The science and technology of nuclear power

P. K. Iyengar

One of the most important resources in a developing society is energy. There is a direct relationship between the per capita energy consumption of a country and its level of development. The per capita energy consumption of the highly industrialized countries like the USA is about 100 times that of the poorest nation. In this scenario, India, although it can lay claim to a place among the industrialized countries, is not very much above the lowest end of the scale. This is the reason for large segments of our population being under the poverty line. To remedy this situation, it is imperative that we substantially increase energy available for consumption in the country.

Around half the energy consumed in India today is in the non-commercial sector, and consists of crop residues, animal waste, firewood, etc. Burning of firewood is a major cause of deforestation, while animal waste is much better utilized as fertilizer. Improved standards of living will thus call for replacement of non-commercial energy by commercial energy and also for an overall increase in energy use. The various kinds of commercial energy sources are coal, oil, hydro-, biomass, solar, geothermal, wind and nuclear power. Of these, biomass, solar, geothermal and wind power can together provide only a small fraction of our needs at current levels of technology. However, R&D on these sources is important and should be pursued. Of the others, hydro-power is a renewable resource and should be fully exploited wherever possible. Dependence on oil should be minimized since we in India have to import it at great cost. Oil exploration should, however, be actively done. India has plenty of coal, but if we look for high power demand, this could get exhausted very fast. Nuclear power, as currently envisaged by us, will give us energy well into the twentysecond or even the twentythird century. India's nuclear energy programme runs in three phases. In the first phase, there will be a chain of pressurized heavy water reactors (PHWR) working on natural uranium. In the next phase, plutonium from PHWRs will be used to install fast reactors that will essentially run on the unused uranium from PHWR, and will produce more plutonium than they consume. The third phase will use a part of the plutonium that comes from the fast reactors to install reactors that run on thorium. Harnessing of nuclear energy is at present mainly through fission reactors.

P. K. Iyengar is Chairman, Atomic Energy Commission, Government of India. This article is the text of the author's Sir Vittal Chandavarkar Memorial Lecture delivered on 8 October 1990 at the Indian Institute of Science, Bangalore.

What is fission?

Nuclear fission is a process in which a nucleus splits into two parts. It is accompanied by the emission of about 200 MeV of energy. Commonly, one can distinguish two kinds of fission, viz. spontaneous fission and neutron-induced fission. In spontaneous fission, as the name indicates, a nucleus breaks up into two without any external stimulus whatsoever. It is caused by the quantum-mechanical phenomenon called 'tunnelling', by which a stable nucleus can overcome the potential barrier due to coulomb repulsion, and is an event of extremely low probability. The number of spontaneous fission events taking place in natural uranium is no more than about 6 to 7 per second in one kilogram. The other type, namely neutron-induced fission, is caused by neutron absorption in a fissile nucleus like ^{235}U, ^{239}Pu, ^{233}U or ^{241}Pu. The absorbed neutron so excites the nucleus that it can overcome the potential barrier and the probability of splitting is enhanced. This splitting process is accompanied by the emission of some neutrons. This number could be 2 or 3 or more (the average number for ^{235}U is typically 2.43). It is this second kind of fission, the neutron-induced fission, that makes nuclear power possible.

There are two characteristics of neutron-induced fission that make it suitable for nuclear reactors: (i) The fission emits a number of neutrons, which can then go on to cause further fissions. It is thus possible to establish a chain reaction. (ii) The energy released in nuclear fission is high compared to that in chemical reaction. Thus fission relases about 200 MeV, while a chemical reaction like coal-burning releases only 3 or 4 eV. About 85% of the energy of fission appears in the form of the kinetic energy of the two fission-product nuclei. Of the rest, which is contributed by the radioactive decay of the fission products, some appears as β and γ, some as kinetic energy of emitted neutrons, while about 6% is totally lost as neutrino energy. The fission products soon come to rest, having lost their energy to the other nuclei around. This raises the temperature of the material in which fission takes place. If the heat is not properly removed, the fissile material can become red-hot and melt, or even vaporize.

Nuclear reactor

The nuclear reactor has fissions taking place under controlled conditions. 'Controlled' essentially means that the rate at which fissions take place can be decided at the will of the operator. In an assembly of materials

(a) Apart from the initiating event, which could be an external cause, or failure of some system, every other part of the reactor functions as designed.
(b) The reactor protection system and engineered safety systems which are to take care of the accident show a failure in that one system or component fails to function as designed.
The intent of this analysis is to ensure that if one system fails, there will be another to compensate for its failure. The totality of trip parameters and engineered safety features is finalized such that the reactor can absorb two failures without damage to itself. The criterion followed is that, for the large majority of initiating events, the reactor should be completely protected even with failures. For some very-low-probability events, although limited damage to the reactor can be accepted, there should be no release of radioactivity to the environment.

Reactor protection

The reactor protection system works by sensing the advent of an accident and immediately shutting down the reactor. The coming accident should be sensed by monitoring a number of parameters. The designer should identify those parameters that are likely to signify a possible accident precursor in case they go out of the stipulated bounds. The number of trip parameters is fairly large and the list given here is not exhaustive. In the case of certain other parameters, their violation is not serious enough to warrant a reactor trip, yet they need to be corrected. This is done by a 'set back', which is an automatic reduction of reactor power.

Trip parameters

1. High neutron power
2. High log rate
3. High PHT pump room pressure
4. High PHT pressure
5. Low PHT pressure
6. Primary coolant channel flow very low
7. High boiler differential temperature
8. Low boiler level
9. PHT storage tank level
10. Moderator level very low
11. Actual power/demand power
12. Deaerater level very low
13. A primary shut-off rod leaving top limit without trip
14. Poising of primary shut-down system within 2 hours of a trip
15. No primary circulating pump running
16. Secondary shut-down system banks unavailable
17. Adjuster rod coolant flow low for more than 3 minutes
18. Manual trip
19. Reactor start-up trip

Set-back parameters

1. Master key removed from door interlock system
2. Alpas tank level very low
3. Reactor outlet header temperature high
4. Steam discharge valve not fully closed
5. Boiler level low
6. Channel outlet temperature very high
7. Bleed condenser level very high
8. Moderator temperature very high and process water pressure low/temp. high
9. Manual set-back switch
10. Active high-pressure PWRB supply header pressure very low and bleed cooler outlet temperature high

Defence in depth. Systems are so ordered that if one fails another one will be available. Thus let us consider a reactivity transient. It should give a trip on high log rate. If that fails, power rise will continue until it reaches the over-power trip level. If power trip also fails, high boiler differential temperature will take over, and so on. If one shut-down system fails, there is another to provide backup. If fuel fails, releasing radioactivity, it is still contained within the PHT system. If PHT also fails, it is held within the containment.

Preplanned actions. For every conceivable failure, scenarios are developed beforehand and the chain of operation actions that are necessary to bring the sequence of events under control are worked out. This is done in such a fashion as to make the best use of the different safety systems provided. Although all important safety actions are initiated automatically, there is also a need for a number of manual actions by the operator. Operators are trained to follow the action plans.

Engineered safety features (ESF). A large number of engineered features are provided. These include emergency core cooling system (ECCS) and containment, including energy management features like vapour suppression pool, containment isolation systems, radionuclide management system, etc.

Emergency preparedness. In the remote possibility of an accident happening despite all these precautions, preplanned schemes for protecting the public are made. The reactor itself is surrounded by an exclusion zone where habitation is not permitted. This is surrounded by a 'sterilization zone' in which existing settlements are not disturbed, but further development is not permitted. Apart from this, there are evacuation plans and emergency drills in which the local civic authorities participate.

The Chernobyl accident

The accident was the outcome of an amazingly large number of violations. A technical description of the accident scenario reads like a saga of repeated attempts by the reactor protective system to prevent the accident, each one foiled by the operator. A very brief description of the accident, giving only the basic essential parts, is as follows.

The accident occurred during a test. This test was supposed to be done at a power level of about 800 MW. The RBMK reactor (the type that existed in Chernobyl) has a positive power feedback coefficient below 600 MW, and so it is not supposed to operate in this range. The operator forgot to enter 'hold power' at 800 MW before power reduction to 800 MW started from its nominal power of 3200 MW. As a result, power fell below 800 MW and went to the unstable range where the positive coefficient could only result in any reduction in power leading to further reduction of power until the power level becomes very low. Frantic attempts were made to prevent this. In the process, more and more control rods were drawn out and many trips were disabled. Finally the reactor was brought up to 200 MW, which is a forbidden zone. The test should not have been attempted in this zone, but the operators went ahead anyway. As part of the test, they shut off steam to one of the turbines. This led to rise in PHT pressure and simultaneously to the running down of the pumps connected to that loop. The flow into PHT decreased. In this highly sensitive region, all these factors combine to give a resultant value for steam quality. In this case, it turned out to be more voidage in the PHT. Since the void reactivity coefficient of RBMK is positive, this started introducing positive reactivity, which was made worse by the positive power coefficient, and the reactor became prompt critical. The proposed test was not a routine test, but a special one attempted in an operating power reactor without proper planning.

Can it happen here?

It will be instructive to compare the designed behaviour of RBMK and PHWR. Both are pressure-tube reactors and have positive void coefficient of reactivity. But, unlike RBMK, the PHWR does not have voids in the coolant during normal operation. In RBMK, the coolant is a two-phase steam-water mixture, which is highly compressible so that the void coefficient is a constantly shifting quantity. This introduces a corresponding fluctuation in core reactivity, which is compensated by a system of control rods constantly acting to correct the reactivity changes caused by changes in void fraction.

In addition to the void coefficient, there is a fuel-temperature coefficient, which is fast-acting and always negative, being mainly constituted by the Doppler coefficient. In PHWR, there is no voiding under normal conditions, and, as the power coefficient is dominated by the Doppler coefficient, it is always negative. In RBMK, too, this is true over the normal operating range; however, below 600 MW, void coefficient dominates, leading to a positive power coefficient. This is thus an unstable region where a small increase in power can trigger a larger increase. This is what actually happened at Chernobyl. In PHWR, there is no range over which the power coefficient is positive. The boiling coolant in RBMK combined with a significant void coefficient leads to a very strong coupling between the physics and thermal hydraulic behaviour. Such an interaction in PHWR is weak, specially during normal power operation. In a system like RBMK, where the void coefficient is positive, this could lead to difficulties in bringing the reactor to a pre-specified power, specially over the range where the power coefficient is positive. Further, a small perturbation to the reactor, either deliberate or otherwise, could lead to a much larger change on account of positive feedback mechanisms.

The shut-off system of RBMK appears to be very slow. This, coupled with the positive power coefficient, can create a situation in which, although the trip signals have been properly received and the shut-down system activated, the rate of reactivity insertion due to the positive power coefficient exceeds the rate of negative reactivity insertion by the control system. Thus, even after 'shut-down' starts, reactivity continues to increase. This, too, happened at Chernobyl. By contrast, the shut-down systems of PHWR—the moderator and dump (Rajasthan and Madras atomic power stations) and the mechanical shut-off rods (Narora atomic power plant) falling under gravity—are relatively fast-acting systems, and, combined with the power coefficient, which is always negative, manage to insert sufficient negative reactivity during the first vital second to control any power excursion.

The shut-off rods of RBMK also act as regulating rods. This can lead to a situation in which the regulatory system, needing extra reactivity, pushes them out beyond a level to which they should not be permitted to go, as it would lead to reduction of their effectiveness when they are called upon to act. This happened at Chernobyl. In PHWR, the mechanical shut-off rods have fixed positions outside the core and cannot be withdrawn further. Moderator dump too starts from full tank, and though it is part of the regulatory system, its efficiency cannot be reduced.

If shut-down systems fail to act and the core overheats, in the worst case it could lead to the hot fuel touching the pressure tube, which also could fail in turn. The hot coolant is now depressurized and escapes into the moderator. In the RBMK, the moderator is hot graphite, at a temperature of about 800°C. Coming into contact with the coolant, this can give rise to water–graphite reaction, creating the highly explosive water gas ($CO+H_2$). In PHWR, the hot coolant bubbling into the cold moderator could increase the pressure in the calandria, leading to failure of the rupture disc. Steam and D_2O will rush out through the opening and terminate the power excursion. The escaping steam will get partially depressurized in the calandria vault. If the depressurization is insufficient, a rupture disc in the vault will give way, and finally depressurization will take place in the much larger volume of the reactor building. The containment will hold, so that the release of radioactivity to the atmosphere would be kept within levels. In RBMK, the fuelling machine is directly above the core, and is located in a hall which does not have any containment capability. In the event of an explosion above the core, it can crash on to the core, thus further damaging it. In PHWR, the fuelling machines are on the sides and operate in thick-walled fuelling-machine vaults. The configuration is thus safer from the point of view of both reactor and containment.

From the foregoing it can be seen that an accident of the type that took place at Chernobyl cannot possibly occur in our reactors. Further, in the absence of availability of large quantities of inflammable materials and limited quantities of explosive gases that can be released, it would appear that the large fire and explosions of the magnitude that occurred during the accident at Chernobyl cannot occur in our reactors. Even in the case of the worst reactivity-induced accident, damage due to release of radioactivity to the surroundings cannot possibly be on a scale as large as what happened at Chernobyl on account of our containment systems and other engineered safety features.

Pressurized heavy water reactors

Figure 4 shows a simplified flow diagram of a PHWR and Figure 5 a cut-away view. It has natural uranium fuel and uses heavy water as both coolant and

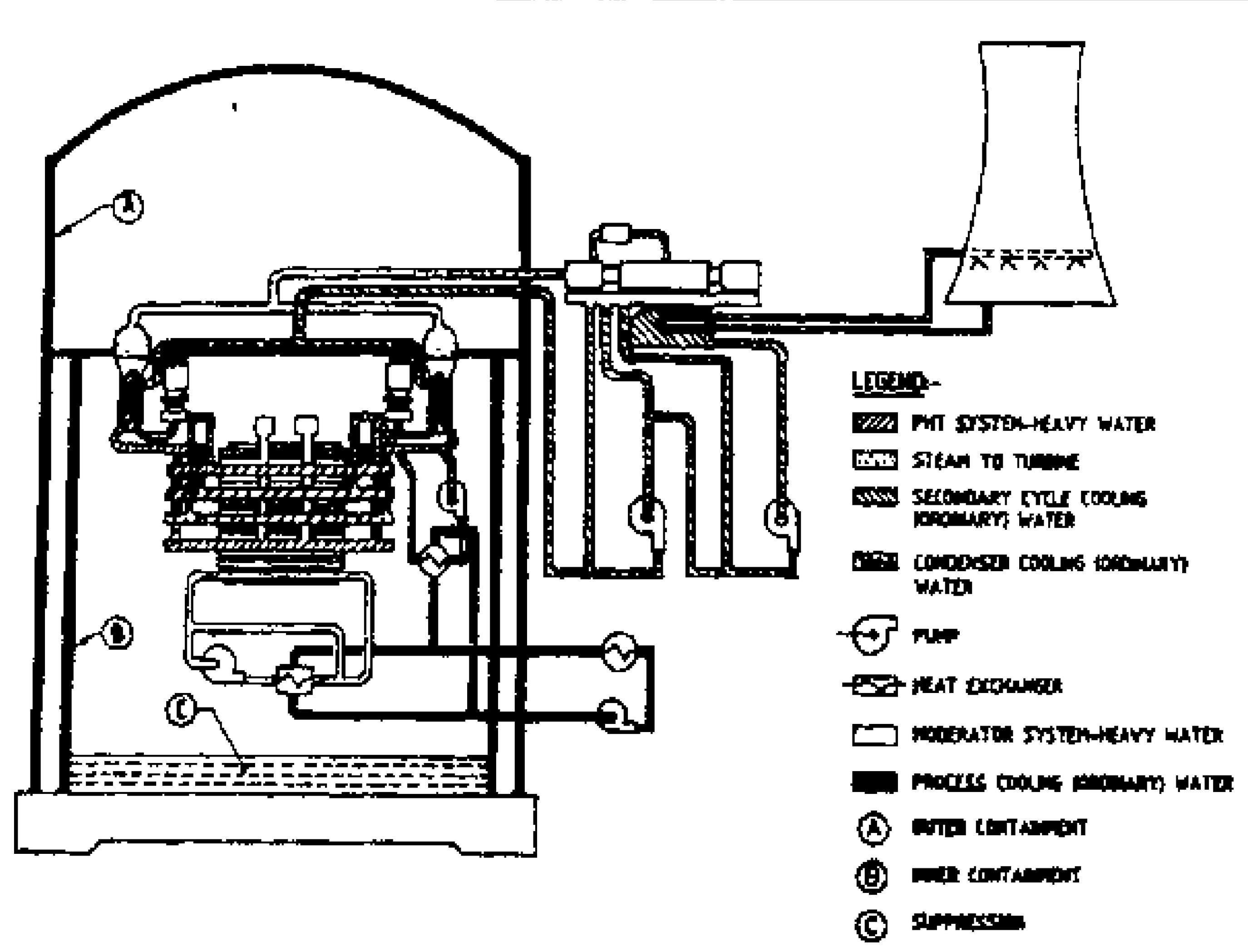

Figure 4. PHWR—closed loop water and steam circuit.

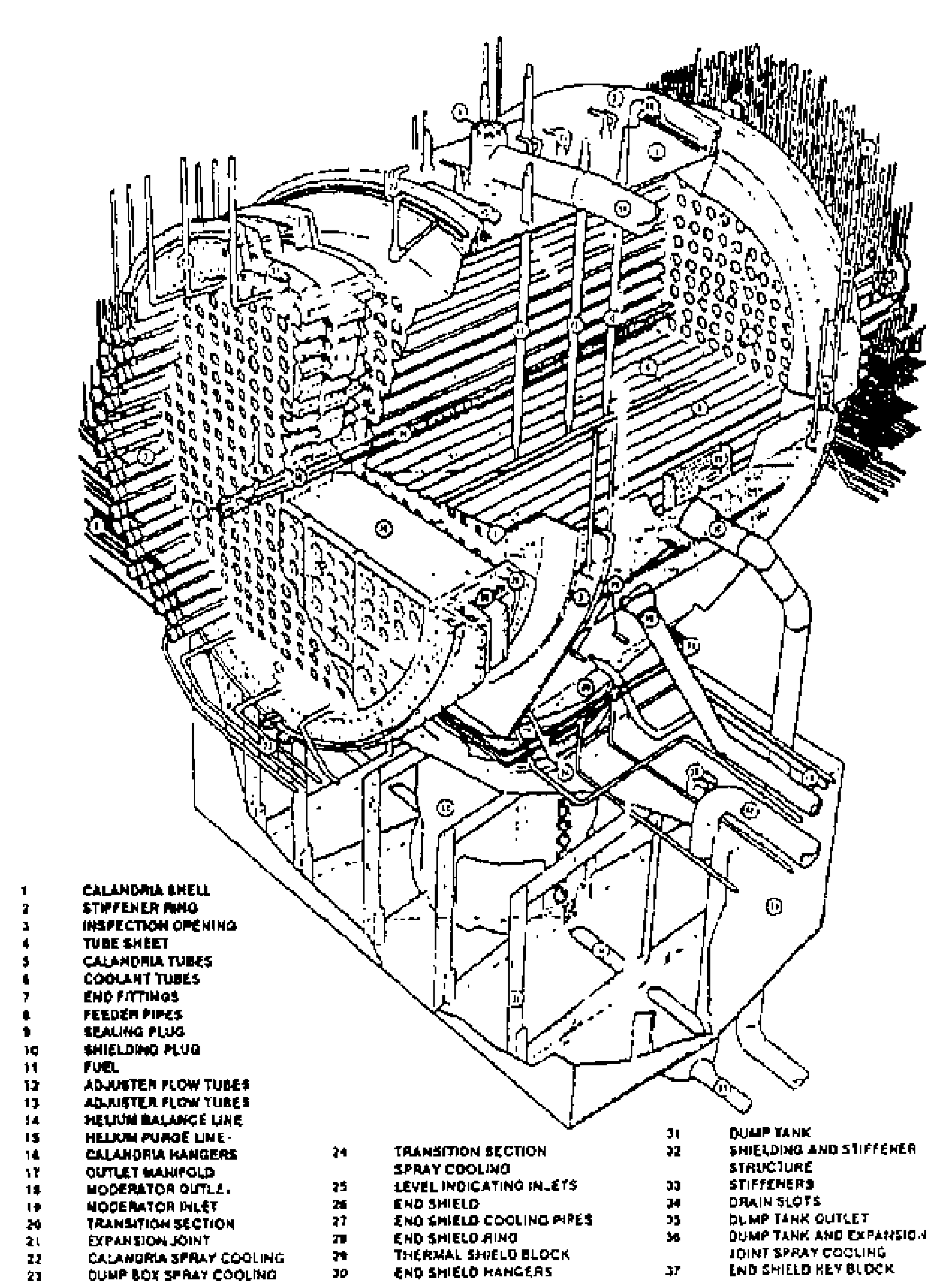

Figure 5. PHWR—cut-away view.

moderator. The fuel is in the form of 19 rod-cluster bundles of 50 cm length. It is cooled by high-pressure coolant flowing through horizontal pressure tubes. Moderator is at low pressure and relatively low temperature (50°C). Fuel utilization is maximized by having on-load refuelling. The neutron spectrum is a well-thermalized one, and conversion ratio is around 0.8. Fast shut-down is achieved by gravity-driven shut-off rods, poison injection under high pressure into the moderator or into tubes in the core, or moderator dumping. Power is maximized by having fuel in the inner region having more burn-up than that outside.

At present, India has about 1450 MW(e) installed capacity of nuclear power reactors. We have two 160 MW(e) units at Tarapur, which are boiling water reactors. At Rajasthan, we have two units of 220 MW(e) PHWRs. Another two units of 235 MW(e) capacity are operating at Kalpakkam near Madras. A fifth 235 MW(e) PHWR has been commissioned at Narora, and is presently operating at 75% power. One more unit at Narora will be ready for commissioning late this year or early next year. Two units at Kakrapar in Gujarat are in an advanced stage of construction, and four more units (two each at Kaiga and Rajasthan) are in various stages of construction. We have also completed the design work of a bigger, 500-MW(e) PHWR. A series of units of these is also planned.

New concepts

The established power reactors today are the light water reactor (LWR) and the heavy water reactor (HWR). LWR form nearly 80% of the total installed capacity in the world and HWR about 15%. To receive serious consideration any other concept should display some superiority over these two. The thrust of new concepts is along three directions: (i) lower energy cost, (ii) better fuel utilization, and (iii) enhanced safety.

Fast reactor

The quantity η, defined as the number of fast neutrons produced by fission for every one neutron absorbed in fuel, is an important parameter in fuel utilization. Of these η neutrons, one is needed for maintaining the chain reaction. Another small number, say x, is lost by leakage, parasitic absorption, and the like. The remainder, $\eta - 1 - x$, is absorbed in fertile material like ^{238}U and is converted into fissile material. Clearly, if $\eta = 2 + x$, exactly as much fissile material is produced as was consumed. Since x is generally around 0.2, an η of about 2.2 is needed if we want to get breeding. For ^{235}U and ^{239}Pu, η is much lower than this in the thermal spectrum. In a fast spectrum, η of ^{239}Pu could go as high as 2.9 or 3.0. This has led to the concept of fast reactors. Power densities in fast reactors are very high. There is also the need to avoid moderating materials in the core. This has led to the concept of liquid metal cooling. Figure 6 shows a schematic of the fast breeder.

A fast breeder test reactor of 40 MW(th) has been constructed and is operating successfully at Kalpakkam. The design of a 500-MW(e) prototype fast breeder reactor (PFBR) has been finalized. The programme is thus making steady progress in commercial exploitation of nuclear energy for electricity generation.

High-temperature graphite reactor (HTGR)

This is an intermediate-spectrum concept whose technology is sharply different from the others. The fuel is small particles of UO_2 coated with pyrocarbon to form balls of $\sim 5\ \mu m$ diameter. Such balls are then packed together to form big balls of ~ 5 cm diameter. The big ball has an outer shell about 1 cm thick with carbon alone and no UO_2. Figure 7 shows a simple view of the reactor. A large number of these balls are put into the reactor vessel, which is funnel-shaped at the bottom. The coolant is helium gas, which flows upward among the balls. The balls are extremely compact and have very high integrity. They are impervious to the coolant and can retain the fission products with very high level of reliability. The gas coolant can take very high temperatures, resulting in very high thermodynamic efficiency. Fuelling is continuous, being done by fresh balls added at the top and old ones removed through the funnel at the bottom. The balls come out through the funnel at the rate of one every few seconds. They are checked for damage and burn-up, and the undamaged ones with low burn-up are fed back into the vessel at the top. This whole procedure is automated. The only disadvantage of this concept is the difficulty in reprocessing and the problem of graphite burning. It is understood that the reprocessing problem has been solved but the safety consideration associated with the likelihood of high-temperature graphite catching fire remains. The solution here is to take great care to avoid the graphite coming into contact with oxygen in any form, notably through water ingress.

Process-inherent ultimately safe (PIUS) reactor

Figure 8 describes the working of this reactor. The reactor is basically an LWR. It is placed at the bottom of a very tall vessel (A), and the entire set-up is immersed in a pool of borated water. The hot water from the reactor rises upwards and is returned to the reactor through a pump (B). A steam generator placed in the circuit (C) will remove the heat and send it to the turbine. C also shows a region of stagnant water below

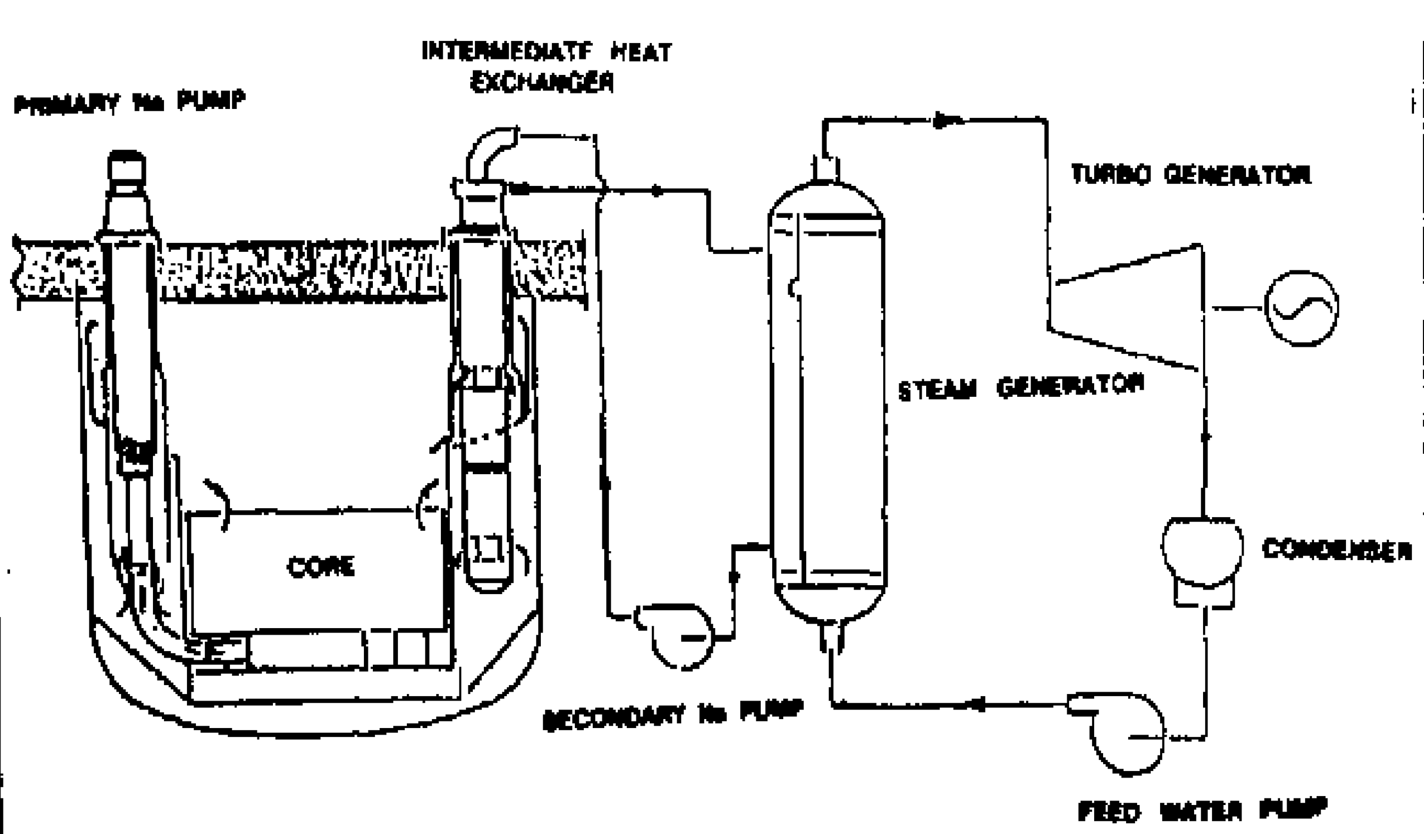

Figure 6. Simplified schematic of 500-MWe PFBR.

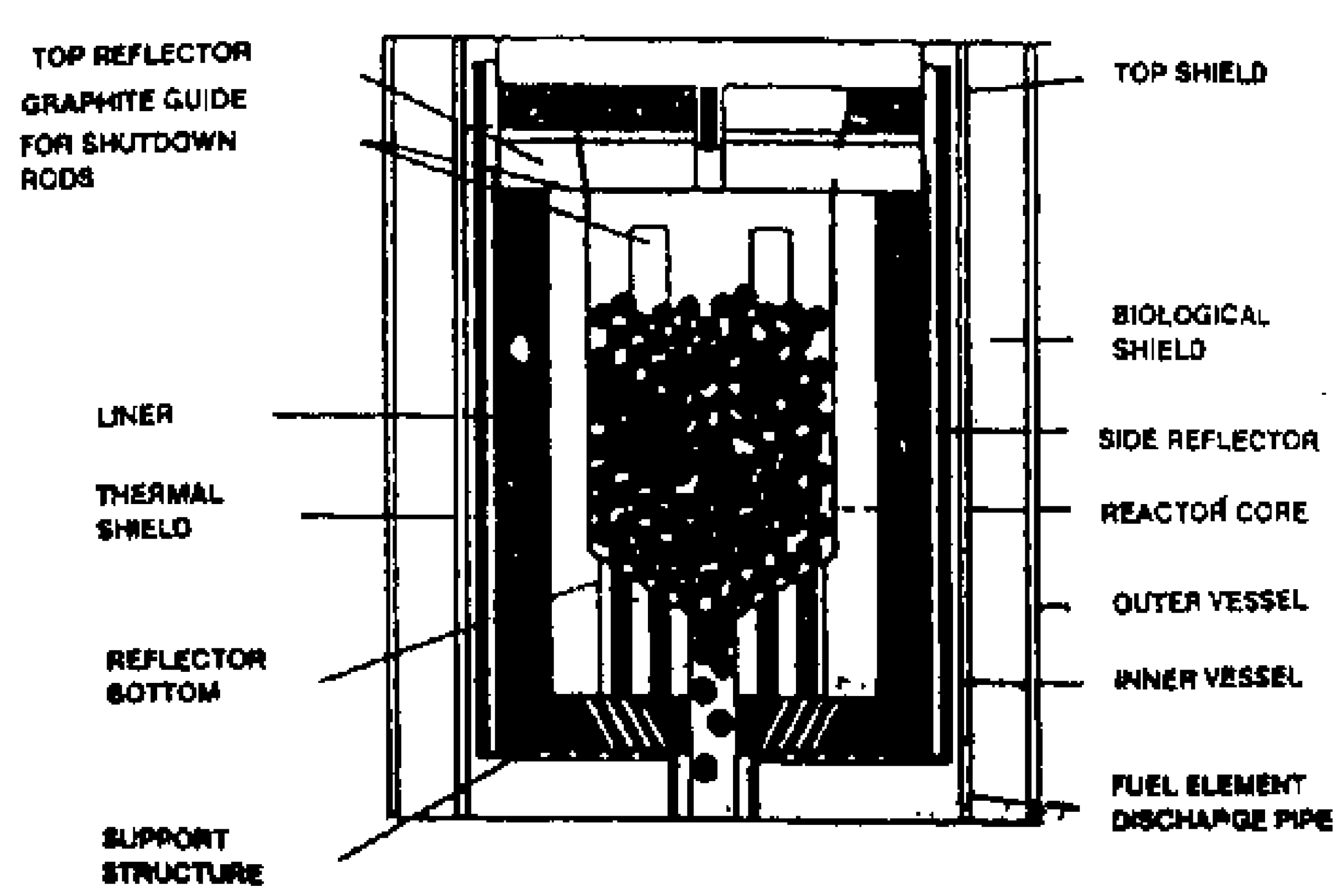

Figure 7. The principle of the pebble bed reactor.

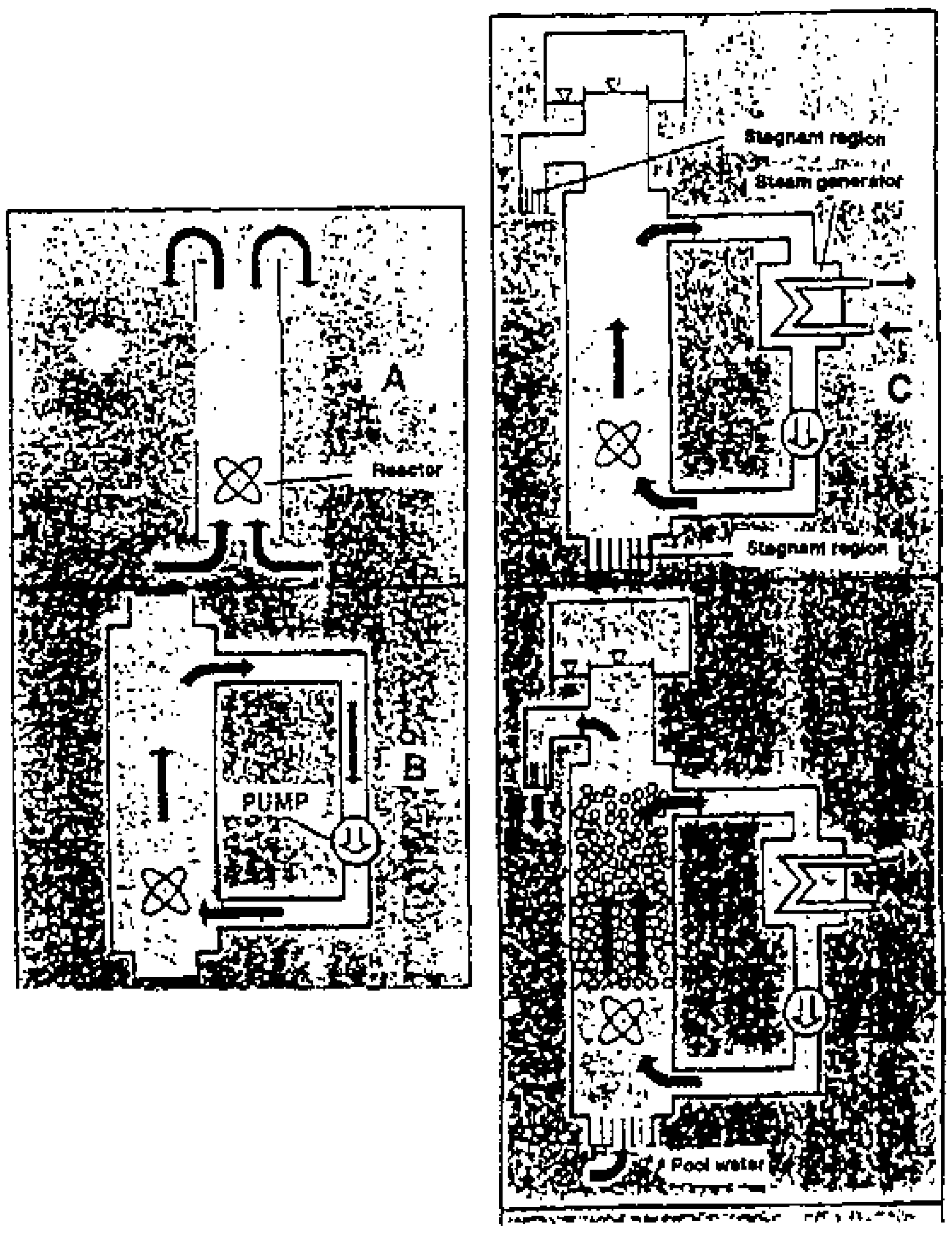

Figure 8. PIUS reactor.

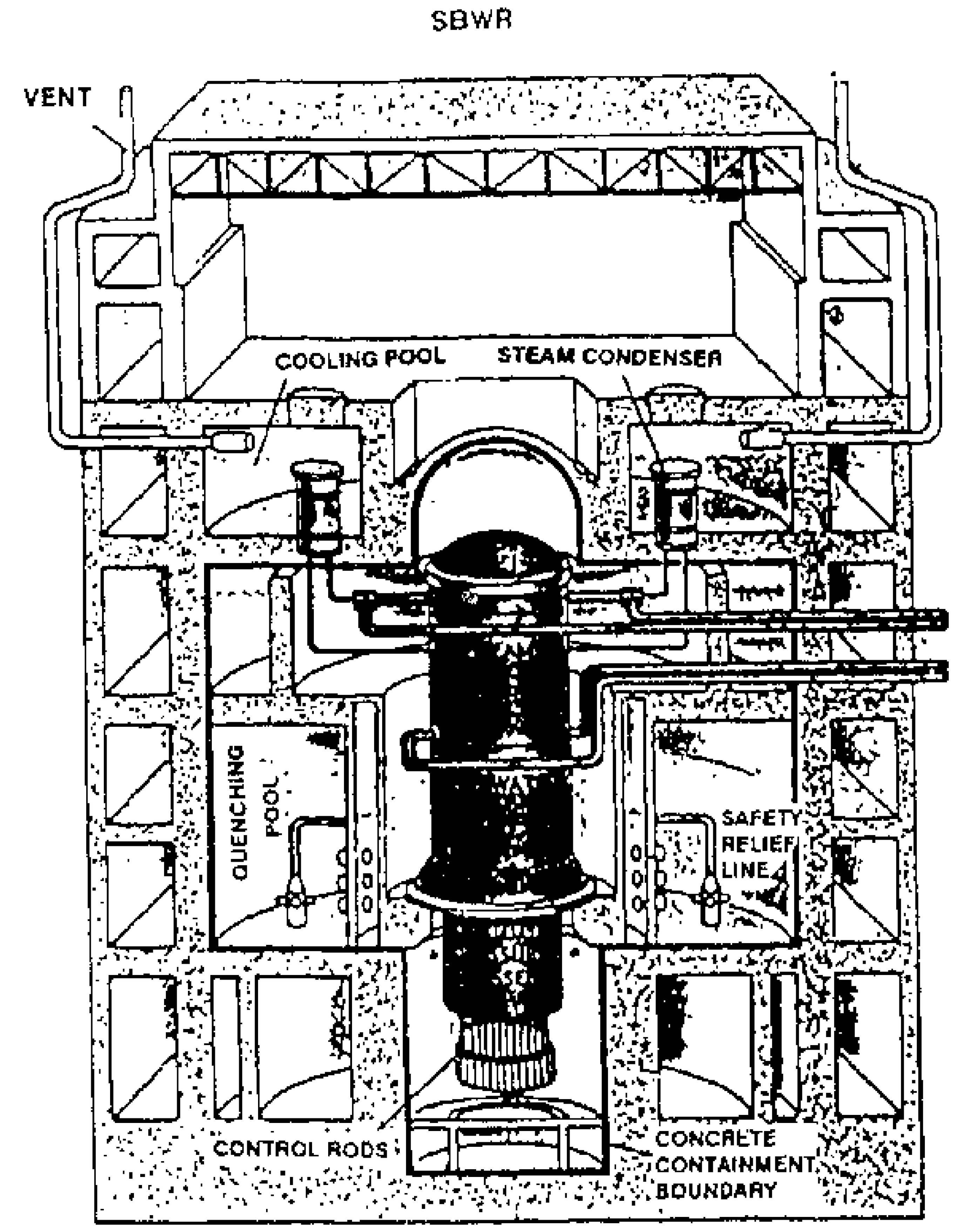

Figure 9. Simplified boiling water reactor.

the reactor. Thermal hydraulic equilibrium is so maintained that the stagnant water does not permit any mixing of the water in the pool outside with the water in the reactor. There is no other physical separation between the reactor water and the pool water. Any kind of accident situation will disturb this equilibrium and cause borated water from the pool to enter the reactor vessel (D). This then can be described as a passively safe reactor since no engineered action is required to ensure safety in the event of malfunction.

Simplified boiling water reactor (SBWR)

The SBWR is a take-off from the ordinary BWR. A schematic diagram is shown in Figure 9. The essential feature of this concept is that it does away with primary heart transport pump altogether for heat removal. The coolant flow is only by natural convection. To achieve this, it is necessary to have a very tall pressure vessel (about 24 m). The steam goes directly to the turbine. There are large pools of water close to the reactor at an elevation higher than the reactor. In the case of an accident these can be brought down by gravity to cool the core. There are steam condensers immersed in these pools. For decay heat removal the steam from the reactor vessel is directly sent to the steam condensers, where it condenses, releasing heat to the pool water outside. The pool water itself is cooled by evaporation from its surface.

Advanced light water reactor (ALWR)

This concept came about from a need to recycle plutonium in reactors since there is a glut of plutonium accumulated from the present generation of LWRs. The conventional LWR has a neutron spectrum that is very unfavourable to plutonium. A harder spectrum would be better. There was also a desire not to change existing technology too much. Attempts are therefore being made to modify the existing LWR technology to suit the requirements of plutonium. This could be done by reducing the quantity of water in the reactor. Thus, by reducing moderation, the spectrum will harden and lead to better fuel utilization of plutonium. In parallel, it also has the effect of making heat transfer less efficient and increasing the neutron absorption in ^{238}U. This has to be compensated by high enrichment. The fuel that has been considered for this concept is MOX with 6–8% plutonium in uranium. The ALWR could be a boiling water reactor or pressurized water reactor.

Plutonium recycling

Plutonium recycling has been considered in concepts other than LWR as well. Notably it has been studied in HWR, the Indian PHWR among others. Fuel utilization in the present PHWR is less than 1%. Recycling in LWR will increase fuel utilization by 2.5%. Recycling in PHWR will increase it five-fold. ALWR can increase the fuel utilization ten-fold. The fast breeder can give practically full utilization. Plutonium is produced by neutron absorption in ^{238}U followed by two beta decays. As already explained in the section on fast reactor, plutonium can breed only in fast reactors, which is why fast reactors are a necessity for obtaining complete fuel utilization with uranium. The scene is different when we consider thorium.

Thorium utilization

Thorium itself does not contain any naturally occurring fissile element. Any cycle in thorium has to be initiated from the uranium cycle in some form or the other. But once initiated, the thorium cycle runs on the fissile material ^{233}U, which is produced by neutron absorption in thorium. This material is superior to both ^{235}U and plutonium in thermal reactors. It has an η value of above 2.2 in the thermal spectrum, in contrast to a value of about 2.0 for both ^{235}U and ^{239}Pu. Because of this it is possible in a thorium system to get practically full fuel utilization in thermal reactors. Many analyses have been carried out to examine various fuel cycles using thorium in the PHWR. It has been found possible to have self-sustaining cycles in which external fissile material is required only till such time as the reactor has reached equilibrium; thereafter, it can run on thorium alone. However, the economics of the fuel cycle will be better if a small, steady supply of external fissile material like plutonium could be made available. This plutonium used in the PHWR will suffer from the consequences of having to be placed in an adverse spectrum. We are currently examining the prospect of doing this while placing the plutonium in a favourable spectrum. This has given rise to a totally indigenous concept which we have named the advanced heavy water reactor.

Advanced heavy water reactor (AHWR)

In this reactor we wish to retain all the desirable features of PHWR and of the pressure-tube construction. We also look for low capital cost and low operating cost. Use of boiling light-water coolant will save the capital cost of the heavy-water inventory by about 30%, reduce the D_2O make-up requirements by 90%, eliminate the need for having extreme leak-tightness in all the seals and valves of the PHT (thus saving on cost), eliminate the tritium problem (thus saving on man rem), and enable the use of direct cycle (thus doing away with steam generator and saving on cost). A schematic diagram of the reactor is shown in Figure 10.

One of the normal disadvantages of light-water coolant in the HWR is the possibility of positive void coefficient. In this concept we have so designed the

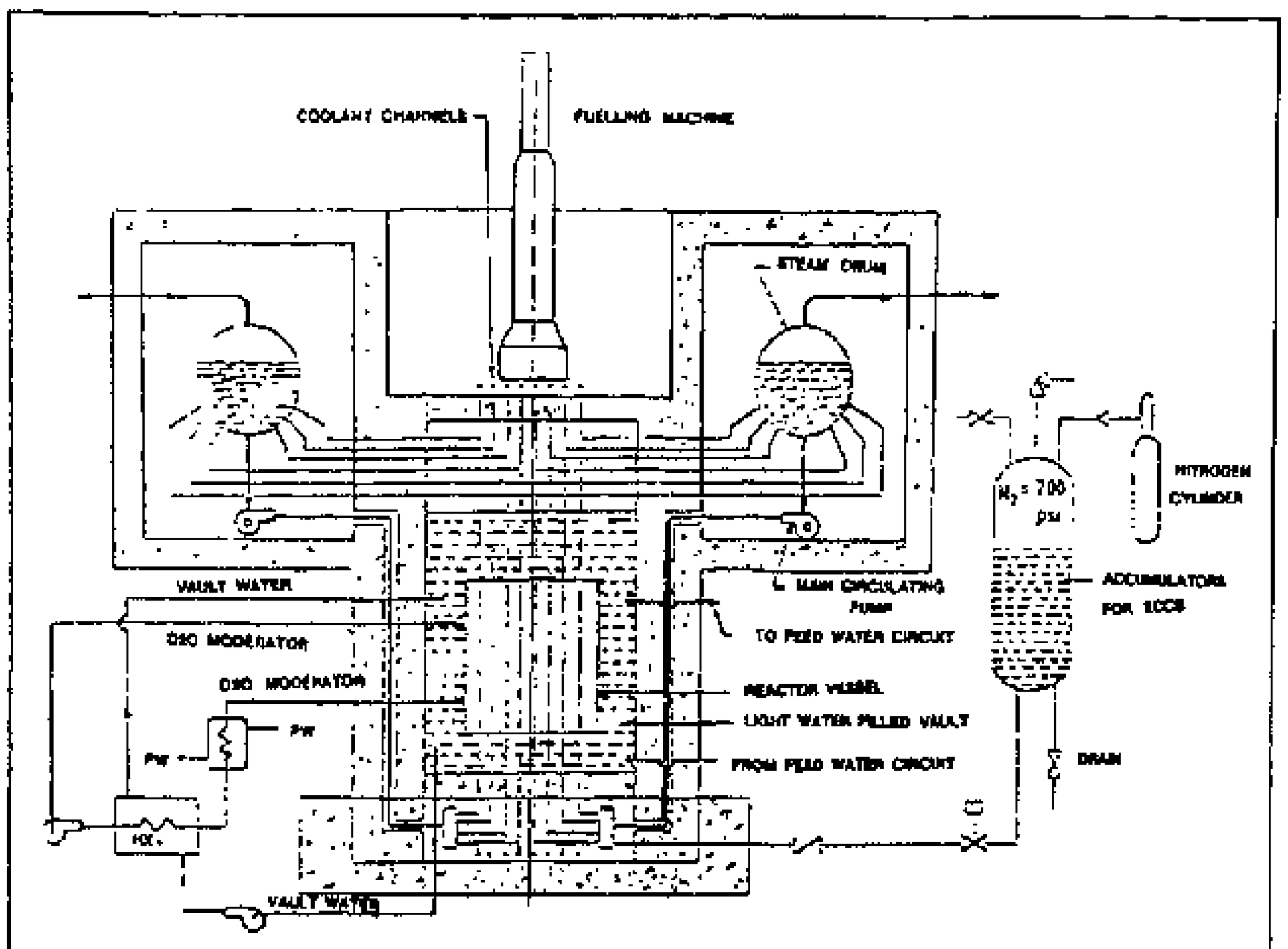

Figure 10. Schematic of advanced heavy water reactor.

system as to avoid the positive void coefficient and at the same time station the plutonium in a favourable spectrum. This has been achieved by having two distinct regions in the core. Figure 11 shows a cross-sectional view of the core. There are two regions, the seed and the blanket. The blanket consists of thorium fuel enriched with ^{233}U. The ^{233}U content is so fixed that the outer region is self-sustaining in ^{233}U. Coolant is boiling light water and this would give a positive void coefficient to the blanket region. The seed region consists of MOX fuel in which natural uranium oxide is mixed with 6–8% plutonium. The seed region is cooled by boiling light water which is in common circuit with the coolant in the blanket region. The size of the seed region is large enough that, over the major portion of the seed, the neutron spectrum is decided by the seed lattice alone. This has been designed to give a hard spectrum that is both favourable to plutonium and also has a negative void coefficient of reactivity. The combined void coefficient of the total core will be

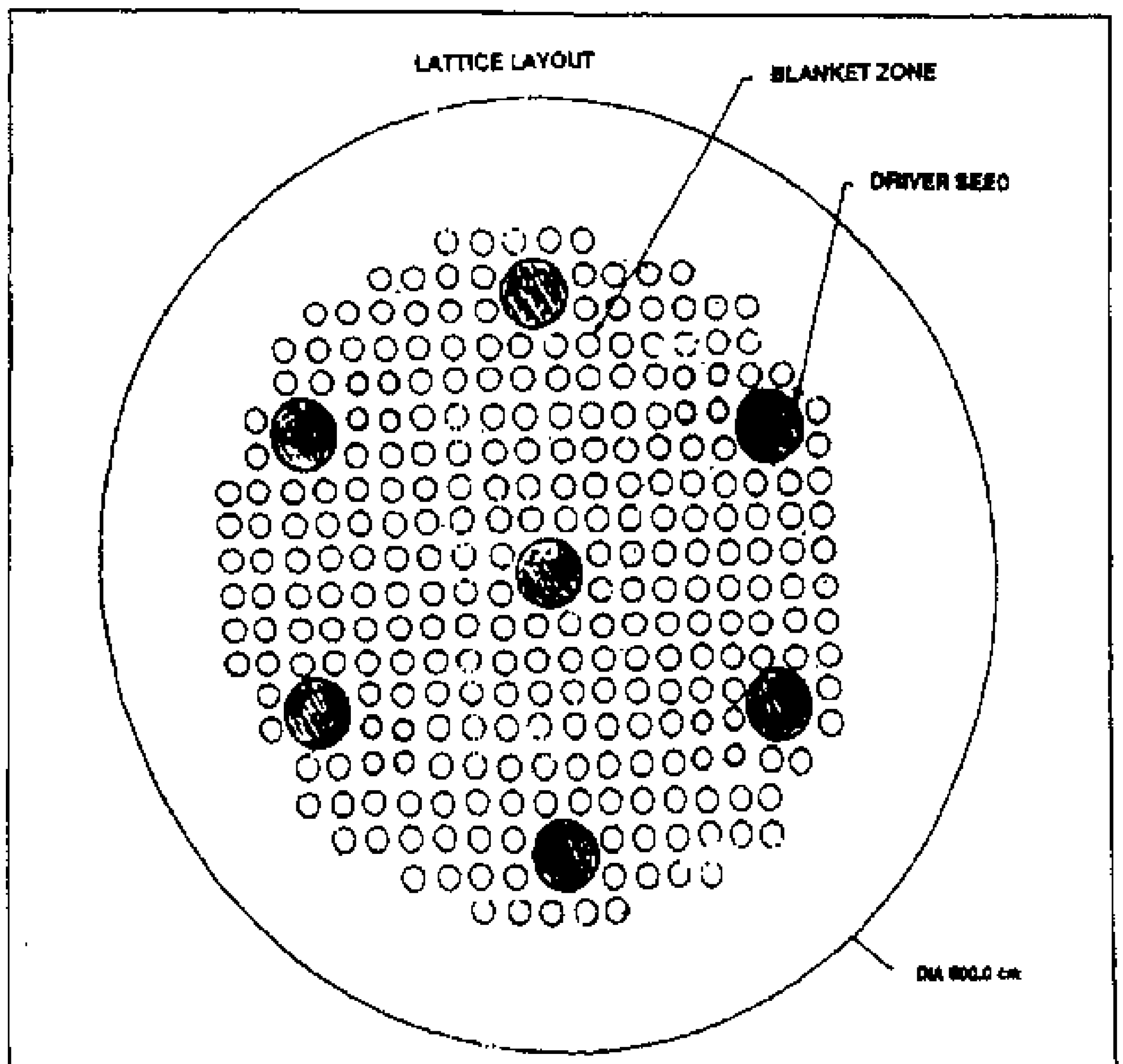

Figure 11. Cross-section of core of AHWR.

negative. One can get about 80% of the total power from the thorium region.

Concluding remarks

It can be seen from the above that fission reactor technology has come a long way since the discovery of nuclear fission in 1939. In the initial days, it was the work of basic scientists, like Fermi, who invented the heterogeneous reactor, to make it possible to sustain a chain reaction. Since then the basic idea has been scaled up to make it a steam generator for turning a turbine and providing electrical power, up to 1200 MW in a single unit. In this process, many complications were introduced, arising out of the need to sustain a power density in the core of the reactor high enough to make steam generation economical. In addition, the need to keep a constant control over the divergence of the chain reaction in different zones and the need to provide decay heat removal after the reactor is shut down have made substantial demands on the engineering of a power station. I have discussed the physics behind these two requirements.

The strong demands made by this technology have forced scientists and engineers to look ahead and incorporate safety devices that will take care of all eventualities and plan for disaster management. This has made the fission nuclear reactor for power generation a fairly complicated device. In recent times, new designs and alternative schemes to mitigate the effects of the two requirements mentioned above are being prepared. Most of them have not yet been incorporated in commercial units, mainly because of the capital cost involved in building prototypes of new designs. After the Chernobyl accident, a series of modifications involving new ideas are being proposed and incorporated. I have mentioned some of these. Looking ahead, one can visualize the benefits from fast reactors in their ability to sustain safe conditions by the thermosyphon effects of the sodium coolant, and by the way the fast reactors have been successfully tested in the last decade. The added advantage of breeding would be welcome in a new phase in which even the resources for nuclear power will get depleted. Thorium utilization is another area that holds promise as a way of utilizing an abundant source of nuclear material in the next couple of centuries. One can therefore conclude that in the area of fission reactor design, one has not exhausted all aspects of science and new technology to make the reactor a reliable, safe and abundant source of electrical energy for the future. However, it requires a fresh input and momentum, which can be provided only by a country that has the need for rapid growth in electrical power, a sustainable scientific manpower, and conviction that it is able to overcome obstacles and establish a new technology. India is ideally suited for this purpose.

Data for several countries (see table) on generation of electricity through nuclear fission reactors and the share of nuclear power in the national grid show that, on an average, 17% of electricity produced throughout the world is from nuclear power reactors. In France, the proportion is as high as 75%. One must recognize the reality that nuclear power has already come to stay as a major source of power and has proved its viability as a safe and ecologically benign source. Perceptions in the public mind might differ for reasons other than scientific; but the fact remains that further progress can be made in attaining greater confidence in safety.

The earth we live in has a concentration of radioactive materials that emit radiation, and we are also bombarded with radiation from outside, with the result that no place on earth is free from radiation. Figure 12 shows the map of India showing the distribution of the background radiation field. One can note the order-of-magnitude variation in the radiation

Nuclear capacity and share of nuclear power in total electricity generation

Country	Nuclear capacity MWe (net)	Share of nuclear electricity in 1989 (%)
France	52,588	74.6
Belgium	5480	60.8
Republic of Korea	7170	50.2
Hungary	1645	49.8
Sweden	9693	45.1
Switzerland	2952	41.6
Spain	7519	38.4
Finland	2310	35.4
Taiwan	4924	35.2
FRG	22,716	34.3
Bulgaria	3538	32.9
Japan	29,320	27.8
Czechoslovakia	3264	27.6
United Kingdom	12,428	21.7
United States	97,606	19.1
Canada	12,185	15.6
USSR	33,060	12.3
Argentina	935	11.4
GDR	1694	10.9
South Africa	1842	7.4
Yugoslavia	632	5.9
Netherlands	508	5.4
India	1374	1.6
Brazil	626	0.7
Pakistan	125	0.2
Italy	1120	—
Mexico	654	—
Total world	317,908	17.0

Data as on 31 December 1989
Source: IAEA

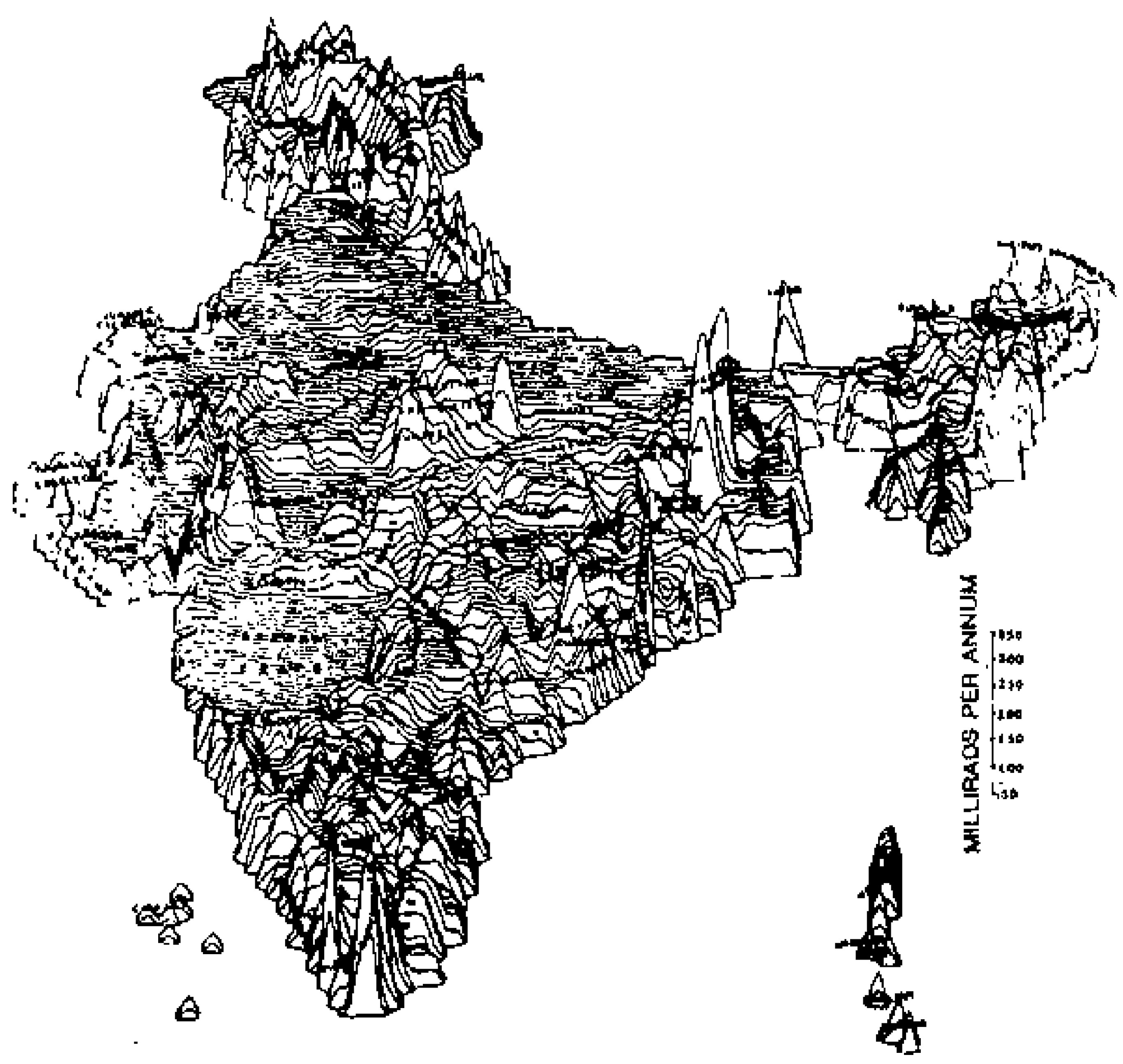

Figure 12. Terrestrial radiation map of India.

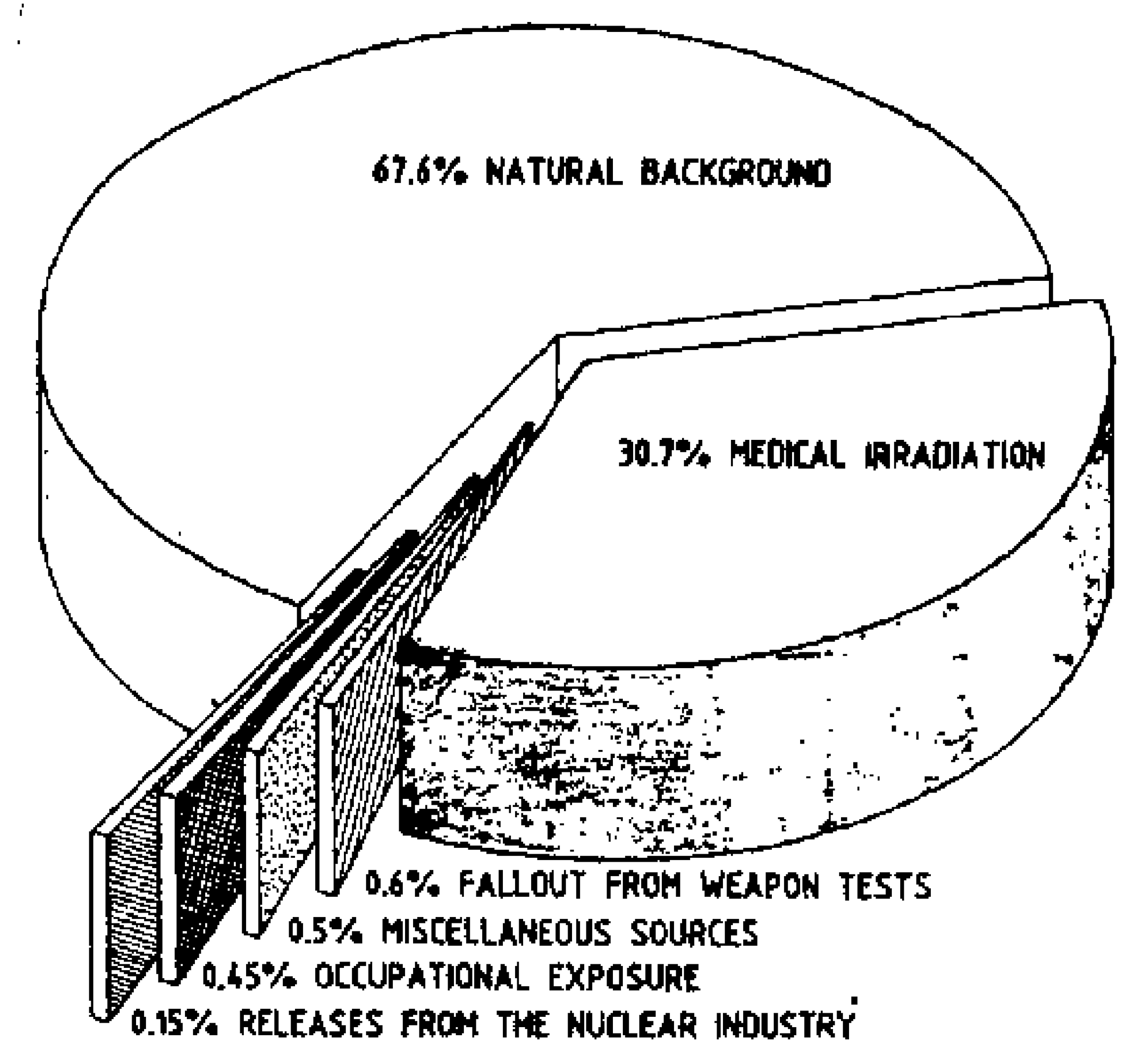

Figure 13. Radiation exposure from various sources.

field. Figure 13 shows the contribution to the radiation dose to the average population from all sources, including medical diagnosis, various industrial activities, fallout from nuclear weapon tests, and nuclear power stations. As can be seen from the figure, the contribution from nuclear power plants is insignificant compared to the others.

Today, high-speed trains do not depend on a signalling system based on manually controlled signals but on more sophisticated, fast-acting signalling systems. Increasing the power of the engine and strengthening the track alone cannot usher in high-speed trains. Similarly, when one builds larger-capacity reactors, many changes are required in order to ensure the same degree of safety. This is what is plaguing the nuclear industry today. Therefore a slower pace of enhancing the power capability of individual reactors is more acceptable and will ensure an organic growth in our nuclear power programme.

Fission power is not the only source of nuclear energy. In nature, the Sun and the stars demonstrate that fusion of light nuclei is indeed feasible and must be a source of abundant energy. Attempts have been made in the last four decades to establish fusion energy in the laboratory. In more recent times, fusion reaction between deuterium atoms in the solid state, which is termed 'cold fusion', has indeed been demonstrated. No doubt the scale of energy produced in this is still very limited. But the discovery is as important as the discovery of nuclear fission, for even the latter did not demonstrate the feasibility of abundant power until Fermi invented the nuclear reactor and proved that a self-sustained chain reaction can be maintained. Similarly, it takes some time before scientists discover the optimum conditions for sustaining cold fusion in metallic lattices and configure an arrangement by which energy can be produced in a sustained manner. Perhaps it will turn out to be much simpler than the problems one has faced in the hot fusion field.