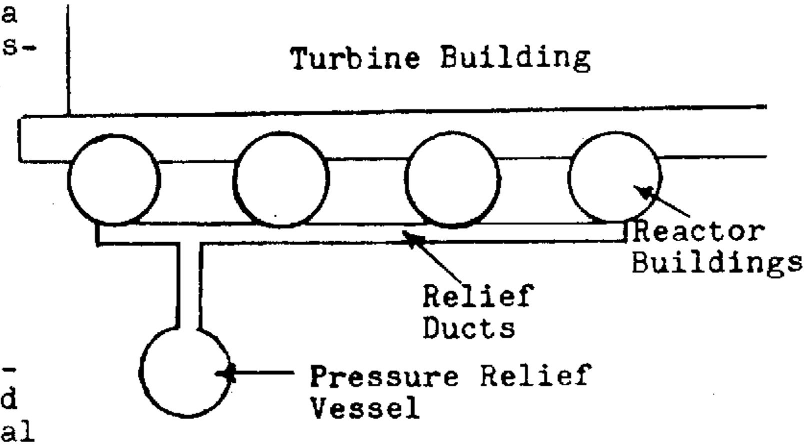


Fig. 3

The various rooms, in the reactor building, are isolated by blowout panels so that increase in pressure in one area will relieve into the rest of the building. This arrangement is also used in Douglas Point. However, in Pickering, a pressure increase in the reactor building will cause relief louvers and valves to open allowing pressure relief into the pressure relief building. This building is maintained at a pressure of 1 psia by vacuum pumps. No such pressure relief exists at Douglas Point.

(b) Use of Pressure Suppression

Since most of the accidents that can be contemplated result in a violent release of steam containing radioactive material, containment of the radioactive material is effected if all the steam can be condensed. Such pressure suppression can be obtained by allowing the pressure buildup to initiate a dousing system which spray-cools the steam. The dousing system may have to be used with a pressure relief system or a containment shell. However, the shell would be thinner and less expensive. It was estimated that, without dousing, a containment system for NPD would have to withstand a 40 psia increase in pressure with 10^6 Btu of heat released. In one Pickering unit around 2.4×10^8 Btu of heat would have to be dissipated during the type of accident envisaged. Yet at Douglas Point, with dousing but no pressure relief, the maximum internal pressure estimated in the containment vessel is only 6 psi. Consequently the containment vessel walls can be made of 4 ft thick concrete with a corresponding thickness of steel in the dome. The leakage rate out of this enclosure can be kept down to 0.1% of the total volume per hour.

A dousing tank, containing 400,000 cu ft of water is located in the roof of the vacuum building at Pickering.

Pressure suppression has an added advantage since some radioactive iodine will be absorbed.

ASSIGNMENT

1. Explain how water from the moderator system is used to provide emergency injection water for the heat transport system.
2. Why is an alternative light water supply also provided?
3. What is the purpose of a containment structure in a nuclear station and what are its design requirements?
4. Explain the use of Pressure Relief Systems and Pressure Suppression Systems in a nuclear station.

R. Kelly

Reactor Boiler and Auxiliaries - Course 133

SAFETY SYSTEM PERFORMANCE

The development of a safety standard, for nuclear power stations was discussed in a previous lesson. The standard applied to NPD was that the annual risk of a nuclear accident should be less than 10^{-5}.

This lesson considers some of the techniques being developed, at NPD and more recent stations, to measure the effectiveness of safety systems against this safety standard.

The Probability of Nuclear Accidents

Nuclear accidents may be defined as circumstances during which sufficient quantities of fission products escape from the fuel to populated areas to cause injuries or deaths.

The escape of fission products is prevented by the various containment barriers, each of which must fail before an accident can occur. If all barriers were completely independent so that no fault could affect more than one barrier then the annual accident probability could be calculated by taking the annual probability of failure of each barrier and obtaining the product of all terms. Thus, if the annual probability of failure of each of 5 containment barriers was 1/10 then the annual accident probability would be $(1/10)^5$ or 10^{-5}, the required standard. If on the other hand a single fault could cause failure of all containment barriers, then the annual accident probability would become the annual probability of the fault.

The faults which can potentially effect more than one containment barrier are less probable than other faults, but they are, nevertheless, of greatest concern in the design and operation of nuclear power stations. The safety systems, as discussed in the previous lesson, reduce the probability of such faults by independently protecting the containment barriers. Thus, nuclear accidents can not occur unless there are simultaneous failures of the following equipment:

(a) The process equipment which normally prevents fuel failures by restricting power production in the fuel and maintaining adequate heat removal from the fuel. Fuel failures are possible if the fuel temperature increases because of failure of this equipment, either as a result of mechanical faults or human error.

(b) The <u>protective equipment</u> fails to limit fuel temperatures either by rapid shutdown of the reactor or by emergency injection. Fuel and fuel sheath failures then occur.

(c) The <u>containment equipment</u> fails to prevent the escape of fission products, which were released into the heat transport system, to inhabited areas.

If the protective equipment and containment equipment are constructed and maintained independent of each other and the process equipment, so that they are not affected by the same faults then the annual accident probability is reduced to the <u>product</u> of the annual probability of failure of each group of equipment. Thus, if an analysis of process equipment indicated that the probable frequency of faults which may cause fuel failures is once per year, then our safety standard would indicate that the annual probability of failure of both the protective and containment equipment should be less than 10^{-5} per year or typically an annual failure probability of 3×10^{-3} for the protective equipment and 3×10^{-3} for the containment equipment.

Safety System Performance

Increased favourable operating experience in nuclear power stations will lead to an increased confidence that a nuclear accident will never occur, as a result of simultaneous failures of process, protective and containment equipment. The safety standard can, however, be used to set an acceptable failure rate for individual process or safety systems and to judge the required effectiveness of the safety systems.

A tentative Reactor Siting and Design Guide, prepared by the Atomic Energy Control Board of Canada sets the following limits for equipment failures:

1. <u>Single Failures</u> - failures of process equipment which could lead to fuel failures but are safely terminated by protective and containment equipment should not exceed 1/3 per year.

2. <u>Dual Failures</u> - failures of process equipment potentially leading to fuel failures combined with failures of <u>either</u> protective or containment equipment should not exceed 1 in 1000 years or an annual risk of 10^{-3}. Dual failures are terminated by correct operation of the remaining containment or protective equipment.

The Siting Guide, which is discussed in Course 131, sets standards for the maximum allowable fission product release following single and dual failures. The allowable releases are limited both by an allowable individual and an allowable integrated population dose. Process failures only are assumed to be frequent enough to form part of the normal operating effluent and the release is limited by the allowable annual exposure of the general public. The release following dual failures is limited below the allowable once per lifetime exposure of the the general public since these would not occur more often than once per 1000 years.

These standards set limits for the required effectiveness of safety systems, (eg, leak tightness of building containment), and for the allowable annual failure probability of the safety systems.

Safety System Unreliability

Safety systems are normally inactive and are only required to operate if there is some process equipment failure. Thus, faults which incapacitate the system may develop and go unnoticed until operation is attempted. To avoid this, safety systems are tested regularly and all observed faults promptly recorded, corrected and their duration estimated.

An estimate of the system reliability can be calculated as the proportion of total time during which the system would have operated correctly if called upon. However, this normally results in numbers which are awkward to use, such as 99.999. To simplify the mathematics the proportion of total time during which the system would have failed to operate is calculated resulting in a number such as .001 or 1×10^{-3}. This portion of total time, during which it was in a failed condition, or the probability of failure, is identified as the system unreliability.

The duration of a fault is frequently difficult to estimate so it is usually assumed that it occurred at the mid point between discovery and the last successful test. Thus, if a system is tested daily and one fault is discovered, during a year, which causes system failure then the system unreliability can be calculated as follows:

$$P = \frac{\text{Duration of Failed Condition}}{\text{Total Time}} = \frac{FT}{2}$$

where P is the system unreliability, F the number of faults per year and T the test interval in years.

In the case quoted; $F = 1$ and $T = 1/365$

$$\text{Therefore,} \quad P = \frac{1 \times 1}{2 \times 365} \approx 0.001$$

Component Unreliability

It may be impractical to completely test all the safety systems after a nuclear power station is operating. It may also be that system failures are so rare that it would take too long to accumulate valid statistics on system unreliability. If the components of each safety system are carefully tested, it is possible to obtain an accurate estimate of the system unreliability from an analysis of the observed component faults.

If the indicated unreliability of a component is too high it may be improved either by reducing the test interval or by reducing the failure frequency.

For a reduction in test interval it is usually necessary that facilities for remote on power tests be available since many components are located in areas that are accessible only during shutdown and shutdowns must occur with a minimum of frequency in base loaded nuclear power stations.

If the component failure frequency cannot be reduced by replacement or maintenance then redundant components should be provided so that system failures do not occur unless there are multiple component faults. If these components are <u>independently</u> arranged so that they are not affected by the same fault then the system unreliability is a function of the <u>product</u> of their individual failure rates.

Suppose, for example, that 3 independent pumps are available for adding injection water following a process failure and that any one pump will provide sufficient flow to prevent fuel failures. If the pumps are tested once each year and records indicate that on the average one pump fails every 3 years then the unreliability of each pump can be calculated as follows:

$$P = \frac{\text{Duration of Failed Condition}}{\text{Total Time}} = \frac{F}{n} \times \frac{T}{2}$$

where P is the unreliability of each pump, F is the average failures per year for all pumps, n is the total number of pumps and T is the test interval in years.

In the example, $F = 1/3$, $n = 3$ and $T = 1$

Therefore, $P = \frac{1}{3} \times \frac{1}{3} \times \frac{1}{2} \approx 0.06$

The unreliability of all three pumps, (ie, the probability that they will all fail at the same time), is approximately P^3 or 0.0002. This is the safety system unreliability due to the pumps.

The analysis of each safety system will show that it may fail for any one of several reasons. Thus, an emergency injection system may fail if pumps fail, or if valves do not open, or if a water supply is not available. The calculated unreliability of the safety system is the sum of the unreliabilities of each of these groups of components. Therefore, the unreliability of each group must be less than the required unreliability of the system.

Operating Considerations

Experience at NPD has indicated that the deterioration of components, which sit idle and unattended for long periods of time in inaccessible areas, tends to be higher than predicted. This applies to both moveable components, such as valves and dampers, and static components such as lines containing stagnant water. On the other hand the performance of components which can be tested frequently has been better than predicted and so provision should be made for remote component tests wherever possible.

If facilities are available for on power safety system component tests, then the facilities must provide either for continued availability of the system during the test or for its rapid return to service. For routine testing these facilities must not interfere with normal operation of the station. Thus, if dousing components are tested, all lines must be drained remotely to avoid spilling of dousing water into the reactor area.

ASSIGNMENT

1. (a) Which faults are of greatest concern in the design and operation of nuclear power stations?

   (b) How do the safety systems reduce the probability of such faults?

2. What limits for equipment failures are set by the AECB Reactor Siting and Design Guide?

3. (a) Why is it desirable to test safety systems regularly?

   (b) A safety system is tested once every six months and one fault was discovered in 4 years. What is the apparent system unreliability?

   (c) Why would the unreliability, as calculated in 3(b), not be too valid?

4. How can component testing provide an estimate of the system reliability and what previous experience suggests that provisions should be made for remote component tests whenever possible?

R. Kelly

Reactor Boiler and Auxiliaries - Course 133

CHEMISTRY OF WATER CIRCUITS

Water circuits in a nuclear-electric generating station are widely varied as is the chemistry associated with them. In most cases we are concerned with corrosion and contaminant control in such systems as the moderator, primary coolant, boiler-feedwater and storage bays and with the production of pure water from a raw water source to supply the light water systems. Also, where municipal water services are not available such as at NPD and Douglas Point, raw water must be treated to meet regulations for domestic consumption. In the same respect sewage treatment must be controlled.

This lesson will, however, deal mainly with the chemistry of the main heavy water systems - moderator and coolant and with the boiler water.

The Need for Chemistry

The need for chemistry arises from a number of problems that are encountered in a nuclear-electric station. Following are some of the more important of these problems with a description of the more significant factors associated with the problem.

1. Corrosion: Corrosion is defined as the destruction of a metal by chemical or electrochemical reaction with its environment. In the corrosion of ferrous metals in contact with moisture, reversion of the metal to the oxide form takes place. The tendency to corrode is electrochemical. However, the rate at which corrosion proceeds is dependent on the resistance to continued attack caused by the products of the corrosion reaction.

In water systems, the principal factors influencing the corrosion of ferrous metals are the water characteristics temperature, rate of water flow and contact with dissimilar metals. The chief variables controlling the corrosive characteristics of a water are its dissolved oxygen concentration, carbon dioxide content, pH and dissolved solids. Other factors may be free mineral acid, hydrogen sulfide, sulfur dioxide, etc. In general, however, dissolved oxygen and carbon dioxide are the chief problems, with pH directly affected by the carbon dioxide concentration. Increased dissolved solids, particularly chloride and sulfate, increase the corrosive effect of oxygen and carbon dioxide.

2. Embrittlement: Embrittlement is usually encountered in boilers or other systems where free sodium hydroxide is present either intentionally or by decomposition of sodium phosphate. It can occur in primary coolant systems if too high a pH is maintained with lithium hydroxide. Three factors must be present for embrittlement.

   (a) Leakage - to permit escape of steam and concentration of water at point of leakage

   (b) Stress - either internal or external

   (c) The concentrated water must possess embrittling characteristics and chemically attack the metal.

3. Heat Transfer Reduction: Heat Transfer Reduction is normally the result of fouling of some sort. It may be due to algae and/or slime growths in heat exchangers or condensers due to improper or insufficient chlorination of the cooling water supply. Chlorine demand can vary from season to season and healthy shots once or twice a day over a 1 to 2 hour period is preferred to continuous chlorination. Heat reduction can also result from deposits of hardness scale on hot boiler surfaces or corrosion products, commonly referred to as crud, depositing on fuel element surfaces causing an increase in fuel temperature.

4. Domestic Water Supplies: Regulations have been established by Ontario Hydro, Departments of Health and the Ontario Water Resources Commission controlling the chlorination of water supplies for human consumption where municipal supplies are not available. In most cases this presents no problem except for changing chlorine dose rates to meet changing water conditions and sending samples to the Department of Health for bacteriological examination. The chemical unit may also be required to provide samples for O.W.R.C. where sewage effluent is being discharged to open bodies of water.

5. Radioactivity: Radioactivity is a natural occurrence in a nuclear-electric station and shielding is designed for the predicted radiation. However, unpredictable impurities in water systems that are subject to neutron radiation will become radioactive and contribute to radiation fields to the extent that if they were not removed, extra shielding would be required or areas would become inaccessible. These impurities may be corrosion products or trace impurities added with other chemicals.

The sodium - 24 detected in NPD heavy water systems is believed to be due to small amounts of sodium in the lithium hydroxide added to the PHT system. Similarly sodium could be introduced at Douglas Point as well as impurities with the boric anhydride. It might be noted here that any impurities with boron are likely to be cations which will necessitate a 100% cation exchange column while boron is being used extensively during early operation.

6. Water Treatment: Water treatment is important in that pure water must be available for boiler-feedwater, storage bays and light water systems subjected to neutron radiation. Closed cooling circuits will also require pure water for corrosion control. Failure to provide sufficiently pure water can lead to more complicated boiler water chemistry as will be shown later as well as a number of other problems such as corrosion and radiation. Water treatment problems will vary as the raw water supply. The humic acids of the Ottawa River resulted in NPD taking its raw water supply for water treatment from a spring. The changing lake conditions appear to be the major problem at Douglas Point for coagulation control.

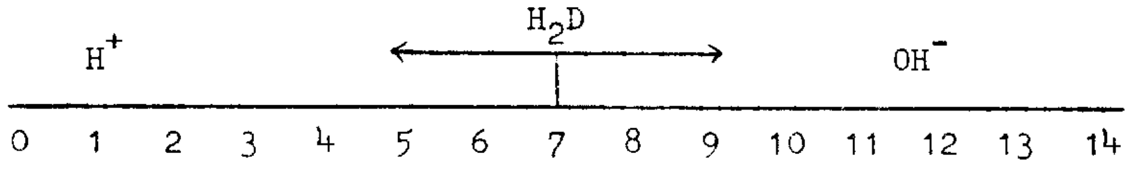
7. Gas: Gases, mainly dissolved, can lead to a number of problems. The corrosion effect of oxygen and carbon dioxide was discussed above. High gas concentrations in high pressure systems such as the Primary Coolant can result in gas locking in the lower pressure auxiliary circuits due to gas coming out of solution at the lower pressure. Deuterium and oxygen in the moderator systems due to radiolysis of heavy water must be controlled to prevent explosive concentrations of deuterium accumulating. This is done by recombining the deuterium and oxygen to heavy water in catalytic recombiners. Corrosion control and protection of the magnatite coating in the high pressure - high temperature systems is maintained with some free deuterium or hydrogen and no oxygen.

pH and Conductivity

Two terms, pH and conductivity, are used extensively when referring to nuclear-electric water systems. These should probably be defined before discussing the chemistry of the main water systems.

pH

Exactly defined, pH is the logarithm of the reciprocal of the hydrogen ion concentration. A more simple explanation is that a number between 0 and 14 denoting various degrees of acidity or alkalinity. Neutral water has a pH of 7. Values below 7 and approaching 0 are increasingly acid while values from 7 to 14 are increasingly alkaline.



To return to the original definition, a pH of 6 would indicate that a solution contained 10^{-6} grams of hydrogen per liter whereas a pH of 4 means 10^{-4} grams hydrogen per liter. Thus a change of 2 in the pH scale means a change of 100 in the hydrogen or acid concentration or alkalinity if we were referring to the upper end of the scale. A 0.5% solution of sulfuric acid would be approximately pH 1 while a 0.4% solution of sodium hydroxide would be about pH 13. It might be of interest to note that where the neutral point of light water is pH 7 the equivalent pH or more correctly pD of heavy water is 7.36 due to difference in dissociation constants.

It might be noted that pH measurements are not reliable in pure or low conductivity water. This is true for in-line pH meters and more particularly laboratory readings where the sample is open to the atmosphere and is affected by atmospheric gases.

Fig. 1 shows the relationship between conductivity and pH (or pD) in D_2O, for a pure acid below pH 7 and lithium hydroxide above pH 7.

Conductivity

Specific conductance is defined as the conductivity of 1 cubic centimeter of solution and is the reciprocal of the resistance of a solution to current flow measured between electrodes 1 sq. cm. in cross section and 1 cm. apart. "Mho" is the basic unit of conductivity and is actually ohm spelled backwards. Since we are usually dealing with pure solutions which would have a high resistance and hence a low conductance, most conductivities will be expressed in micromhos which is millionth of a mho or is equivalent to a specific resistance of 10^6 ohms. Conductivity provides a rapid means of measuring the purity of the water either by in-line instruments or by taking samples to the laboratory. It is not specific for any one ion but rather a measure of the total ionic concentration. Changes in conductivity will indicate changes in the system and can serve as a warning that further investigation is required. An example of this took place at NPD where a lithium hydroxide concentration

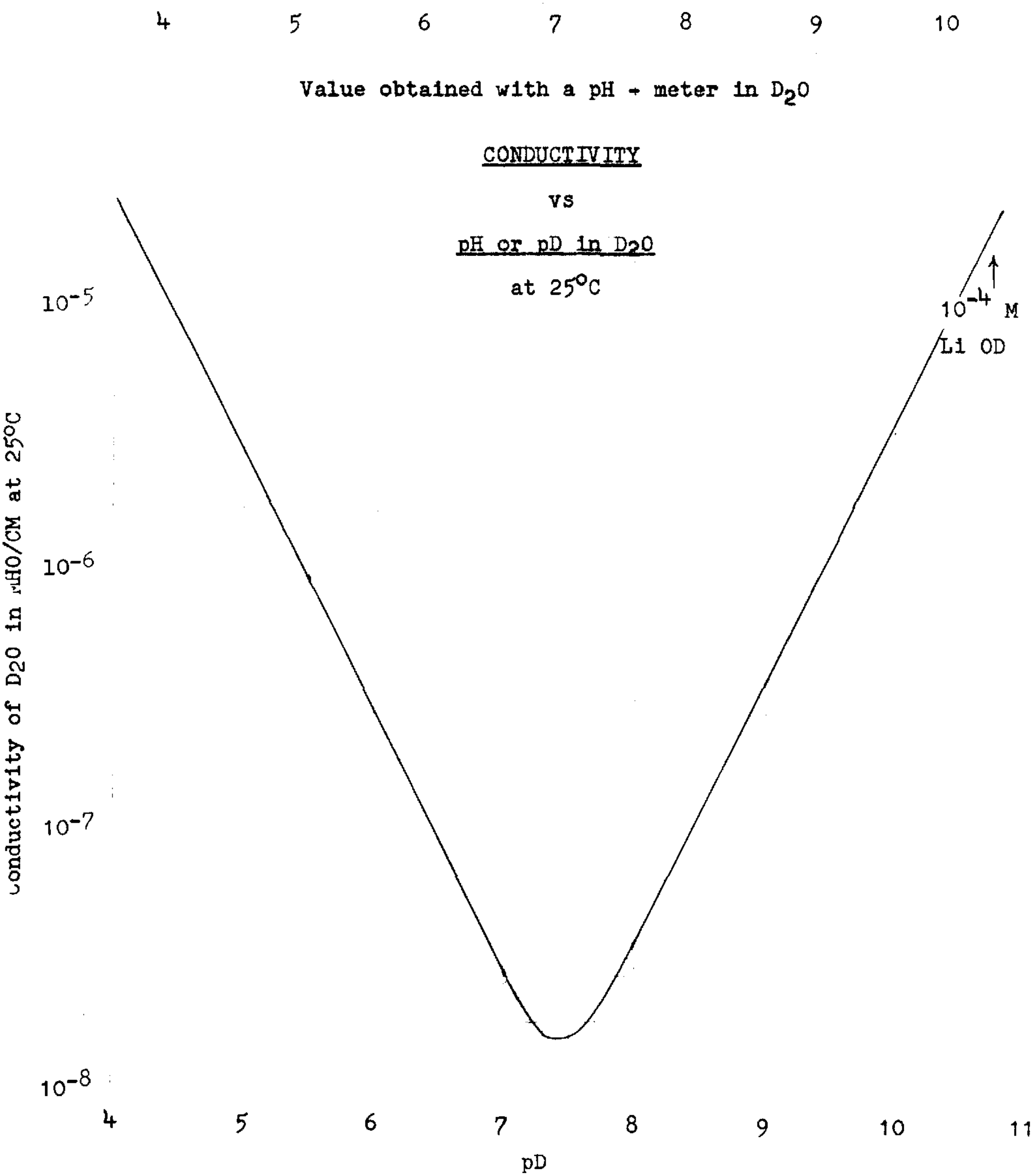

4
5
6
7
8
9
10
Value obtained with a pH - meter in D2O
CONDUCTIVITY
vs
pH or pD in D2O
at 25°C
10-5
10-6
10-7
10-8
10-4 M
Li OD
onductivity of D2O in MHO/CM at 25°C
4
5
6
7
8
9
10
11
pD

Fig. 1

of about 10^{-4} molar is maintained in the PHT system to give a pH of about 10. The conductivity was observed to gradually drop whereas no change in the lithium concentration could be detected. The cause was found to be the formation of lithium bicarbonate and carbonates from the radiolytic breakdown of oil or chemical grout. The effect of the carbonates on conductivities is shown below where the specific conductances of the three lithium solutions are listed. In all cases the lithium concentration is 10^{-4} molar.

$LiOH$	(hydroxide) ----------------	23.7 micromhos
$LiHCO_3$	(bicarbonate) ---------------	8.3 micromhos
Li_2CO_3	(carbonate) ----------------	10.3 micromhos

HEAVY WATER CHEMISTRY

The chemistry of the two main heavy water systems, moderator and pressurized coolant is similar in that the three basic problems, radiolysis, corrosion product transport and purification are common to both systems although one problem may be more significant to one system than the other.

HEAVY WATER MODERATORS

1. Radiolytic Reactions in Moderators

Radiolytic reactions may involve either water alone or both water and other components of a system. These reactions have an important bearing on the design of systems and the choice of materials.

(a) Water Decomposition

When heavy water is irradiated equilibrium concentrations of dissolved D_2 and O_2 are rapidly established. The D_2 and O_2 concentrations depend on the ionizing density of the radiation, the temperature and concentrations of impurities. The amount of D_2 is usually slightly more than twice that of O_2 because O_2 is taken up in corrosion reactions and in the oxidation of any organic impurities. The helium present strips the D_2 and O_2 from the water and if no action were taken, their concentrations in the gas phase would eventually reach equilibrium with the dissolved gas concentrations and net decomposition would stop. The gas, however, would contain 10-50 per cent D_2 and 5-25 per cent O_2 and would be potentially explosive. The helium is, therefore, recirculated through a palladium catalyst bed to convert the D_2 and $O2$ to D_2O at a rate to maintain the D_2 concentration in the system below 4 per cent. Should a high D_2

concentration occur with insufficient O_2 for complete recombination, O_2 is usually added.

(b) Nitric Acid Formation

When mixtures of nitrogen, oxygen and water vapour are irradiated, nitric acid is formed. Where the helium is passed through carbon adsorbers to remove nitrogen, nitric acid production is small. However, in NPD with no adsorbers, air leaking into the subatmospheric gas space causes high nitrogen concentrations in the helium system. The nitrogen concentration is normally kept in the range 15-30 per cent by purging with helium and acid formation is about 1 gram mole/day. An economic balance has been struck for NPD between the cost of helium to purge the nitrogen from the system and the cost of ion exchange resin columns to remove the nitric acid formed.

(c) Organic Decomposition

The presence of oil or other organic material in the water will result in the production of carbonates by oxidation of the organic material. Normally this oxidation will be slow in the cold moderator but the presence of excess air due to air inleakage will increase the rate of production. The presence of oil has caused no operating problems in NPD other than an increase in the rate of consumption of ion exchange columns. Where carbonates and nitrates are present together as in NPD, it is difficult to remove the carbonates completely even though the source of organic decomposition has been eliminated, without sacrificing an expensive ion exchange column. When a new column is placed in service, the carbonates are removed from the system in about two days. However, as nitric acid is produced the nitrates displace the carbonates from the resin and at the end of the useful life of the column for nitrates all the carbonates are back in the system.

2. Corrosion Product Transport in Moderators

Release of corrosion products to the relatively cold moderator is very small except for aluminum in aluminum calandrias and presumably iron in stainless steel units. A small amount of alumina may accumulate in D_2O if neutral conditions are maintained but this is not troublesome. There are some activation products of aluminum and stainless steel but these have given no difficulties.

3. Purification of Moderators

Heavy water moderators are readily maintained at a very low impurity level by a continuous by-pass flow through a mixed bed ion exchange resin column. These beds remove all ionic materials and also have a high efficiency for removing suspended solids. Where large amounts of anions such as carbonates and nitrates are to be removed two beds may be used in parallel, one being 100% anion resin and the other mixed or 100% cation. Alternately the mixture may be varied in a single column to suit the operating conditions. In aluminum systems it is better to operate at a pD of 5.5 to 6 rather than with pure water and so minimize corrosion.

Stainless steel systems are normally operated with a pH, neutral to slightly alkaline. However, the boron removal being a determining factor with CANDU reactors it may be necessary to maintain the moderator in the pH range 5 to 6 for more efficient boron removal. In which case evidence of corrosion will have to be watched closely. No significant increase in corrosion is expected from the boric acid used for poison control since it is a weak acid. A solution of 2 grams per liter (2000 ppm H_3BO_3) would have a pH of about 5.2.

Chloride ion must be controlled in stainless steel systems since oxygen in combination with very small concentrations of chloride (0.1 ppm) can cause severe stress corrosion cracking problems.

PRESSURIZED HEAVY WATER COOLANT

1. Radiolytic Reactions in Coolants

Water Decomposition

At the operating temperatures of 250 to 300 C, there is negligible decomposition of water and only low concentrations of dissolved D_2 and O_2 form. However, any dissolved O_2 should be avoided to prevent pitting of carbon steels or alkaline leaching of chromates, manganates, etc., from stainless steels. Oxygen attack is more liable to occur at ambient temperatures than at operating temperatures. Oxygen concentrations are readily maintained below the detectable limit (0.01 ppm) by adding excess dissolved D_2. Hydrogen (or deuterium) concentrations in the range 10-50 cm^3 (STP)/kg water are commonly employed in American PWR's to prevent coolant decomposition. However, NPD and CRNL loops have operated with deuterium concentrations in the order of 5 cm^3/kg D_2O, with no adverse effects even at 1 cm^3/kg D_2O. That is - no conversion of the iron oxide carried by the water from Fe_3O_4 (magnetite) to Fe_2O_3 (ferric oxide) has been detected. If required, deuterium can be added from cylinders but this is expensive ($1.00 per liter in 100 liter quantities). A hydrogen generator suitable for deuterium is now on the market for less than $5.00 U.S. that will produce pure deuterium at pressures up to 60 psig. Operating cost would be less than $0.05 per liter deuterium including the cost of the heavy water.

Organic Decomposition

Organic matter such as oil, resin beads or chemical grout as in the case of the December 3rd, 1962 incident at NPD will oxidize more rapidly in the high temperature pressurized systems than in the cooler moderator. Here, H_2 (or D_2) will be formed in amounts equivalent to the CO_2 produced. Although the H_2 (or D_2) is beneficial to the system, the CO_2 makes pH control difficult as will be discussed later under purification.

2. Corrosion Product Transport

High temperature water reacts with metals in varying degrees to produce corrosion products or what is referred to as "crud". In the case of steels this product of the corrosion is magnetite, Fe_3O_4, which also contains oxides of the other constituents of steel, Cr, Ni, Mn, etc. Some of the corrosion products remain on the surface of the metal and form a protective coating which reduces the corrosion rate, while some are released to the water and are transferred throughout the system in particulate or colloidal form. This crud release to the water can create several problems.

(a) it can accumulate in places causing reduction in flow rate. If this occurs in fuel channels the temperature of the fuel will rise.

(b) it can cause wear and malfunction of moving parts made to close tolerances.

(c) it can deposit on fuel surfaces causing a change in heat transfer properties and increasing the temperature of the fuel.

(d) it can be activated in the neutron flux and then deposited in the out-reactor parts of the system causing radiation problems when maintenance is required.

The most serious of these problems is that of deposition on fuel element surfaces. When a system is faced with a crud problem care should be exercised during start-ups due to crud bursts from stagnant parts of the system. It may even be necessary to hold power, i.e., fuel temperature down until sufficient crud is removed from the system by filtration and settling, to prevent large amounts of crud depositing on the hot fuel surfaces.

It has been found that by operating pressurized water systems, stainless steel and carbon steel, with 10^{-4}M lithium hydroxide in the water to give pH 10 (or pD 10.7) the concentration of corrosion products in the water remains low and their deposition of fuel-sheath surfaces is small.

Crud concentrations are obtained by in-line filtration of a cooled side stream through a 0.45 micron (1.7×10^{-5} in) filter. Soluble and colloidal corrosion products are difficult to determine. Some may be determined by gamma spectrometry if their activation products have sufficiently long enough half-lives and are not masked by other radionuclides.

Lithium hydroxide has been selected for pH control in pressurized water systems since it presents no radiation problems. The resulting tritium from the $Li^6\ (n,\alpha),{}_1H^3$ reaction is not significant in heavy water systems. Sodium or potassium hydroxides would result in high energy gamma radiation from Na-24 or K-42. Unforturnately, a small amount of light hydrogen is added with each lithium hydroxide addition. Lithium oxide is not commercially available and lithium metal is too difficult to handle being more reactive than sodium metal. The use of ammonia for pH control has been investigated but crud levels were found to be considerably higher than with lithium hydroxide and deposition of crud on the fuel was somewhat greater.

3. Purification of Coolants

The purification of pressurized water systems is normally a small by-pass flow which is depressurized and cooled before passing through the ion exchange purification columns. For pH 10 operation the exchange sites of the cation resin are in the lithium form instead of hydrogen (or deuterium) in the case of moderator columns. In this way not only is lithium exchanged for any soluble cations removed but the lithium added to the system for pH control is not removed by the ion-exchange resin. The resin bed removes impurities such as ammonia, carbonate, silica, etc., and any soluble corrosion products. Also, soluble fission products should they occur. Normally the ion-exchange bed is a 1:1 mixture of anion and cation resin. This can be adjusted to meet existing requirements. For example, a 100% anion resin bed may be on standby to remove carbonate periodically when there is sufficient lithium in the system but the pH is low due to lithium carbonate. Such a column would only be used during such times as it was necessary to remove carbonate.

The synthetic organic ion-exchange resins currently in use are strictly temperature limited to prevent damage to the resin. The cation type resin being used in nuclear stations has a maximum operating temperature of 250 F and anion resin a maximum of 140 F. Thus a mixed bed is limited to a maximum temperature of 140 F. High temperature operation ion exchangers such as zirconium phosphate and zirconium oxide which have been operated up to 590 F may, in the future, allow for high temperature high pressure purification.

To get the maximum life from a resin column it should be preceded by a filter to remove any crud from the water. The resin column itself acts as an efficient, although expensive, filter and if particulate material is not removed from the coolant before it enters the column it will collect on the resin, reducing its exchange capacity and shortening its effective life. Also, a strainer should be installed downstream from the column to trap resin beads in the event of a resin bed screen failure. Such a failure occurred at NPD a very short time after a strainer was installed.

BOILER WATER CHEMISTRY

Boiler water chemistry must meet two requirements. Corrosion to the boiler metal and associated piping must be prevented or held to a minimum and contaminants such as hardness, silica, copper and iron must be treated to prevent scale formation, silica deposits and further corrosion. The only 100% safe way to operate a high-pressure boiler is to keep out all contaminants and keep the boiler water absolutely pure. Few plants can afford to approach 100% safety because expensive polishing equipment would be required in the condensate. Where more than one contaminant is present, the optimum treatment must be selected to control contaminants likely to give the most trouble if ignored.

The first step towards long-run boiler economy is to reduce corrosion in the pre-boiler cycle. This is done by keeping oxygen out of the system by means of tight shaft seals on pumps or removing it by deaerators. The small amount of oxygen that does enter the system or which cannot be removed is treated by feeding sufficient hydrazine into the feedwater cycle to react with the dissolved oxygen.

Boiler water chemistry has been under considerable investigation in recent years to meet the exacting demands of high pressure boilers. As pressure has increased, specifications have become more stringent with less and less solids and chemicals being permitted in boiler water. Table 1 illustrates the limits for total solids and suspended solids as related to operating pressures.

TABLE 1

Solids Content in Boiler Water

Operating Pressure psig	Total Solids ppm	Suspended Solids ppm
0-300	3500	300
301-450	3000	250
451-600	2500	150
601-750	2000	100
751-900	1500	60
901-1000	1250	40
1001-1500	1000	20
1501-2000	750	10
2001 and higher	500	5

1. Pre-Boiler Chemistry

Hydrazine removes dissolved oxygen in accordance with the following reaction:

$$N_2H_4 + O_2 = 2H_2O + N_2$$

hydrazine + oxygen = water + nitrogen

Since the products of the reaction are water and nitrogen, no solids are added to the boiler water. Excessive hydrazine can result in ammonia and nitrogen by decomposition. The ammonia is alkaline and therefore will not attack steel. However, if present in sufficient quantity, it can attack copper bearing alloys when oxygen is present.

Theoretically, 1.0 ppm hydrazine is required to react with 1.0 ppm of dissolved oxygen. In practice hydrazine is usually fed at 1.5 to 2.0 parts of oxygen. To avoid excessive ammonia concentration in the steam, the residual hydrazine concentration in the boiler water is usually maintained at 0.1 ppm or less.

In some instances both sodium sulfite and hydrazine is used. In these cases a very low sulfite excess is carried in the boiler water so that the potential for corrosion from acidic gases is minimized but a reserve of deaerant is available to react with an oxygen surge. The reaction of sulfite with dissolved oxygen is:

$$2Na_2SO_3 \quad + \quad O_2 \quad = \quad 2Na_2SO_4$$

sodium sulfite + oxygen = sodium sulfate

The removal of 1.0 ppm dissolved oxygen theoretically requires 7.88 ppm of chemically pure sodium sulfite. Both the sulfite and sulfate contribute solids to the boiler.

The effect of carbon dioxide towards corrosion is overcome by the addition of one of the volatile neutralizing amines. The fact that the amines are volatile results in protection to the entire boiler feedwater system since the concentration in the steam is the same as the condensate. Neutralizing amines, however, do not protect return lines against oxygen attack. Of the neutralizing amines available, morpholine is the most commonly used. Approximately 1.5 ppm morpholine is required for each 1.0 ppm carbon dioxide to elevate the pH to 7.0.

Filming amines are a more recent means of boiler feedwater protection. Unlike the neutralizing amines, the filming amines do not neutralize the carbon dioxide, but form an impervious, non-wettable film on the metal surfaces that act as a barrier between the metal and condensate thus protecting against both oxygen and carbon dioxide attack. The film formed is substantially monomolecular thickness and does not increase in thickness with continued treatment. This amine will penetrate corrosion products present on a metal surface and deposit a protective film on the metal.

2. Boiler Chemistry

There are few cases where contaminants are not present to some degree. Although the contaminant may not seem to be significant in the make-up water it can become concentrated in the boiler to the extent that difficulties can be encountered. Where contaminants are present the type of boiler water treatment will be determined by the contaminant most likely to give trouble if ignored.

In Table 11 are listed the various factors requiring boiler water treatment and the relative merits of each of the methods. These methods will be described later. Table 111 gives six hypothetical situations where one type of contaminant predominates, plus the most suitable treatment based on the ratings in table 11. The disadvantage of the particular method is also listed.

There are four basic types of boiler water treatment as shown in Tables 11 and 111. These are:

Table 11

Relative Boiler Water Treatment

Significant factors	Precision control	Coordinated phosphate-pH control	Congruent control	Pure water
Corrosion	4	3	2	1
Carryover	3	4	2	1
Buffering ability	1	1	1	4
Silica volatility	1	2	3	4
Turbine deposit solubility	1	2	3	4
Scale formation	1	1	2	4
Sludge adherence	1	3	2	4
Oxide formation rate	1	2	3	4
Aluminum solubility	4	3	2	1

(1 is most favourable, 4 is least favourable)

Table 111

Recommended Boiler Water Treatment for Various Contaminants

Predominant Contaminant	Best Treatment	Disadvantages of Treatment
Aluminum	Precision control Coordinated phosphate - pH control	Promotes gouging Promotes gouging and carryover
Silica	Precision control	Promotes gouging
Cation leakage	Congruent control	No protection against silica or sludge
Anion leakage (other than silica)	Congruent control	No protection against silica or sludge
Condenser leakage	Precision control	Promotes gouging
Corrosion products	Pure water	No protection against silica or scale
None	Pure water	None

1. Precision control
2. Coordinated phosphate - pH control
3. Congruent control
4. Pure water

The first three methods employ the use of orthophosphates and sodium hydroxide to varying degrees while the fourth as the name implies involves no chemical treatment other than corrosion control chemicals fed into the feedwater, namely, hydrazine and morpholine. Pure water control has been proposed for Douglas Point.

Precision Control

This is the oldest of the chemical treatments for modern high pressure boilers. A residual of 2-4 ppm orthophosphates and 15-50 ppm sodium hydroxide is maintained in the boiler. This method keeps the generating surfaces free of magnesium phosphate sludge while reducing soluble silica concentration in the boiler water. However, as boiler pressure increased the metal temperatures also increased and it was found that when solidum hydroxide concentrated on hot surfaces or beneath corrosion products, it not only failed to restart corrosion but accelerated it. This caustic attack was characterized by large pits or grooves on the metal.

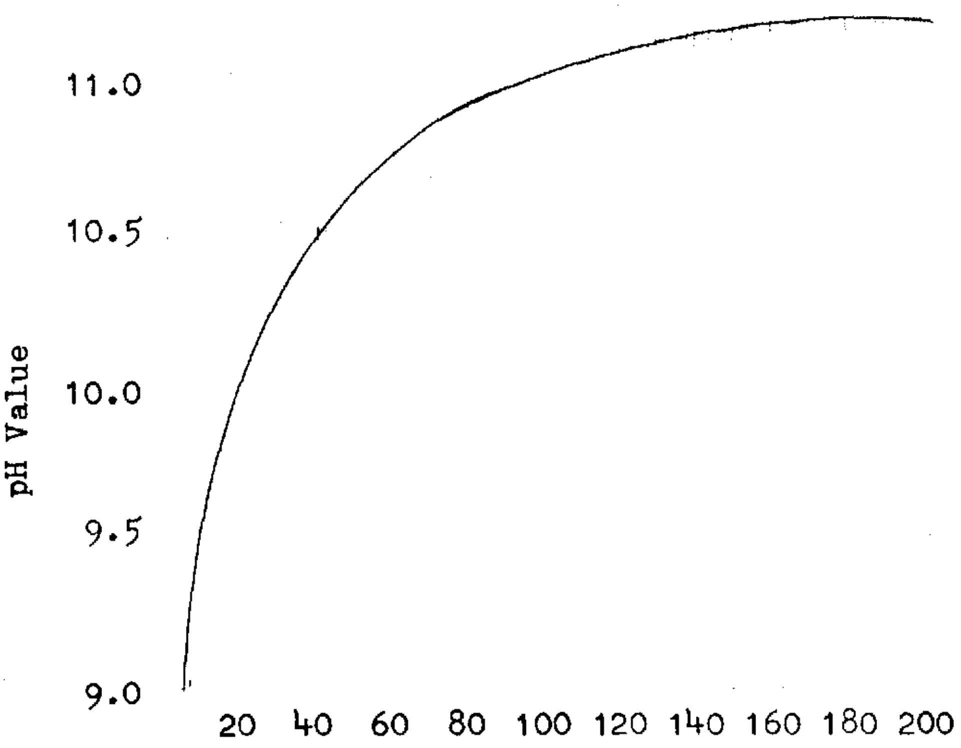
Coordinated Phosphate - pH Control

This method of treating boiler water was developed to prevent caustic embrittlement. The boiler water is kept below a curve, Fig. 2, representing phosphate - pH coordinates for trisodium phosphate. Theory is that caustic attack cannot occur without free caustic. However, caustic damage did show up and carryover was encountered in the 100 to 200 ppm phosphate concentrations that were originally used. Phosphate concentrations have been reduced so that the recommended upper phosphate level is between 10 and 15 ppm. However, it has been shown that being just below the theoretical curve may not be enough since sodium hydroxide can be produced from sodium phosphate under conditions of hideout.

Congruent Control

Congruent control - where the composition of both the liquid and solid phases are the same, has been developed to avoid the the production of sodium hydroxide under hideout conditions. In

Approximate pH Values of Trisodium Phosphate Solutions



Phosphate as PO_4 (ppm)

Fig. 2

this method of control, the mol ratio of Na-PO_4 is maintained between 2.85 and 2.6. The mol ratio of Na-PO_4 in trisodium phosphate is 3 and the upper curve in Fig. 3 is actually the low end of the curve in Fig. 2. A 2-4 ppm phosphate residual is maintained in the boiler and using an arbitrary minimum pH of 8.5 the shaded area in Fig. 3 designates the operating region for congruent control. Depending on the contaminants entering the boiler, control can usually be held within the area by adding phosphoric acid, monosodium phosphate, disodium phosphate, trisodium phosphate, sodium hydroxide or a mixture of these materials.

Pure Water Control

Pure water control is employed when only corrosion products are the contaminants. It simplifies boiler water treatment in that no chemicals are added other than morpholine and hydrazine to the feedwater cycle. However, this method provides no protection for contaminants entering the cycle. Condenser leakage leads to scale and silica problems, cation leakage to the

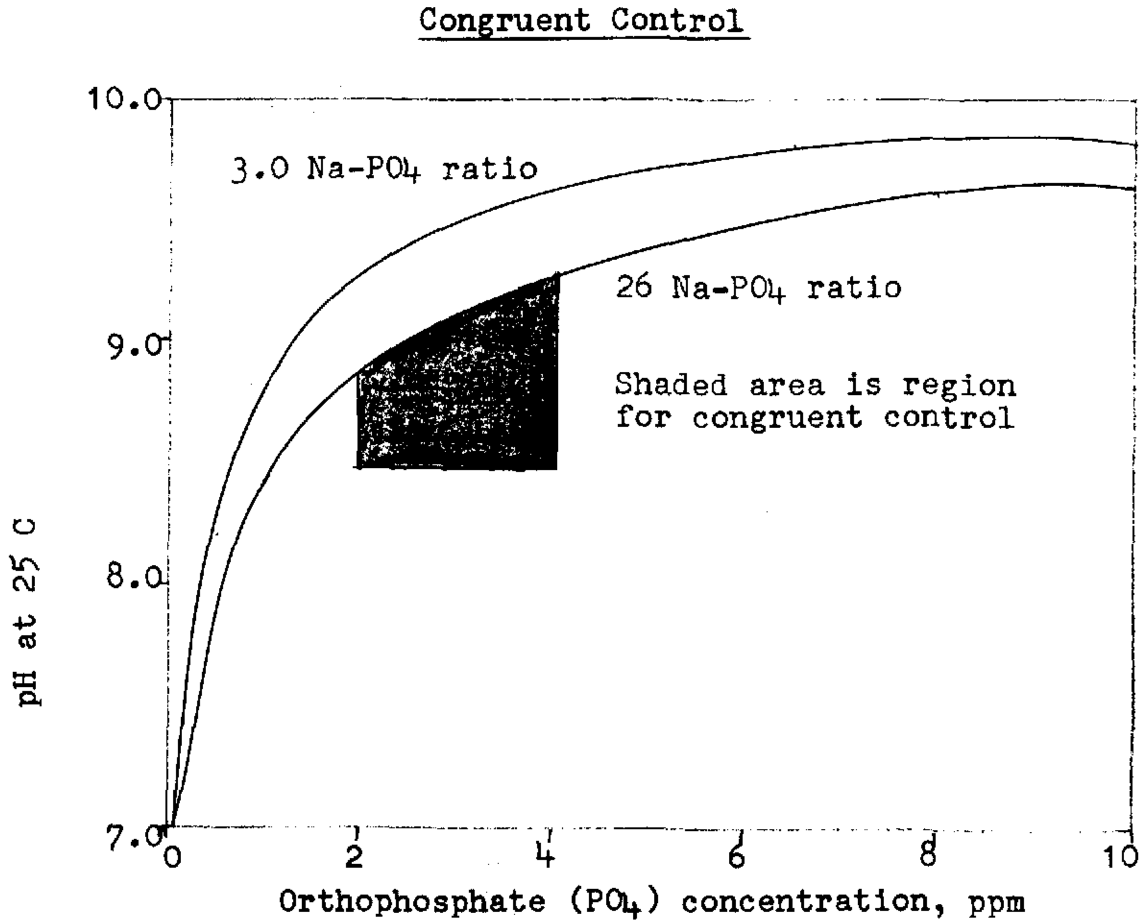


Fig. 3

development of caustic and anion leakage to acidic boiler water. With this method not only **must** chemicals be withheld but extreme care is necessary to keep the water pure.

Mention might be made of two contaminants that must be carefully controlled, namely, silica and chloride. Silica from certain compounds can volatilize from the boiler and deposit on the turbine blading. The water treatment plants must be operated as directed, observing the maximum gallonage for the cycle even though the effluent from the unit appears satisfactory as far as conductivity is concerned. Silica compounds have a very low specific conductance and are therefore not detected by conductivity cells in trace quantities. Fig. 4 shows the maximum silica concentration in boiler water to limit silica in steam to 0.02 ppm as related to boiler pressure.

Silica can be controlled by blowdown or by one of the phosphate treatment procedures discussed previously.

Silica VS Steam Drum Pressure

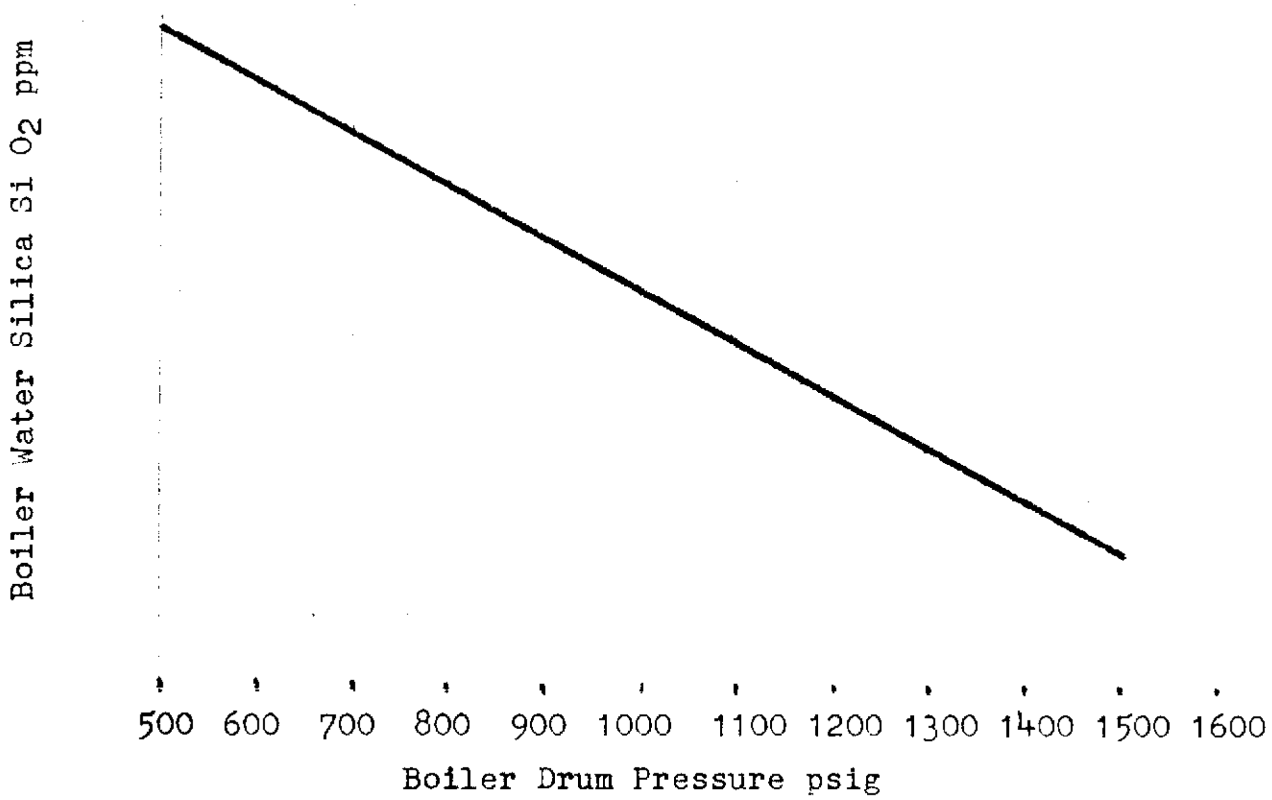


Fig. 4

Chloride is a contaminant that must be given serious consideration due to its corrosive properties. Control is by blowdown since there is no chemical means of complexing the chloride in the boiler. The source of chloride is usually condenser leakage.

In summary, what high pressure boiler operations need is a good, inexpensive, high-temperature corrosion inhibitor, giving the advantages of phosphates and caustic without having their disadvantages.

CHEMICAL SAMPLING AND ANALYSIS

Proper chemical conditions for the satisfactory operation of the water circuits in a nuclear-electric generating station can only be maintained by chemical sampling and analysis of the variables involved, either by laboratory analysis or by monitoring with in-line instruments.

Regardless of how much care is taken in carrying out the analysis or the accuracy of the method, the result can only be as good as the sample. Unless a truly representative sample is taken, the result will have little meaning and can, in fact, lead to adjustment of chemical conditions that are not necessary.

Sample points should be provided so that good samples can be taken from all major systems, and where ion exchange columns are being used, before and after sample lines are required.

Three types of samples and analyses will be discussed, heavy water, light water and gas.

HEAVY WATER SAMPLING

Special techniques are required in heavy water sampling for two reasons:

(a) High grade heavy water is very hygroscopic and can be easily downgraded with light water vapour if exposed to air. This is especially true when sampling for isotopic purity.

(b) Heavy water exposed to a neutron flux in a reactor will contain tritium which is an internal radiation hazard as the heavy water vapour can be easily inhaled or absorbed through the skin.

(1) Manual Sampling

Manual samples are required where the sample must be taken to the laboratory for analysis or processing. Manual sampling may be by:

(a) Hypodermic Syringe

Heavy water is allowed to circulate through a stainless steel pot which is sealed with a neoprene rubber diaphragm. By this method sample lines can be flushed through a closed system without loss or downgrading. Samples are taken with a standard hypodermic needle and syringe by puncturing the diaphragm and drawing

(a) Hypodermic Syringe (cont'd)

the samples into the syringe. The diaphragm is self-sealing when the needle is withdrawn. When larger samples are required from such a sample point, 3-way stopcocks are available that can be fitted between the needle and syringe to which is attached a length of dry tubine.

(b) Bomb Sampling

Bomb samples are taken when the amount of dissolved gas is to be determined at system pressure. For primary coolant samples, the water must be cooled before passing through the bomb for handling purposes. It is preferred that high pressure couplings similar to those being used with the Crud Filter, Fig. 5, be used for high pressure bombs instead of swagelok fittings. The former can be disconnected by hand or with a minimum of tools. Couplings for the lower pressure bombs, such as Moderator, are readily available and are in common use at C.R.N.L. Samples are normally processed in a vacuum system where the gas is extracted from the water and then either analyzed in the system or by gas chromatography.

(c) Miscellaneous Sampling

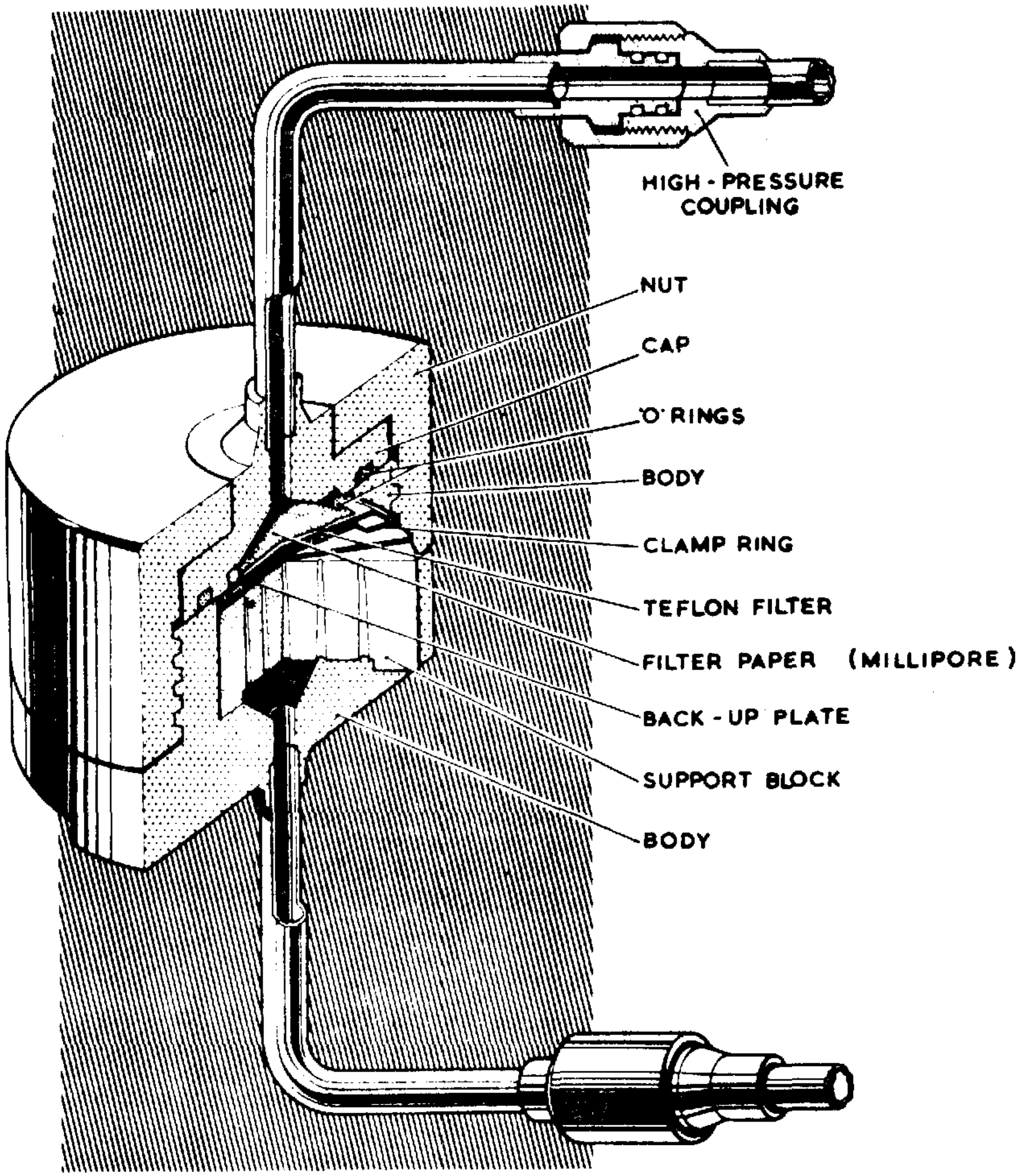
Other heavy water samples are taken as required for a variety of analyses including aluminum, iron, chloride, boron, nitrate by spectrophotometric methods, pH, carbonate, oil, radioactivity, etc. In many cases the heavy water is contaminated with chemicals to the extent that recovery is not practical. In some cases archive samples of Moderator and Coolant are kept for comparative purposes.

2. In-Line Sampling

Some variables are measured continuously by instruments installed in the system and recorded or indicated in the Control Room or on local charts. These include pH, conductivity, boron, and isotopic purity. Crud and dissolved oxygen are considered to be in-line but a certain amount of manual manipulation is required.

(a) pH

As mentioned previously, pH measurements of pure water are not reliable and this refers particularly to Moderator. During



CRUD FILTER

Fig. 5

early operation of Douglas Point when the Moderator will have boric acid for poison, pH could be of some value. Later, when the Moderator is operating "clean" pH will have little meaning. However, a continuous pH measurement of the primary coolant where lithium is added for pH control will, along with a conductivity measurement, give a good indication of the condition of the system chemistry. PH meters should be installed upstream of ion exchange columns to remove any chloride ions that may leak from the potassium chloride solution of the pH system.

(b) Conductivity

Conductivity measurements are the best indication of Moderator purity. This applies to NPD and should apply equally well to Candu reactors since the boric acid is only slightly ionized and is not expected to contribute appreciably to the conductivity. In coolant systems, conductivity measurements are used in conjunction with pH to control the system chemistry.

Normally, conductivities are measured before and after ion exchange columns to determine the effectiveness of the resin. This is valid for clean systems such as the Moderator where the effluent conductivity will be lower than the influent if the resin is operating satisfactorily. If the water is very clean and contamination is slight, there will be very little difference. In coolant systems, conductivities are much higher and effluent conductivity from the ion exchange column may even be higher than the influent if anions such as carbonates and bicarbonates are being removed. If the coolant is clean, except for lithium hydroxide, there will be little or no difference between inlet and outlet conductivities and an alternative method must be used to determine when the resin requires changing.

(c) Isotopic Purity

The isotopic purity of a heavy water stream can be determined by passing a cooled sample through a quartz or calcium fluoride cell. The instrument compares the infrared light absorbed by the sample to that absorbed by a known standard and reads out % D_2O. In the 99.7% D_2O range the instrument will indicate within 0.005% D_2O.

(d) Boron

An in-line boron analyzer is to be installed in Douglas Point for poison control. The instrument operates on the colorimetric or spectrophotometric principle. It has been checked out at CRNL but there has been no operating experience with this instrument.

(e) Crud

The Crud Filter, Fig. 5, is installed so that flows may be continued for extended periods of time, e.g., overnight if the crud level is low. Associated equipment includes a cooling coil, flow orifice, D.P. cell, transmitter and counter to record volume flow. It is preferred to have the filter mounted in a dry box because of tritium hazard as it is impossible to remove crud samples without spilling some water. The processing of the crud sample and final weighing is carried out in the laboratory. The crud colour is a good indication of the condition of the coolant piping. Black crud indicates the desired Fe_3O_4 or magnetite, whereas a reddish crud indicates Fe_2O_3 of the presence of oxygen in the system. Colours varying from brown, red, green, yellow and black were seen during start-ups at NPD after chemical grout had entered the system.

(f) Dissolved Oxygen

Dissolved oxygen concentrations down to 0.005 ppm O_2 can be detected by a thallium type analyzer used for loop chemistry by CRNL and a laboratory conductivity meter. Any oxygen in water passing through a column of thallium turnings reacts with the thallium to form soluble thallium hydroxide. The difference in conductivity before and after the thallium column is due to thallium hydroxide and thus directly related to the oxygen. A small ion exchange column removes soluble ions in the influent to increase sensitivity while a second column removes thallium to prevent it going into the system.

LIGHT WATER SAMPLING

The precautions required for D_2O sampling are not applicable to the light water systems in our station. However, certain precautions must be taken for specific samples. For example:

(a) Air bubbles must be excluded from samples for dissolved oxygen

(b) Samples for sulfite and hydrazine must not be exposed to air because of oxidation

(c) Conductivity measurements on low conductivity systems should be taken locally in flow cells to avoid contamination from atmospheric absorption.

1. Manual Sampling

Manual samples are normally taken in clean plastic or glass bottles that are well rinsed with the sample. Since many light water samples can not be circulated as in the case of D_2O sample pots, care must be taken that all hold-up is flushed from the line and that the sample collected is actually that of the system at the time. Samples are taken manually for a number of analyses including hardness, hydrazine, organics, iron, aluminum, chloride, silica phosphate, morpholine, copper, solids, sulfate, etc.

2. In-Line Sampling

In-line sampling of light water systems is similar to that of heavy water and includes pH, dissolved oxygen, conductivity and isotopic purity.

(a) pH

PH of the feedwater cycle is measured and recorded. At Douglas Point, in-line pH measurements will be used to control morpholine injection. PH measurements are also recorded for the water treatment at Douglas Point. The pH measurements of the raw water or filter inlet samples will be valid but the measurements of the anion and mixed bed effluents will have little meaning because of low conductivities as was stated previously.

(b) Conductivity

Continuous conductivity measurements are provided extensively in the condensate and feedwater circuits as well as the water treatment plant. Here conductivities are a good indication as to how well the systems are operating chemically.

(c) Dissolved Oxygen

Dissolved oxygen in the feedwater is measured and recorded. The hydrazine dose rates are based on these measurements.

(d) Isotopic Analyzer

Whereas the isotopic analyzers in D_2O circuits measures D_2O purity to detect H_2O leaks into D_2O, similar analyzers are provided in light water circuits to detect D_2O leaks by measuring H_2O isotopic purity. Circuits being monitored include secondary side of the boilers, heavy water heat exchangers, gland seals, etc

3. Gas Sampling

Certain gas systems require frequent or continuous sampling. Deuterium - oxygen concentrations before and after recombination units indicate whether the units are operating satisfactorily, whether the deuterium level is dangerously high or if oxygen should be added to the system. Nitrogen concentrations will indicate air leakage into the system. The CO_2 in the reactor vault atmosphere and hydrogen for generator cooling must be checked for purity.

(a) Manual Sampling

Manual gas samples may be taken in a number of ways depending on the method of analysis. Samples for an Orsat or Burrell type analysis may be taken in a glass sampling bulb or sampled directly into the apparatus. Gases to be analyzed by gas chromatography in the laboratory can be sampled with special gas syringes similar to the hypodermic syringes used for heavy water, or in special glass sampling bulbs that are specially made to adapt to the instrument.

(b) In-Line Sampling

A process gas chromatograph may be set up to sample and analyze a number of gas streams on a timed cycle. The helium streams before and after recombination are the more important of these. Helium is used as a carrier gas in the instrument and precautions must be taken to ensure a continuous supply to prevent damage to the instrument. In addition, purity meters will monitor the CO_2 in the vault atmosphere and hydrogen in the generator cooling system.

HEAVY WATER ANALYSIS

The analysis of heavy water for isotopic purity is one of the more important analyses performed in nuclear electric stations where heavy water is used as the moderator and coolant. There are several methods for determining the isotopic purity of heavy water including density, mass spectrometry, refractive index and infrared analysis. Of these, infrared and refractive index are the two more commonly used for routine work.

The infrared method is based on the absorption of infrared light at a certain frequency. It is best suited for D_2O concentrations greater than 99.5% or less than 1%. Intermediate concentrations can be determined with decreasing sensitivity as 50% is approached. In the high range concentration maximum absorbance is read at a frequency of 3400 cm^{-1} which actually determines that light water in the sample. At the other end of the range absorbance is measured at 2500 cm^{-1} and this is the D_2O in the light water.

Refractive index is based on the fact that with reference to the D-line of sodium, heavy water has a refractive index of 1.32844 while light water has a refractive index of 1.33300. Although this difference may seem small, a good refractometer can distinguish a difference of slightly more than 1% D_2O over the entire range, providing the samples are clean. The main disadvantage to the refractive index method is that soluble salts will alter refractive index readings. Samples from unknown sources are usually distilled before being analyzed.

MISCELLANEOUS

Previous mention was made to the fact that conductivity measurements can not always be used to determine whether or not an ion exchange column is spent and requires replacing. This is applicable to primary coolant ion-exchange columns in particular. In this case the decontamination factor across the resin is measured for gross activity or for an active nuclide which is normally removed efficiently by the resin. The decontamination factor is the ratio of the influent concentration to the effluent concentration of the nuclide considered. The resin is usually changed when this factor is less than 10.

The chemical conditions of the water circuits in a nuclear-electric station change with changing reactor technology and operating conditions whether light or heavy water systems. However, the basic requirements of chemical control are minimum corrosion and minimum contaminants to enter the circuits.

C.A. McKerlie

ASSIGNMENT

1. The pD of the heavy water moderator was found to be 4.5
   (a) Would you consider this to be serious in an aluminum system, and a stainless steel system? Why?

   (b) What chemical would you likely add and how much to raise the pD to 6.5, if there were 200,000 lbs. of D_2O present at pD 4.5?

2. The analysis of the helium cover gas showed 12% D_2, 6% O_2 present and the remainder helium. Suppose that the flow through the recombination units was 20 cfm of gas and the efficiency of the unit was 80%. How long would it take to reduce the concentration of D_2 to 3% if the cover gas system held 4000 cu. ft. of gas?

3. In stainless steel systems chlorides are to be kept at very low concentrations. Why?

4. Suppose the boiler in your station held 20,000 lbs of water and had an O_2 concentration of 6 parts/million. Show by calculation the theoretical amount of hydrazine required to reduce this O_2 concentration to zero.

5. The boiler contain 20,000 lbs of water, how much trisodium phosphate must be added to precipitate 5 ppm of Mg^{++} and 7 ppm of Ca^{++} ions?

A. Williams