PHYSICS ASPECTS OF
CANDU NUCLEAR REACTORS

by

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ABSTRACT

Certain scientific-technical aspects of the CANDU nuclear energy system are discussed. Major emphasis is placed on the fuel cycle, thorium breeding, heat and mass transfer, safety, and environmental impact.

The analysis is based on a critical review of the published literature. Both qualitative and quantitative aspects are discussed and subsequently summarized.

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1. INTRODUCTION

The essence of this project is the survey of five physics aspects of the Pressurized Heavy Water CANDU nuclear reactor system (CANDU PHW). The treatment of these aspects does not aim at a fundamental and detailed analysis of the theoretical principles to which they are associated but rather the purpose is to give an analytical view of these subjects hoping to show in this way that they are closely interrelated.

The CANDU PHW nuclear system has shown to date to be an indisputable success by its realizations as well as by its possibilities even though it represents the latest entry to commercial nuclear energy activities. The origin of this success is due to a design which permits the utilization of nautral uranium as fuel and which gives a neutron economy for the attainment of high fuel burnups. The versatility of the CANDU reactor also permits its adaptation to other fuel cycles, therefore giving to the Canadian nuclear system a very strong future economic viability.

The physics aspects which we develop here are respectively the fuel cycle, thorium breeding, heat and mass transfer, safety of CANDU, and its environmental impact. Each of these aspects has been discussed separately but because of the consequences that one of them may have on the others these aspects are basically difficult to separate. The thorium cycle, for example, may be considered as one of the possible different fuel cycles of the system but it has been treated in the second

chapter in order to provide more clarity. Also, this thorium utilization aspect has been treated in less detail than the other aspects because of its complexity and its economical consequences. The subject is nevertheless important since it represents one of the future steps of the CANDU system evolution.

2. FUEL CYCLE OF CANDU

2.1 Nature of Fuel and Difference with Other Fuels

The nuclear fuel used in CANDU reactors is uranium dioxide (UO_2) extracted from natural uranium ore. A fuel bundle is formed with end plates that are welded to the end of several fuel pencils; each of them containing high-density cylindrical pellets of UO_2. Spacers are brazed to the sheaths to keep the desired separations between the pencils. A bundle consists of only the fuel material and a minimum containment envelope.

The Pickering bundle as an example, shown in Figure 1, is 92 wt% UO_2; the remaining 8 wt% is made up of the sheaths, end caps, structural end plates and spacers. Since the structural material accounts for only 0.7% of the thermal neutron cross-section of the bundle, this gives an assembly which has a high efficiency for neutron utilisation.

A number of changes in nuclear fuel design have taken place since the first charge of the NPD reactor 17 years ago. Recent fuel elements possess short bearing pads and spacers, both attached by brazing and the spacers are fastened at an angle to prevent interlocking. Other changes are in the diameter of the fuel elements and in the number of pencils in a fuel bundle; this may also vary for each reactor depending on the power and heat transfer requirements. But whatever variations have been performed or could be performed on the CANDU fuel, there are some basic performance requirements which must be taken into consideration in its

design and which may be stated in the following way:

(i) There must be sufficient fissile material (U-235) to maintain the chain reaction for a practical period of time.

(ii) The heat produced must be taken away without encountering burnout or meltout.

(iii) The endurance of the fuel must allow it to last the required life without failure which may be caused by fission gas, coolant pressure, irradiation damage and fuel swelling.

(iv) The structure must be strong enough to withstand coolant flow compression and forces imposed by fuelling machines.

(v) The dimension specifications must allow the bundles to go in and out of the coolant channels without difficulty even after long irradiation.

Some fundamental differences between the CANDU nuclear fuel and the enriched fuel used in the PWR USA reactors are listed in Table 1. Without going into too much detail, we do note that enriched fuels are more expensive by a factor 3 or 4 in total fuel costs. This difference in the fuel costs is mainly due to the enrichment process in the PWR fuel cycle. Figures 2 and 3 show both cycles for natural and enriched uranium. As can be deduced, considerable amount of unused fissile uranium still present in the PWR spent fuel has to be reprocessed. The PHW cycle on the other hand has fewer steps and does not normally claim credit for the plutonium in the spent fuel. Because of its simplicity,

the PHW cycle is also called OTTO cycle or Once-Through-Then-Out cycle. However, if the future market justifies it, the CANDU spent fuel could be sold or the plutonium recycled to be used with thorium as we will discuss in the next chapter.

We may note that since PWR fuel is full length, the whole assembly has to be discharged if any part becomes defective. For CANDU, the short fuel bundle and on-power refuelling characteristics permit the discharge of the defective bundle only.

2.2 General Features of CANDU Reactors

The main characteristics differentiating CANDU reactors from PWR or BWR systems is the use of heavy water as moderator instead of light water. Even if heavy water has a much higher moderating ratio than light water, its slowing down power is however smaller because of the higher mass of deuterium compared to hydrogen. As a direct consequence of this, the use of natural uranium for fuel implies that a larger moderator volume has to be provided in CANDU reactors than in light water reactors. The ratio of moderator to fuel volumes for CANDU is typically 20:1.

One pressure tube in the Canadian design contains nine to twelve fuel bundles placed end to end. Each pressure tube has a calandria tube around it to insulate the moderator from the hot coolant which flows inside the pressure tubes. An advantage of this type of design is that refuelling may be done at power.

Because of its large volume, the overall power density in a heavy water core is relatively low, typically 9 kWt/litre. However, the use of short fuel bundles which can be shifted in the axial direction to

flatten the flux distribution, gives a high power per unit of fissile material; approaching 3 MWt/kg of fissile material which is higher than in either fast or light water reactors. Also this procedure gives a high burnup for natural uranium fuel: typically 10,000 MWd/TeU. Since the amount of energy which can be derived from burning all the U-235 in a tonne of natural uranium is only 6000 MWd, this means that more than half the energy comes from Pu-239, Pu-241 and U-238 fissions. The fuel bundles removed from the core after producing 10,000 MWd/TeU in energy contains about 3 g of plutonium per kg of uranium. It may then become economical in the future to reprocess this plutonium, reducing furthermore the fuelling costs for CANDU reactors.

2.3 Nuclear Fuel, Structural Materials and Coolants

An ideal material as a fuel would be one with high uranium density. The UC density is higher than UO_2 but also has a higher corrosion rate in water cooled reactors; the UC utilisation as a fuel is then preferred in conjunction with organic coolants which have much lower corrosive effects. Other fuel materials which would be acceptable for water cooled reactors and having higher uranium density than UO_2 is the binary alloy U_3Si and ternary alloy uranium-silicon-aluminum. Even though all CANDU reactors to date have used natural UO_2, it could be expected that these alloys or others, will be developed and used in the future as a higher density fuel. Also, to date, fabrication and irradiation of UO_2-PuO_2 and ThO_2-UO_2 have revealed no major difficulties for future thorium, uranium, and plutonium fuel cycles.

The zirconium alloys, Zircaloy-2 and -4 used as the basic

structural material in the construction of the fuel, have been originally developed by the USA for their naval reactor program. Table 2 shows the composition of Zr-2 and Zr-4. Zr-4 has a slight corrosion and hydrogen pick-up performance advantage over Zr-2 under our coolant conditions. Other zirconium alloys have been tested as fuel sheaths for CANDU-BLW where local overpower transients could cause dry-out of the sheath with its temperature rising to 500°C. Also, ozennite, a Russian alloy of Zirconium, is being developed as a sheathing material used with organic coolants because of its better performance with these coolants.

The predominant coolant used in CANDU reactors is pressurized heavy water (PHW). Boiling heavy water (BHW) has been used in NPD for two years as an experiment and boiling light water (BLW) is used in the Gentilly-1 power reactor, where the average exit quality for the vertical core is 16.5 wt% steam. The HB-40 organic coolant, being used in the WR-1 test reactor at Whiteshell allows a higher thermal station efficiency: 39% vs. 29% for PHW. Organic coolants can be operated at higher temperatures because they have a much lower vapour pressure than water.

2.4 Refuelling and Burnup

2.4.1 Effect of Burnup

We previously mentioned that one of the most important features of the CANDU system is the high burnup attainable from the natural uranium fuel. In fact this feature is highly desirable from an economical point of view because the fuelling costs are roughly inversely proportional to the burnup. Consequently, the objective of fuel scheduling is to move fuel in and out of the reactor core in such a way that the highest average

burnup from all the fuel is achieved.

From a physical point of view, we know that if a reactor is initially loaded with fresh fuel that is kept untouched, the burnup at any point in the core after some period of operation will be proportional to the flux at that point. Hence, the burnup will be higher in the central regions where the flux is equally higher. Therefore, since the statistical weight is higher in the centre regions and since reactivity falls as burnup progresses, a point will be reached where at least some of the old fuel must be replaced with new fuel if the reactor is not to become subcritical. Then, in order to obtain the maximum burnup from the fuel, it must be distributed in such a way as to minimize the effect of burnup on reactivity. So, on-power refuelling is the answer to the non-uniform burnup caused by a non-uniform flux. In practice, 1.5-2% of the total fuel has to be replaced every week for full power operation. The on-power refuelling also helps to even-out the power distribution with time because the fuel can be shifted when the statistical weight of a region gets lower than desirable; the small length of the fuel bundles ($\sim$0.5 m) facilitates this operation.

2.4.2 On-Power Refuelling

The original concept of refuelling for CANDU reactors is to refuel one or more fuel bundles at a time by means of a pair of fuelling machines that could be attached to the ends of the horizontal fuel channels while the reactor is at full power. Another aspect of the concept is to move the fuel through alternate channels in opposite directions so that the fuel near both reactor ends is an equal mixture of fresh and nearly-spent

fuel, and thus of approximately the same average reactivity as the partially-spent fuel at the centre. As a result, this bi-directional fuelling associated with the bi-directional coolant flow, averages-out the axial lattice characteristics throughout the reactor.

This procedure puts the highest burned-up fuel elements at the ends of the reactor where they have the least effect on reactivity. For this reason, the bi-directional fuelling scheme results in a higher burnup for the spent fuel. For a CANDU PHW reactor at equilibrium, since the core ends consist of fresh and nearly-spent fuel, the burnup of the spent fuel discharged from the reactor is nearly twice the average burnup of the fuel in the core. By comparison with the case of single batch type of loading in a reactor, the average burnup of the discharged fuel is equal to the average burnup of the core.

The radial flux distribution is also controlled by the refuelling scheme. As for the axial flux distributions, if the radial distribution is also flattened in the centre of the core, the total power may be increased and a higher overall burnup is achieved. This is realized by increasing the burnup in the central channels and decreasing it in the remaining outer channels. Since burnup depends on the time spent by the fuel in the reactor, the burnup in the central channels is increased by changing fuel in these channels less frequently than if no flattening were required.

2.4.3 Transition Fresh-Equilibrium Fuel

A fresh core contains a considerable margin of excess reactivity. On the other hand, when the fuel has attained its equilibrium, the total

reactivity may be much lower due to the following effects:

(i) temperature effect on reactivity;

(ii) equilibrium of the poison loads (mainly Xe and Sm);

(iii) fuel burnup;

(iv) poison override.

Until these reactivity losses become manifest, they must be compensated. One of the solutions used to overcome these initial difficulties is to use boron as a water soluble neutron poison injected into the moderator to allow the complete filling of the calandria. As reactivity decreases the poison is eliminated by the moderator purification system. Another solution is to charge the central channels of the core with fuel depleted in U-235. This method was used for the first charge of NPD and Douglas Point. For certain reactors, the flux flattening provided by the adjuster rods may be sufficient to help the reactivity loss compensation.

Typically, for a fresh fuel charge in a CANDU reactor, during approximately the first 2 months at full power, the reactivity has increased little because of the build-up of plutonium. This may require an increase in the boron concentration until the reactivity begins to drop. After about 6 months the boron is almost completely removed but the equilibrium is not reached before nearly 1 year of full power operation. Then the refuelling rate is such that the fuel has to be replaced as fast as it is being burnt up.

The different effects on reactivity that we have mentioned above are also present during the equilibrium life of the fuel. As a matter of fact the reactivity of a reactor is almost constantly varying and the

effects of burnup, poison presence or temperature on reactivity are the basis of fuel management that we will discuss in the next section.

2.5 Basis of Fuel Management

The effects on reactivity in a reactor core are of two types: the long term and the short term effects. The long term effects arise due to changes in the fuel caused by prolonged exposure to neutrons. These changes are mainly:

(i) burnup of U-235 and buildup of plutonium;

(ii) accumulation of neutron absorbing fission products in the fuel.

The short term effects are relative to the temperature change in the reactor and will be discussed subsequently in the heat and mass transfer chapter. In this section we will consider the long term effects and their relation with fuel management.

2.5.1 Long Term Reactivity Change Due to Burnup

As we mentioned previously, the burnup of fissile U-235 and the conversion of U-238 into fissile plutonium are the two predominant effects that will change the composition of the fuel. Since the rate dN/dt at which material is destroyed by neutron capture is given by

$$\frac{dN}{dt} = -N\sigma_a\phi \quad , \tag{2.1}$$

this will depend on the neutron flux. The product $\sigma_a\phi$ in this equation is equivalent to a decay constant and the term $0.69/\sigma_a\phi$ may then be considered as the half-life of the material against removal by neutron capture.

The depletion of U-235 is offset by the conversion of U-238 into Pu-239 as shown in Table 3. The conversion ratio also depends on burnup, but for fresh fuel in CANDU reactors the initial conversion ratio is about 0.8.

The Pu-240 which may be formed by non-fissile neutron capture of Pu-239 has properties very similar to U-238; then, if it captures another neutron, it will form fissile Pu-241. Therefore, after a long period of reactor operation, power will be produced from fission of U-235, Pu-239 and Pu-241. Table 4 shows the complete formation of the plutonium isotopes with rates of formation calculated for a thermal flux of 10^{13} n/cm^2 sec.

In general, the rate of change in the concentration N_i of the i^{th} isotope in the fuel in a flux ϕ, is given by

$$\frac{dN_i}{dt} = N_{i-1}\sigma_{i-1}\alpha_{i-1}\phi + N_j\lambda_j - N_i\sigma_i\phi - N_i\lambda_i \quad , \tag{2.2}$$

where we define σ_i as the absorption cross-section, λ_i the radioactive decay constant and α_i the probability of an absorbed neutron to produce a (n,γ) event. The first term on the right hand side is therefore the production of the i^{th} isotope by neutron capture in the $(i-1)^{th}$ isotope. The second term is the production from radioactive decay of a parent j^{th} nuclide. The last two terms give respectively the losses due to neutron absorption and radioactive decay.

Figure 4 shows the calculational results of a set of equations like Eq. (2.2) for the Pickering units. We note that for a burnup of about 7700 MWd/TU the Pu-239 concentration begins to exceed the U-235 concentration.

Since Pu-239 (and Pu-241) has a higher fission cross-section, most of the power comes from plutonium fissions at this stage.

The effect of burnup on reactivity will be related to the changes in the reactor physics factors η, ϵ, p and f as the fuel changes its composition. Since the fast fission factor ϵ and the resonance escape probability p are dependent on the U-238 concentration in the fuel, both of these will not change significantly with time. The change in fuel composition with time will mainly affect the thermal fission factor η and the thermal utilization f; hence, the product ηf will be given by

$$\eta f = \frac{N_5\sigma_{f5}\nu_5 + N_9\sigma_{f9}\nu_9 + N_1\sigma_{f1}\nu_1}{N_5\sigma_5 + N_8\sigma_8 + N_9\sigma_9 + N_o\sigma_o + N_1\sigma_1 + \frac{\bar{\Phi}_m}{\bar{\Phi}_f}(N_m\sigma_m)} , \qquad (2.3)$$

where the subscripts 5, 8, 9, 0 and 1 refer respectively to U-235, U-238, Pu-239, Pu-240 and Pu-241. The absorption in material m other than fuel isotopes is included in the last term of the denominator, and will also include the fission products poisons. The change in reactivity after a specified burnup can be determined by Eq. (2.3) where the N_i's have been obtained from Eq. (2.2).

Figure 5 shows the reactivity contribution made by the various fuel components as a function of burnup, without including xenon poisoning. It can be seen that initially, there is an increase in reactivity due to the higher cross-section of the newly formed Pu-239 relative to U-235 leading to a higher value of f. At higher irradiations, the build-up of Pu-239 becomes less rapid until its production will be equal to removal due to absorption. Consequently, at higher burnup the reduction

in the number of fissile isotopes causes a reduction in reactivity. The buildup of Pu-240 which has a large absorption resonance at 1 eV produces a large negative contribution. However, to some extent this reduction is off-set by the production of fissile Pu-241 atoms formed by resonance capture in Pu-240.

Although Fig. 5 is typical of many reactors, the actual shape of the reactivity curve depends on the conditions in the reactor. For enriched fuel reactors, the plutonium transient is less noticeable because the conversion factor is usually smaller. Figure 6 shows a reactivity curve for Pickering Unit 1 as a practical example.

2.5.2. Reactor Dynamics

As we know, the small fraction of delayed neutrons in a reactor substantially increases the reactor period for a given reactivity. The Pu-239 has lower neutron yields than U-235, the effect is then to reduce the mean lifetimes and to make the reactor behaviour more dynamic. In Table 5 we show the delayed neutron data for Pu-239 and U-235. The weighted average lifetime of all neutrons, prompt and delayed is, for these two isotopes

$$L_5 = \frac{8.5}{100} + 0.001 = 0.086 \text{ sec.}, \tag{2.4}$$

and

$$L_9 = \frac{3.3}{100} + 0.001 = 0.034 \text{ sec.}, \tag{2.5}$$

where 0.001 sec. is the diffusion time in the moderator.

For equilibrium fuel, this lifetime will be around 0.06 sec.; this means that a 1 mk insertion will cause a more rapid power increase

for equilibrium fuel than for fresh fuel. Therefore, the design of the control system must take into consideration this decrease in reactor period as the plutonium concentration increases.

The plutonium buildup also affects the prompt critical condition. Referring to Table 5, for U-235 this occurs when $\Delta k = \beta_5 = 0.0065 = 6.5$ mk while for Pu-239, it already occurs at $\Delta k = \beta_9 = 0.0021 = 2.1$ mk. Since for equilibrium fuel this condition may be reached at around 4 mk, the control systems must be designed to prevent reactivity values even close to this. The prompt critical condition implies a very short reactor period, of the order of 1 sec., and the CANDU systems have been designed to trip a reactor when its period becomes less than 10 sec.

2.5.3 Fission Products and their Effects on Reactivity

Most fission products may be considered to some extent as reactor poisons because they absorb neutrons. However, Xe-135 and Sm-149 are the main poisons because they both are produced in large quantities and have rather large thermal neutron capture cross-sections, around 10^4 and 10^6 barns respectively for Sm and Xe. Since their effects on reactivity occur on different time scales, we shall discuss them separately.

(A) Xenon 135

(a) Buildup of Xenon

Xe-135 is produced in the fuel in two ways:

(i) directly as a fission product, which accounts for about 5% of the total Xe-135,

(ii) indirectly from the decay of I-135 which may be produced as a fission product or from the

decay of the fission product Te-135 as follows:

$$Te^{135} \xrightarrow[11s]{\beta^-} I^{135} \xrightarrow[6.7h]{\beta^-} Xe^{135} \xrightarrow[9.2h]{\beta^-} Cs^{135} . \qquad (2.6)$$

Xe-135 is also removed in two ways:

(i) by decaying to stable Cs-135 as shown above,

(ii) by a neutron capture and formation of Xe-136 which has a much lower cross-section.

Consequently, the rate of formation of Xe-135 will depend on:

(i) the fission rate or the thermal neutron flux, and

(ii) the yield of I-135 and Xe-135 from these fissions.

And the rate of removal will depend on:

(i) its own decay rate, and

(ii) its thermal neutron capture cross-section.

The equation describing the rate of formation of Xe-135 will therefore be given by:

$$\frac{dN(Xe)}{dt} = Y(Xe)\bar{\phi}\Sigma_f(F) + \lambda(I)N(I) - \lambda(Xe)N(Xe) - \bar{\phi}N(Xe)\sigma_a(Xe), \qquad (2.7)$$

where the different terms are defined as follows:

N(I): I-135 concentration per unit volume of fuel.

N(Xe): Xe-135 concentration per unit volume of fuel.

Y(I): fractional yield of I-135 per thermal fission (= 0.061 for fresh fuel).

Y(Xe): corresponding yield of Xe-135 (0.003).

$\lambda(I)$: decay constant for I-135 (28.7×10^{-6} s^{-1}).

$\lambda(Xe)$: decay constant for Xe-135 (21×10^{-6} s^{-1}).

$\sigma_a(Xe)$: microscopic absorption cross-section of Xe-135 (3×10^6 b).

$\Sigma_f(F)$: microscopic fission cross-section of fresh UO_2 fuel ($0.1\ cm^{-1}$).

$\bar{\phi}$: average thermal flux in the fuel.

Looking at Eq. (2.7) we note that when the reactor is started up for the first time, since $N(Xe) = N(I) = 0$, the term $dN(Xe)/dt$ will be small but positive. After a while, the formation of iodine will increase $dN(Xe)/dt$ up to the time at which the xenon removal and decay terms will become more important. The rate of Xe formation will therefore slow down and finally will be zero when

$$Y(Xe)\bar{\phi}\Sigma_f(F) + \lambda(I)N(I)_o = \lambda(Xe)N(Xe)_o + \bar{\phi}N(Xe)_o\sigma_a(Xe). \qquad (2.8)$$

We conclude that the equilibrium value of Xenon, $N(Xe)_o$ will depend on the equilibrium of iodine, $N(I)_o$. The iodine buildup will be given by

$$\frac{dN(I)}{dt} = Y(I)\bar{\phi}\Sigma_f(F) - \lambda(I)N(I) \quad , \qquad (2.9)$$

where the production of Te-135 is assumed to decay to I-135 immediately and is therefore included in the Y(I) term. Also, we ignore the neutron capture removal of I-135 because $\sigma_a(I)$ is small, about 7 b.

The equilibrium value of iodine, from Eq. (2.9), is then given by

$$N(I)_o = \frac{Y(I)\bar{\phi}\Sigma_f(F)}{\lambda(I)} \quad , \qquad (2.10)$$

which is directly proportional to the average flux $\bar{\phi}$. Replacing $N(I)_o$ of Eq. (2.10) in Eq. (2.8) gives the equilibrium concentration of Xe-135:

$$N(Xe)_o = \frac{[Y(I) + Y(Xe)]\bar{\phi}\Sigma_f(F)}{\lambda(Xe) + \bar{\phi}\sigma_a(Xe)} \quad . \tag{2.11}$$

Figure 7 shows the buildup of Xe-135 as described by Eq. (2.7); note that the equilibrium of Xe-135 cannot exist until equilibrium of I-135 has been attained.

(b) Equilibrium Xenon Poisoning

The presence of xenon atoms in the fuel reduces the value of the thermal utilisation factor f because of the absorption of neutrons which would otherwise have been available for the fuel. A useful parameter describing this effect, the poisoning factor R is defined by

$$R = \frac{\text{(Thermal neutrons absorbed by Xe-135)}}{\text{(Thermal neutrons absorbed by the fuel)}} \quad , \tag{2.12}$$

or

$$R = \frac{\bar{\phi}\Sigma_a(Xe)}{\bar{\phi}\Sigma_a(F)} \quad , \tag{2.13}$$

where $\Sigma_a(F)$ is the total absorption cross-section of the fuel; fission plus capture.

Therefore, using Eq. (2.11), at equilibrium we have

$$R = \frac{\sigma_a(Xe)[Y(I) + Y(Xe)]\bar{\phi}\Sigma_f(F)}{[\lambda(Xe) + \bar{\phi}\sigma_a(Xe)]\Sigma_a(F)} \quad . \tag{2.14}$$

Substituting the various constants for natural uranium gives

$$R = \frac{0.104 \times 10^{-18}\bar{\phi}}{21 \times 10^{-6} + 3 \times 10^{-8}\bar{\phi}} \quad . \tag{2.15}$$

A graph of R as a function of the average thermal flux is shown in Fig. 8. The xenon poisoning factor is negligible for very low fluxes and is usually neglected when $\phi \leq 10^{-11}$ ncm^{-2}sec^{-1}. For very high fluxes

($> 10^{14}$ $ncm^{-2}sec^{-1}$) we see that R tends toward the limit value of R_∞ = 0.035, implying that 35 neutrons would be absorbed by Xe-135 for every 1000 absorbed by the fuel.

(c) Equilibrium Xenon Load

The xenon load is the reduction in reactivity caused by the presence of xenon in the fuel. This xenon load, due to the change in f, can be related to the poisoning factor in the following way. For fresh fuel, when there is no Xe-135, the thermal utilisation is given by

$$f = \frac{\Sigma_a(F)}{\Sigma_a(F) + \Sigma_a(M)} \quad ; \tag{2.16}$$

but the presence of Xe-135 decreases f to f':

$$f' = \frac{\Sigma_a(F)}{\Sigma_a(F) + \Sigma_a(M) + \Sigma_a(Xe)} \quad , \tag{2.17}$$

where $\Sigma_a(M)$ is defined as the absorption cross-section of all the material that is not fuel or xenon. The reactivity due to xenon will therefore be expressed by

$$\rho = \frac{k'_{eff} - k_{eff}}{k'_{eff}} = \frac{f' - f}{f'} \quad , \tag{2.18}$$

or

$$\rho = -Rf \quad . \tag{2.19}$$

This means that when a reactor is critical with its xenon load, k'_{eff} = 1 and the product -Rf represents the reactivity loss due to xenon. For a CANDU reactor such as Pickering, which has a thermal averaged flux of about 7×10^{13} $ncm^{-2}sec^{-1}$, the reactivity loss caused by equilibrium xenon poisoning is of the order of 28 mk.

The fact that k'_{eff} is equal to 1 at equilibrium with the poisons load means that the reactor has to be supercritical without these poisons, i.e. $k_{eff} > k'_{eff}$. Consequently, when the reactor is started up with fresh fuel, the excess of reactivity due to the calculated equilibrium poison load has to be compensated as the poisons buildup. One of the approaches to this problem for the CANDU reactor has been to use the moderator level which was kept below its normal operating level for a fresh core and increased as the poisons were building up. However, this causes a distorted radial flux distribution and some of the fuel channels do not contribute to power production during the poison buildup period. A better solution to this problem is the addition of boron into the moderator system to allow a full tank and hence full power operation. One ppm of boron is equivalent to a poison load of about 8 mk. As the poisons buildup to the equilibrium, the boron is removed at a corresponding rate by ion exchange columns. The boron, as mentioned previously, also performs the additional function of simulating the reactivity loss due to fuel burnup compared to fresh fuel.

(d) Xenon Buildup During Reactor Shutdown

The real problem presented by xenon begins after a shutdown or a cut-back in power. If a reactor is at equilibrium, for a reactor trip at t = 0 we may approximate the flux as dropping to zero, and Eq. (2.7) becomes:

$$\frac{dN(Xe)}{dt} = \lambda(I)N(I) - \lambda(Xe)N(Xe) \quad , \tag{2.20}$$

where no xenon or iodine are produced or removed by the neutron flux which

is almost non-existant. The term N(I) in Eq. (2.20) is $N(I)_o \exp[-\lambda(I)t]$; therefore,

$$\frac{dN(Xe)}{dt} = \lambda(I)N(I)_o e^{-\lambda(I)t} - \lambda(Xe)N(Xe) \quad . \tag{2.21}$$

If the right-hand side of this last equation is positive, the xenon load will increase; i.e. the decay of the remaining iodine will produce xenon more rapidly than the xenon decay itself. The solution of Eq. (2.21) is given by:

$$N(Xe) = N(Xe)_o e^{-\lambda(Xe)t} + \frac{\lambda(I)N(I)_o}{\lambda(Xe) - \lambda(I)} [e^{-\lambda(I)t} - e^{-\lambda(Xe)t}], \tag{2.22}$$

and the corresponding graph is shown in Fig. 9. Let us note that after the xenon concentration has reached its maximum value, the rate of decrease is then essentially governed by its own half-life of 9.2 hours.

For Pickering and Douglas-Point the transient maximum is about 80 mk above the equilibrium xenon load. The time t_m at which this peak occurs will be given by

$$t_m = \frac{1}{\lambda(Xe) - \lambda(I)} \ln\{\frac{\lambda(I)}{\lambda(Xe)} - \frac{[\lambda(Xe) - \lambda(I)]}{\lambda(Xe)} \frac{N(Xe)_o}{N(I)_o}\} \quad . \tag{2.23}$$

For an average thermal flux of 10^{13} $ncm^{-2}sec^{-1}$, t_m turns out to be about 6 hours and it increases to a limiting value of 11.3 hours for very large fluxes.

The rate of rise of the xenon load after a trip from full power is, for CANDU reactors, typically 24 mk per hour. It means that if a reactor has a maximum available excess reactivity of 16 mk, the poison override time is about 40 minutes. If the reactor is not brought back to full

power within this period of time, the poison out time, during which the reactor could not be started up again, may be as high as 32 hours.

The desired reactivity for poison override may be provided by the removal of absorbers or the insertion of the boosters, but the design of a reactor that needs a longer override time is expensive. In practice, the cost of providing the excess reactivity is optimized with the energy production that would otherwise have been lost during the poison out time.

(e) Xenon Spatial Oscillations

As described above, a shut-down or a change in power (or flux) affects the xenon concentration in the reactor. This effect may however also be applied to a small flux perturbation in one part of the reactor and increase this perturbation causing spatial oscillation in the flux distribution.

Let us assume that a core has a fixed flux distribution and has equilibrium concentration of Xe and I. If a small perturbation increasing the flux at one side of the reactor occurs, the fixed power of the reactor will force the flux on the other side to decrease symmetrically. On the side with the slight increase in flux, the I-135 production rate will go up but its own half-life will delay its decay to Xe-135. This increase in flux will also decrease the present Xe-135 concentration because the neutron capture by Xe-135 will go up. Thus both effects combined will cause the flux to rise further because the heavy parasitic absorber xenon quantities will have been reduced. Similar arguments for the other side of the reactor will cause a further decrement of the flux.

Therefore, the flux perturbation will initially be amplified, but

the point will be reached where the surplus of I-135 decaying to Xe-135 will turn the process around and drive the fluxes in the other direction for both sides of the reactor thus causing oscillations with cycle lengths of the order of 24 hours. All types of large thermal reactors are susceptible to this form of instability which has to be controlled by constant monitoring and feedback systems.

(B) Samarium-149

Like Xe-135, Sm-149 is a strong neutron absorber but it is a stable poison. Samarium is mainly formed by the decay of the fission products Neodymium-149 and Promethium-149,

$$Nd^{149} \xrightarrow[1.7\ h]{\beta^-} Pm^{149} \xrightarrow[53\ h]{\beta^-} Sm^{149} \quad . \tag{2.24}$$

Since Sm-149 is stable, the only removal process for it is neutron capture. And since it has a much lower cross-section (4.2×10^4 b) than Xe-135, the equilibrium will be attained after a longer time.

(a) Buildup of Sm-149

Using symbols analogous to those used for xenon analysis, with (Sm) and (Pm) referring respectively to Sm-149 and Pm-149, the differential equations governing the behaviour of N(Pm) and N(Sm) are

$$\frac{dN(Pm)}{dt} = Y(Pm)\bar{\phi}\Sigma_f(F) - \lambda(Pm)N(Pm) \quad , \tag{2.25}$$

and

$$\frac{dN(Sm)}{dt} = \lambda(Pm)N(Pm) - \bar{\phi}N(Sm)\sigma_a(Sm) \quad . \tag{2.26}$$

In Eq. (2.25) we neglected the Pm-149 removal by neutron absorption because its cross-section is small. Also, in Eq. (2.26), since the yield

of Sm-149 production directly as a fission product is small, it has been neglected.

Setting the derivative to zero in Eq. (2.25), gives for the equilibrium Pm-149 concentration,

$$N(Pm)_o = \frac{Y(Pm)\bar{\phi}\Sigma_f(F)}{\lambda(Pm)} \quad . \tag{2.27}$$

Similarly from Eq. (2.26), the equilibrium concentration of Sm-149 is, after replacing N(Pm) by $N(Pm)_o$,

$$N(Sm)_o = \frac{Y(Pm)\Sigma_f(F)}{\sigma_a(Sm)} \quad , \tag{2.28}$$

which is independent of the neutron flux. The equilibrium samarium poisoning factor will therefore be

$$R = \frac{N(Sm)_o\sigma_a(Sm)}{\Sigma_a(F)} = Y(Pm)\,\frac{\Sigma_f(F)}{\Sigma_a(F)} \quad . \tag{2.29}$$

Using typical values for CANDU reactors, the equilibrium samarium load, Rf, turns out to be around 5.5 mk. However, this reactivity loss of 5.5. mk takes a long time to appear. The Pm-149 concentration takes about 4 half-lives ($\sim$ 9 days) to reach equilibrium; the buildup of Sm-149 is then exponential with a time constant of $[\sigma_a(Sm)\bar{\phi}]^{-1}$. With a flux of about 10^{13} $ncm^{-2}sec^{-1}$ it takes about 40 days before the Sm-149 approaches its equilibrium value.

(b) Samarium Growth After a Shutdown

After a shutdown, the Sm-149 concentration will increase since none is being removed ($\phi \sim 0$) and some is still produced by Pm-149 decay. In a time t the promethium will have decayed from $N(Pm)_o$ to $N(Pm)_o\exp[-\lambda(Pm)t]$

and the Sm concentration will have increased by the difference. Therefore,

$$N(Sm) = N(Sm)_o + N(Pm)_o[1 - e^{-\lambda(Pm)t}] \quad . \tag{2.30}$$

Using the equilibrium values of Eqs. (2.27) and (2.28) this gives,

$$N(Sm) = N(Sm)_o\{1 + \frac{\sigma_a(Sm)\bar{\phi}}{\lambda(Pm)}[1 - e^{-\lambda(Pm)t}]\} \quad . \tag{2.31}$$

And the maximum concentration of samarium $N(Sm)_m$ after the shutdown will therefore be

$$N(Sm)_m = N(Sm)_o[1 + \frac{\sigma_a(Sm)\bar{\phi}}{\lambda(Pm)}] \quad . \tag{2.32}$$

We show in Fig. 10 a samarium transient after a shutdown for a considerable flux of 10^{14} $ncm^{-2}sec^{-1}$. For this rather high flux the Sm load has increased from 5.5 mk to 12 mk, in other words only 2.2 times bigger than that before the shutdown.

For CANDU reactors, it is interesting to note, that although the equilibrium samarium load has to be allowed in the design, the shutdown load may be ignored. One of the reasons for this is that by looking at the time scale of Fig. 10 we see that the increase in samarium load during the xenon poison override time is negligible. Consequently, the maximum Sm load will not appear until the Xe transient has long been and gone. Secondly, the half-life of Pm-149 from which Sm is formed (53 hours), corresponds almost exactly to the rate at which Pu-239 is formed after a shutdown (56 hours) from Np-239 decay. And it turns out that the increased reactivity from this Pu-239 transient more than compensates for the increased Sm load.

(C) Other Fission Products

Although Xe-135 and Sm-149 are the most important fission products, there are a number of others which when added together produce a significant reactivity change. However, since for these additional fission products the associated reactivity changes take place over a much longer period of time, they are usually considered in conjunction with the reactivity changes due to the fuel burnup.

2.6 Fuel Management Calculations

In fuel management calculations the optimum fuel cycle must be used, that is the maximum reactor power must be maintained with minimum fuel costs.

The practical way to achieve this purpose is by using computer programs capable of following the histories of the bundles in the core. These programs calculate the expected radial and axial power distribution, the burnup of each bundle and the excess reactivity available. The physics data of the program may in practice be modified in order to minimize the discrepancies. The running of such a program helps to derive the fuelling rates in the different zones of the core, but even so, no rigid pattern is used because some basic criteria have to be considered. For instance, the priority must be given to channels known to contain failed fuel, the refuelling should be symmetrical and one must consider the effects of neighbouring channels.

If the axial flux distribution in the reactor is fairly flat, it might be well expedient to fuel in so called 8 or 10 bundle shifts. If the axial flux distribution is more peaked in the centre, it may be necessary to use 4 or even 2 bundle shifts to achieve economic burnup.

3. THORIUM BREEDING IN CANDU

3.1 Breeding and Near-Breeding in the CANDU System

Up to now the Canadian Nuclear Industry has used only natural uranium as a fuel for their nuclear reactors. However, other fuel cycles are possible and those which are particularly interesting from an economical point of view, are near-breeding or breeding fuel cycles. For example, in the USA, fast breeders using plutonium as a fuel are presently under study. Since the Canadian approach has been to deal only with thermal heavy water moderated reactors, studies are initiated to attain a breeder thermal reactor. One advanced fuel cycle that has been considered uses a mixture of thorium and fissile material in order to get near-breeding by conversion of Th-232 into fissile U-233.

The introduction of the thorium cycle into the CANDU system would lead to a major gain in resource utilisation. The near-breeder thorium fuel cycle would have to reprocess the spent fuel and separate uranium-233 which would be combined with fresh or recycled thorium plus additional fissile material (U-235 or Pu-239), and fed back into the reactor. This type of cycle could possibly yield 50,000 MWd/Te and more from uranium, and the neutron economy can be so good that the known supplies of uranium seem adequate for many centuries.

It has also been shown that under certain circumstances, it is possible to attain the self-sufficient equilibrium thorium cycle. In this breeder mode of operation, no external supply of fissile material

would be required and only the uranium from the spent fuel would be fed back with thorium for each pass. Recent studies have shown that taking into consideration the future economic trends, this equilibrium cycle would incur a penalty of some 20% in electrical energy costs compared to the optimum thorium cycles. However, since the uranium reprocessing and fabrication costs are expected to decrease in the future, this will tend to reduce the percentage penalty.

The uranium availability, or its price, is the criteria that will mainly determine when it will become economical to introduce the thorium cycle into CANDU reactors. However, the most important point to note is that the CANDU type of reactor is particularly suitable for almost any fuel cycle and may theoretically be used with thorium cycle without any major change in its basic conception. Consequently, we will treat in the following pages some aspects of the physics of thorium utilization.

3.2 Comparison Between U-233 and U-235 as Fissile Materials

It is known that U-233 has some reactor physics properties that are better than U-235 for thermal reactors. For instance, the excess of neutron produced in fission, η-1, is about 20% better for U-233 than U-235 in the thermal range. Roughly, of 100 neutrons absorbed in U-233, this will induce 92 fissions while for U-235, 84 of them will result in fissions.

We present in Fig. 11 a comparison between the infinite cell multiplication constant k_∞ for a standard natural uranium CANDU bundle and for a bundle in which the U-235 atoms have been theoretically replaced

by U-233 atoms, as a function of the burnup. Neglecting leakage, the fuel may be kept in a reactor until the integral above $k_\infty = 1$ equals the integral below it. Thus, for natural uranium bundles, one can extract about 7500 MWd/Te while for an idealized thorium fuel, we could be able to extract roughly 80% more energy.

This comparison is useful to note the superiority of U-233 as fissile material, but such a theoretical fuel, in which we would replace U-235 by U-233 and leave everything else the same would not be useful because without including Th-232 no breeding is possible. In fact, before entering in more details, a very idealized advanced fuel cycle for CANDU reactor could be briefly described in the following way:

(i) replace uranium in CANDU fuel by Th-232,

(ii) add a certain amount of U-233,

(iii) place these bundles in CANDU D_2O moderated reactors.

It has been assumed that this fuel can be placed in a CANDU reactor of standard design but in fact, even if the basic conception of CANDU is suitable for thorium utilisation, some modifications would have to be done. For instance since the power density of thorium fuel is higher, smaller fuel pencils and different lattice pitches would be required to obtain better heat transfer. Other modifications would involve the kind of coolant to use, organic, light water or heavy water, and the core itself would probably have to be smaller.

3.3 Thorium Cycle in the CANDU System

3.3.1 Production of U-233 by Thorium Activation

Basically, fissile U-233 is produced by the absorption of neutron

and decay of Th-232. We show in Fig. 12 the sequence of events that follow the capture of a neutron by a nucleus of Th-232. This capture first transmutes Th-232 to Th-233 which has a half-life of 22 minutes and then forms Pa-233 by beta decay. Fissile U-233 is produced by a beta decay of Pa-233 of half-life equal to 27 days. As may be expected, the fairly long half-life of Pa-233 can have interesting effects on the fuel management.

This long half-life gives a significant probability that Pa-233 after its formation, will absorb a neutron, and eventually form U-234 which is not thermally fissile. This process therefore bypasses the production of U-233, creates a new material to absorb neutrons and captures a neutron. The higher the flux level, the more neutrons are absorbed and the less U-233 is formed. Consequently, the highest burnups will be attained with the lowest values of thermal flux.

The net effect of Pa-233 is then to make thorium fuels sensitive to the neutron flux level and to their irradiation history. This complicates fuel management computations because in the center region of the core where the flux is greatest, the U-233 content will be depressed which will reduce the reactivity and then, acting as a feedback will reduce the flux. The opposite effect near the core edges will help to produce flattened power distribution but this will increase leakage.

3.3.2 The Equilibrium Fuel Cycle

For an ideal fuel composed of U-233 added to Th-232, the concept of equilibrium thorium cycle may be visualized by the Fig. 13 where we show k_∞ for such a fuel as a function of the amount of U-233 added to Th-232. We see that because Th-232 has roughly 3 times the absorption

cross-section of U-238, much more fissionable material must be present before there is enough reactivity to sustain the reaction. Neglecting leakage we can see that we need a U-233 enrichment of about 1.25% just to make the reactor critical while in natural uranium, the U-235 is only 0.7%. So, in a reactor containing such a fuel, its concentration must be greater than about 1.3%.

However, since we have breeding, the concentration of U-233 will vary as the burnup increases. Figure 14 shows us this variation for two possible initial concentrations. Starting with 1.5% of U-233 in Th-232, the energy that can be extracted is 8600 MWd/Te leaving about as much U-233 as we began with. Starting with a concentration of 2% we obtain roughtly 5 times the previous energy but we will be left with only 75% of our initial U-233. This latter option seems to be of less interest because U-233 does not occur naturally but helps to illustrate the concept of the equilibrium cycle.

We show in Fig. 15 a simple cycle based on this concept. Theoretically, the fuel is enriched to about 1.5% U-233, the energy is extracted and out comes 1.5% U-233 which is reprocessed and refabricated into new fuel to be recycled indefinitely. A way to further increase the burnup would be to spike the fuel with small amounts of plutonium or enriched uranium. Calculations have shown that the introduction of less than 0.5% plutonium would greatly increase the available energy from each cycle and still would require about 8 times less uranium than needed in Pickering. It seems that the economic optimum lies in this direction.

3.3.3 Concepts of U-233 Production

Up to this point we have discussed the possible equilibrium thorium-cycle, where theoretically one may obtain the same amount of U-233 out the reactor as was put in. Obviously, to start such a system, enough U-233 would have to be provided.

A possible way to produce this U-233 is shown in Fig. 16. The fissile material, plutonium or enriched uranium, is mixed with thorium introduced in the reactor which gradually phases in U-233 as it becomes available. The first charge may be pure Pu-Th, and more and more U-233 is gradually mixed while less and less Pu (or U) is used until equilibrium is achieved. Another concept is shown in Fig. 17 where a reactor "A" is designed to burn Pu-Th or enriched U-Th fuel and to collect the U-233 until enough of a stockpile has been accumulated to start up a second equilibrium reactor "B".

In these two possibilities we present the equilibrium cycle as an immediate objective but, as mentioned previously, since the equilibrium thorium cycle may incur some economic disadvantages, a short term solution would be to use an optimum thorium cycle. This type of cycle could still be represented by Fig. 16 where the complete cycle, if not completely breeding, needs much less uranium for a given energy production than the usual natural cycle previously presented in Fig. 3.

4. HEAT AND MASS TRANSFER

4.1 Flexibility of CANDU System to Various Coolants

The PHW CANDU reactor, as it is commercially known, uses natural uranium as fuel, heavy water as moderator and pressurized heavy water as coolant. We mentioned in the previous chapter that the CANDU system could use different kinds of fuel and may be made to breed; but another feature of the system is its flexibility to the type of coolant that may be used. The Canadian Nuclear Program is essentially based on three different coolants: heavy water, ordinary water, and HB-40, an organic substance which has the appearance of a light oil.

The most efficient liquid coolant from a standpoint of neutron economy is heavy water. The use of ordinary water as a coolant requires special design of the reactor in order to have as little neutron absorption as possible. This is achieved by allowing the coolant to boil to reduce its density as is done in the Gentilly-1 boiling light water reactor (BLW). The most attractive point of the BLW CANDU type of reactor is that the vapour generated in the pressure tubes is directly fed to the turbine without having to pass through heat exchangers. The thermal efficiency of such a system is theoretically higher but its main disadvantage is that the interface between all liquid coolant and two-phase coolant in the core, which is vertical, tends to move up and down. The effect is then to expose the fuel elements in certain parts of the core to variations in heat transfer and neutron flux, increasing therefore their

failure probability.

Organic coolants such as HB-40 have conisderably lower vapour pressure than any form of water at any temperature near operating values. Typically, heavy water at 299°C is under a pressure of 8.56 MPa(a), whereas HB-40 at the same temperature has a vapour pressure of only 0.28 MPa(a). The vessels may therefore be considerably thinner and leakage is more easily controlled. Also, organic coolant is much less corrosive than hot water which slowly dissolves the walls of pipes and vessels, carrying away impurities in the coolant. The main problems involved with organic coolant are its neutron absorption, its inflammability at high temperature and its low vapour pressure advantage itself. Firstly, HB-40 is not as transparent to neutrons than heavy water but is nevertheless better than ordinary water. Secondly a leak from a pipe at high temperature may produce an explosive mixture of hydro-carbon vapours in the area. Since very highly reliable defences against explosion exist, this hazard in no way rules out organics but precautions must be taken. Thirdly, the low vapour pressure of organics has as a consequence that the pressure on the steam side of the steam generators, is much greater than the pressure on the organic side. A tube failure will produce a leak of water into the organic coolant with very undesirable results. Again this difficulty is not insuperable, but perhaps it is indicative of the array of problems confronting the designers.

The present situation with respect to coolants in Canada is that the PHW reactors are not mere prototypes; they are proven systems and numerous plants of this type are in operation in Canada and abroad. The

organic cooled CANDU reactor is seriously under study in Whiteshell, Manitoba; and a light water cooled reactor would have prototype aspects if it were larger than G-1 (250 MWe) or if it employed enriched fuel.

The rest of this chapter will be devoted to the PHW system and we will try whenever possible, to give the Standardized CANDU 600 MWe PHW reactor as an example.

4.2 Description of the Heat Transport Systems

4.2.1 Main Circuit

The pressurized heavy water cooling system for PHW CANDU reactors is called the primary heat transport system (PHT) while the associated steam water system is commonly referred to as the secondary system. The heat transport system includes a standby cooling system to remove decay heat from the fuel following a unit cooldown and a system to provide high pressure heavy water to the fuelling machines.

The complete steam-water system includes the boilers and related auxiliaries, steam transfer piping and valves, the turbine and condenser, the reheat system and the feedwater system. We show in Fig. 18 a simplified flowsheet of the heat transport system for Pickering reactors and Table 6 gives the principal data associated with it.

Some of the main features of the CANDU heat transport system may be stated as follows:

(i) A continuous circulation of coolant is maintained at all times during operation, shutdown, and maintenance.

(ii) Adequate heat transport system flow for shutdown

heat removal is maintained by natural convection flow following pump rundown.

(iii) The heat transport system pressure is controlled at the reactor outlet headers.

(iv) A separate shutdown cooling system is provided which is independent of the boilers.

(v) Purification of coolant by filtering and pH control of coolant.

(vi) Collection of D_2O leakage from potential leak points in the system.

4.2.2 Boilers

For the 600 MWe Standardized CANDU reactor, four identical steam generators with integral preheaters, transfer heat from the D_2O reactor coolant on the boiler primary side to H_2O feedwater on the boiler secondary side. The boilers mainly consist of an inverted vertical U-tube bundle installed in a shell. Steam operating equipment is housed in the upper end of the shell (see Fig. 19).

The practical range of secondary side steam pressures is between about 4.14 MPa(a) and 5.52 MPa(a). Higher pressures would improve cycle efficiency and total station output but the cost of the boilers would increase because of their larger size and greater heavy water holdup in the U-tube bundles. An optimisation program reflecting these and other relevant factors is therefore employed for each unit to determine the optimum secondary steam pressure.

4.2.3 Heat Transport Pumps

A typical heat transport pump is shown in Fig. 20 for the Standardized 600 MWe CANDU reactor. The pumps are centrifugal, single suction and double discharge.

Each pump is driven by a vertical, total enclosed, air-water cooled squirrel cage induction motor; the motor is supplied with a flywheel to prolong pump operation after a loss of motor power. Also, as mentioned before, the design is such that natural circulation will maintain adequate cooling of the fuel after the pump stops.

4.2.4 Pressurizer

A common pressurizer controls the pressure in the two heat transport circuits. As shown in Fig. 21 for the 600 MWe CANDU system, the pressurizer is a cylindrical pressure vessel installed vertically and is connected to lines linking the outlet headers at one end of the reactor. A valve is provided in each of the lines, so that the two heat transport circuits can be isolated from one another.

The pressure is provided by electric heating of heavy water stored in the pressurizer. The cushioning effect of the D_2O steam volume in the pressurizer is supplemented by two 100% steam bleed valves which open when the pressure rises above a set point. These steam bleed valves also provide overpressure protection for the pressurizer when the valves in the lines linking the headers are closed. The discharge from these bleed valves goes into the degasser-condenser tank.

4.2.5 Shutdown Cooling System

The system consists basically of a pump and a heat exchanger at each end of the reactor connected between the inlet and outlet headers at both heat transport circuits. The system is normally full of D_2O and is normally isolated from the heat transport system by 8 sets of duplicated valves.

For the 600 MWe CANDU reactor, the shutdown cooling system is provided to cool the heat transport system from 177°C down to 54°C and to maintain this last temperature for an indefinite period of time. The initial cooldown to 177°C is provided by blowing off steam from the main boilers. For cooldown from 177°C to 54°C, the main isolating valves at the reactor headers are opened and a portion of the total core flow is circulated through the shutdown cooling heat exchangers. The flow through the heat exchangers is selected to give a cooldown time of approximately 4 hours.

4.3 Coolant Condition

4.3.1 Coolant Temperature, Pressure and Flow

The coolant temperature and the associated pressure are very important parameters in a reactor design. The higher is the coolant temperature, the higher will be the thermal efficiency of the cycle and hence the lower will be the fuel cost. But an increase of temperature implies an increase in pressure and thicker pressure tubes and higher conversion rate. It is shown in Fig. 22 how the coolant temperature fixes the steam temperature in the boilers.

The limit on the coolant flow is related to the core design; pressure drop considerations and fuel vibrations set the upper limit of flow. In CANDU reactors with typical fuel elements of about 50 cm long and 1.5 cm in diameter held together by metal plates at the end, the fluid velocity at 9 m/sec seems to be a practical limit. Beyond this velocity, the elements may chafe against each other or may wear upon the pressure tubes.

The distribution of the heat generated in the fuel is shown in Table 7. We note that the coolant removes about 94.6% of the total heat generated in the fuel and 31.4% is for electrical generation.

4.3.2 Thermal Hydraulic Regimes

Since the conception and design of the CANDU BLW reactor Gentilly-1, the major heat transfer research effort has been devoted to the fundamental study and prediction of the various flow regimes that may occur in two-phase flow.

Referring to Fig. 23 it can be noted that there are mainly three critical hydrodynamic phenomena associated with heat transfer in a two-phase flow:

(i) Starting of nucleate boiling or starting of significant void, depending on what degree of void formation is considered undesirable.

(ii) Departure from nucleate boiling (DNB); a vapour blanketing which occurs at low or negative qualities and leads to severe surface temperatures.

(iii) Dryout or loss of liquid film from the heat surface

in the annular flow regime which leads to less severe surface temperatures.

Eventually a single phase high steam quality condition may develop in the coolant but as coolant enters in the channels, bubbles form in it which subsequently form slugs of void creating large variations in the void fraction along a channel. We may note that at the dryout point, there is a sudden increase in the wall temperature. This obviously represents a most undesirable condition because of the associated thermal effects on the sheaths.

In the axial dimension, the operating powers are very similar in all PHW CANDU reactors, nearly sinusoidal in shape. However, for BLW, due to boiling, the axial distribution of sheath and coolant temperature differs from those in PHW reactors.

We show in Fig. 24 the Bruce PHW reactor axial distribution, typical of a central channel. We note that there is no net quality at the outlet because the average exit coolant temperature equals the saturation temperature. Because of different hydraulic characteristics of the subchannels within the fuel, the coolant temperature may vary from subchannel to subchannel and the differential can be enough to produce net boiling in certain subchannels.

The fact that the fuel sheath temperature follows the coolant temperature fairly closely, with only 20°C maximum difference for Bruce, is characteristic of the rather good heat transfer existing in PHW CANDU reactors.

4.4 Heat Transfer in the Fuel

The temperature variations in the core that we have previously discussed are very small relative to the large variations existing inside the fuel itself. This arises because of the thermal conductivity of uranium dioxide fuel which is extremely low; hence a very large temperature differential is necessary to support the flow of heat.

The temperature profile in the fuel can be determined using the steady state heat transfer analysis for a solid cylindrical fuel element or heat source, surrounded by the fuel sheath and a coolant. The power or source density in the fuel region, defined in terms of the fission process, is for a thermal reactor,

$$q'''(\underline{r}) = G\int_0^{\infty} \Sigma_f(\underline{r},E)\phi(\underline{r},E)dE \quad , \tag{4.1}$$

where $q'''(r,z)$ has the dimension of Watt/cm^3 and G is the energy per reaction in watt-sec.

The conservation of energy and the steady state general heat conduction equations are expressed by:

$$q'''(r,z) = \nabla\cdot\underline{q}''(r,z) \quad , \tag{4.2}$$

and

$$\underline{q}''(r,z) = -k(r,z)\nabla T(r,z) \quad , \tag{4.3}$$

where we have defined:

$T(r,z)$ - the local medium temperature (°C),

$k(r,z)$ - the local thermal conductivity (W/cm^2 °C),

$q''(r,z)$ - the vector heat flux (W/cm^2).

Combining Eqs. (4.2) and (4.3) gives us the general form for which a solution is sought:

$$q'''(r,z) = -\nabla\cdot[k(r,z)\nabla T(r,z)] \quad . \tag{4.4}$$

Referring to the definition sketch given in Fig. 25, it is noted that the problem can be solved by specifying the thermal conductivity in each region and the temperature at the boundaries. Restricting our analysis to a specific z-coordinate and assuming that the conductivity k_f and the power density q''' are constant, Eq. (4.4) in the fuel region reduces to:

$$q''' = -k_f\left(\frac{\partial^2 T_f}{\partial r^2} + \frac{1}{r}\frac{\partial T_f}{\partial r}\right) , \tag{4.5}$$

or

$$\frac{\partial^2 T_f}{\partial r^2} + \frac{1}{r}\frac{\partial T_f}{\partial r} + \frac{q'''}{k_f} = 0 \quad . \tag{4.6}$$

And its general solution is,

$$T_f = -\frac{q'''}{4k_f} r^2 + A\ell nr + B \quad . \tag{4.7}$$

The two following conditions:

$$\text{at} \quad r = r_1 \quad , \quad T_f = T_1 \quad , \tag{4.8}$$

and

$$\text{at } r = 0 \quad , \quad \frac{\partial T_f}{\partial r} = 0 \quad , \tag{4.9}$$

permit us to evaluate the constants A and B and hence, to obtain for the radial temperature in the fuel region, the following expression:

$$T_f = T_1 + \frac{q'''}{4k_f}\left[r_1^2 - r^2\right] \quad . \tag{4.10}$$

In the sheath, since the source term becomes zero, the Eq. (4.4) reduces to the steady state with no heat generation Laplace equation:

$$\nabla^2 T_s = 0 \quad , \tag{4.11}$$

or

$$\frac{\partial^2 T_s}{\partial r^2} + \frac{1}{r}\frac{\partial T_s}{\partial r} = 0 \quad , \tag{4.12}$$

which has the general solution

$$T_s = C\ell nr + D \quad . \tag{4.13}$$

The boundary conditions T_1 at r_1 and T_2 at r_2 permitting the evaluation of the constants C and D, Eq. (4.13) therefore becomes:

$$T_s = T_1 - \frac{(T_1 - T_2)}{\ell n(r_2/r_1)} \ell n(\frac{r}{r_1}) \quad . \tag{4.14}$$

The heat flux at radius r_1, given by Eq. (4.3) is:

$$q''(r_1) = \frac{k_s(T_1 - T_2)}{r_1 \ell n(r_2/r_1)} \quad ; \tag{4.15}$$

and finally, the temperature distribution in the sheath becomes:

$$T_s = T_1 - \frac{r_1 q''(r_1)}{k_s} \ell n(\frac{r}{r_1}) \quad . \tag{4.16}$$

In the coolant, the mean temperature will be defined by the Newton law of cooling,

$$q''(r_2) = h(T_2 - T_c) \quad , \tag{4.17}$$

where h is the film coefficient. Using the assumption of the constant value of q"' in the fuel medium, we have:

$$q''(r_2) = \frac{q'''r_1^2}{2r_2}, \tag{4.18}$$

and hence, the mean coolant temperature becomes:

$$T_c = T_2 - \frac{q'''}{2r_2h} r_1^2 \quad . \tag{4.19}$$

We show in Fig. 26 a calculated temperature distribution inside a fuel pellet. We may note that this temperature distribution represents the hottest pencils of the bundle in the hottest region of a core. The temperature distribution over the majority of a core is well below the one shown. In the above analysis we have neglected the contact resistances at the interfaces which contribute to small temperature drops across these boundaries. The peak central temperature shown is around 2100°C; if the power were increased by about 45%, the central region of the fuel would begin to melt.

Our analytical approach has been made for an ideal case where certain effects were neglected. A more detailed thermal analysis would provide a temperature profile different in some respect from the precedent representation. The differences may be attributed to the following considerations:

(i) k_f is both space dependent and temperature dependent. The temperature dependence is often stated as to be $k_f = k_o(1 + \alpha T)$.

(ii) q''' is space dependent as shown previously by the definition of $q'''(\underline{r})$.

(iii) Presence of contact resistance between the fuel and sheath usually defined by $R = (T_a - T_b)/q''$ where $(T_a - T_b)$ is the temperature decrement between the two surfaces and q'' is the flux density.

(iv) Boundary layer phenomena of the sheath-coolant interface.

(v) Dependence of the heat transfer coefficient h upon the flow conditions of the coolant.

The axial temperature profiles can be determined on the basis of a similar analysis when the axial flux can be assumed to have a shape defined by a regular function. The axial sheath and coolant temperatures have the form previously shown in Fig. 24, and the temperature profiles along the axial dimension of uranium oxide fuel inside the channels possess a form similar to the one shown in Fig. 27.

For the case of a non uniform flux distribution caused by the insertion of control rods, or by prolonged operation with control rods inserted, or by the fact that a portion of the core has been poisoned by xenon, numerical methods have to be used for the calculations. In any case, the two following parameters are used in the calculations.

(i) The proportion of the total amount of heat transferred in the channel up to the position of the maximum surface temperature of the fuel element,

$$\delta_c = \frac{\int_0^{\ell_c} q' d\ell}{\int_0^{L} q' d\ell} \quad . \tag{4.20}$$

With q' being the rate of heat generation per unit length parallel to flow in the coolant channel, which is usually defined by,

$$q' = 4\pi \int k dT \quad , \tag{4.21}$$

ℓ_c being the length of coolant passage to point of maximum surface temperature and L being the total length of the coolant channel.

(ii) The proportion of the total energy added as heat to the coolant up to the point of maximum internal temperature,

$$\delta_o = \frac{\int_0^{\ell_o} q' d\ell}{\int_0^{L} q' d\ell} \quad , \tag{4.22}$$

where ℓ_o is the length of coolant passage to the point of maximum fuel center line temperature.

These two parameters are being used to obtain the temperature rise of the coolant in the channels.

4.5 Effects of Temperature Changes in the Reactor

We have previously discussed in Section 2 the long term effects on reactivity as being mainly the burnup and the buildup of poisons. An

effect which affects reactivity on a smaller time scale is temperature changes in the reactor. We will discuss these effects in the present section not only because the origin of the effects may be a heat and mass transfer one, but to show how the control of a reactor may be closely related to the heat transfer aspect of the system.

A temperature change in the moderator, the fuel and coolant affects the core reactivity in different manners. However, to illustrate the fact that the net effect is viewed as a short term effect we may mention that when the NRX reactor experimented its power execution in 1949, the power initially increased exponentially with a period of 33 seconds. This increase in reactivity did not last because as the temperature in the fuel rods increased it slowed down. Finally, the reactivity decreased at a faster rate as the heavy water got warmer and the reactor went subcritical. This shows that the net temperature coefficient on reactivity is negative for the NRX reactor and this self-regulating feature is very desirable for a reactor in general.

The effects causing changes in reactivity when the temperature is increased may be analysed by looking at the changes produced in the four factor formula:

$$k_{eff} = \eta \varepsilon p f P_f P_t \quad , \tag{4.23}$$

which, when written in a logarithmic form becomes:

$$\ell n k_{eff} = \ell n k_{\infty} + \ell n P_f + \ell n P_t \quad . \tag{4.24}$$

A change from temperature T to T + dT gives, by differentiating Eq. (4.24),

$$\frac{1}{k_{eff}}\frac{dk_{eff}}{dT} = \frac{1}{\epsilon}\frac{d\epsilon}{dT} + \frac{1}{\eta}\frac{d\eta}{dT} + \frac{1}{p}\frac{dp}{dT} + \frac{1}{f}\frac{df}{dT} + \frac{1}{P_f}\frac{dP_f}{dT} + \frac{1}{P_t}\frac{dP_t}{dT}, \quad (4.25)$$

which is a form depending on the changes in each of the factors.

Before explaining the variation of each factor let us first mention the main effects produced in the core with a temperature increase; these effects may be divided into three categories:

(i) Thermal Expansion Effects

Since the density of the moderator will decrease by an increase of temperature, this will cause the absorption and scattering cross-sections Σ_a and Σ_s to decrease and therefore the chances of neutron escaping will increase. The fuel element expansion with a temperature increase will also reduce the coolant volume around these elements. This will tend to increase reactivity since there will be a decrease in the neutron capture between the fuel elements.

(ii) Direct Nuclear Effects

High temperature in the fuel will cause Doppler broadening of the resonance capture peaks in the cross-section curve. The neutrons will therefore be captured on a wider energy band than with colder fuel and the effect will be a decrease in reactivity caused by a decrease in the resonance escape probability factor p.

(iii) Indirect Nuclear Effects

The fact that the Maxwell-Boltzmann distribution for the thermal neutron spectrum will be shifted toward higher energies, will tend to decrease the cross-section for 1/v absorbers. Therefore, the absorption in the fuel will be reduced and the thermal flux depression in the fuel

will become less severe. The average fuel flux will increase and consequently the thermal utilisation factor f will increase also. Another indirect nuclear effect is one associated with the burnup of the fuel or the quantity of Pu-239 present in the fuel. As a matter of fact, the cross-section of Pu-239 does not behave like U-235 and has a fission resonance at about 0.3 eV. So, as the neutron temperature is raised more and more neutrons will be found in this resonance peak and as a result, there will be quite a significant increase in the thermal fission factor η for Pu-239, while for fresh fuel with only U-235 as fissile material, η normally decreases as the neutron energy increases. For fresh fuel and low burnups this effect will be negative and for high burnups it will be positive.

Since temperature changes in the fuel, coolant and in the moderator affect the reactivity in different ways, we will now consider them separately and thus consider their influence on the factors of Eq. (4.25).

4.5.1 Fuel Temperature Coefficient of Reactivity

Two primary effects are encountered in the fuel. First, an increase in fuel temperature results in increasing resonance capture by Doppler broadening of the resonance peaks. Secondly, the neutron temperature being increased this affect the reaction rates as explained above under indirect nuclear effects. Consequently, as we can see by Table 8, the net effect will vary depending on the burnup of the fuel but still, the predominant term is the resonance capture term which tends to reduce reactivity.

4.5.2 Heat Transport Coefficient of Reactivity

Since a greater number of collisions occur in the coolant than in the fuel, a temperature increase in the coolant will cause a greater effect than in the fuel. The effects will be due to a greater increase in neutron temperature. Also, an associated reduction in coolant density will cause a slight increase in f (less absorption), an increase in ε (less slowing down in the coolant), an increase in fast and thermal leakage and an increase in p. We present in Fig. 28 the overall temperature coefficient for the Pickering units as it has been calculated from the design data. The determination of such a curve from direct measurements is obviously very difficult because one would have to isolate coolant temperature changes from fuel temperature changes.

4.5.3 Moderator Temperature Coefficient of Reactivity

Again the two main effects are the change in density which is this time the moderator density, and the increase of the neutron temperature. However, the temperature of the moderator affects the neutron spectrum much more than the fuel or coolant does because the moderator temperature has the biggest influence on the thermal spectrum. Consequently, the magnitude of the moderator coefficients are greater than the other two as it is shown in Table 9.

4.5.4 Net Effect of Temperature Change on Reactivity

As we have already mentioned, it is very desirable for the temperature coefficients to be negative so that the self-regulating feature is provided. However, the net effect is not dictated by the algebraic sum of the coefficients but by their time constants. For example, a reactor

with a fuel temperature coefficient of -10 μk/°C and a coolant temperature coefficient of +15 μk/°C will still be inherently stable because the fuel temperature change due to a change in neutron flux will be rather fast in comparison with the change in coolant temperature; the latter change being delayed by 7 or 8 seconds due to greater heat capacity of the coolant. The change in moderator temperature will still take a longer time, and its temperature coefficient is irrelevant from this point of view. This means that by the time the moderator responds, the action is all over.

To further illustrate the net effect of temperature on reactivity we may mention an experiment which has been performed on Pickering Unit 3 when it contained fresh fuel. This was in order to measure the change in reactivity due to a change in temperature of both the coolant and the fuel, and the measurements extended over a period of 13 hours so that it may be assumed that the fuel temperatures kept in step with the coolant temperatures. The result was that the negative effect caused mainly by Doppler broadening in the fuel more than compensates for the positive effect of the coolant, and the reactivity was decreasing uniformly as the temperature was increased.

The CANDU PHW reactor may, on the basis of these arguments, be considered to have a negative temperature coefficient of reactivity. For the case of a large power excursion accident, the broadening effect of the resonance peaks would probably act as an ultimate shutdown system for the reactor. The case of the Boiling Light Water reactor Gentilly-1 is quite different and is the only Canadian reactor to have a positive temperature

coefficient of reactivity. The reason for this is primarily due to the fact that an eventual loss of coolant which is light water, would cause a big decrease in the neutron absorption by the coolant and hence, an increase in reactivity.

5. SAFETY OF CANDU

5.1 Design Philosophy of Safety Systems

In the Canadian reactor safety approach, the purpose of the safety systems is to limit radioactive releases to the public for two classes of events: a single failure in a process system combined with the coincident failure of one of the safety systems, a dual failure.

For the case of the 600 MWe CANDU reactor, all the systems may be divided into two groups for the purpose of safety assessment. Group one systems comprise the normal plant operating systems which are the process systems and the first line of defense safety systems. More specifically, they are:

(i) all process systems except auxiliary moderator cooling;

(ii) shutdown system number one;

(iii) emergency core cooling.

Group two systems comprise systems that, acting together will be capable of independent reactor shutdown, radioactivity release prevention and decay power removal. In particular, they are:

(i) shutdown system number two;

(ii) containment (including dousing, isolation, etc.);

(iii) auxiliary moderator cooling;

(iv) emergency power supply and water supply.

Since it is probable that any one of the safety systems may fail

to perform its function to counteract a single process system failure, the design has to ensure that the possible release to the environment remains under the limits derived from the Atomic Energy Control Board Siting Guide. We show in Table 10 these limits for single failures and dual failures.

The existence of two reactor shutdown systems reinforces the probability that at least one of them will operate following any single process failure for which both are designed to be effective. However, credit is not taken for operation of both shutdown systems acting together. In the case of a single process failure for which only one shutdown system is designed to be effective, if the shutdown system fails to operate, the releases are within the limits derived by the Siting Guide for dual failures.

In addition to the dose limits, the AECB Siting guidelines specify two requirements of the safety systems which are based on a risk frequency approach.

(i) The safety systems must be independent of the process systems and independent of each other.

(ii) Each safety system has to be designed for an unavailability of less than 10^{-3} year per year.

The availability requirement of the safety systems is met by the fact that each process and nuclear measurement loop essential for the operation of a safety system is triplicated. Therefore, a single loop component or power supply failure will not incapacitate the operation of the safety system. The isolation between loops of different channels and between the different safety systems is achieved by the use of

unique transmitter mounting racks, electrical cubicles and power supplies for each channel.

5.2 Different Safety Systems

Before giving a description of the main safety systems of the CANDU PHW reactor, we may note at this point, even at the risk of repeating ourselves, that the main philosophy of the safety systems is to minimize the environmental impact that could be caused by the possible accidents which may happen in the other process systems. From this point of view, the set of safety systems represents the interconnection between the operating aspects of a nuclear power plant like the heat and mass transport systems, fuel management, reactor control, etc. to the environmental aspect that will be treated in the next chapter. This shows us the importance attached to the safety systems and also that the conception of each aspect of a nuclear plant is directly a function of the other aspects through the safety of the whole system.

5.2.1 Shutdown Systems

The shutdown systems are not basically designed to provide the protection of the owner's investment but to safely stop any power excursion initiated by a process failure. Both loss of control and loss of coolant can increase the core reactivity and start a power excursion. The reactivity rates following such an excursion are much larger than the control system can provide and this activates the shutdown system to operate.

Basically, a shutdown system puts a neutron absorbing material

into the core to decrease the reactivity and turn off the power generation. The systems commonly used in recent designs are shutoff rods which gravity drop into the core, shutoff rods which are gravity-drop assisted by springs, and liquid poison injection into the moderator.

For the 600 MWe Standardized CANDU reactor, the first shutdown system is the release of 28 spring-assisted gravity-drop shutoff rods. When the requirement for reactor trip is sensed by the independent triplicated logic system, this de-energizes the direct current clutches to release the shutoff rods. The second shutdown system, used when certain combinations of very unlikely failures occur, is the rapid injection of a concentrated gadolinium nitrate solution into the bulk moderator through six horizontally distributed nozzles. Another independent triplicated logic system senses the requirement for this emergency shutdown and opens fast-acting helium pressure valves to inject the gadolinium poison into the moderator.

We may note that the Pickering reactor uses a partial dump of the moderator to augment the negative reactivity effect of the shutoff rods. The shutdown reactivity worth in Pickering is 24 mk from the shutoff rods plus 6 mk from partial dump, totalling 30 mk.

5.2.2 Containment System

The containment system is designed to withstand the overpressure that could be created by a loss of coolant accident. The three types of containment systems that have been used for CANDU reactors are:

(i) the pressure relief systems,

(ii) the pressure suppression systems,

(iii) the vacuum systems.

The pressure relief system is only applicable for only remote areas, it was used for the NPD and Whiteshell reactors. In this case the reactor vault room is designed to withstand an internal pressure of 68.9 kPa(g) and the boiler room to withstand 34.5 kPa(g). A combined pressure relief duct and dousing system are used to keep the pressures within these limits. The condensing action of the dousing system helps to keep the reactor vault pressure below 68.9 kPa(g). Before the dousing system is fully operative, the pressure relief duct is used in turn to keep the boiler room pressure below 34.5 kPa(g) in case of the fracture of a 40.6 cm pipe.

The pressure suppression system is used at Douglas Point and Gentilly-1 and 2 where a dousing system similar to the NPD system is used for pressure suppression. For these reactors, the pressure relief system is not employed and the leakage rate out of the enclosure must be limited to about 0.1% of the total volume per hour.

For Pickering and Bruce multi-unit stations, the vacuum containment system consists of a separate building which is maintained at a vacuum. A dousing system is also provided in this vacuum building to condense incoming steam following a loss of coolant accident.

5.2.3 Emergency Core Cooling System

The emergency core cooling system is a safety system intended to provide light water cooling to the reactor fuel in the event of a loss of coolant accident. The initial light water supply used by the system is taken from the dousing tank in the reactor building and a continuing supply

is provided by pumping the water from the reactor building basement to the heat transport system.

The purpose of the emergency core cooling system is to maintain the fuel sheath near its normal operating temperature after a loss of coolant accident. Any increase in the normal sheath temperature would decrease its strength and the internal pressure caused by the gaseous fission products inside the sheath, accentuated by the loss of the outside coolant, could lead to the release of these fission products in the containment system.

We should mention to this point that a possible failure of the emergency core cooling system does not lead to a core meltdown because of the presence of the moderator which acts as a heat sink having a considerably lower temperature. We can see that this distributed heat sink in the core is an important safety feature of the CANDU reactors which is not the case in pressure vessel reactors because both coolant and moderator are discharged through a pipe break.

5.2.4 Water and Power Supply

The emergency water supply may provide an alternate source of water in case of the unlikely possibility that feedwater to the primary heat transport system, or to the steam generators should fail, or that service water to the moderator heat exchangers should fail. The system consists basically of two 100% diesel engine driven main pumps which supply either demineralized water from the demineralized water storage tank or raw cooling water from a separate intake structure. This water can be supplied through a simple distribution system to the boilers as

feedwater to the heat transport system for makeup or to the moderator heat exchanger for cooling.

The emergency power supply system may supply the necessary power to the emergency water supply system valves and provide power to the group two safety and control systems. A connection to the moderator cooling system also ensures that the moderator can always be cooled.

5.2.5 Multi-layer Defense Against Radioactive Releases

The different safety systems mentioned above can be used in case of an accident susceptible to lead to radioactive releases but the original conception of the CANDU reactors itself represents, to some extent, a safety feature against accidents.

The most important radioactivity to contain, relatively to accident analysis, comes from the fission products in the fuel. The most significant of the fission products are the noble gases such as krypton and xenon and the isotope of iodine, cesium and strontium. The CANDU system provides, by its conception, some barriers which prevent these fission products from reaching the public in any significant concentration.

Most of the fission products remain trapped in the UO_2 matrix of the fuel but a small fraction of the gaseous fission products may be released from the matrix. The fuel sheath acts as the first barrier in withstanding the pressure resulting from UO_2 expansion and from these gaseous fission products. The coolant surrounding the fuel bundles is contained in a closed heat transport system forming the second barrier to fission products release. The zirconium alloy pressure tubes inside the core are designed with a security factor of about three between the

working and ultimate stresses. The containment system itself is the next barrier to fission product release and is designed to attenuate these releases from the heat transport system by a factor of 10^6. Finally, the exclusion of the public from a zone of about 900 m radius from the plant provides the reduction of any release from the containment system by a factor of 10^2 to 10^3.

This multi-layer defense between the public and the fission products produced in the fuel provides a total attenuation of between 10^8 and 10^9.

5.3 Operational Aspects of the Safety Systems

In the design of the operational sequences of the different safety systems it is important to distinguish the failures of control systems and failures of process systems. First, the failures of the control systems are treated in such a way that the worst failure mode is identified and it is assumed that if adequate protection is provided against this then all the other less serious failure modes are covered. The worst failure is some combination of events which drives all reactivity devices positive at their maximum speeds. This produces the highest rate of reactivity increase possible from the control system.

Failures in the process systems are treated in a more complicated manner because each process system is considered in turn. Component failures are postulated and the consequences have to demonstrate agreement with the single failure dose criterion. In the single failure cases, the operation of all four safety systems is credited.

Once the single process failures have been analysed, a coincident failure of each safety system is then postulated in combination with each process failure to demonstrate compliance with the dual failure dose criterion. For instance, the worst failure mode of the control system is combined with coincident failure of shutdown system No. 1, then shutdown system No. 2, then emergency cooling, and finally containment. In each case, the other safety systems are assumed to operate.

The accident which is much more severe than all the others is the loss of coolant accident. This accident has not happened yet in the CANDU system but such a postulated failure sets the design requirements for all four safety systems so that safety systems designed to these requirements are more adequate for all other process failures.

5.4 Radiation Exposure Management

During reactor operation, the exposure of the surrounding population is limited by exclusion from the plant area used by preventing, in accordance with AECB requirements, any habitation nearer than 915 m (3000 ft). The release of all affluents, liquid and gaseous, which might possibly carry significant activity is monitored and controlled. Active solids are disposed of in a form which prevents release of activity. Thus, any activity which may reach the public through the air or in the water can be maintained below permissible concentration levels.

Exposure of plant personnel to radiation hazards is limited by shielding and controlling access to areas of high activity or of possible contamination. In addition, radioprotection devices like protective

clothings, air masks and decontamination facilities are available for use when required.

The considerations involved in routine radiation protection may be divided under two basic categories:

(i) those relating to protection from direct radiation, i.e. layout, shielding, area radiation monitoring, access routes and access control;

(ii) those relating to active contamination in all forms, i.e. waste management, zoning, ventilation, decontamination, protective clothing.

The principles underlying the approach of the designers in providing radiation protection for a nuclear facility are mainly the following:

(i) Segregation or isolation of sources of radiation as much as possible. The interior of a nuclear plant is usually divided into three main areas: the accessible areas, the shutdown areas only accessible when the reactor is shut down and the permanently closed areas.

(ii) Shielding by distance and by use of dense and neutron absorbing materials. The areas throughout the plant have allowable radiation levels based on the regulations and adjusted for occupancy.

(iii) Control of exposure time and use of activity decay.

(iv) Removal of contamination and dispersal or remote storage of the resulting active waste products.

(v) Supply of adequate clothing and clean air for workers.

The standards of exposure dosages are based on those published by the International Commission of Radiation Protection (ICRP) and adopted by the AECB, and are used together with any additional provisions set up by the plant owner. The objective of the designers is that by using normal procedures, the workers will not receive a total exposure in excess of ten times the public allowable values shown in Table 11.

5.5 Licensing and Regulations

In Canada, all atomic energy matters are governed by the Atomic Energy Control Act of 1946. This act clearly specifies that it is to govern the development and control of atomic energy and it provides explicitly for the creation of the Atomic Energy Control Board. The licensing and regulation of nuclear power plants is one of the many functions that fall under the Atomic Energy Control Board responsibilities by this act.

Before deciding to grant a license for a nuclear plant operation, some requirements must be satisfied:

(i) The necessary competence for the design and procurement of a nuclear station was available to the applicant.

(ii) Adequate information has been submitted concerning the design and the site.

(iii) Appropriate assurance has been given for the

performance of the reactor.

(iv) Sufficient controls have been applied in the design, component manufacture and installation phases to ensure the desired performance.

(v) The administrative, operating and maintenance discipline appropriate to the use of the plant are documented and will be put in effect.

Two types of licenses are required from the AECB for the operation of a nuclear facility: prescribed substance licenses and a facility operating license. The prescribed substance licenses authorize the acquisition and possession of nuclear fuel, heavy water and radioactive materials used at the station. The operating license authorizes reactor operation and specifies operating limits and restrictions. Before these licenses are granted, a construction permit must be obtained from the AECB.

The major hazard arises from the large inventory of radioactive fission products produced and contained in the fuel. Therefore, the criteria imposed on the safety by the AECB are directed toward minimizing the chance of mechanical failure of the fuel and minimizing the escape of fission products from the plant if fuel failure occurs. The chance of fuel failure mainly depends upon the adequate control of heat removal and power produced by the fuel. The escape of fission products is prevented by the presence of the high integrity barriers previously described; the most important being the final containment.

To specify the requirements to be met by the designer and operator, the AECB uses the concept in which the nuclear plant is assumed

to consist of three systems: the process system, the protective system and the containment system. If these systems are independent of one another and if each is of sufficient reliability, the chance of a significant release of radioactive material to the public domain can be kept extremely small.

During the initial years of operation of a nuclear plant, the AECB keeps its resident inspectors at the site. They monitor the operation, verify the performance of tests, assess the safety systems reliability and approve any significant modification. They also maintain surveillance over the release of radioactive effluents in consultation with provincial and environmental control agencies.

Before giving a construction permit for a nuclear plant, the AECB requires the study of the future siting of the plant. Special items such as meteorology, seismic characteristics, geological data, land use and population distribution must be thoroughly analysed. The nuclear power station is designed to withstand much more than the maximum earthquake movement that could be anticipated at the site. Similarly, stations are designed to prevent damage from the severest of storms. This also applies to the impact of debris and other objects that could be propelled by storms. An interesting figure which illustrates the solidity of a reactor building is that in case of the impact of an aircraft such as a fully loaded jumbo jet (360 tonnes), the plane would have to be travelling at 1600 km/h to penetrate a Pickering type reactor building. Also, the probability of the impact of any kind of aircraft on a plant is about once in a million years.

The interest that always has been given to the safety of nuclear plants requests that an acceptable report on a site must be presented to the licensing authorities before their construction. Another interest, more recent than the safety one, is the public interest in concern for the environmental impact that we shall discuss in the next section. In the nuclear industry, of course, that has always been a serious consideration but among the population, when a nuclear plant may be very welcome in a particular locality for economic reasons, environmental groups are likely to be anxious about the effects they have on the local ecology.

6. ENVIRONMENTAL IMPACT

6.1 Release of Radioactive Materials to the Environment

The complete removal of radioactivity from effluents discharged to the environment is practically impossible to achieve, so that these effluents will always contain some amounts, although small, of radioactive material. The radiation dose standards for these effluent releases, set by the Atomic Energy Control Board, are based on the recommendations of the International Commission on Radiological Protection (ICRP). These standards are that individual members of the public shall not receive a radiation dose in excess of one tenth of that permitted to an occupational worker; and that the population surrounding the station will not receive an integrated dose of 10^4 man-rem due to station operation. The reasons why the allowed dose for the workers is 10 times the allowed population dose, as previously shown in Table 11, are (i) because of the presence of growing children in the population, (ii) because workers have more often medical examinations than the average population, and (iii) because workers dose is constantly measured by film badges and other measurement devices.

In addition to the gaseous and liquid effluent monitoring, an environment surveillance program is carried out around nuclear stations to ensure that regardless of the derived release limits, primary dose standards are not being exceeded. This also permits us to gain knowledge of environmental radioactivity levels and the impact on the environment so

that corrective actions may be taken if required. These data, obtained over a number of years, may be used as a rational basis for adjusting the permissible release levels.

6.1.1 Source of Radiation

The principal radioactive materials generated in the plant are, as already said in the previous chapter, the fission products. Possible escape of some fission products through minute imperfection in the fuel element cladding account for some radioactivity picked up by the coolant. These fission products, amounting approximately 0.001% of the fission products inventory of the station, are principally the gaseous and more easily vapourized solid constituents of the fission product mixture. Another source of radioactivity is the activation products. These include activation products formed in the water, most of which have very short half-lives, activation products that are formed in reactor structural material and enter the coolant through corrosion and erosion, and activation products formed in the air around reactor core which is the only part of the reactor where neutrons are present. To maintain the purity of the water and to limit the amount of radioactivity in these systems, the heat transport system and the moderator system are purified.

Referring to Table 12, we note that the gaseous effluents of importance are I-131, tritium and radioactive noble gases. For the liquid effluents of importance, since the reactor systems in the CANDU type reactors are closed systems, the only radioactive liquid which is released from the station is very low level wastes from laboratories, decontamination facilities, etc. The radioactive liquid waste water is collected in

one of several large tanks and monitored before being released to the large condenser cooling water flow.

6.1.2 Exposure of the Public to Radioactive Releases

The Canadian safety approach recognizes that the risk people will accept is dependent on the frequency of the event which puts them at risk. For the purpose of safety analysis, this recognition is quantified at two levels depending on if an eventual accident is a single failure or a dual failure.

In setting the dose limits for single failures, the AECB judged that the overall frequency of a serious process failure requiring the intervention of a safety system should not exceed one per three years. The dose limit for failure conditions have been mentioned in Table 10 in the preceding chapter but they are dose limits not to be exceeded. These doses are not harmful and still, they are higher than those associated to continuing releases related to normal plant operation. At the present, the releases based on effluents from a single station are satisfactory; however at some time in the future, with a larger number of nuclear power stations, the limits may have to be adjusted downwards to take into account the possibility of the buildup of a radionuclide. Presently there is very little release due to CANDU stations; these releases from all CANDU stations in operation are generally less than one percent of the derived release limits. The results from Pickering are shown in Table 13.

It is interesting to mention that in North America, on the average an inhabitant receives about 100 mrem (0.1 rem) per year from various natural sources. We also show in Table 14 that the average man-generated

radiation dose, not necessarily coming from nuclear power, received by a person in Canada is about 80 mrem per year and 0.003 mrem if it comes from nuclear power.

6.1.3 Effects of Radiation

To illustrate furthermore the safety of the recommended dose limits we will discuss in this section the effects of radiation on humans and at what dose these effects are encountered. Briefly, if a dose received is large there may be immediate effects and there may also be delayed effects, depending upon whether the dose is large or small.

A large dose of radiation received over a short period of time may produce a variety of immediate effects depending on the severity of the dose. For gamma radiation below 25 rems, generally, there are no observable immediate effects but at the other extreme, a dose of 1000 rems received within a few hours will result in death within 30 days of anyone who receives it. Between these two values there is a progressive worsening of effects which include sickness, diarrhea, loss of appetite, loss of hair, and bleeding. At 450 rems, about half the people exposed would likely die within 30 days and it is unlikely that death would be caused by doses less than 200 rems.

The biological effects against which people are to be protected are the acute effects and the late effects. The acute effects result within a few hours or days, from doses of tens or hundreds of rads delivered over a few hours or less. The late effects which result years later, from external and internal radiation may be of two types:

(i) Somatic - malignancies (leukemia or cancer) in

the irradiated individual;

(ii) Genetic - death before birth or handicaps in the live born of the off-spring of the irradiated individuals.

We list in Table 15 some of the acute effects of large doses of ionizing radiation to the whole body. Acute effects frequently seen from irradiation of the skin are epilation from about 300 rems or more and erythema from about 1000 rems or more. If all doses to the skin or the whole body are kept below 25 rads, one could be reasonably certain of avoiding all acute effects. This means that there is a definite threshhold for acute effects.

The situation with the late effects is quite different because one cannot see in an individual the effects of doses of the size that are considered. An important fact is that both the malignancies and the genetic harm produced by radiation are indistinguishable from those occurring naturally in people, radiation only increases their frequency. Since no threshold was assumed, one cannot speak of a "safe level" for late effects. And since one cannot know if a late effect was produced in any one individual by radiation or by something else, one can only refer to the probability of a late effect.

To summarize the concepts adopted by the ICRP it can be said that for acute effects there is a threshold dose, at a specified dose rate for each early effect. At a dose and dose rate sufficiently below the threshold the risk of acute effect can be safely concluded to be zero. For delayed effects it is assumed that:

(i) the increment in biological effect per unit increment in dose is the same at all doses, i.e. there is a linear non-threshold dose relation, and

(ii) the biological effect of a given dose is independent of the rate at which the dose is delivered.

These hypotheses permit two things without which it would be impossible to assess compliance with dose limits:

(i) the simple addition of all doses received by the same tissue of an individual at any time and at any dose rate;

(ii) the simple addition of all doses received by the individuals in a population to give the population dose as well as the averaging of these to give the mean dose to individuals in the population.

It is important to note that the above assumptions about delayed effects have two consequences:

(i) The risks of harm from low doses and low dose rates will, in most cases, be overestimated.

(ii) It is not possible to think of a safe dose, i.e. a dose with no risk of harm. Rather, one must consider a dose sufficiently low to reduce the risk of harm to an acceptable level.

6.2 Radioactive Waste Management

Among the radioactive materials present in the spent fuel of a

nuclear reactor, some of these, the fission products and the actinides americium and curium, still do not have any economical value and are considered as wastes. In this sense, plutonium is not a waste material but any fuel reprocessing or fuel fabrication facility will, however, produce waste streams containing small residual amounts of plutonium; radioactive wastes thus include plutonium.

Radioactive waste management embraces all the activities that are being done to ensure that these radioactive materials are always handled and stored so carefully that only insignificant amounts could ever escape to the environment. Because of the long radioactive decay time of some of these materials, this is a demanding exercise. Also, the time scales involved raise questions about the responsibilities toward future generations.

6.2.1 Objectives of Waste Management

For the reasons mentioned above, it is useful to identify two objectives for radioactive waste management:

(i) The health and safety objective: radioactive materials should be managed so that the health hazards are negligible.

(ii) The responsibility objective: the wastes should be managed so that trouble and concern to future generations will be minimized.

It should be noted that these two objectives can conflict. For any kind of container in which these materials could be kept, it is very difficult to give an absolute guarantee that none of the material will ever escape or that the container will never require monitoring or main-

tenance. The objectives of zero release and no future responsibility are essentially unattainable because they are ideal and absolute. In this sense the two objectives should be restated as to be practical minimization of both the health hazards and the future responsibilities.

6.2.2 Classes of Wastes

Three classes of radioactive wastes arise from CANDU reactors:

(i) The dispersed wastes. Very small amounts of radioactive effluents, gaseous and liquid which are carefully monitored and well below the permissible limits.

(ii) Stored wastes. These recovered solid wastes must be stored in an isolated place. Some of them may contain relatively short-lived isotopes and in these cases the storage period can be fairly short.

(iii) Spent fuel. This is a special category and can be considered a waste which must be stored, or a valuable by-product from which plutonium can be recovered. In either case, a highly radioactive solid containing 99+% of the fission products must be stored in a special way.

Table 16 shows the solid wastes that could be expected from a mature CANDU-PHW station. The non-combustible waste is mainly metallic piping, failed component, etc. Since the activity is low, it is stored at the waste management site.

The spent fuel contains neutron capture products and 99+% of the

fission products and is giving off considerable heat. First the radioactivity must be contained for a very long period of time because of the long decay time of some radioactive elements. Also, the heat must be dissipated because a bundle, when it first arrives in the fuel bay is generating about 6 kW of heat. This however decays rapidly and after one year in storage it is giving off about 85 watts.

6.2.3 Options for Wastes

In the CANDU-PHW system, no financial value is presently assigned to the spent fuel, a condition which allows considerable freedom in the methods available for dealing with it. Figure 29 shows the possible options in schematic form.

With the natural uranium fuel cycle, spent fuel is currently in interim storage in water filled pools. If it comes to be decided never to reprocess then the fuel would eventually have to be moved to an ultimate storage site. It if were decided to reprocess then one would have waste streams containing fission products and actinides and two product streams, plutonium and uranium. In Fig. 29 we show the fission products and actinides separated with the actinides possibly recycled to the reactor to be transmuted into shorter lived material; this procedure would however cause a burnup penalty of about 15%. As there would always be some material stored in inventory at a reprocessing plant awaiting shipment to the ultimate store, both interim and ultimate storage are shown for waste streams.

A major step in waste management is the conditioning of the wastes into a form suitable for ultimate storage. As the main objective is the

isolation from the environment, the idea is in general to convert wastes to a solid insoluble which is then incorporated into another inert insoluble matrix such as bitumen or inorganic glass. Waste solidification and conditioning is an essential part of the flowsheet of a reprocessing operation and would require a large fraction of the effort in a reprocessing development program.

6.2.4 Storage of Wastes

The two possible kinds of storage are interim storage and ultimate storage. We will discuss them separately by giving their possible way of application.

(A) Ultimate Storage

There are four general possibilities for ultimate storage or disposal of radioactive waste:

(i) ejection through outer space for the fission products or the actinides;

(ii) transmutation into shorter lived or inactive material of the fission products or actinides;

(iii) storage on or near the surface;

(iv) burial at some depth beneath the earth's crust.

The ejection into space has been studied by NASA and the USAEC and is technically conceivable today. These studies show that with some further developments of space technology, it could be economically feasible for separated actinides (americium and curium) but not for separated fission products or for irradiated fuel.

The transmutation of fission products into inactive material does

not appear to be feasible because the cross-sections of the required nuclear reactions are not large enough. The surface storage considered as ultimate storage is an idea which conflicts with the responsibility objective; the concept however, may be utilized on an interim basis.

It seems that presently, the deep burial scheme represents the most safe and economical way in which ultimate storage could be applicable. The idea is to bury the wastes into a deep rock formation which should have the qualities of (i) having low economical value, (ii) being a large formation, (iii) being a geologically stable and of low seismic activity formation, and (iv) isolated from moving ground water. These qualities can be met, for example, in salt beds which are well known as having been stable for hundreds of centuries.

(B) Interim Storage

For interim storage, the design of the first facilities will have to guarantee that the fuel is under control at all time no matter what unexpected happening might take place. Since interim storage will be applied to spent fuel, the fuel must be recoverable in case it has to be removed to another location for permanent storage or reprocessing. The safety of the storage requires the presence of barriers between the radioactive material and the environment and that the monitoring of the releases should be continuous. The economical aspect of interim storage implies that the storage system must be simple with a minimum of equipment which can fail; the annual operating cost should therefore be low because such a facility would be required for a certain number of years.

The different facilities that have been studied for interim storage

consist mainly in wet storage and dry storage. The wet storage uses the concept of large water filled pools for spent fuel at a central storage site. The dry storage is studied essentially as a back-up to the pool concept. Of several dry storage concepts studied by CAFS (Committee Assessing Fuel Storage), the so-called "canister" appeared most attractive because of its simplicity and the relatively small cost of development and demonstration.

6.3 Thermal Effects

All methods of energy conversion are thermally inefficient and the amount of heat rejected depends on the efficiency of the power plant. For nuclear plants the efficiency of the system is about 30%, this means that approximately 70% of the thermal energy generated is dissipated to the receiving water or atmosphere as heat. At present, modern fossil fuel stations are somewhat more efficient than are nuclear stations but also, they reject some of the heat through smoke stacks which contribute to some extent to air pollution, (see Fig. 30).

Typically, in Canada, the discharged water of nuclear plants is about 9°C warmer than its intake temperature. The volume of cooling water is also considerable, for example, the 3000 MWe nuclear station at Bruce is designed to use 9100 m^3 of cooling water per minute. It has been estimated that all the thermal discharge from all the electric power stations expected to be in use by the year 2000 on the Great Lakes would cause an average surface temperature rise of 0.22°C in the summer. This means that the lake wide temperature increase would be less than the normal year to year variations in the Great Lakes.

In practice, however, the heat is discharged near the shoreline where most biological activity such as fish spawning takes place. Since the temperature in a body of water can determine the species that will live and reproduce, ecologists are deeply concerned with the threat to fish and other aquatic life forms which can occur in the future. The effects of thermal discharge are of two types: the physical effects on the mass of water and the biological effects on the living species.

6.3.1 Physical Effects

Basically, three physical processes operate to determine the resulting pattern of temperature increase within the receiving water and the area over which this increase will be observable. The processes being (i) mixing, (ii) stratification, and (iii) atmospheric transfer, are interrelated and subject to a variety of determining influences.

Mixing implies the diffusion or spreading out of the heat within the receiving water, such that temperature effects on the environment tend to become more evenly distributed throughout the water body. However, where such physical mixing forces as current and wave action are weak or absent, mixing is significantly slowed and thermal stratification becomes an important factor.

Stratification occurs when less dense liquids rise to the surface. Since the density of water increases as the temperature decreases, at least within the range considered here, the heated effluents have therefore a natural tendency to stratify and raise to the surface. The third process, atmospheric transfer, is the phenomenon by which the excess heat flows to the atmosphere by evaporation, convection and radiation in order to restore

the original equilibrium temperature of a water body. Since the heat loss to the atmosphere is more rapid when the surface water is warmer, thermal stratification tends to increase the atmospheric transfer process.

Another physical effect that we may mention is the effect of temperature on gas solubility and gas diffusion in water. The solubility of oxygen for instance, decreasing with temperature, is important by its role to sustain many forms of aquatic life.

6.3.2 Biological Effects

One of the responses of aquatic organisms to elevated temperatures is a shift in the population structure of the ecosystem. Any stresses caused by temperature changes will tend to reduce the number of species by elimination of those which are thermally intolerant. In the case of fish, the lethal temperature is a function of many factors but observations have shown that for many of them, a large increase in mortality can result from a small change in temperature near the tolerance limits.

Extreme temperature may kill but for temperatures within the zone of tolerance the physiological responses which force a species to a more unfavourable competitive position may be termed sublethal. Such responses may include growth rate, increase susceptibility to disease and predation, reduced rate of reproduction and decreased resistance to toxic substances.

It is known that the upper limit of the heat tolerance range is not an unalterably fixed point since most organisms can be acclimated to live at higher temperatures by intermittent or gradual exposure. In general, young animals are more easily acclimated than more mature animals.

Observations of heated water discharges have indicated that fishes

tend to move toward them in colder months and to move away during summer months. The phenomenon of temperature selection by mobile organisms is a type of protective response by which these organisms tend to congregate in a definite narrow range of temperature.

In short, there is no doubt that some changes will occur in the ecosystem of a water body but the biological problem is to determine exactly the degree of these changes both on the long term and short term and if they significantly affect water uses. A lot of research is presently being done on these thermal effects and it is hopeful that the results will help to find a way to use thermal discharges for human benefit. Some interesting prospects are, for instance, fish raising or marine plants culture as source of proteins.

6.4 Public Attitudes

The concerns of society and public attitudes to nuclear power have become an important factor in planning for future nuclear generating capacity in Canada. While we have escaped the strong public reaction against nuclear power evidenced in U.S. and other countries, with the 1970's the situation has changed and since this time Canadians are showing an increasingly active interest in nuclear matters. A number of new groups opposed to nuclear power developments have formed in 1974 and there have been attempts to form a federation of such groups across the country and regionally.

In order to know what fraction of the population agrees with the opinion of these groups, in 1974 Ontario Hydro has conducted research into

public attitudes as part of a preliminary study in two areas of the province where future nuclear generating stations could possibly be located. These two areas were Huron County in southwestern Ontario and the North Shore area of Lake Huron. In both these studies, random samples of the population were drawn and interviewed. Some of the specific observations obtained from the attitudinal data relating to nuclear power are the following:

(i) The proportion of the population which is opposed to nuclear power appears to be quite small, 4% in the North Channel study compared with 17% for a U.S. national study in 1974.

(ii) There are large blocks of uncommitted individuals often ranging in size from 1/4 to 1/3 of the population.

(iii) Women are generally less well informed about nuclear power than men and are more opposed to it.

A conclusion that can be drawn from these results is that it indicates a lack of information. One of the major problem areas in informing the Canadian public stems directly from the U.S. information flow which exerts great influence even though it often has little relevance to the CANDU system.

Among the groups opposed to nuclear power development those with an environmental or ecological orientation are probably the most important to the nuclear industry. The CANDU system has been very successful and is now a well proven system. Future education programmes about nuclear power

in Canada will however have a very difficult task to discount the arguments of the U.S. information flow and to present information about the CANDU system. Since future expansion in Canada is oriented to nuclear energy and since by the year 1990 about 28% of the electric production in Canada will come from nuclear plants, it is suggested that the groups opposed to nuclear development should present the relevant informations and employ their creativity in trying to reach acceptable solutions rather than simply organizing for opposition.

7. CONCLUSION

A quotation of A.J. Mooradian taken in one of his papers entitled "Energy for Five Thousand Years" shows fully the reliability of the Canadian nuclear system. Speaking of the CANDU reactor he says: "It is the most efficient burner of uranium yet produced. The spent fuel leaving the reactor contains less fissile uranium than the tails from diffusion plants which supply enriched uranium to the light water and the current generation gas cooled reactors".

The Atomic Energy of Canada Limited, relying on a long experience dating from their first prototype reactors in Chalk River up to the success obtained with the Pickering plant, has recently decided to develop a standardized form of the Canadian reactor, already known as the Standardized 600 MWe CANDU reactor, which would have general application in Canada and offshore. In Canada, the first plant of this type to be committed, will be the Gentilly-2 reactor (G-2). This does not mean that such a realization is a final product of the nuclear Canadian industry; on the contrary, future plants being presently designed by AECL will firstly be built in Ontario, with an electric power output of 1200 MWe per reactor.

The fuel cycle will see changes by thorium utilisation and plutonium recycling when the economical trend will be favourable to it. Further, organic coolants will improve the thermal efficiency of the system because, when used with adequate mixtures of fuel like uranium carbide, higher temperatures may be reached. This will also considerably decrease the

necessary mass of heavy water in the reactor, reducing in this way the costs associated with it. The reliability of the nuclear components will be improved since the more plants will be in operation, the best will have to be the safety of each of them in order to keep the failure risk to a low level. This also applies to the environmental aspect because the total radioactive releases must not exceed a certain maximum value, independently of the number of nuclear plants in operation.

There is consequently no doubts that the versatility of the CANDU nuclear system places it in a good competitive position as an energy source in Canada and abroad. The five reactor physics aspects that we have treated in this project show by their contents, that their relevancy to the system considered as a whole is essential but that these relations can not be conceivable without, firstly considering these aspects as to be strongly interrelated to each other.

8. TABLES

TABLE 1

Differences Between CANDU and PWR Fuel
(From: AECL Nuclear Power Symposium, 1974)

	CANDU PHW	PWR
Fissile Materials	Natural Uranium 0.7% U-235	Enriched 1.5-3%
Relative Total Fuel Cost	1	3 to 4
Relative Length of Elements	1	8
Relative Diameter of Elements	1	0.7
Relative Sheath Thickness	1	1.45
Relative Diametral Gap	1	2.3
Relative UO_2 Density	1	0.95
Fuelling	On Power	Off Power

TABLE 2

Composition of Zircaloy-2 and -4
(From: AECL Nuclear Power Symposium, 1974)

	Zircaloy-2	Zircaloy-4
Tin	1.20-1.70 wt%	1.20-1.70 wt%
Iron	0.07-0.20 wt%	0.18-0.24 wt%
Chromium	0.05-0.15 wt%	0.07-0.13 wt%
Nickel	0.03-0.08 wt%	-
Total Fe + Cr + Ni	0.18-0.38 wt%	0.28-0.37 wt%
Carbon	150-400 ppm	150-400 ppm
Oxygen	900-1400 ppm	900-1400 ppm
Zr + Permitted Impurities	Balance	Balance

TABLE 3

Conversion of U-238 into Pu-239 by Neutron Absorption

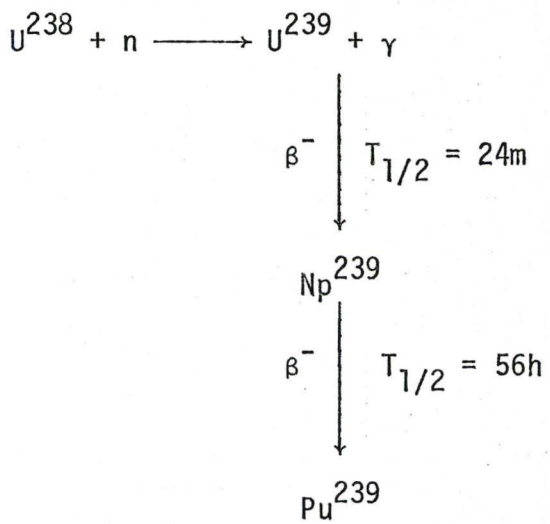

TABLE 4

Formation of Plutonium Isotopes
(The half-lives due to neutron capture or fission refer to a thermal flux of 10^{13} n/cm^2-sec)

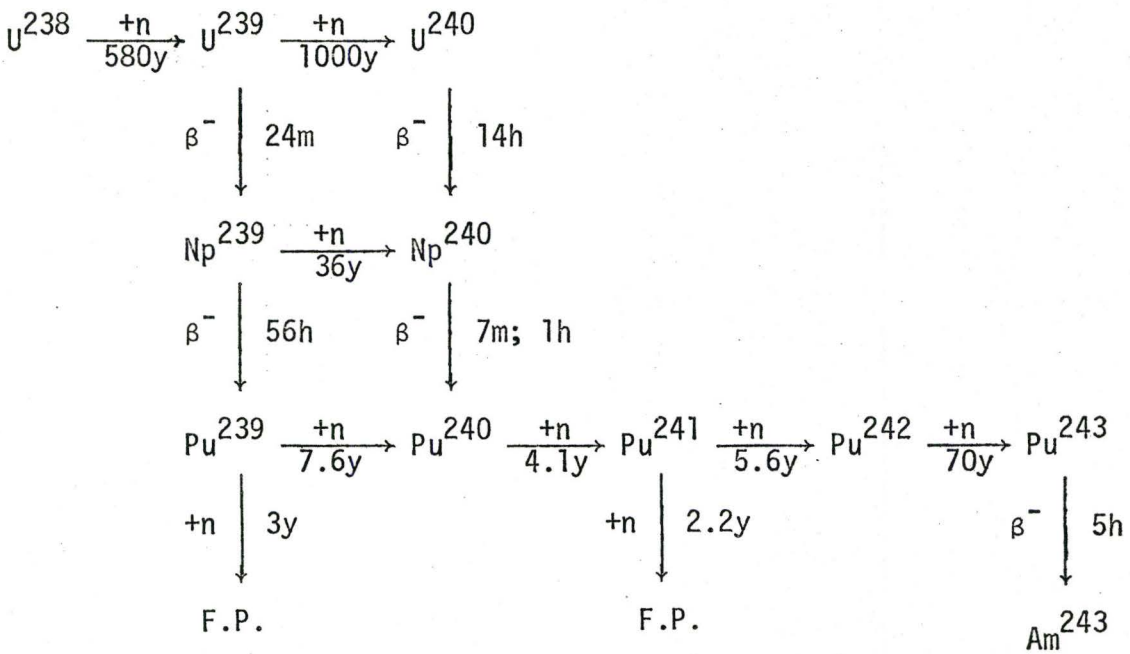

TABLE 5

Delayed Neutron Data for Thermal Fission of U-235 and Pu-239
(From: Ontario Hydro, "Nuclear Training Centre Course 127", 1973)

Group	$t_{1/2}$(sec)	$\bar{t}$(sec)	U-235		Pu-239	
			β_j % yield	$\beta_i \times \bar{t}$	β_j % yield	$\beta_i \times \bar{t}$
1	55.72	80.40	0.0215	1.729	0.007	0.563
2	22.72	32.78	0.142	4.655	0.063	2.065
3	6.22	8.98	0.127	1.140	0.045	0.404
4	2,30	3.32	0.257	0.853	0.069	0.229
5	0.61	0.88	0.075	0.066	0.018	0.016
6	0.23	0.33	0.025	0.009	0.009	0.003
Totals			0.65	8.5	0.21	3.3

TABLE 6

Heat Transport Systems Data for Pickering
(Source: Ontario Hydro, "Pickering G.S. Design Description", 1969)

General	
Volume of system	
- Main circuit	130.2 m^3
- Standby cooling and auxiliary circuits	22.8 m^3
- Storage tank	7.08 m^3
Total D_2O in system	∿158 Tons
Main Circuit	
Coolant flow rate (Tot.)	27.82 x 10^6 kg/hr
Reactor inlet temperature	249.5°C
Reactor outlet temperature	293.4°C
Mean pressure, outlet headers	8.82 MPa(a)
Mean D_2O temperature	268.7°C
Mean D_2O density	0.847 g/cc
D_2O temperature at low power	265.6°C
Design pressure (ext. coolant tubes)	11.03 MPa(g) (298.9°C)
Boilers	
Total steam output	2.93 x 10^6 kg/hr
Feed water inlet temperature	171°C
Minimum temperature difference between D_2O and H_2O	5.0°C
Steam pressure at drum	4.09 MPa(a)
Steam temperature at drum	251.7°C
Steam pressure at low power	5.13 MPa(a)
Design pressure	
- Shell side	5.38 MPa(g)
- Tube side	10.00 MPa(g)
Design temperature	
- Shell side	269.7°C
- Tube side	299.4°C
Steam quality at drum outlet	99.78%

TABLE 7

Distribution of Heat from Fission in PHW CANDU Reactors
(From: AECL Nuclear Power Symposium, 1974)

Total fission heat (MW)	100
Heat generated in fuel assemblies	93.9
Heat generated in coolant tubes	0.3
Heat generated in calandria tubes	0.1
Heat generated directly in coolant	0.5
Heat generated in moderator	5.0
Heat generated in shields	0.2
Heat loss coolant to moderator	0.07
Heat removed by moderator	5.15
Heat removed by coolant	94.6
Heat to coolant from pumps	0.68
Piping losses	0.33
Heat to turbine cycle	95.0
Electrical generation from heat	31.4

TABLE 8

Fuel Temperature Coefficients for Pickering at Nominal Operating Conditions (Units are μk/°C)
(From: Ontario Hydro, "Nuclear Training Centre Course 127", 1973)

	Fresh Fuel	Equil. Fuel
$(1/\epsilon)\ d\epsilon/dT$	0	0
$(1/p)\ dp/dT$	-9.33	-9.29
$(1/f)\ df/dT$	-0.79	+0.34
$(1/\eta)\ d\eta/dT$	-4.04	+5.33
$(1/P_f)\ dP_f/dT$	0	0
$(1/P_t)\ dP_t/dT$	-0.83	-0.43
Total	-14.99	-4.05

TABLE 9

Moderator Temperature Coefficients for Pickering
(Units of μk/°C, calculated for $\Delta T = -13°C$)
(From: Ontario Hydro, "Nuclear Training Centre Course 127", 1973).

	Fresh Fuel	Equil. Fuel
$(1/e)\ dE/dT$	0	0
$(1/p)\ dp/dT$	-24.0	-23.9
$(1/f)\ df/dT$	+55.4	+67.1
$(1/\eta)\ d\eta/dT$	-59.2	+76.0
$(1/P_f)\ dP_f/dT$	-13.0	-13.0
$(1/P_t)\ dP_t/dT$	-28.7	-22.0
Total	-69.5	+84.2

TABLE 10

Radiation Dose Limits for Failure Conditions
(From: Ontario Hydro, "Nuclear Training Centre Course 127", 1973)

	Individual (rem)	Population (man-rem)
Single Failures		
External, Whole Body	0.5	10^4
Thyroid, I-131	3	10^4
Dual Failures		
External, Whole Body	25	10^6
Thyroid, I-131	250	10^6

TABLE 11

Population Radiation Dose Limit
(Equal to one tenth of the exposure limits for occupational workers)
(From: AECL Nuclear Power Symposium, 1974)

Organ	Annual Dose Limits
Whole body, gonads, and bone marrow	0.5 rem
Skin, bone, and thyroid	3.0 rem (1.5 rem to thyroid of children up to 16 years)
Other single organs	1.5 rem
Extremities	7.5 rem

TABLE 12

Main Radioactive Materials Produced in Reactor

(From: Canadian Nuclear Association, "Nuclear Power in Canada, Questions and Answers", 1975)

Radionuclide	Source	Half-life
In Gases		
Tritium	Activ. Prod.	12 y
Ar-41	Activ. Prod.	1.8 hr
Kr, Xe	Fission Prod.	hrs to day
I-131	Fission Prod.	8 days
In Liquid		
Co-60	Activ. Prod.	8.3 y
Tritium	Activ. Prod.	12 y
Cs-137	Fission Prod.	30 y
Cs-134	Fission Prod.	2 y

TABLE 13

Releases from Pickering, 1974

(From: Canadian Nuclear Association, "Nuclear Power in Canada, Questions and Answers", 1975)

Maximum dose from gaseous emission	
- Continual residence at boundary	2.0 rem
- 1.6 km from boundary	0.7 rem
- 8 km from boundary	0.1 rem
Maximum dose from liquid effluent to a person drinking effluent and eating 18 kg of fish raised in effluent.	2.0 rem

TABLE 14

Natural and Man-Made Radiation Doses Received on the Average by Population
(From: AECL, "The Standardized CANDU 600 MWe Nuclear Reactor", G.L. Brooks,1974)

Average Dose in North-America from Natural Sources	mrem/year
Cosmic Rays (average)	44
Terrestrial Radiation	
- External	40
- Internal	18
Total	102

Average Dose in Canada from All Sources	mrem/year
Environmental	
- Natural	102.0
- Global Fallout	4.0
Medical	
- Diagnostic X-rays	72.0
- Radiopharmaceutical	1.0
Nuclear Power	0.003
Miscellaneous	2.8
Total	182.0

TABLE 15

Effects of Acute Ionizing Radiation Doses on the Whole Body
(From: Canadian Nuclear Association, CNA'73-302)

Range	0 to 100 rems Subclinical Range	100 to 1000 rems Therapeutic Range		
		100 to 200 rems	200 to 600 rems	600 to 1000 rems
		Clinical Surveillance	Therapy Effection	Therapy Promising
Incidence of vomiting	none	100 rems: 5% 200 rems: 50%	300 rems: 100%	100%
Delay time	-	3 hours	2 hours	1 hour
Leading organs	none	hematopoietic tissue		
Characteristic signs	none	moderate leukopenia	severe leukopenia; purpura; hemorrhage; infection; epilation	
Prognosis	excellent	excellent	good	guarded
Convalescence period	none	several weeks	1 to 12 months	long
Incidence of death	none	none	0 to 80% (variable)	80 to 100% (variable)
Death occurs within	-	-	2 months	
Cause of death	-	-	hemorrhage; infection	

TABLE 16

Solid Wastes from a CANDU-PHW Station
(per reactor-year)

(From: Canadian Nuclear Association, CNA'73-303)

	Volume (m^3)	Approximate Activity (Curies)
Combustibles	95	1
Non-Combustibles	2	0.02
Spent Ion Exchange Column	7.5	400
Spent Filters	2	100

9. FIGURES

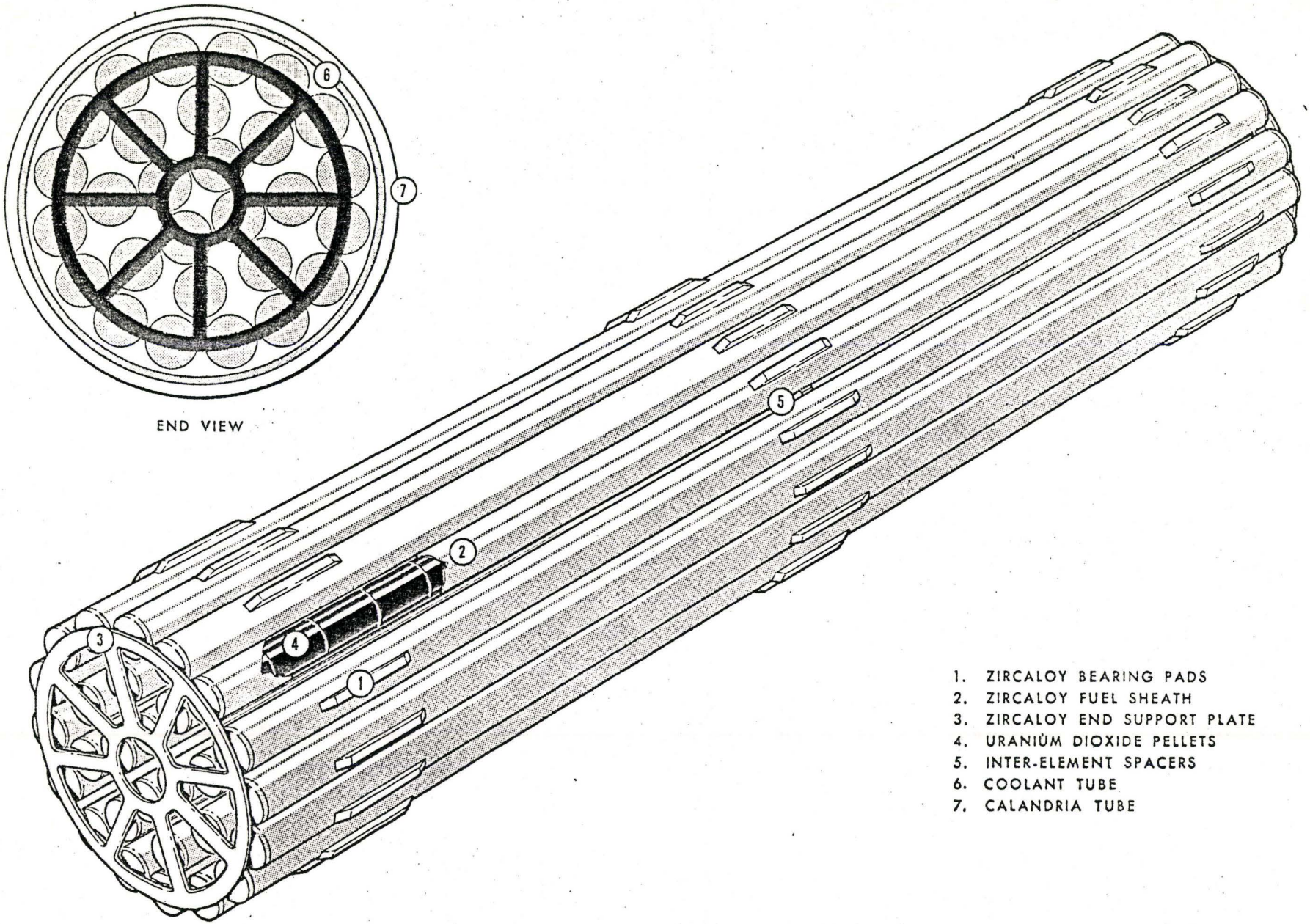

Figure 1. Pickering Fuel Bundle.

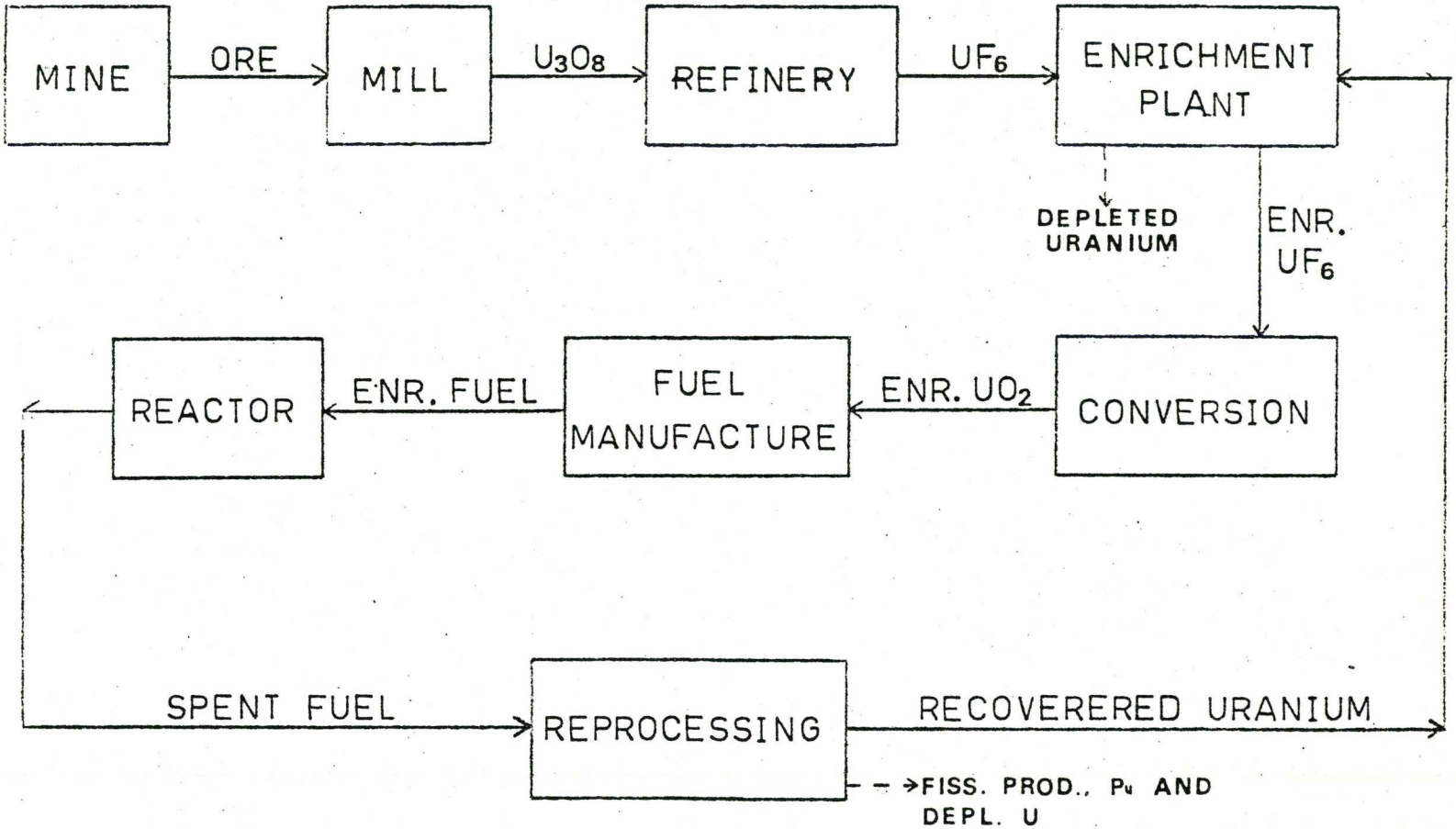

Figure 2. Enriched Uranium Cycle.

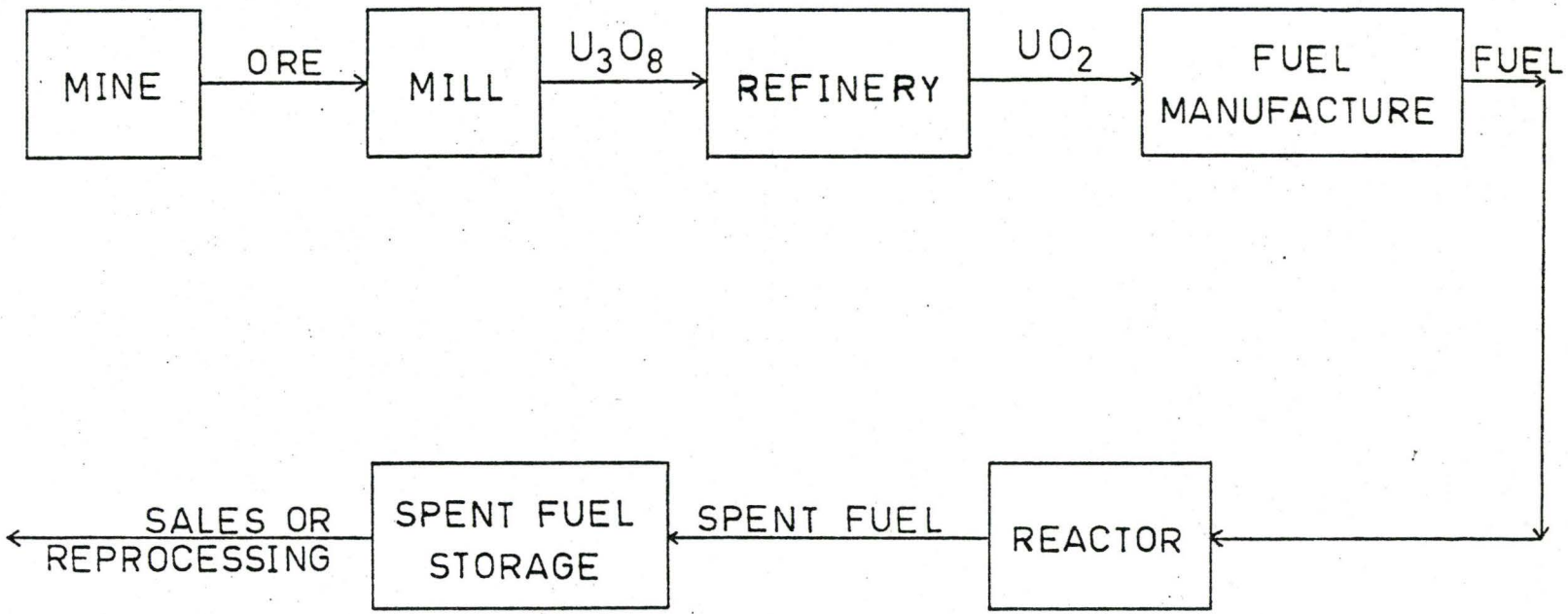

Figure 3. Natural Uranium Cycle.

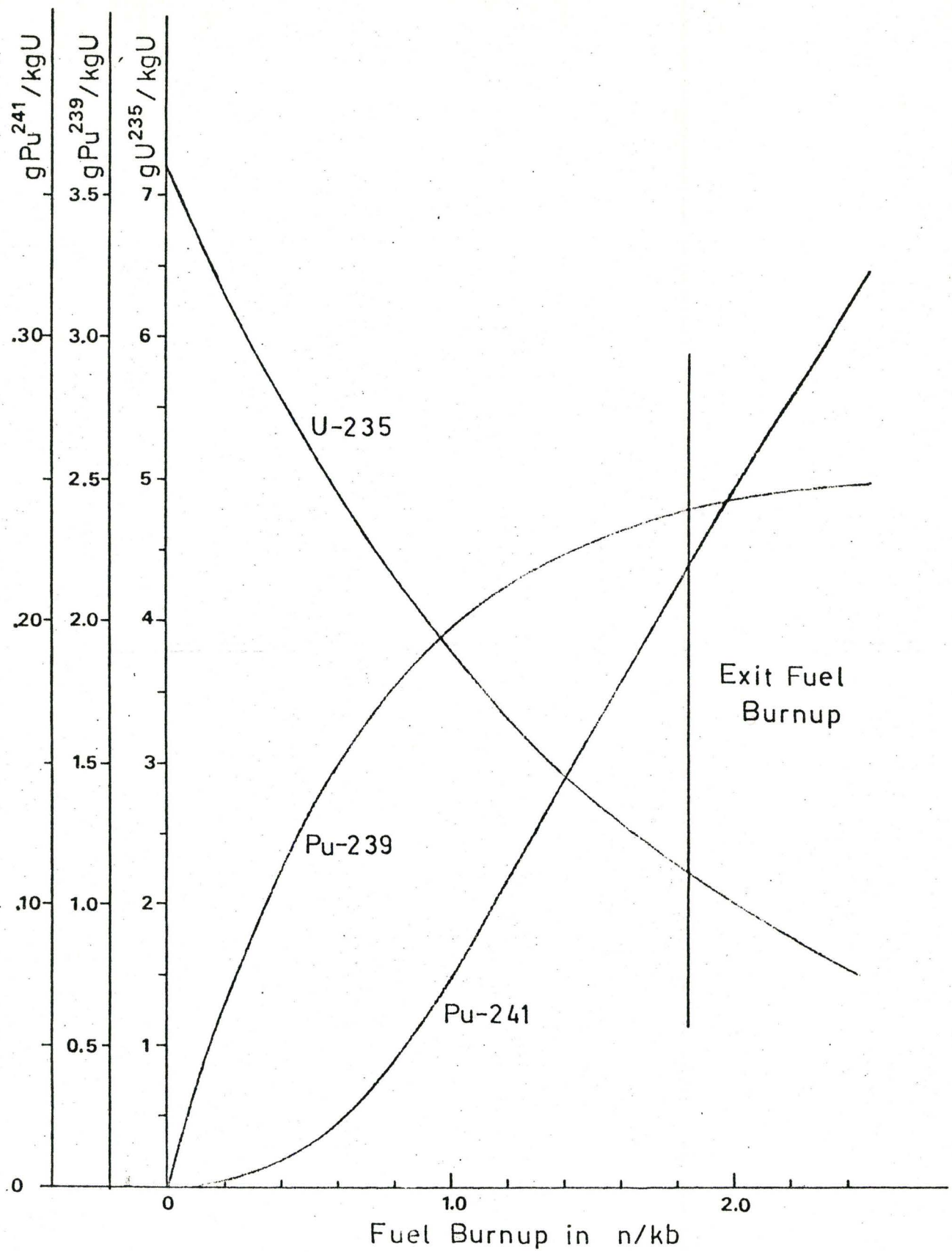

Figure 4. Fissile Content of Fuel vs Burnup.

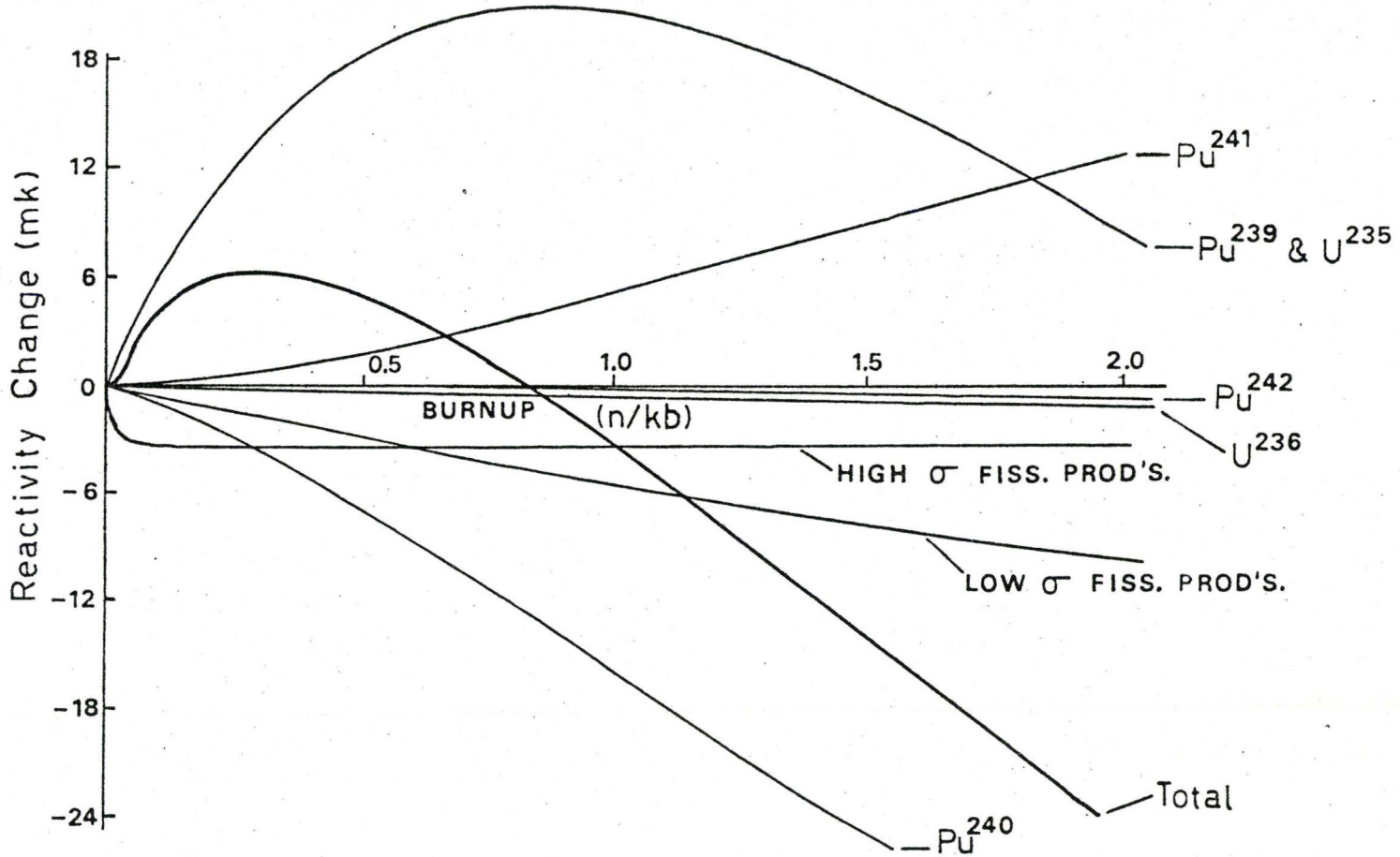

Figure 5. Reactivity Changes with Burnup.

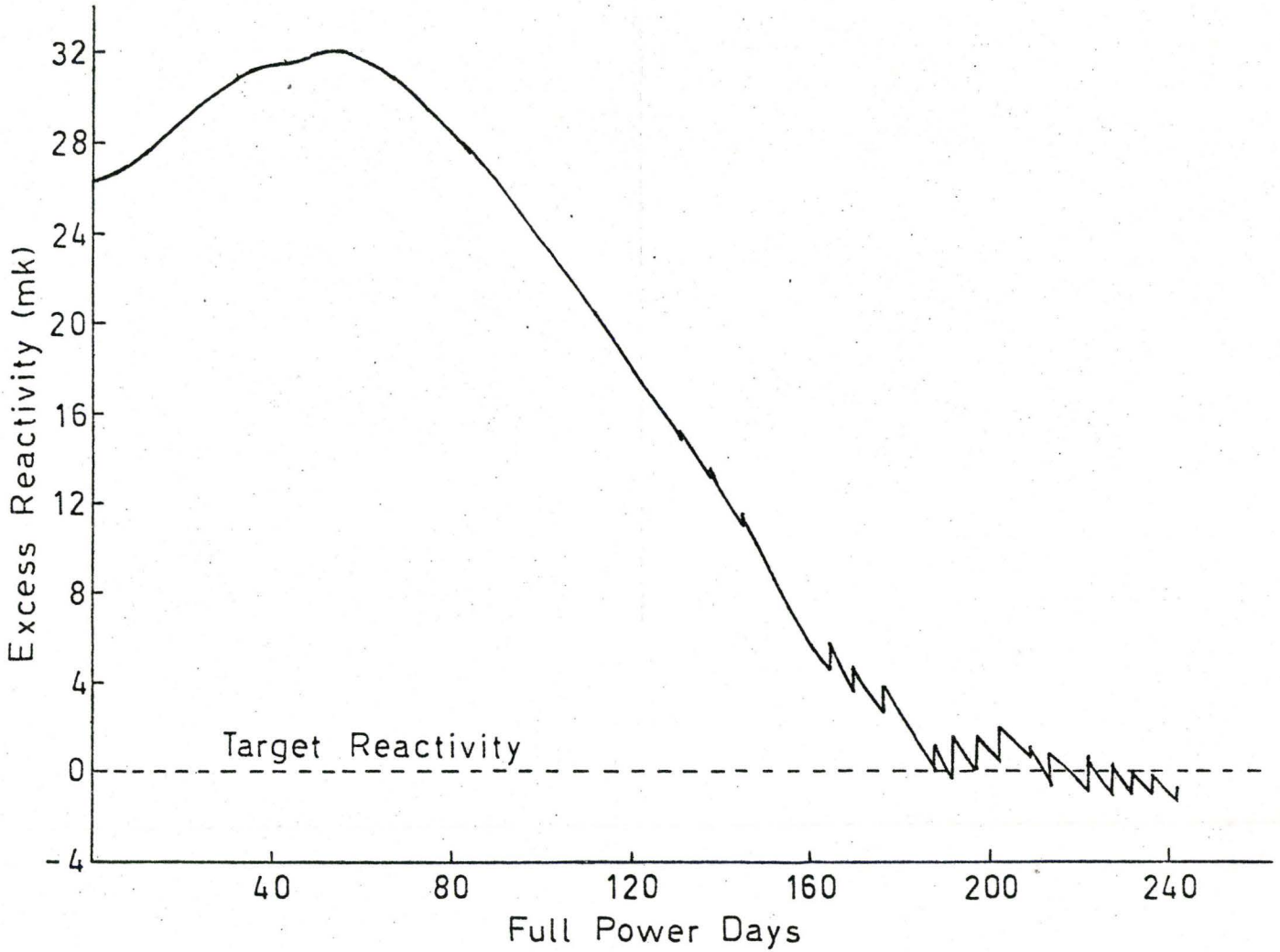

Figure 6. Reactivity vs Burnup (Pickering No 1).

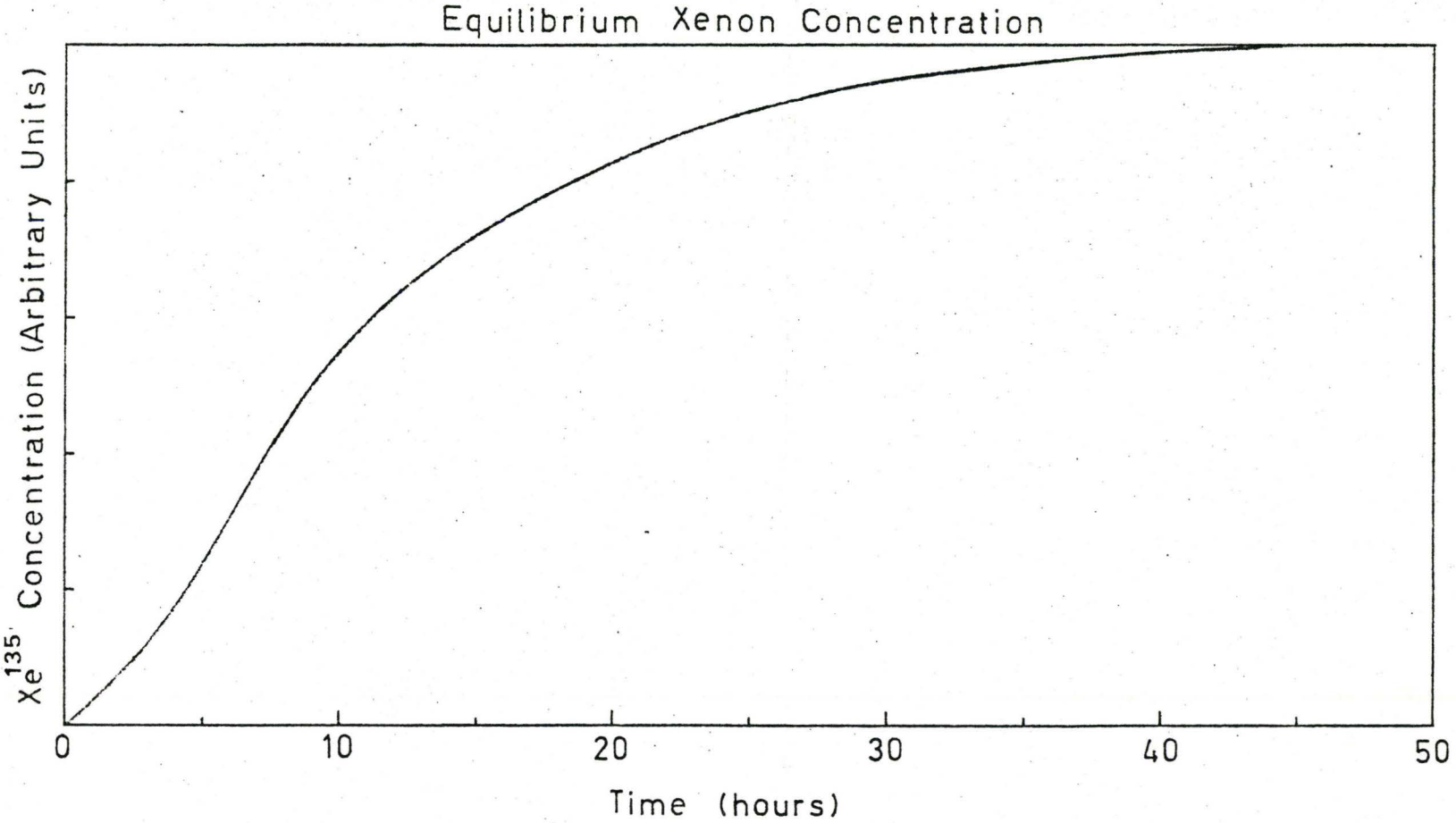

Figure 7. Xenon-135 Buildup.

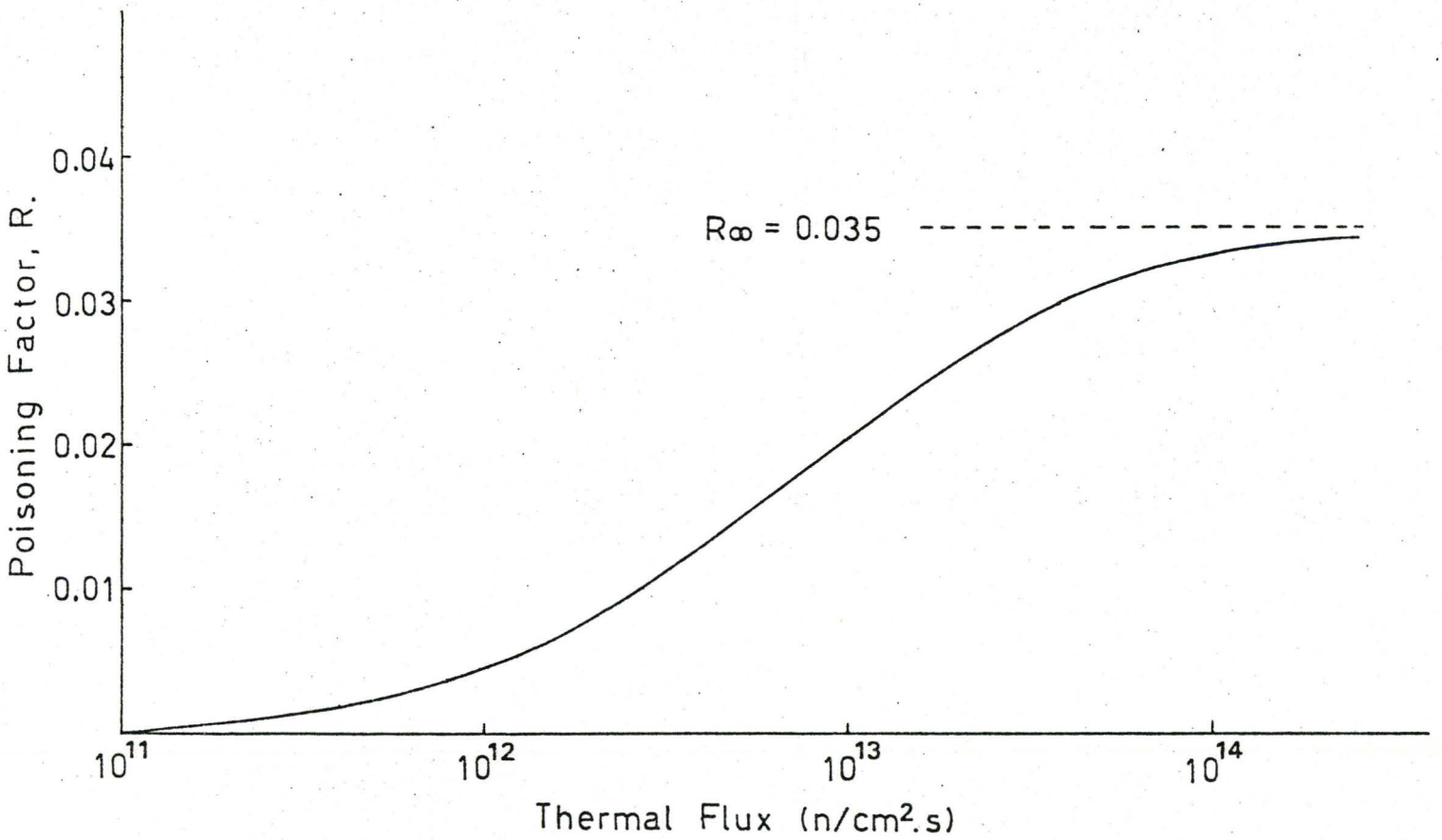

Figure 8. Equilibrium Xe Poisoning vs Thermal Flux.

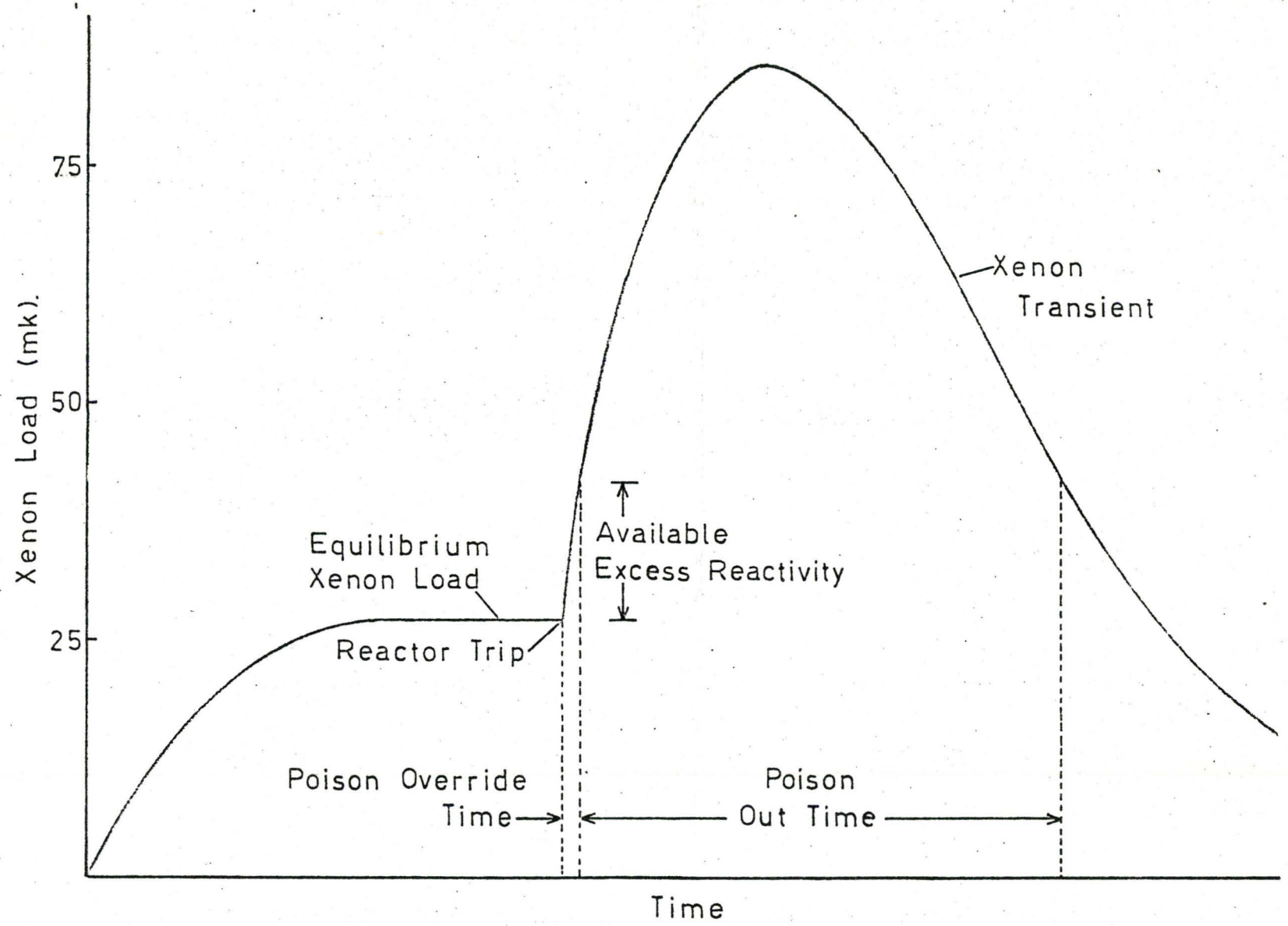

Figure 9. Typical Xenon Transient.

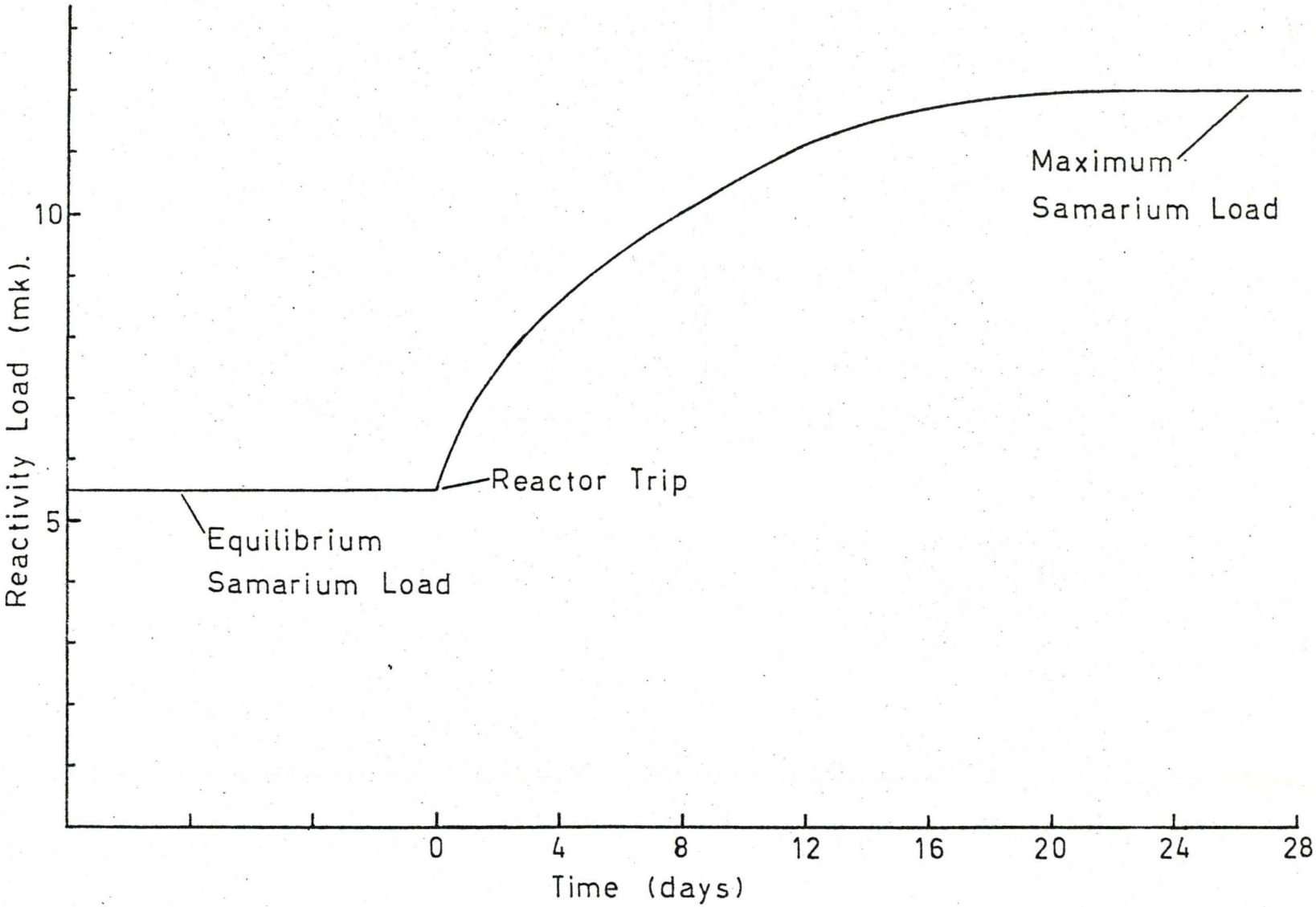

Figure 10. Samarium Transient After a Shutdown.

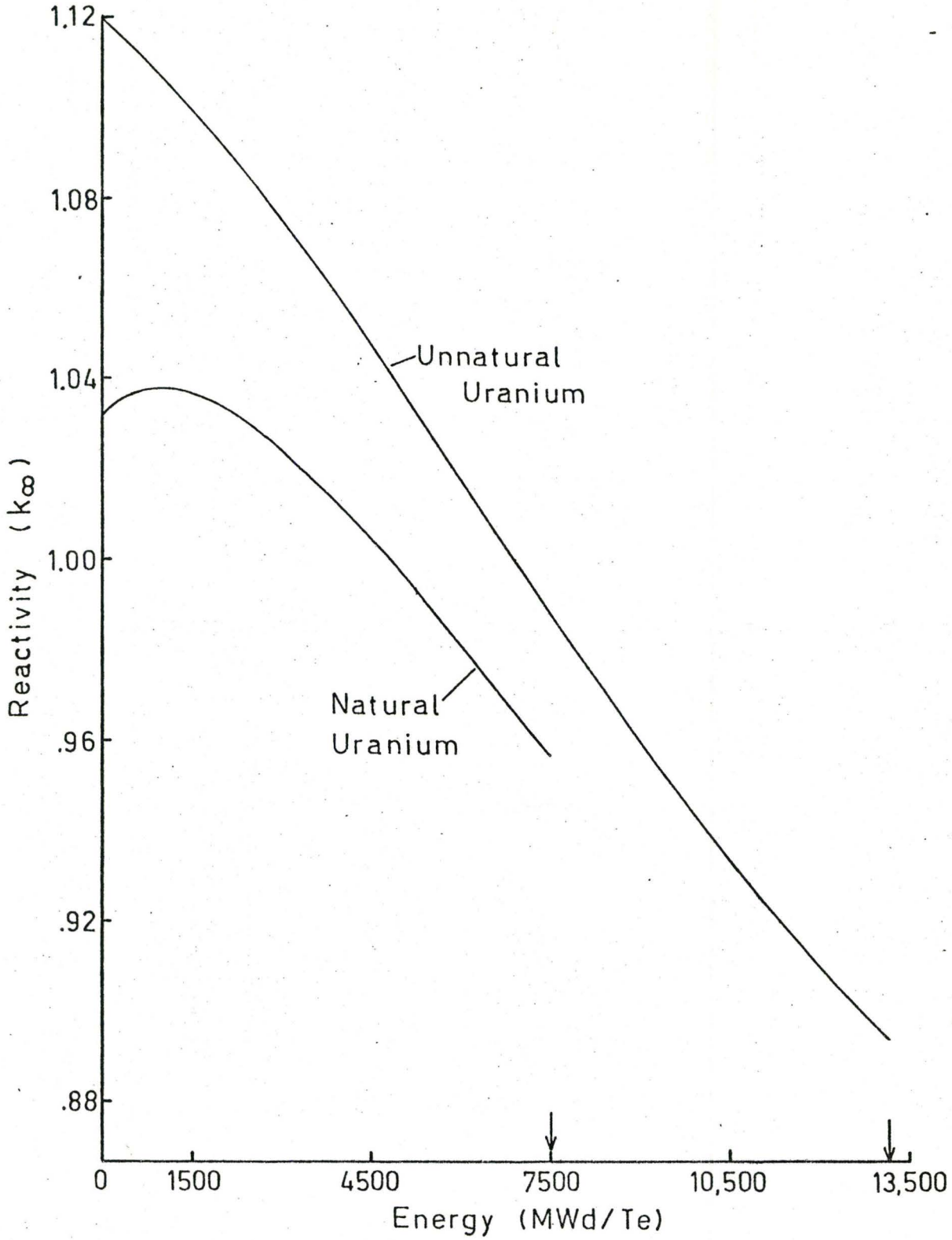

Figure 11. k_∞ as a Function of Burnup.

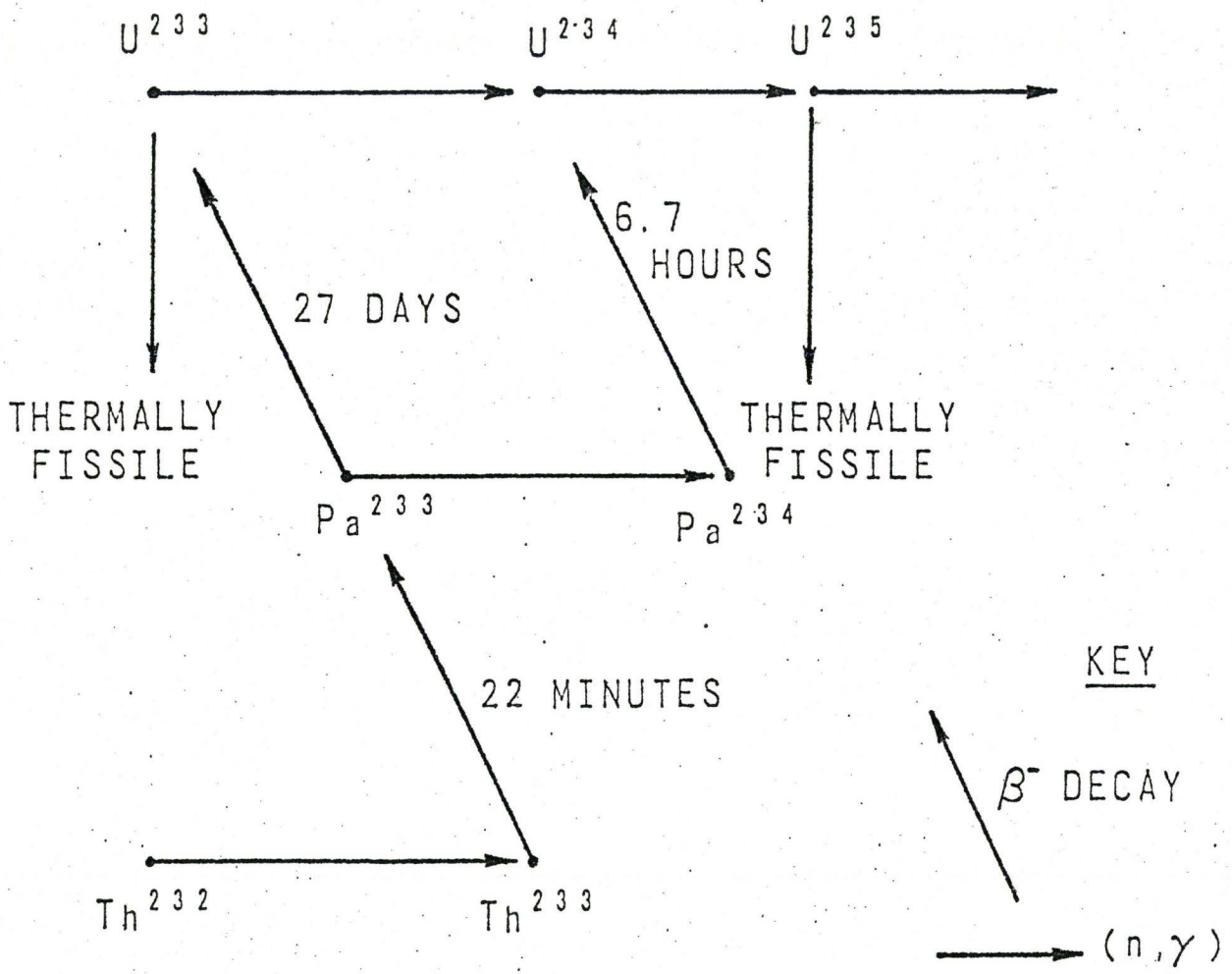

Figure 12. Thorium Activation Scheme.

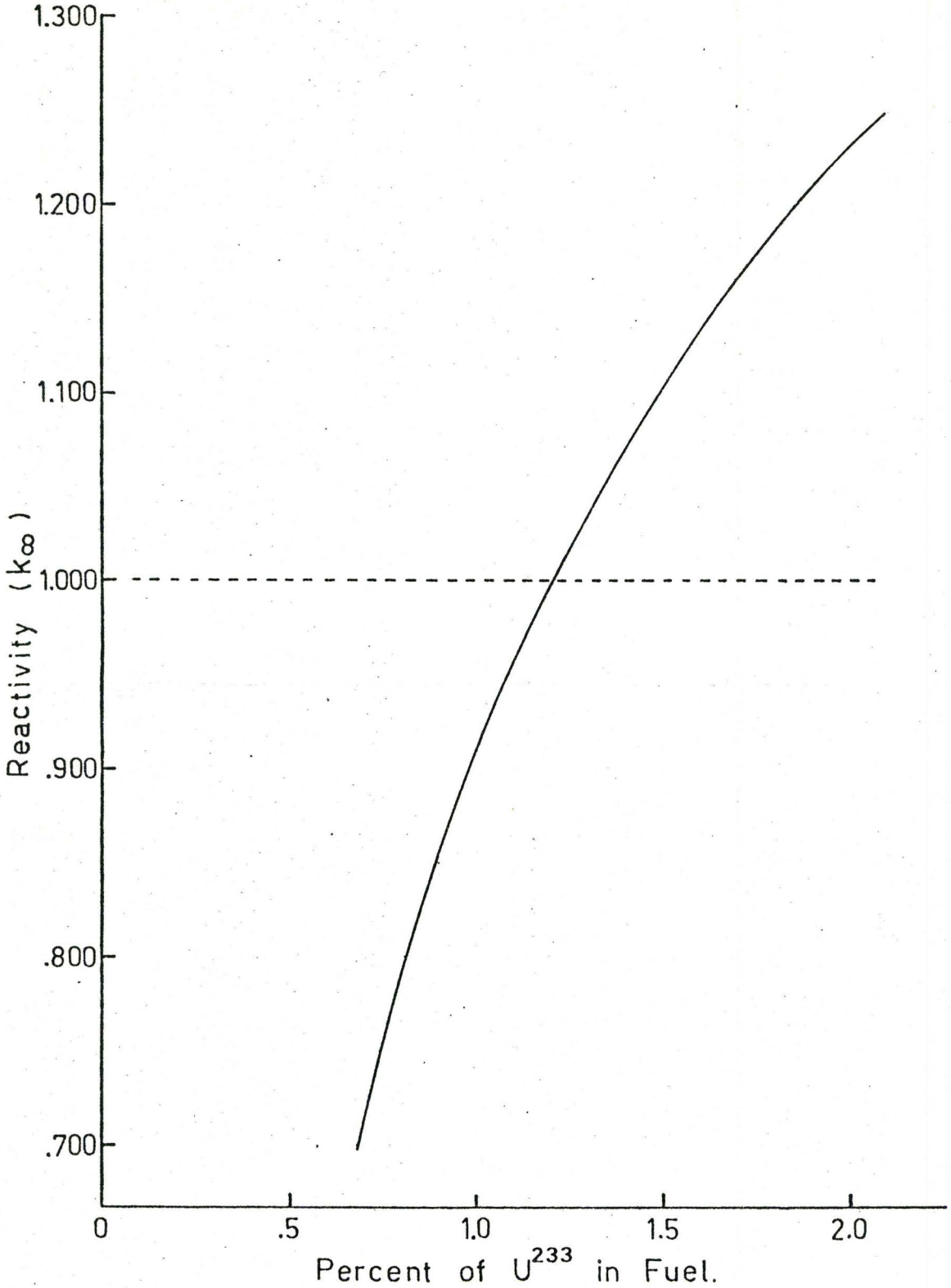

Figure 13. Reactivity of a U^{233}-Enriched Th^{232} Fuel Bundle.

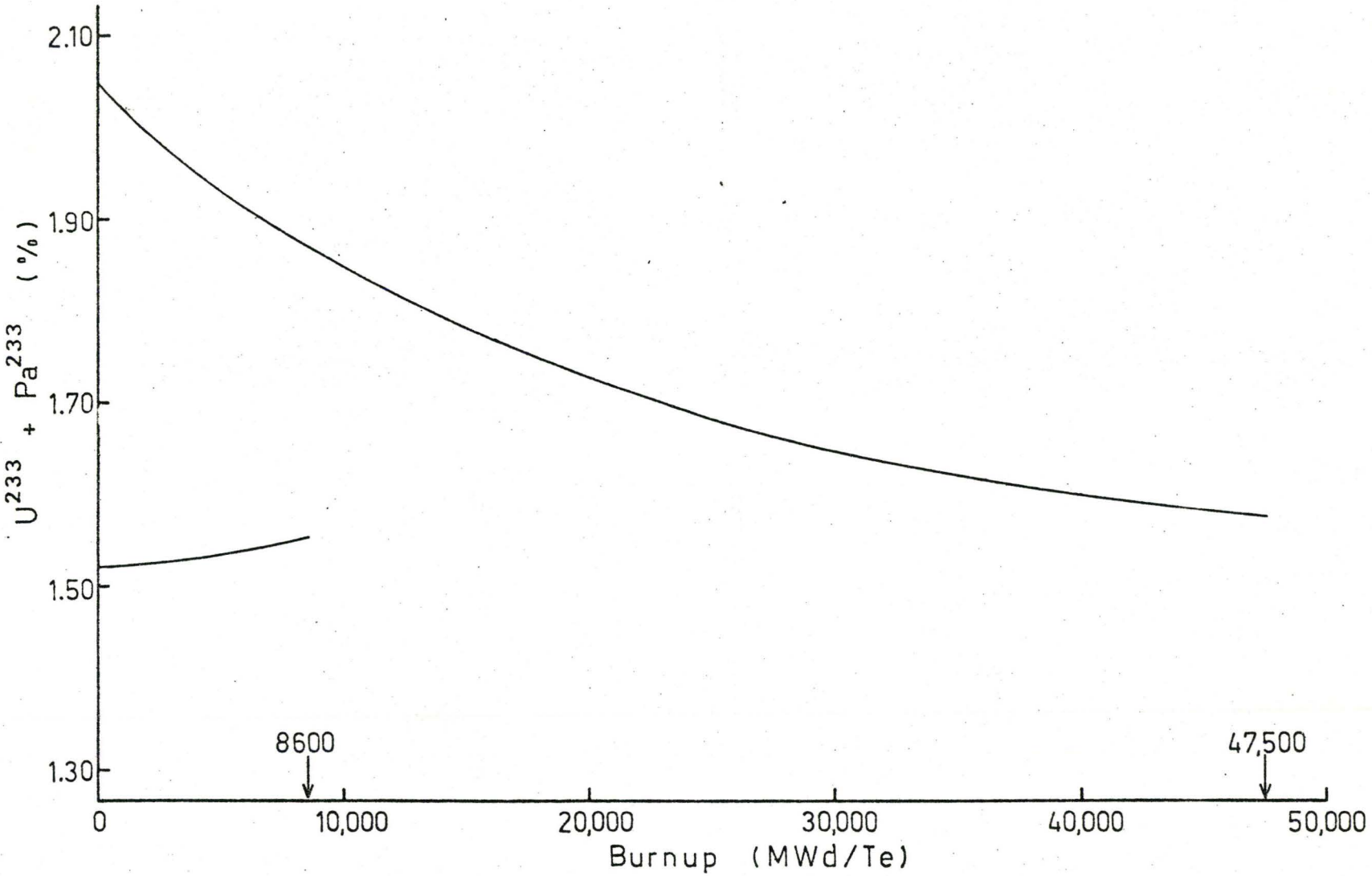

Figure 14. Variation of U^{233} with Burnup.

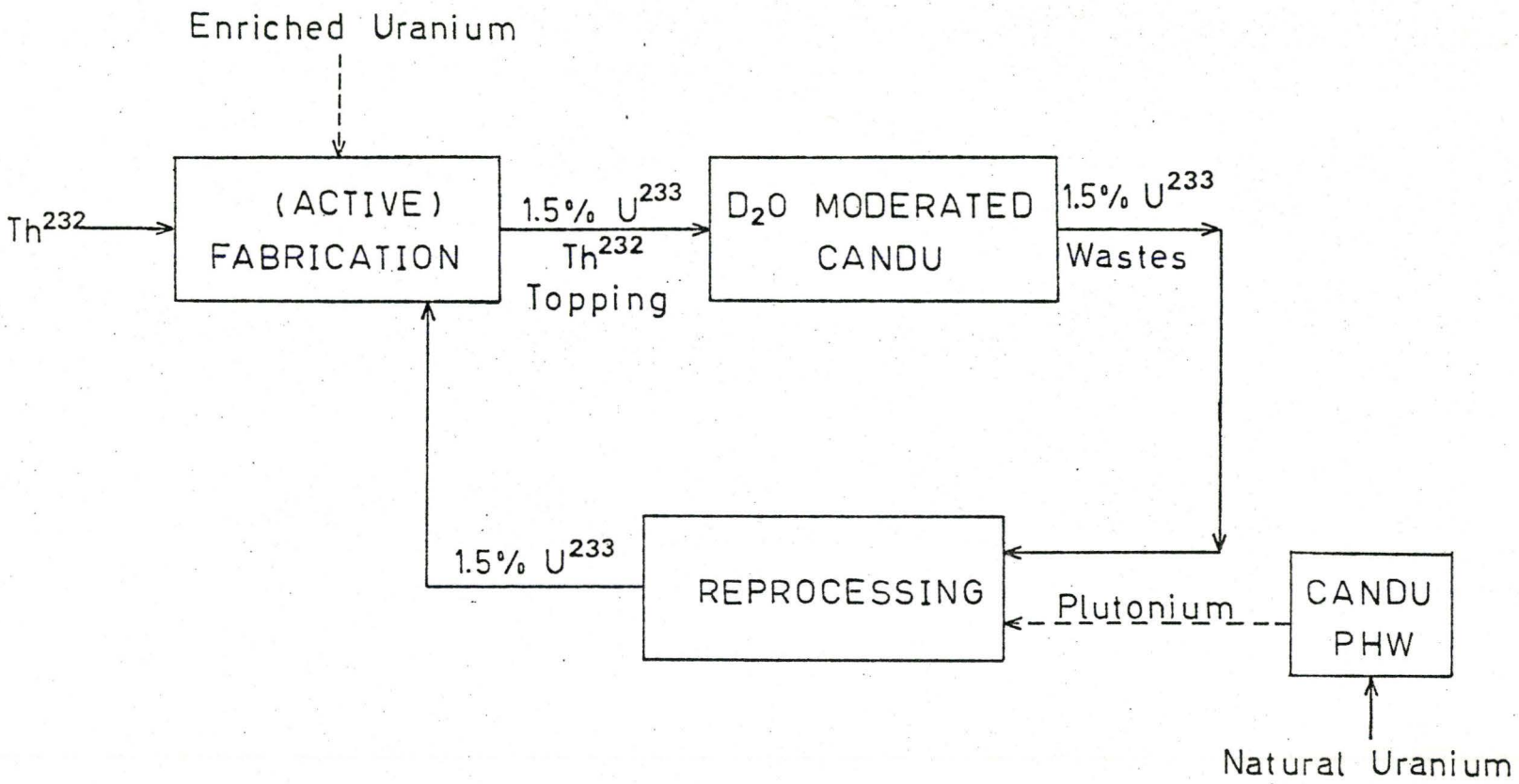

Figure 15. Equilibrium U^{233} - Th Fuel Cycle.

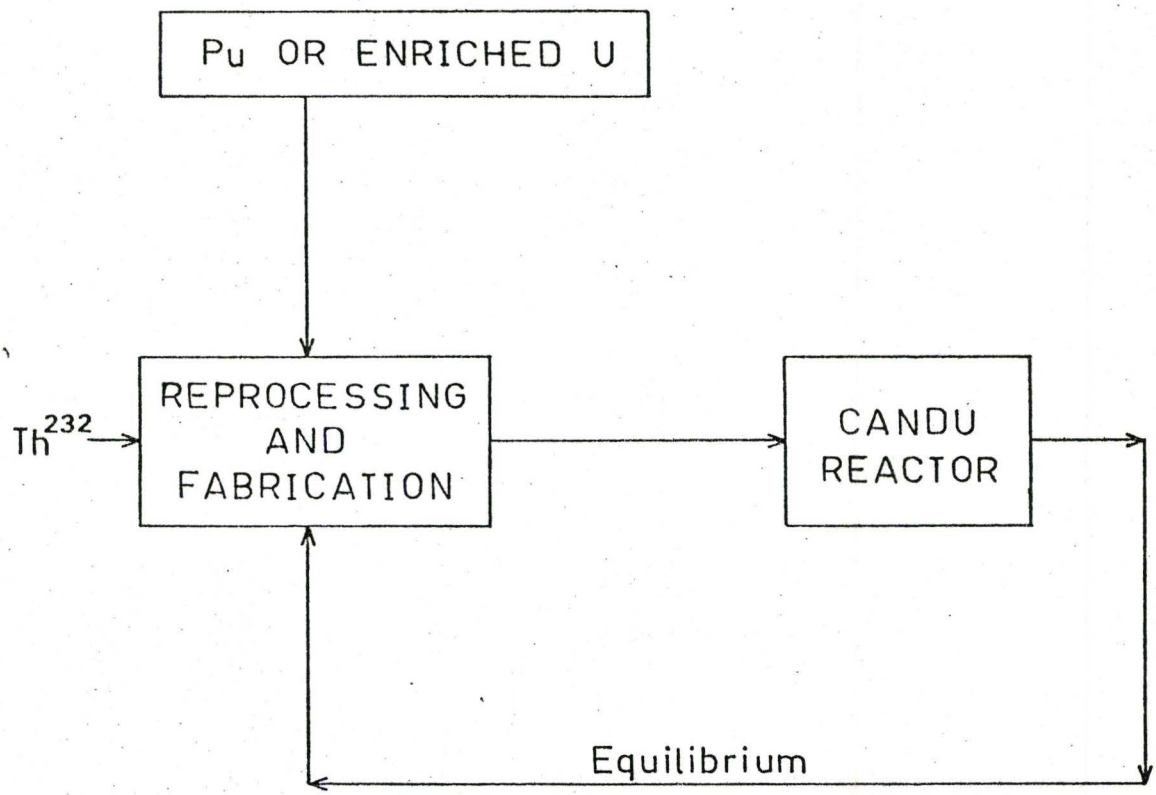

Figure 16. Thorium Utilisation Cycle.

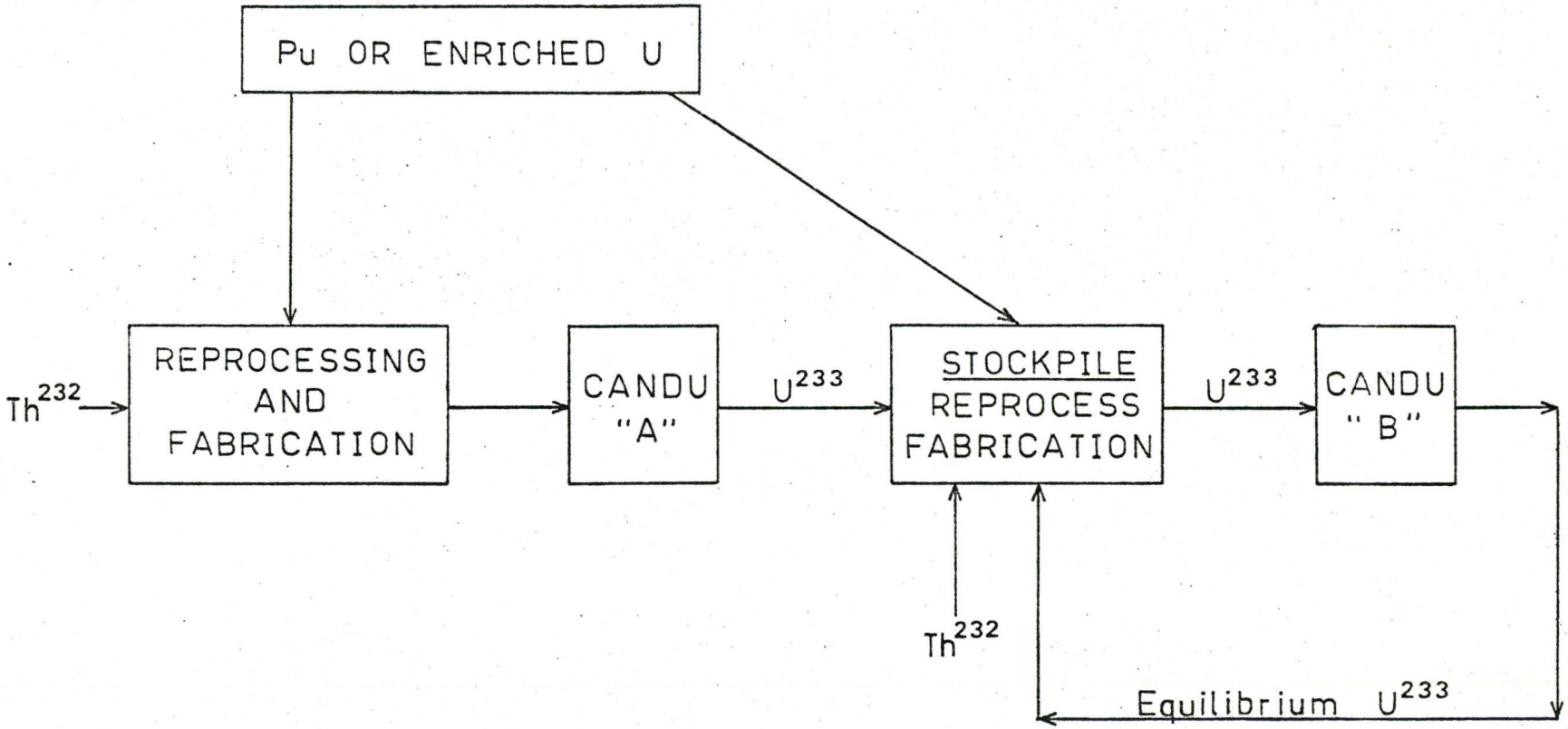

Figure 17. Thorium Utilisation Cycle.

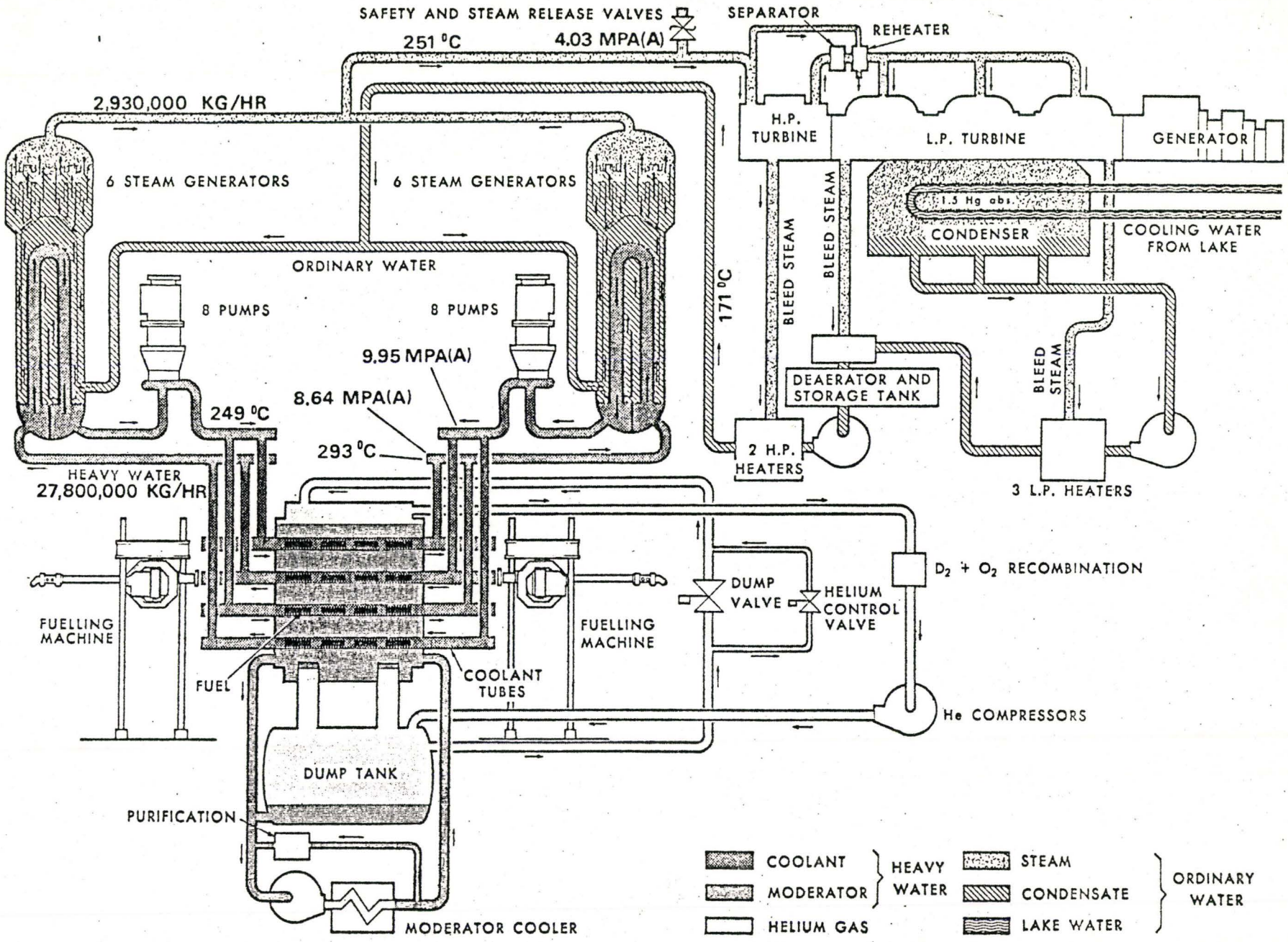

Figure 18. Flowsheet of Coolant System.

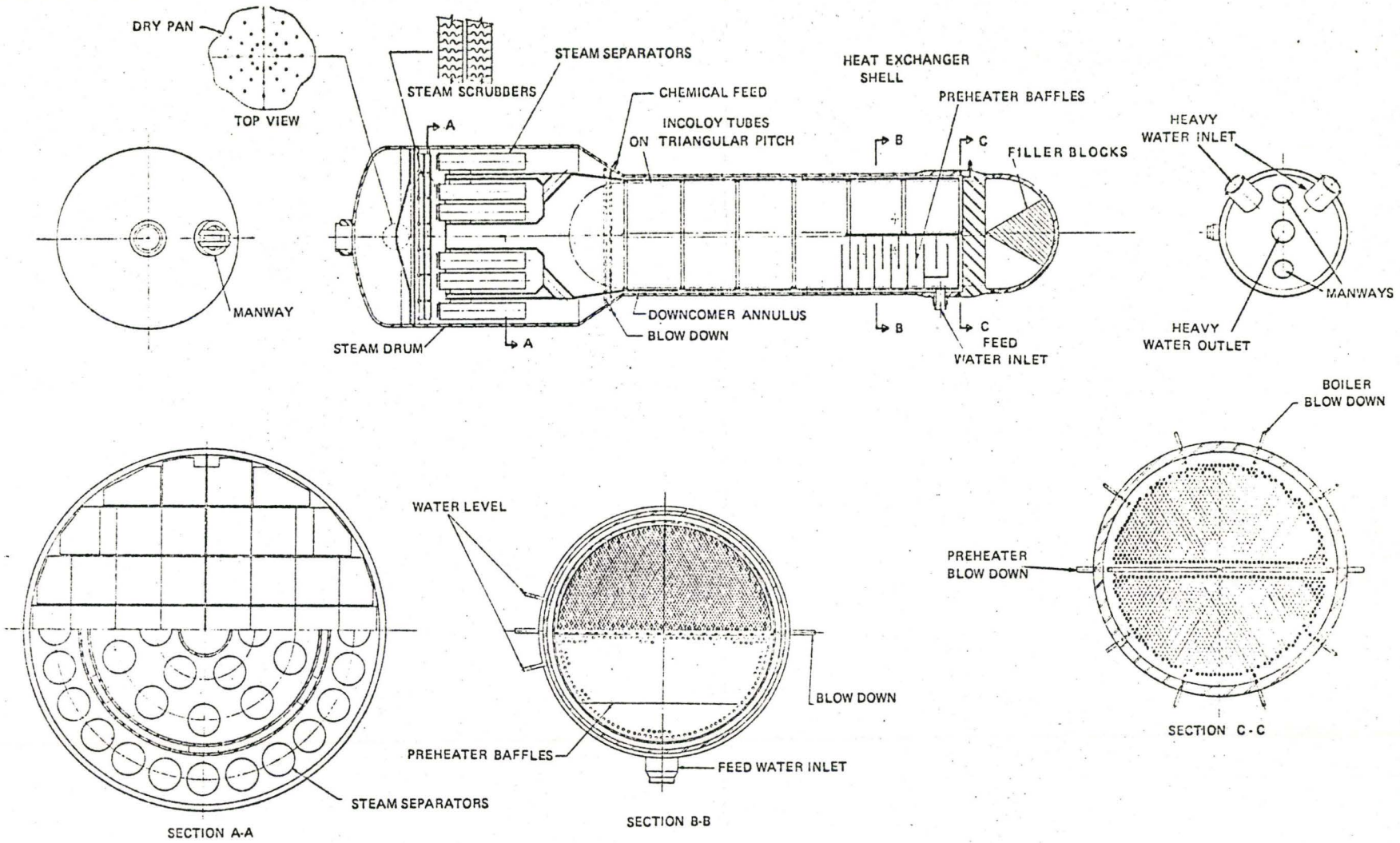

Figure 19. Typical Steam Generator.

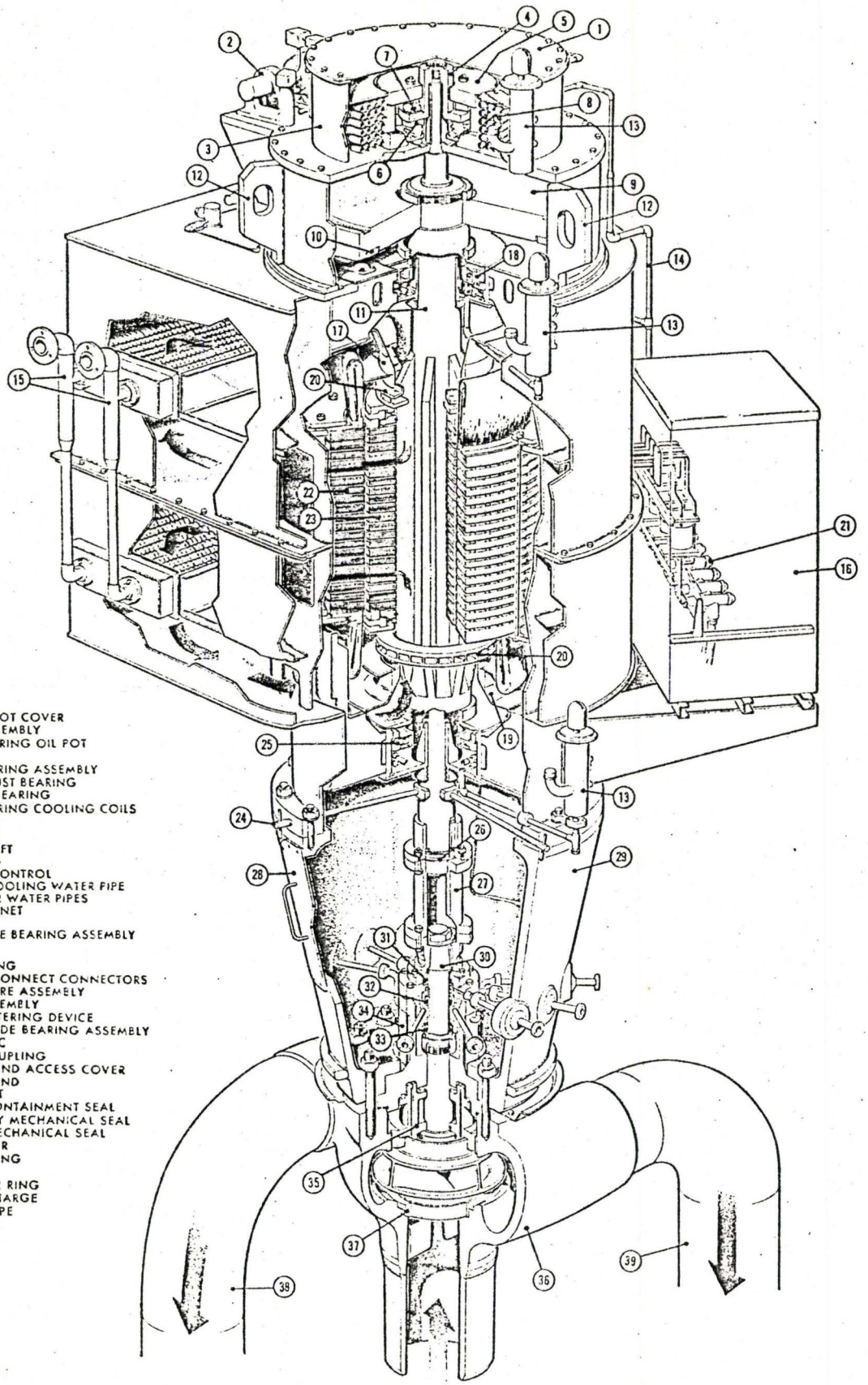

Figure 20. Typical Heat Transport Pump.

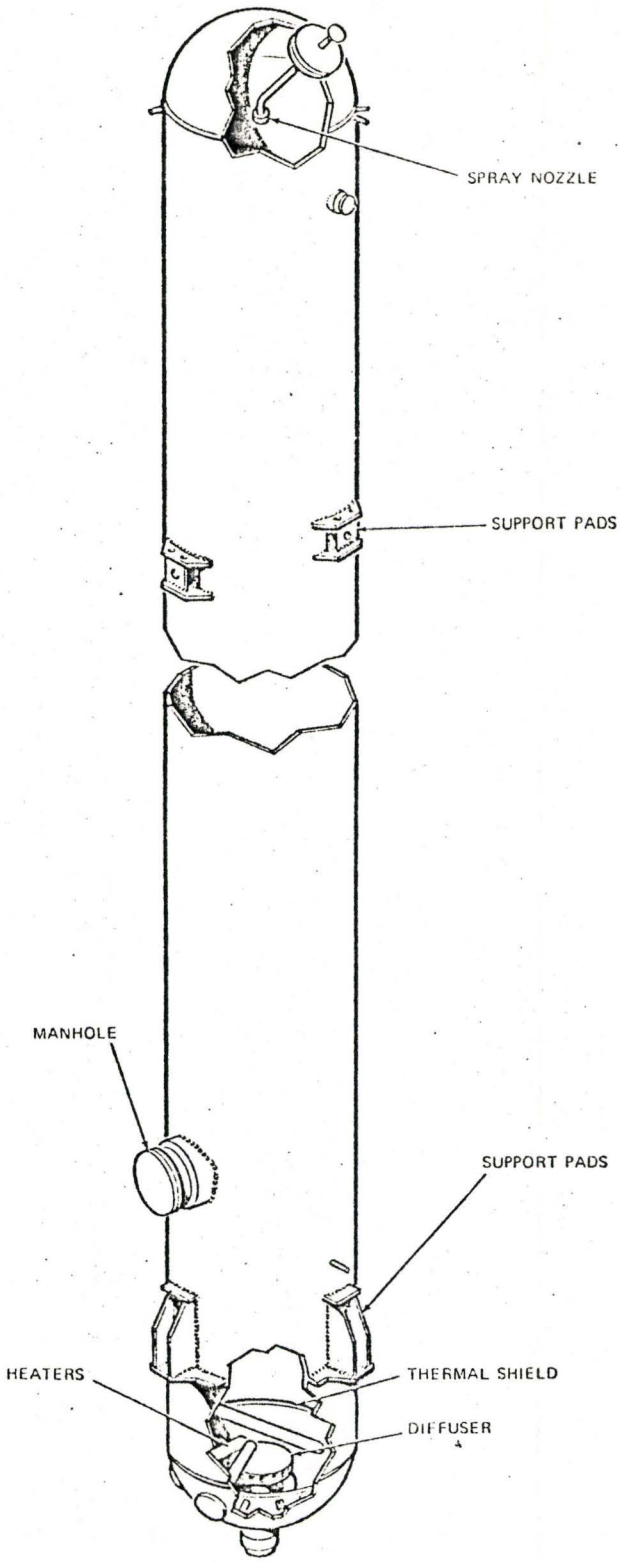

Figure 21. Pressurizer Arrangement.

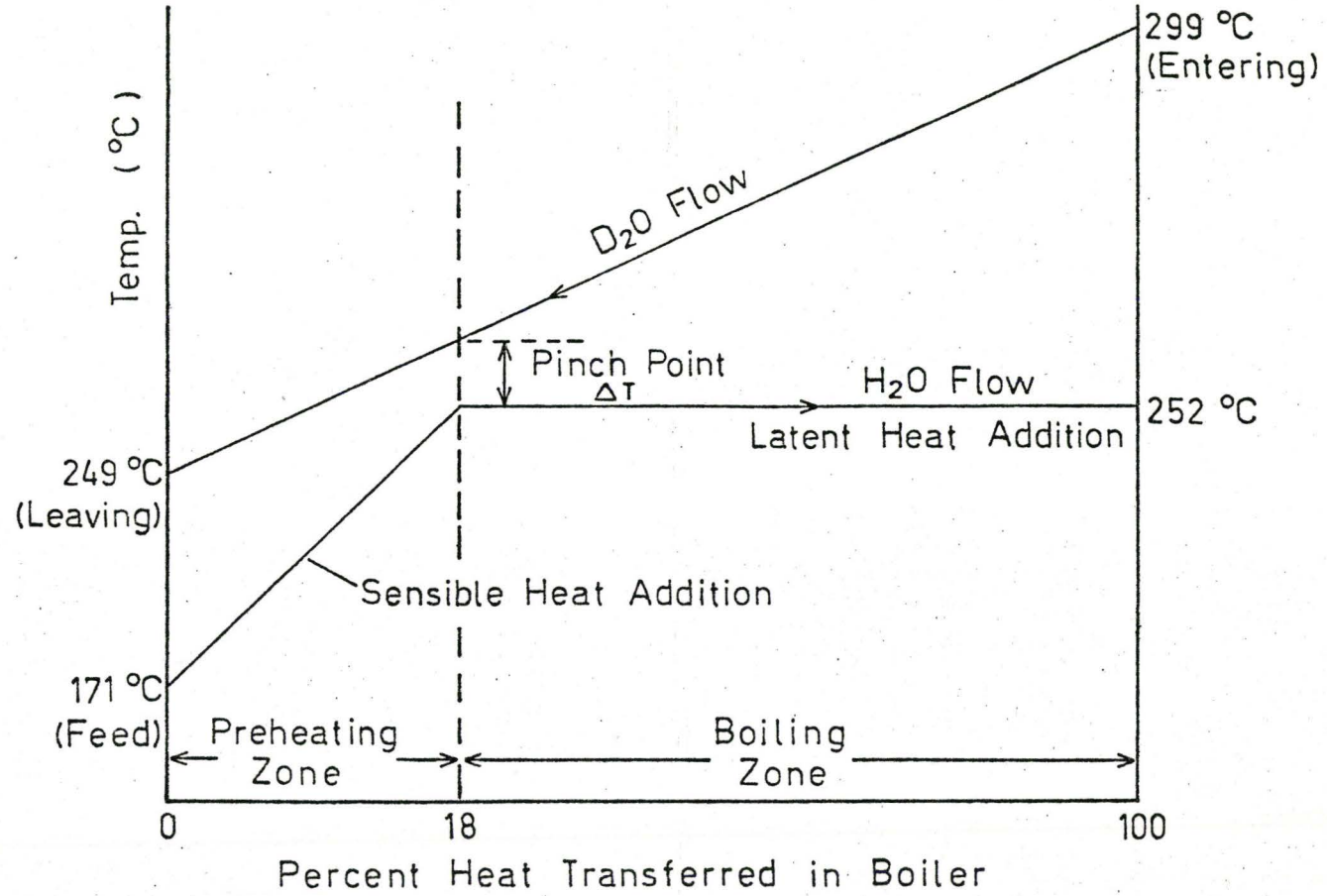

Figure 22. Relationship of Coolant Temperature to Steam Temperature.

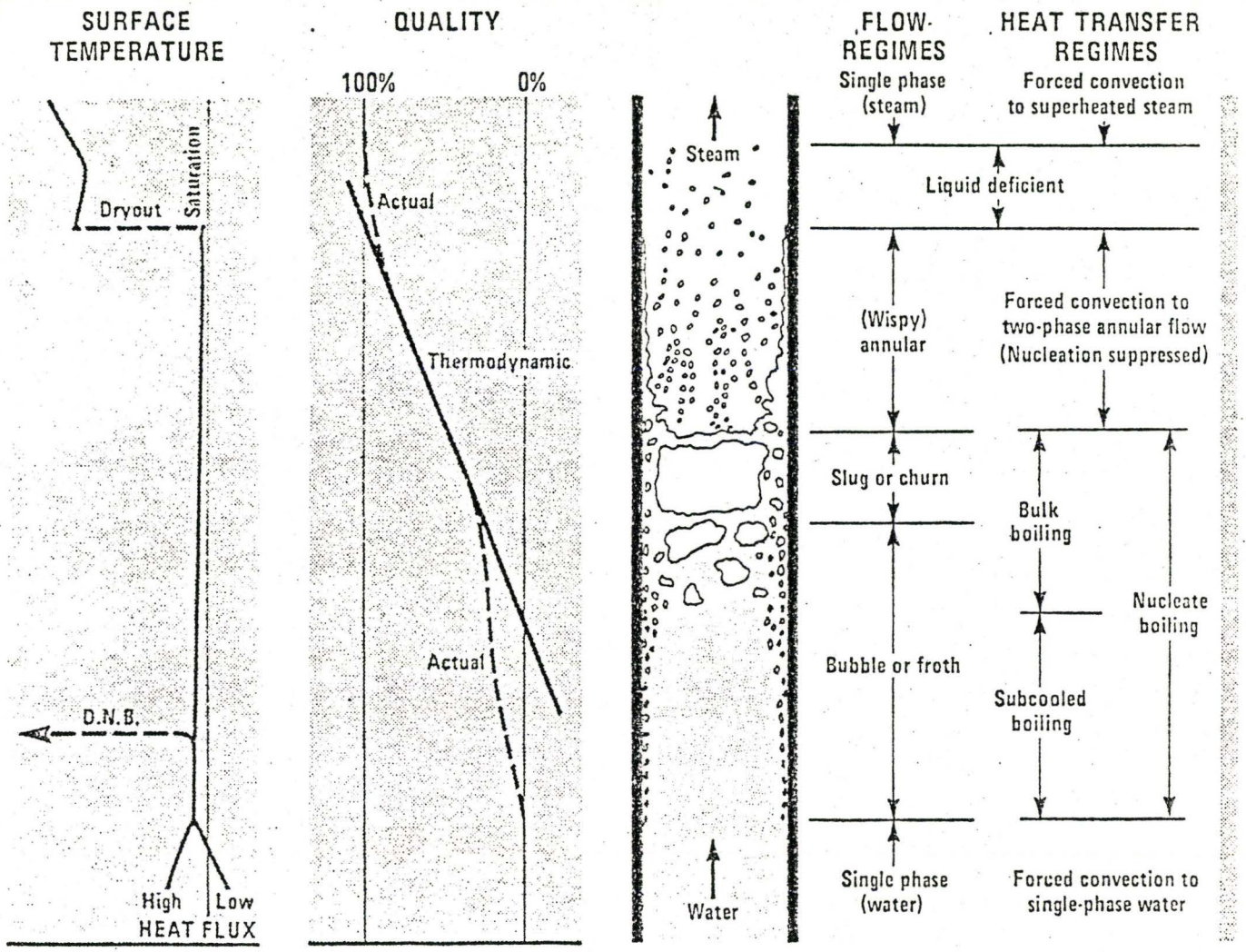

Figure 23. Thermal-Hydraulic Regimes in a Coolant Channel.

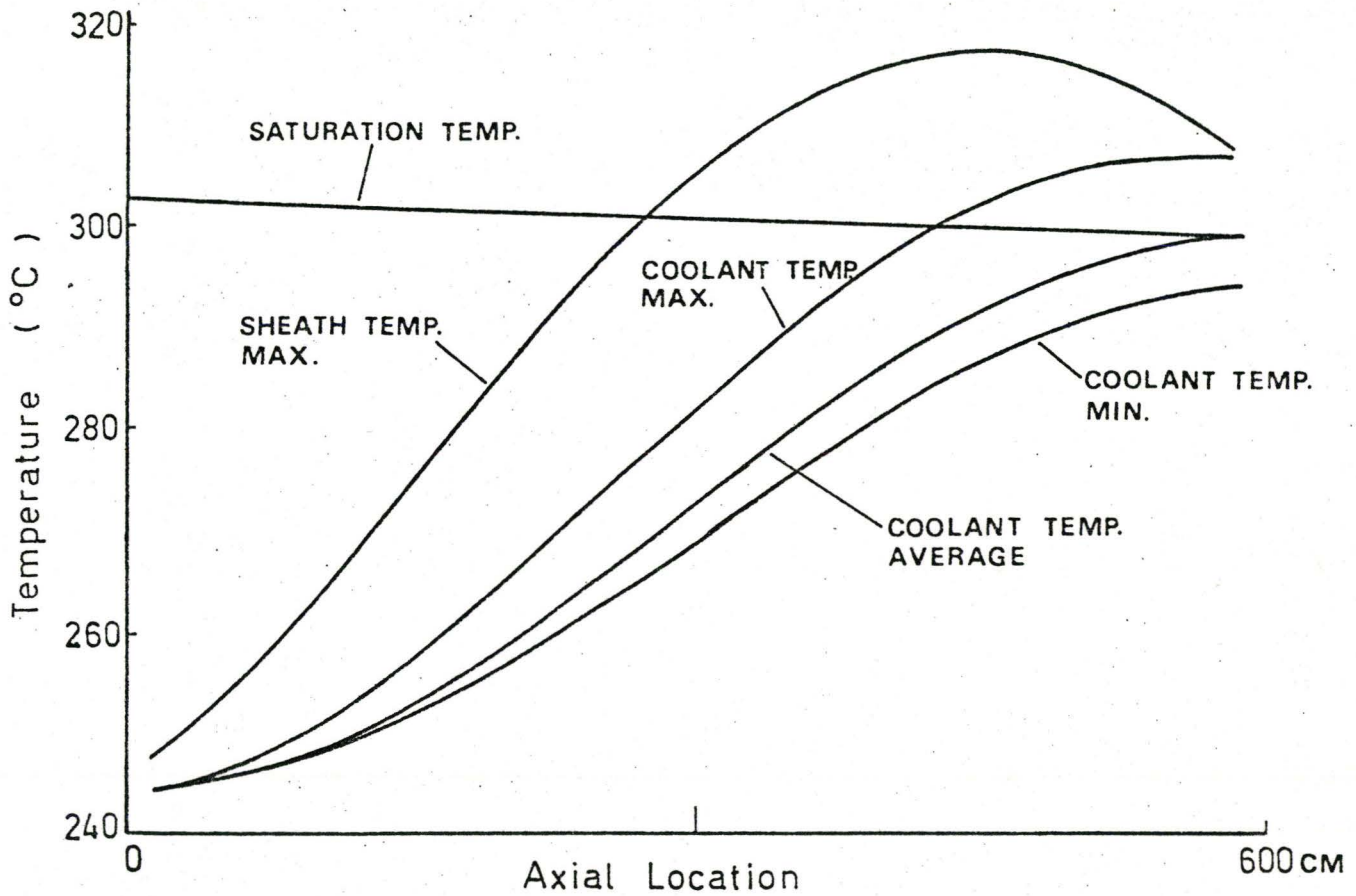

Figure 24. Axial Sheath and Coolant Temperatures in Bruce.

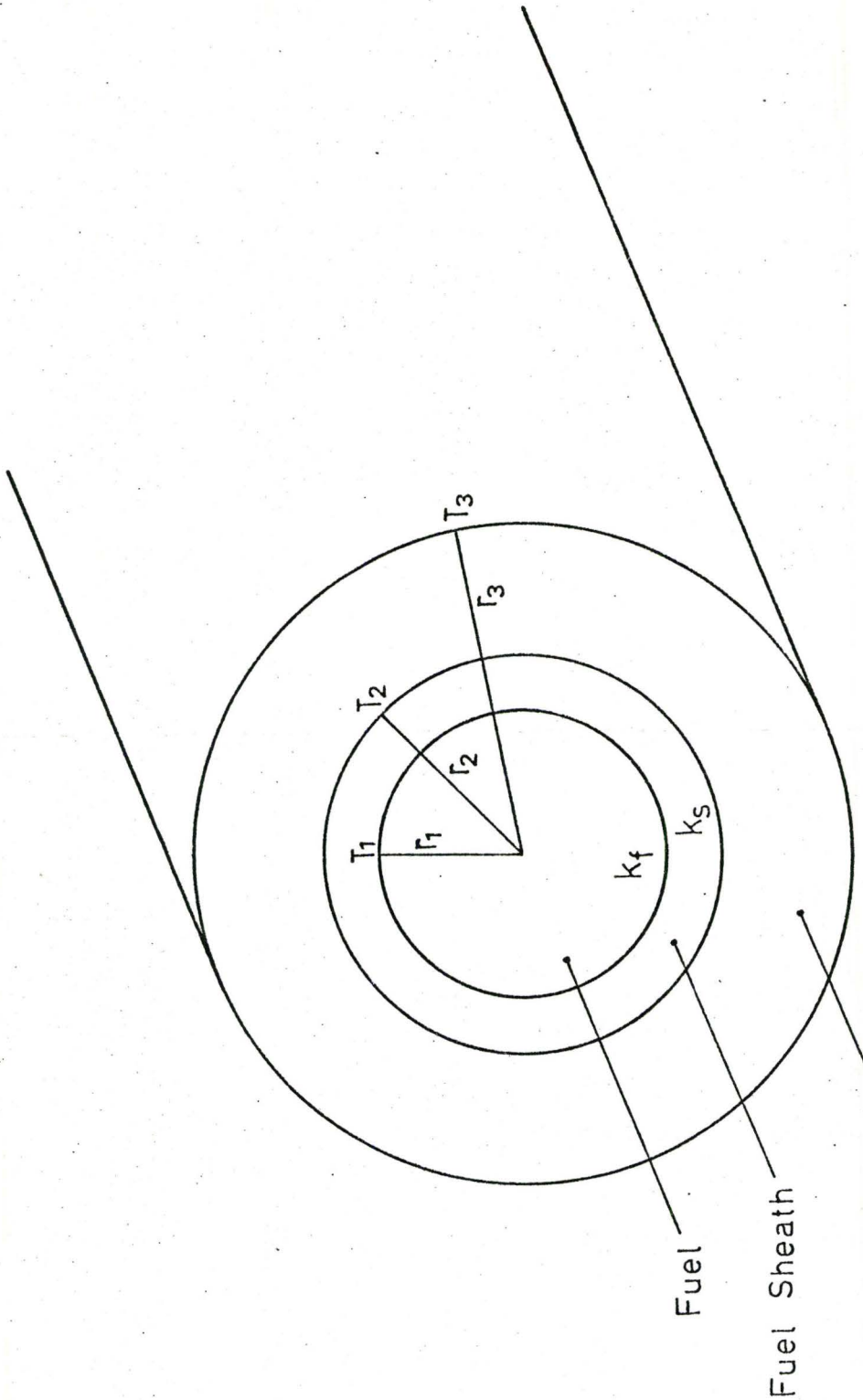

Figure 25. Parameters Definition.

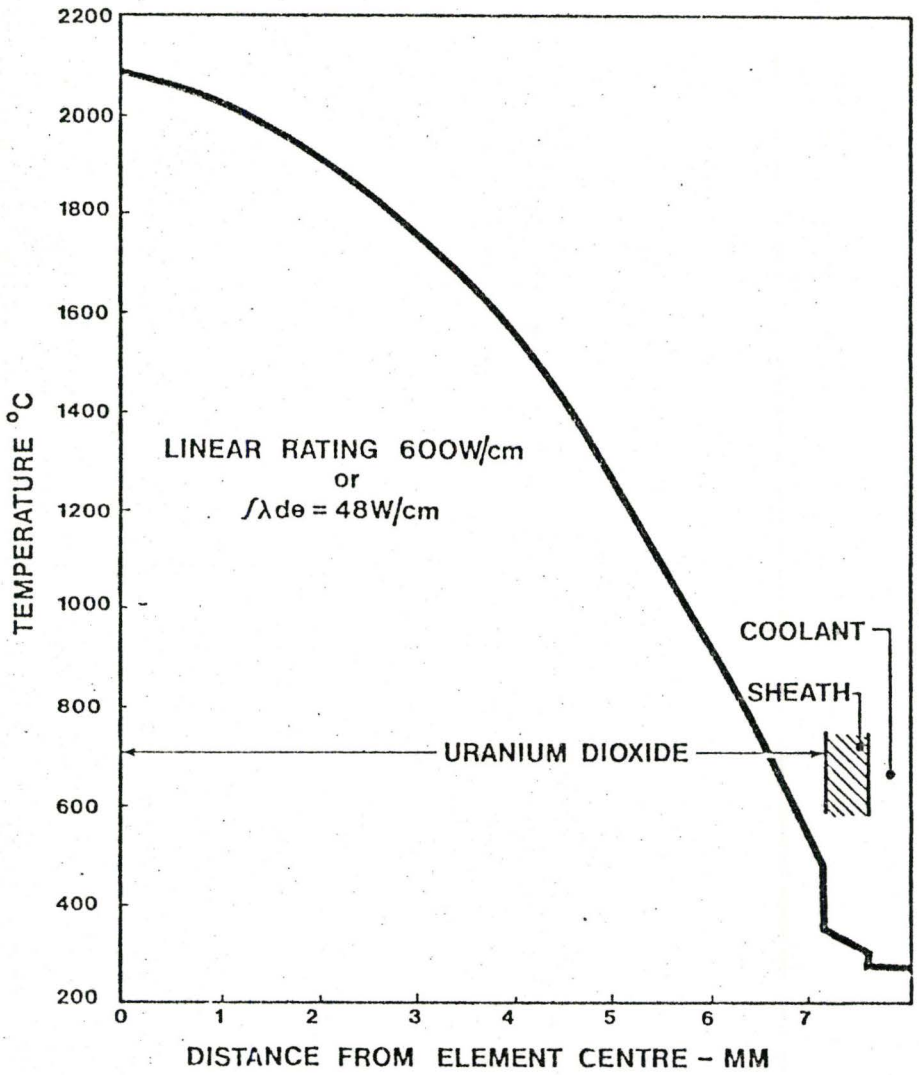

Figure 26. Radial Fuel Pellet Temperature.

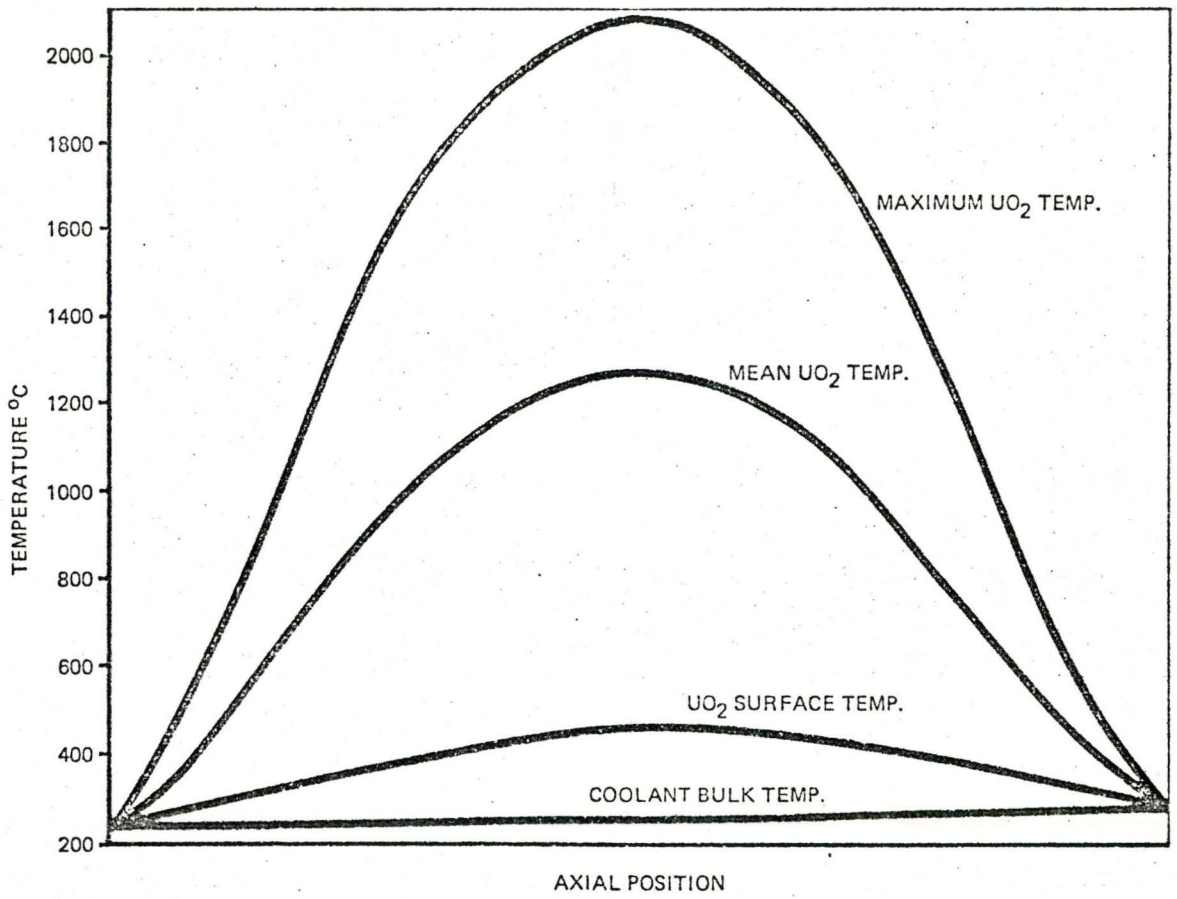

Figure 27. Axial Fuel Pellet Temperatures.

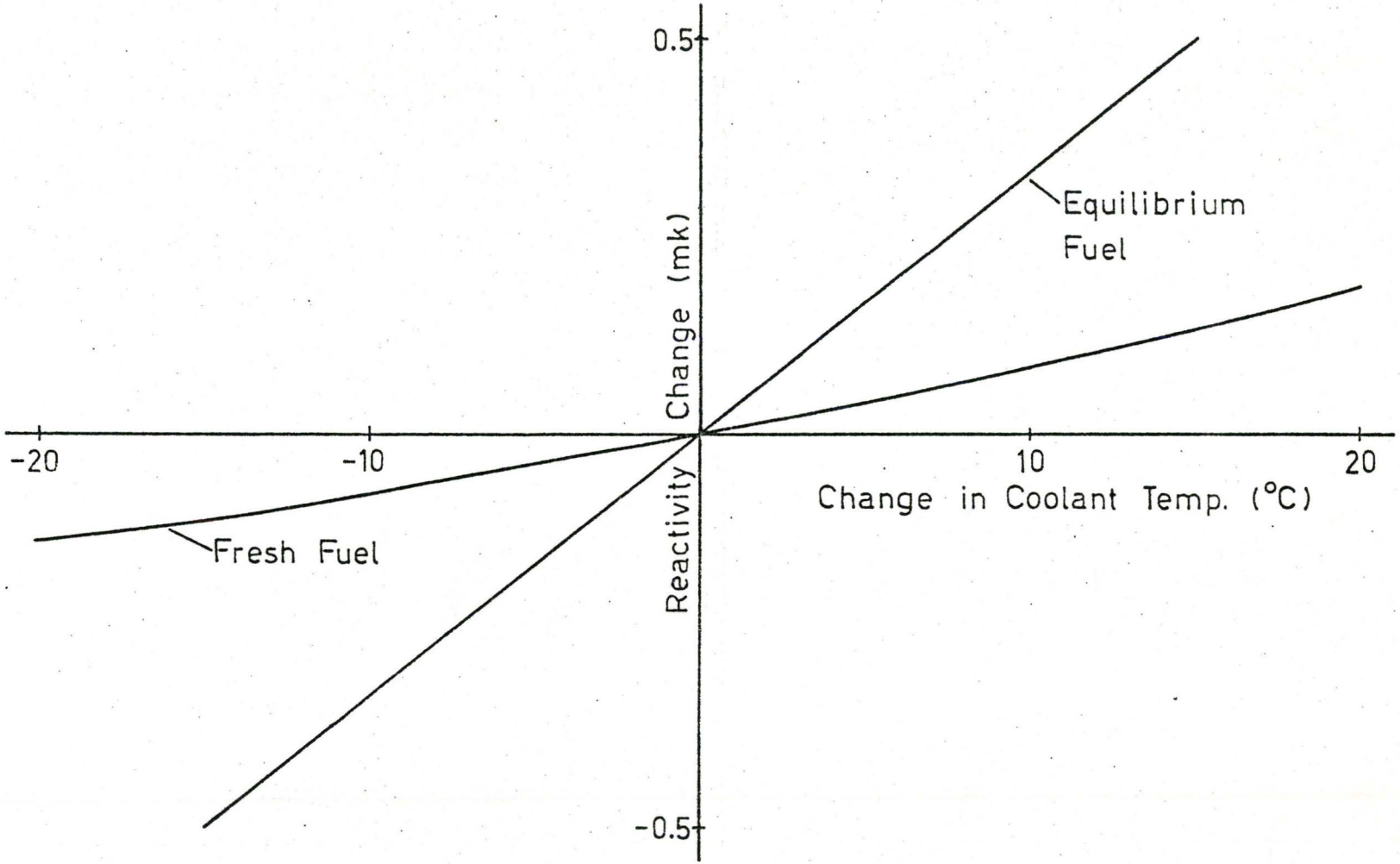

Figure 28. Change in Reactivity vs Change in Coolant Temperature (Pickering 1-4).

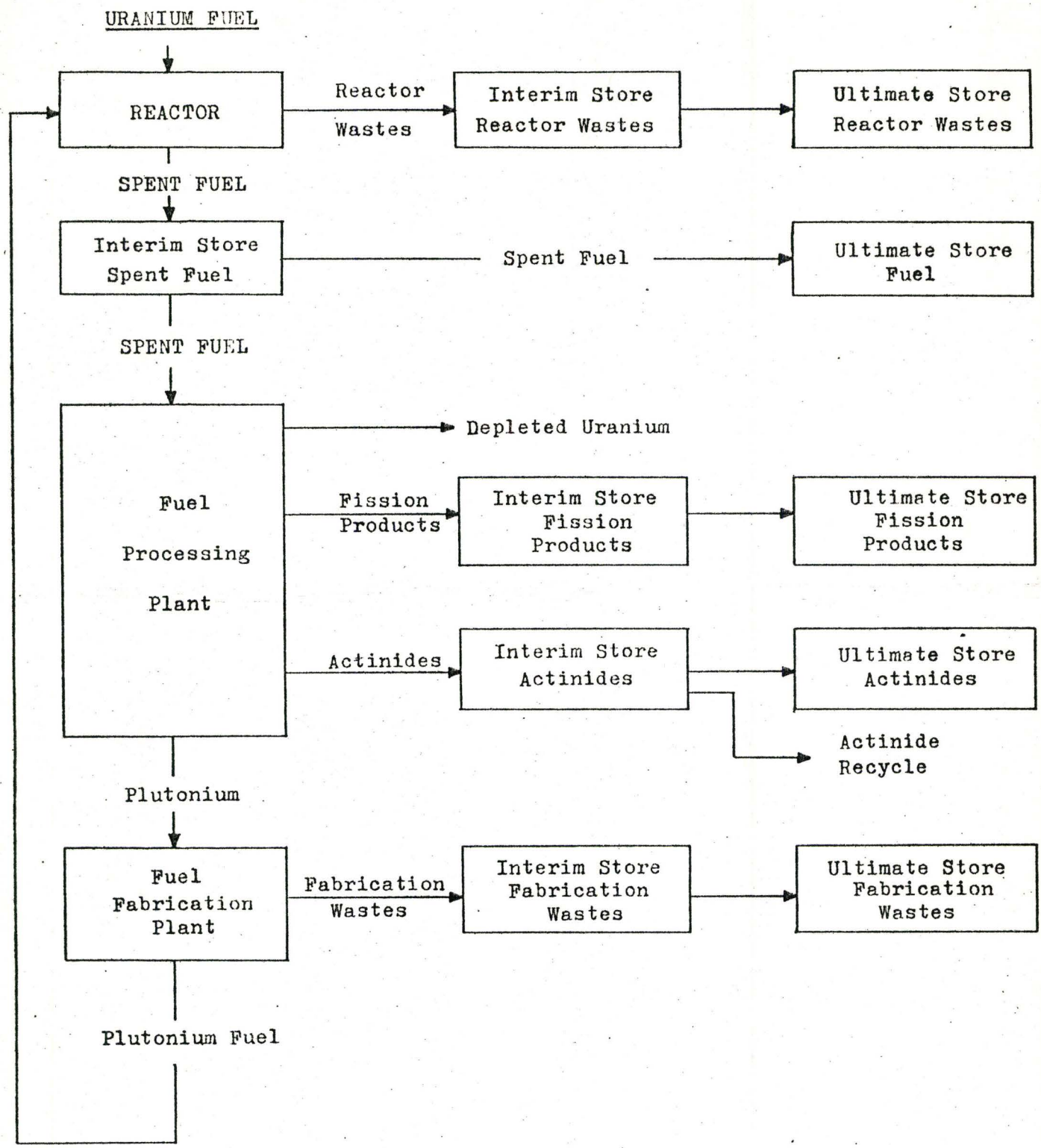

Figure 29. Nuclear Waste Management.

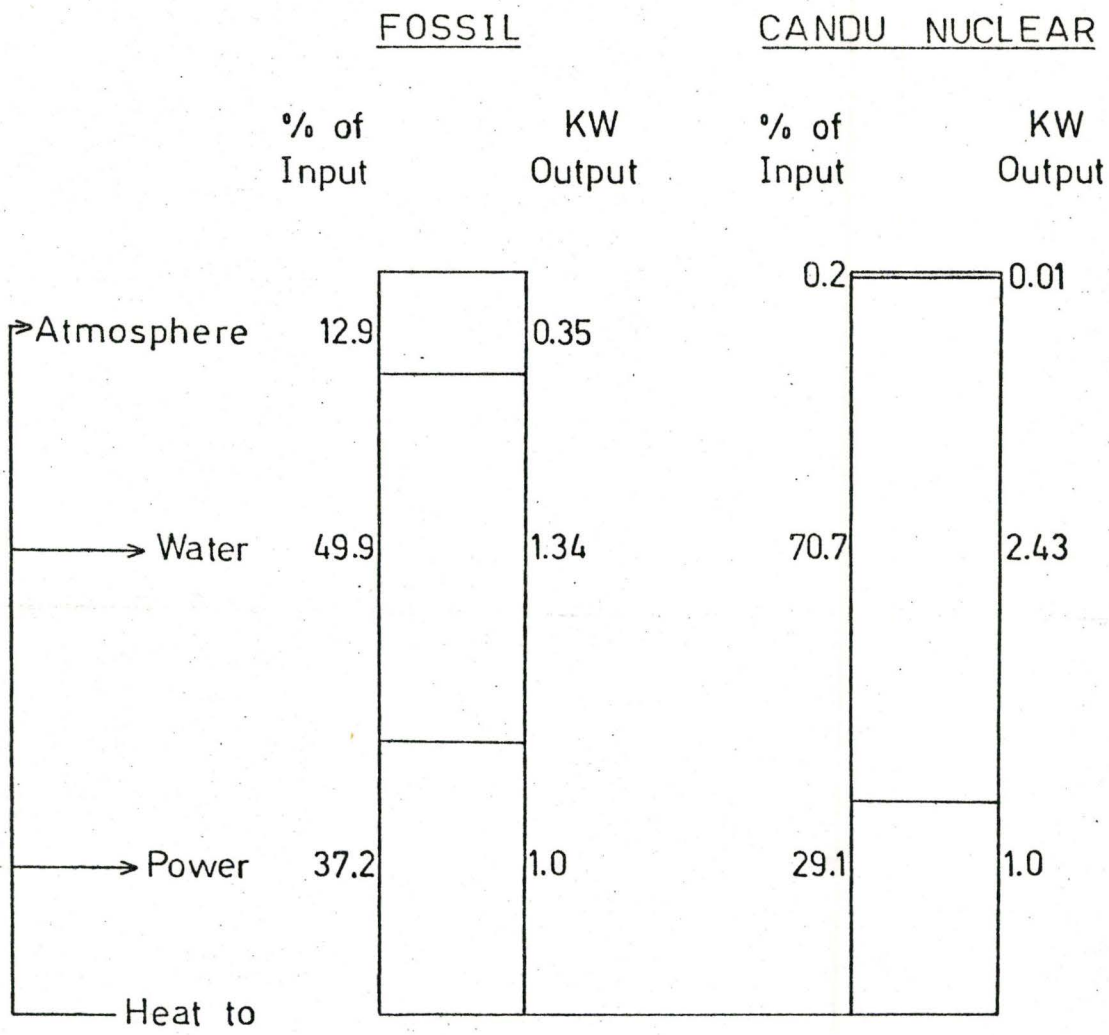

Figure 30. Fossil and Nuclear Heat Distribution.

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