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A Guidebook to Nuclear Reactors

Anthony V. Nero, Jr.

May 1976

Lawrence Berkeley Laboratory University of California/Berkeley

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A GUIDEBOOK TO NUCLEAR REACTORS*

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May 1976

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PREFACE

The purpose of this guidebook is to provide an introduction to nuclear reactors that is more direct and descriptive than a textbook on reactor physics or nuclear engineering. For this reason, the focal point of the book is the second chapter, which gives details on a number of reactors as they have recently been available. The first chapter includes introductory material on the basic physical and design features of reactors and on their resource utilization and economic/environmental interaction. The third chapter discusses the implications of limited uranium resources in terms of more advanced reactor types, some of which are improvements of those of the previous chapter; others are distinct types. The appendices give a more detailed discussion of topics which could not appropriately be discussed at length in the first chapter. In spite of this auxiliary material, this work cannot treat the physics and engineering of nuclear reactors at the depth found in numerous textbooks that are currently available. It is still hoped that the discussion and description provided herein can provide a "feel" for the various reactor types, and for some of the fundamental concepts underlying them, that is useful to individuals who are not specialists in some field of nuclear engineering. It is further hoped that the necessarily simplified treatment still manages to be accurate. The reader who requires a fuller treatment is referred to several useful and more complete references given at the end of each chapter. Anyone desiring a working, rather than passing, knowledge of reactor design should depend on more complete treatments than this guidebook.

The reason for assembling this material has been a frustrating inability to find a convenient source of information on reactors as they exist. There are of course a number of good treatments of the principles underlying reactor design, but they remain curiously isolated from the commercial practice. Since commercial reactors are highly relevant from the point of view of many people--from aspiring nuclear engineers to active environmentalists--a publication emphasizing reactors, rather than their physics or engineering, would seem useful. As it turns out, some of the physical principles are discussed here, but primarily as background for understanding the general structure of a nuclear reactor and the features which distinguish one reactor type from another. We do not intend to explain in detail how the particular parameters of a given reactor are calculated, but do hope to describe quite

graphically what solution was chosen. Valuable as a generalized schematic diagram might be, it does not serve to portray a reactor as it really is. Accordingly, the reader will find a proliferation of figures and tables, largely from government and industry publications, serving to introduce and compare various reactors, lending this work to use as a source for generic descriptions of reactor types. Hence the title, a guidebook to nuclear reactors.

This report was prepared in large part while the author was assistant professor of physics at Princeton University. Review and editing of the present draft was completed while a member of the Energy and Environment Division of the Lawrence Berkeley Laboratory. The author is grateful to various individuals who encouraged this work, and expresses special thanks to R. Budnitz, H. Hamester, W. Loewenstein, A. Sesonske, F. von Hippel, and G. Yadigaroglu, who generously made comments on portions of the manuscript. Further comment will be welcomed and should be addressed to the author at Building 50, Room 208, Lawrence Berkeley Laboratory, Berkeley, CA 94720, telephone 415-843-2740, Ext. 6377. Responsibility for any errors lies exclusively with the author.

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CHAPTER 1. GENERAL INTRODUCTION TO REACTORS

1.1. Physical and Design Features1.1.1. The nuclear chain reaction.

The most basic fact of a nuclear reactor is the release of a large amount of energy (about 200 MeV) from each fission that occurs in the reactor's core. This is enough that if one pound of uranium 235 fissions completely, it releases energy equivalent to about 6000 barrels of oil or 1000 tons of high quality coal. (The amount of energy available from a typical chemical reaction is measured in eV rather than MeV or million electron volts.[†])

The number of fissions occurring in a reactor is a good standard by which to measure energy release, since it is nuclear fission that provides the backbone of the nuclear "chain reaction". These fissions are induced principally by absorption of a neutron by a "fissile"^{*} nucleus. This absorption results, a large percentage of the time, in the breakup of the nucleus into two major fission fragments (which are actually nuclei such as strontium or cesium) plus lesser "pieces" such as neutrons and gamma rays. The neutrons produced can cause subsequent fissions, thus establishing a chain reaction. As suggested by Fig. 1-1, once a reactor is up to power, we only require that--on the average--one of the neutrons produced per fission cause another fission. (This condition is called "criticality".) Since any average fission event produces two or three neutrons, there are additional neutrons available which are "captured" in a way that does not induce fission--either by the fuel itself, or by the reactor structure or coolant--or are lost by leakage from the core.

The probability that one or another type of interaction (between, say, a neutron and a nucleus) will occur is usually measured in terms of the "cross section" for that particular interaction (see Appendix A). This probability (and the associated cross section) depends on the particle types involved and their state of motion with respect to one another. To cite an example, a slow neutron passing through a region containing fissile nuclei has a higher probability of inducing fission if the target nuclei are plutonium 239 than if they are uranium 235 (see Table B-3). The converse is true if the

[†]See table of "Miscellaneous Units, ...", prior to appendices.

^{*}One which can be caused to fission by a slow neutron (see below and Appendices A and B).

neutron has certain higher energies. (The cross section for interaction with either of these nuclei is reduced at higher neutron energies.)

The fact that at low neutron energy the fission cross section rises accounts for a basic feature of power reactors in this country--that is, that the fuel is surrounded by moderating material which slows down neutrons rapidly to improve their chances of inducing fission and thus continuing the chain reaction. In a typical U. S. reactor, the moderator is ordinary water, which also serves as the coolant (see next section). Other choices of moderator are possible and, in some respects, preferable. For example, water (H_2O) contains hydrogen, which can "capture" a neutron, forming deuterium (and, incidentally giving off a gamma ray γ , which has a small amount of energy compared with that given off by a fission event). This loss of neutrons is not as severe if carbon or "heavy" water (in which the ordinary hydrogen is replaced by deuterium) is chosen as the moderator (see Appendix A). On the other hand, light water is both plentiful and dual purpose (it cools the core).

Neutrons are also captured (i.e., without fission) by the fuel material itself. This is not as unfortunate as it might seem; it can actually be turned to an advantage, because capture of a neutron by uranium 238 leads to the following series of reactions:

(See Appendix B for more details.) $^{239}_{94}\text{Pu}$ is itself fissile. Further it is produced from $^{238}_{92}\text{U}$ by the capture of one neutron. In principle it is possible to build a reactor that produces more fissile material than it destroys.

---

* Gamma rays are one example of electromagnetic radiation; light is another. A nucleus may be represented by a chemical symbol, such as U for uranium and Pu for plutonium, preceded by the "mass number", the total number of neutrons and protons in the nucleus. The chemical element is determined by the number of protons. Beta (β) decay amounts to the change of a neutron into a proton or vice versa, accompanied ordinarily by emission of an electron (either negatively or positively charged).

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Fig. 1-1. A SCHEMATIC NEUTRON ECONOMY, showing some of the mechanisms for absorption of fission-produced neutrons.

So the only productive ends of neutrons are in a fission event (yielding subsequent neutrons and a large amount of energy) and in a conversion event (turning a "fertile" nucleus such as ^{238}U into a fissile nucleus, one which can be induced to fission by slow neutrons). Various materials may be characterized by a number of properties (and this list is far from exhaustive):

- the fission cross section (both at thermal^{**} and higher energies),
- the number of neutrons resulting from fission events (this will vary from one event to another, so that the number, like a cross section, is actually an average),
- the capture to fission ratio (the ratio of the capture cross section, wherein a neutron is absorbed and a gamma ray is given off, to the fission cross section),

etc. Appendix B contains additional information on nuclear materials.

Others ends to neutrons than fission or conversion may be regarded as a loss. These include capture by moderator or coolant, reactor structural material, ^{235}U , and the products of fission (such as xenon 135). A careful reactor design minimizes these losses to the extent compatible with other requirements. Among the more notable of the latter is the need to provide enough moderator to slow the neutrons down. (But an increase in the amount of water or other moderator means greater losses due to capture.) Similarly, enough coolant must be provided to keep the core at safe operating temperature.

The actual design of a reactor requires careful treatment of two basic interacting quantities to which we have alluded. These quantities are the neutron and heat fluxes, which are embedded in a common medium, the reactor core. In general a flux describes the flow of some material or other measurable, such as thermal energy, and the quantities of direct interest in a reactor are the neutrons which permeate the core and the heat which is produced, primarily as a result of fission induced by these neutrons. It is clear that there is a strong connection between these quantities. Not only does the reaction rate affect the amount of heat that is generated, but the temperature, whose value is directly controlled by this heat and the manner

^{**}

The energy that neutrons have after they have been permitted to collide a large number of times with their surroundings, which are at some particular temperature. The molecules constituting the surroundings will also have this same average energy (see Appendix A).

in which it flows, affects the reaction rate. (See, for example, the Doppler effect discussed in Appendix D.)

Implicit in our discussion above is the fact that a neutron "born" at one point in space and time will eventually be absorbed at another point. The equations which describe in some average way the manner in which neutrons move from their point of appearance to their point of absorption contain information on all the ways in which neutrons can be created and destroyed, and on the various interactions they can have in between, such as collisions which "moderate" their energy. Moreover, for many purposes it is not sufficient for these equations to describe the materials which make up the core (since these are the materials with which the neutrons interact) in some average way; the equations must actually describe the spatial distribution of these materials and of the neutrons and thermal energy which exist in the materials. It is particularly clear that this spatial description must be employed at the boundary of a reactor core, since any neutrons which pass beyond this boundary, and are not returned by some sort of reflector, are lost to the system. In rough terms, this loss depends on the surface area of the reactor core, whereas the total number of neutrons being produced depends on the volume of the core. Therefore, because the ratio of surface area to volume decreases as the scale of the core increases, the relative importance of boundary losses decreases. Because of this, for any given average core composition, there will be a minimum size reactor for which criticality can be achieved, assuming that the composition is such that criticality is possible at all.

Spatial variations are important not only at the boundary, but also within the core, since the core does not typically consist of a homogeneous mixture of fuel material, structural components, and coolant. As a result, some fuel elements or portions of fuel elements may run hotter both in reaction rate and temperature than others. The spatial distribution of fissions or, alternatively, heat production is an important design consideration, affecting not only fuel burnup, but also the ability of the cooling system to cool the core adequately.

1.1.2. Core cooling and energy conversion.

The reactor cooling fluid serves a dual purpose. Its most urgent function is to remove from the core the heat that results when the energy released in nuclear reactions is transformed by collisions into random molecular motion (heat). The coolant then transfers this heat for use outside the core, typically for production of electricity. The basic elements of a nuclear generating system are indicated in Fig. 1-2. The designer provides for a nuclear core in a container through which a cooling fluid is pumped. This fluid may be used directly, or it may heat a secondary fluid, to drive a turbogenerator. In all present commercial systems, the final fluid is vaporized water, i.e., steam, although this is not the only possibility. In the process of using the heat energy contained in this fluid to provide the mechanical force driving the turbine, most of the heat is rejected at a lower temperature to a waste heat sink, typically a body of water or a cooling tower. Such rejection of heat to a low-temperature reservoir is necessary for any process which converts heat to mechanical work. (See Section 1.2 and Appendix C.)

The effectiveness of the coolant in preventing core overheating is directly related to the particular core design. In almost every reactor design in use or under consideration, the fuel material is in solid form, normally contained in metal cans around which coolant flows. (If not in cans, the fuel is maintained in some sort of matrix which may also serve a function other than structural, such as moderation. The only exception of note is the molten salt breeder reactor, which would have a liquid core.) Most of the energy produced in the reactor is deposited in the fuel material itself by fission fragments, since these fragments lose all their energy over a very small distance. The neutrons themselves may deposit much of their energy in the moderator (in the case of thermal reactors), but they have relatively little energy to begin with (a few MeV as opposed to roughly 85 MeV per fission fragment). Similar comments apply to gamma rays. As a result, most of the heat from the reactor must flow from the fuel "pin" or pellet to the surrounding material. Portions of the fuel will be substantially hotter than the nearby cooling fluid and one of the prime design considerations is that heat transfer be adequate to prevent temperature increases which may violate the integrity of the fuel or its container. The parameters at the disposal of the designer are many, but the ones most obviously connected

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Fig. 1-2. A SCHEMATIC NUCLEAR POWER PLANT. The system shown has only a primary coolant loop, although many power plants employ a secondary loop in addition. Input and output for the plant are also indicated.

with this heat transfer requirement are the heat conductivity of the fuel and surrounding material (the container, the coolant and/or moderator) and the character of the boundary layer between the coolant and the structure it cools. This boundary layer depends on flow rate, pressure, surface roughness, etc. Complicated as it is, it is the most basic and perhaps most variable element in the cooling system.

This contrasts with the other end of the nuclear power generating system, the turbogenerator, where the use of steam to generate electricity is common to much of power engineering. In fact it has become conventional to divide the nuclear system into two parts (although this division may tend to obscure the fact that there are important interdependences between the two parts): the nuclear steam supply system (NSSS), which includes the reactor itself and any associated equipment (such as heat exchangers) necessary to produce steam, and the turbogenerator system that is driven by this steam. Design of the NSSS is a task for the nuclear engineer; details of the heat-to-electricity conversion are handled on the basis of standard engineering thermodynamics and generator design. It is true, however, that basic changes in turbogenerator design may be prompted by requirements of a particular nuclear system. For example, it may be possible to achieve unusually high thermal efficiencies using a high-temperature gas-cooled reactor in conjunction with a gas (rather than steam) turbine. Design of a gas turbine that can withstand very high temperatures is presently being vigorously pursued, although not just for its potential application to nuclear systems.

1.1.3. Control

Keeping a reactor operating at constant power level requires maintenance of a delicate balance between neutron production and absorption. Roughly speaking, the rate of energy production, i.e., the power level, of a given reactor is proportional to the number of neutrons available in the reactor, and it is, therefore, essential that this number remain constant. As a result, we require that--on the average--for every neutron absorbed or lost, precisely one be produced.

It is straightforward to see how this balance can be approximately maintained. Neutrons are produced predominantly by fission events. (The few that arise otherwise imply only a small correction to what follows.) Such an event, on the average, yields somewhat more than two neutrons. To

maintain a constant power level, exactly one of these, on the average, must induce another fission event. The remainder must be absorbed in other ways. As we have indicated previously, the situation is happiest if most of these are invested in conversion of fertile material to fissile. Neutrons are the currency of the reactor economy, and as few of them should be wasted as possible. However, some are lost through other processes, the most useful of which is absorption by control elements in the form of rods or other mechanisms.

Why is control necessary? First, because it is not possible to design a reactor so that the multiplication factor (the ratio of the number of neutrons in successive "generations"--see Appendix D) is exactly 1. Therefore extra fissile material is included in the fuel, and control elements rob the system of enough neutrons to maintain a balance. This extra fissile material is needed, in any case, because over a long period of time, enough fissile material would be destroyed to turn the reactor off. Control elements are needed because, in their absence, the extra fissile material at the beginning of operation would cause too great a portion of the neutrons to induce fission. As fissile material is "burned", control is deliberately withdrawn. This control is necessary for other reasons, the most prominent of which is the buildup of neutron poisons as a result of reactor operation--many of the nuclear reactions occurring in the core, in particular fission, produce nuclei which absorb neutrons with no useful result. Enough control must have existed when the reactor was turned on, so that it can be reduced as this buildup of absorptive capacity occurs. A particularly important and interesting example of the need for control due to fission product poisoning is the so-called "xenon transient" to which reactors are susceptible during changes in power level (see Appendix D).

Having established the need for control, we can ask whether such control can be implemented. Some doubt might arise from one basic consideration: the very short time between fission events occurring in a chain reaction. In a thermal reactor, the time needed for a neutron to slow down and induce a subsequent fission may be about 10^{-4} sec. Were this the actual time between all of the fissions, we would reason as follows: even if the multiplication factor k were only very slightly greater than 1, the power level would rise very rapidly. This is easily seen. For $k = 1.001$, in 1000 generations (taking a total of only 0.1 sec), the number of neutrons, and hence the power level, would rise by a factor of 1.001 to the thousandth

power, or 1.001^{1000} , which is 2.7. This would be intolerable; on the other hand, moving control rods in a length of time much smaller than 0.1 sec does not seem feasible. A solution is provided, it turns out, by the simple fact that not all neutrons generated in the reactor appear "promptly". (Most neutrons appear within times like 10^{-17} sec of the time of fission--which is what we mean by prompt.) A small portion of them, about 0.5% for uranium-fueled thermal reactors, result from the decay of fission products which have half-lives (the time it takes for half of a sample to be transformed as a result of radioactive decay) on the order of seconds. This is sufficiently greater than the 10^{-4} sec mentioned above that this small number of "delayed" neutrons substantially reduces the rate at which the neutron population changes. As a result, control can be achieved. (See Appendix D for more detail.)

One universal form of control is to provide a number of rods, loaded with neutron-absorbing elements such as boron or cadmium, which can be moved in and out of the core to select the portion of neutrons absorbed. In addition, reactors may have "burnable poisons" as part of the core--nuclei which absorb neutrons and, having done so, are neutralized. Moreover, poisons are commonly added to the cooling fluid. A number of other control mechanisms are sometimes available, particularly for use in emergencies, a need that is discussed in the next section.

1.1.4. Emergency features and containment.

For every reactor, "normal operation" implies, among other things, that enough cooling is supplied to maintain the structural integrity of the reactor system, particularly the core, and that radioactivity generated in the core does not escape into the general environment. Within these last two constraints, there is much latitude for a variety of designs. The same is true of the backup systems which come into play should some sort of abnormality occur. Even so, some general comments can be made. Before doing so, it is appropriate to point out that reactor features may be divided into two categories: intrinsic safety features, those which are inherent in the physical nature of the reactor concept being considered, and engineered safety features, systems which are added to the basic reactor concept. An example of "intrinsic" mechanisms is that, in a light water reactor, overheating of the coolant caused by an abnormal rise in the reaction rate tends to reduce the water density and thereby shut down the chain reaction

due to insufficient moderation of neutrons. Alternatively, the emergency shutdown-control-rod system may be regarded as an "engineered" safety feature. A specific reactor design will attempt to maximize the safety advantages offered by intrinsic physical characteristics and back them up by independent engineered systems.

In a previous section, we noted that the fuel pellet itself sustains the highest temperatures in the reactor. Under normal operation, this temperature is always well below the melting point of the fuel. For many reactors, however, it is the melting point of the fuel can or "cladding" that is more critical, since the cladding may melt at a lower temperature than the fuel and it is the cladding which provides structure to the core (especially in reactors where the core is cooled by a liquid). Should the melting point of the fuel can be approached, it may bend enough to disturb the coolant flow pattern, possibly accelerating, if only locally, the difficulty of cooling. Actual disintegration of the can would even be worse in this respect, and could also result in release of large amounts of radioactivity into the cooling fluid (see next paragraph). For these reasons, the cladding should maintain its integrity at temperatures considerably higher than the design operating temperature.

The cladding provides, in addition to structure, one of the important barriers against radioactive releases. Because of the large quantity of radioactivity generated as the reactor runs, it is important that it be contained at the reactor site. The most obvious means is to keep it in the fuel material itself. To a large extent this occurs automatically since most nuclear reactions occur in the fuel and the fuel is a solid, so that the reaction products lodge in the fuel material. However, some of the reaction products are gaseous (especially at typical operating temperatures), some portion of which will permeate the fuel material. The cladding (or other fuel coating) can prevent escape of such radioactivity from the fuel to the surroundings. To prevent release of gases, it is necessary for metal cladding (such as is used in light water reactors) to be welded shut, then leak-checked.

In spite of welding and checking, a small percentage of the fuel pins or rods leak gases into the reactor coolant. Moreover, some radioactivity is generated in the coolant itself. (For example, a small portion of the tritium--the mass 3 isotope of hydrogen--released from light water reactors is

formed by capture of neutrons by minor amounts of deuterium in the water. More significantly, neutrons may produce ^{16}N from the oxygen in the water coolant). Such radioactivity may be removed from the coolant by a cleaning system of some sort. The collected material may then be saved for later disposal or released in a controlled manner into the general environment.

Although a generic description of systems for dealing with abnormalities is much more difficult, a few examples would be useful. The most basic system is incorporation of "control" as discussed in the last section. In many reactors, there are both "control" and "shutdown" rods, the latter being specifically designed for rapid and failsafe shutdown of the chain reaction. The actual speed with which shutdown occurs (i.e., the rate at which the fission rate decreases) depends on the amount and rate of control added. A normal operational change in power level may occur over a period of hours. An emergency shutdown may take seconds, a significant time interval in some circumstances.

As mentioned before, there may also be recourse to dumping neutron absorber in other forms in the reactor. (In some cases, this technique is also used for routine control.) It is generally assumed that there are enough backup systems to guarantee shutdown when it is necessary.* Some abnormalities of themselves guarantee shutdown. The most notable of these is that, if the cooling water from a light-water reactor is lost, the reactor is reduced below criticality because neutrons are no longer moderated to the low energies where the fission cross section is higher. As a result, the chain reaction would come to a halt, even without the insertion of control rods.

We cannot pretend to outline all the circumstances that would warrant a shutdown (called a "scram" if conditions require a very rapid shutdown). Ordinarily, any abnormality (operation out of predetermined tolerance) in coolant temperatures or flows, fuel pin temperatures, neutron flux, etc., or any abnormality in the instrumentation checking these quantities, or any malfunction of a system that would be needed in an emergency, may provoke

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* This guarantee may not be unambiguous in cases such as the fast breeder reactor, where the possibility of recriticality must be considered. Reliable shutdown systems for breeders such as the LMFBR are in the process of development.

shutdown of the reactor. (As a result, most shutdowns are not due to malfunctions of the basic reactor system, but rather of support and monitoring systems.)

When the chain reaction is stopped by insertion of control, we are still not in a position to shut down other functions. Most importantly, the coolant system must continue to run, although a smaller capacity is needed than under full power operation, because the reaction products continue to generate heat through radioactive decay (see Fig. 1-3). For this reason, immediately after shutdown the power level is still about 6% of the level that was sustained before shutdown, and this 6% is enough to require continued cooling of the core, albeit at a lower cooling capacity.

It is, therefore, necessary that cooling be available under all conditions. In particular, even if the reason for the shutdown is loss of cooling, enough must remain or be supplied to prevent serious damage to the core. In the absence of cooling, portions of the core would actually melt, thus resulting from the operational point of view in destruction of the reactor and possibly leading to a serious release of radioactivity. Because of the consequences of failure, components in the cooling system may be duplicated and have independent sources of power. Moreover, in light-water reactors, there is an emergency cooling system that is largely independent of the main system. This philosophy of redundancy and independence is typical of any reactor design. It is particularly important in the shutdown and cooling systems. Even so, adequacy of these systems has been a matter of dispute.

To conclude this section, we should note that--to some extent--the possibility of breach of the reactor's primary system itself must be considered. This could come in the form of a break in a cooling line, or even in the reactor vessel. Emergency systems are provided to cope with the more credible of such events. Typically, the reactor vessel and primary cooling system will be enclosed by a containment building designed to withstand a significant overpressure. Inside this building will be systems to clean the internal atmosphere of radioactivity prior to any possible release to the external environment. (The most serious, but presumably least likely, event would be a core meltdown which breached the reactor vessel, then the containment building. The course of such an event is difficult to predict; some protection would be afforded by the filtering action of the materials underlying the building.)

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Fig. 1-3. DECAY-HEAT RATE AS A FUNCTION OF TIME AFTER SHUTDOWN of an LWR operating at 3077 megawatts core power (from WASH-1250).

Many of the systems mentioned above are examples of engineered safety features, features designed to limit the extent of any abnormality and to mitigate radioactive releases and damage to the reactor system from operation under both normal and abnormal conditions. Consideration of the possibility and probability of various events is an intimate part of reactor design. It is fair to say that the designer himself is typically satisfied that he has sufficiently provided for the safety of the public and of the reactor itself. Whether others are so satisfied is another matter.

1.2. Input/Output Characteristics of Reactors

1.2.1. The basic notion

An instructive point of view is to consider what goes in and out of reactors and of the nuclear power system as a whole. This will obviously depend to some extent on the particular type of reactor and on the manner in which the nuclear power system develops. Some general comments are still possible. Referring to the schematic representation of a reactor given in Fig. 1-2, we see that the basic input to the reactor is fissionable material and the basic output is heat, a portion of which is transformed by the turbogenerator into electricity. Other products of the transformation from nuclear potential energy to electricity are irradiated fuel, other radioactive discharges, and reject heat. A more general type of representation of the input/output characteristics of a nuclear reactor, in particular a light-water reactor, is shown in Fig. 1-4. The numbers given are input and output for operation of a plant at 1000 MWe power level for 1 year. This representation often turns out to be convenient since it can easily be incorporated into a flowsheet for an entire "fuel cycle" (see Appendix E).

In certain respects, all of the commercial thermal reactors now operating will give similar results. This is true because (1) the energy liberated from a fission event is largely independent of the type of nucleus fissioning, (2) the ratio of fissile material produced to that consumed (the conversion ratio) does not vary greatly among these reactors at the present time, and (3) the efficiency with which heat is converted to electricity is always between 30 and 40% for present reactors. There is a substantial potential for improvement of conversion ratios, as we shall see, and lesser potential for increases in thermal efficiency.

Figure 1-4 gives a basic description of what goes in and out of one type of reactor, but for many purposes this information is not enough since it

MATERIAL FLOW and ENVIRONMENTAL RELEASE
(one year of full-power operation)

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Fig. 1-4. MATERIAL FLOW AND ENVIRONMENTAL RELEASE for one year of full-power operation of a 1000 MWe light-water reactor power plant (adapted from "Comprehensive Standards . . .").

does not describe how the reactor fits into the nuclear power system--and specifically where the fuel comes from and how irradiated fuel is disposed of. In principle, some input/output characteristics can be treated on a plant (i.e., single reactor) basis, others on a fuel cycle basis, and some on a still broader basis. An example of the latter would be the question of the "net energy" available from a growing number of nuclear power plants (see end of this chapter). Unfortunately, the broader the basis for discussion gets, the more assumptions are needed.

1.2.2. The predominant reactor types

A brief introduction to the reactors that have been commercially available in North America is given in Table 1-1. Some of their structural, physical, and input/output characteristics are provided. More detailed descriptions are the subject of the next chapter.

Briefly, the predominant commercial reactor in the United States is the "light-water reactor", one in which ordinary water, in which the hydrogen is almost entirely the mass 1 isotope (H or ^1H), is the moderator and coolant. The light water reactor (LWR) is found in two types, the "pressurized-water reactor" (PWR), sold in the United States by Westinghouse, Combustion Engineering, and Babcock & Wilcox, and the "boiling water reactor" (BWR), sold by General Electric.* The BWR is somewhat simpler conceptually than the PWR in that steam is actually produced in the reactor vessel itself, and this steam drives the turbogenerators. In the PWR, the coolant is kept under greater pressure, so that a higher temperature is needed for formation of steam in the primary coolant circuit. The coolant is maintained below this temperature, preventing boiling, and the water runs through "steam generators" in which the heat absorbed from the core is used to raise steam in a separate secondary fluid, also water. The primary water is circulated back to the reactor, and the secondary fluid, now steam, drives the turbines.

Atomic Energy of Canada, Ltd., has developed a reactor which, as presently available, uses "heavy" water as the moderator and coolant. As noted earlier, the deuterium in heavy water captures substantially fewer

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* These manufacturers are the principal makers of a broad range of electrical generating equipment, including turbogenerators, boilers, and other components of large power systems. Westinghouse has also been heavily involved in the program for nuclear propulsion of Navy vessels, which program accounts for most United States nuclear reactors.

Table 1-1. Characteristics of power reactors (1000 MWe capacity).

	PWR Pressurized Water Reactor	BWR Boiling Water Reactor	CANDU Canadian Deuterium- Uranium	HTGR High-Temperature Gas-Cooled Reactor
Coolant	Ordinary Water	Ordinary Water	Heavy Water (D ₂ O)	Helium Gas
Moderator	Ordinary Water	Ordinary Water	Heavy Water**	Graphite
% ²³⁵ U Enrichment	2-4	2-4	0.7 (Unenriched)	93 (Initial Load)
Fertile Nuclide	²³⁸ U	²³⁸ U	²³⁸ U	²³² Th
Lifetime Uranium* Requirements (Tons of Natural U ₃ O ₈)	4100 (3500 or less)	4020 (3500 or less)	4160 (1440)	2980 (1920)
Thermal Efficiency	32	33	28	39
Approx. Once- Through External Cooling-Water Requirements (gal/ min with 15°F temp. rise)	1,000,000	960,000	1,220,000	740,000
Core Type	Fuel Rods (Bundled into Assemblies)	Fuel Rod Assemblies	Fuel Rod Assemblies (Individually Pressurized)	Fuel Particles Dispersed in Graphite Blocks
Coolant Pressure (lb/in. ²)	2250	1020	1490	700
and Temperature (°F) at Exit From Core	620	545	590	1370

* As run according to present practice. Number in parentheses estimates requirements if reactor is optimized to use as little as possible on a thorium cycle. In either case, a complete fuel cycle is assumed (except for the 4160 CANDU result). All these results are abstracted from Table 3-2 except for the BWR and PWR thorium results, which are taken from EPRI Report NP-2.

** The D₂O moderator system is separate and is at essentially atmospheric pressure (15 lb/in.²).

neutrons than the hydrogen of light water, so the CANDU (for "Canadian deuterium-uranium" reactor) is able to use natural uranium, containing only 0.7% ^{235}U , as a fuel. (This contrasts with LWR's, which require that the uranium fuel be "enriched" in ^{235}U --see Table 1-1 and Appendix E.) Furthermore, the moderator and coolant systems are actually separate, the moderator filling a large low-pressure vessel, through which are built a large number of individually pressurized and cooled fuel channels. As a result, the reactor vessel has a noticeably different structure than that of an LWR. Heated heavy water from the pressurized channels runs through steam generators, producing ordinary steam for driving the turbogenerator.

We also show parameters for the high-temperature, gas-cooled reactor (HTGR) marketed, until recently, by General Atomic. Instead of fuel pins bundled into assemblies which are then immersed in water (a feature common to the above reactors), the HTGR has a core consisting primarily of stacked blocks of carbon. Small regions of each block contain fuel particles,* individually coated with hard ceramic. The carbon moderates neutrons, and the core is cooled by helium gas which flows through and around the carbon blocks. The heated helium raises steam, which powers the turbines, as in water-cooled designs. The core, plus the steam generators and coolant pumps are all contained in a single concrete vessel, which causes this reactor to have a very different layout than those above.

1.2.3. Operational flexibility.

A major difficulty in trying to characterize reactors is that their input/output parameters will change substantially, depending on how they are built and operated. Perhaps the most fundamental of such parameters is the amount of uranium required to produce a specified output of electrical energy. This required feed will clearly depend on whether irradiated fuel is being reprocessed to recover the remaining fissile material. The feed is also directly influenced by the "conversion ratio" at which the reactor runs, i.e., how much fuel is converted to fissile material as the fissile material already there is "burned". Were this ratio 1, the reactor could run indefinitely on one fuel load, except for loss of fertile material

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* This structure of carbon blocks and clumps of fuel is reminiscent of Fermi's 200 watt reactor of 1942.

(due to conversion), buildup of fission products, and other effects. (With reprocessing, these difficulties can be side-stepped.) The conversion ratio may be strongly influenced by many design and operational factors. One example is the frequency with which a portion of the core is replaced with fresh fuel: replacing one-sixth of the core every 6 months, instead of one-third yearly, would reduce the amount of control needed.* Any improvement in the thermal efficiency of the system (the percentage of heat energy that is converted to electrical energy) reduces the required amount of uranium feed. This efficiency is often fixed by materials limitations, i.e., their ability to withstand the higher temperatures that would give better efficiency (see Appendix C). However, a less-than-optimum efficiency may be dictated by external constraints, such as a requirement that the low-temperature heat sink be a dry-cooling tower.

Choosing how any reactor is actually operated depends on a delicate balance of cost and technical feasibility. A reactor vendor may, for example, design the fuel (and thus specify the conversion ratio) to obtain a balance of short-term and long-term costs that best suits the financial situation of the utility purchasing the reactor. Raising the conversion ratio, thereby decreasing the amount of uranium required for the lifetime of the reactor, may require increasing the amount of uranium (and hence money) initially invested in the reactor. The parameters given in Table 1-1 (with the exception of the lifetime uranium requirements given in parentheses) are typical of reactor specifications as they have been. For a given reactor type, variations around these numbers have not been large, except that greater uranium feed has been required because reprocessing has generally not been available. In the next chapter, we will discuss these particular reactor types in more detail.

1.2.4. Energy investment in a reactor system.

In addition to the material and energy flows through a nuclear system, one might consider the investment in the reactor itself. There are enough similarities among the various reactor types that we would not expect the amounts of concrete, steel, enrichment effort, etc., to change radically

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* With less control, fewer neutrons are absorbed without benefit, and the conversion ratio rises.

from one reactor to another. However, these amounts may differ substantially from the corresponding results for other types of energy systems, particularly fossil-fueled electrical generating plants. There has lately been some controversy over the "net energy" from a nuclear power plant, the energy obtained from the system less the energy invested in it. This has arisen largely because of the substantial amount of energy needed to run the plants that enrich the uranium fuel. This amount is roughly 4% (at present conversion ratios) of the energy that will ultimately be derived from the fuel. Moreover, a nuclear plant is a large device itself, the construction of which takes perhaps 1% of the total energy that will be derived from the plant during its lifetime. These are large amounts of energy, but not enough to create difficulty unless the nuclear program were to double in size yearly (an unlikely prospect). And indeed, these results seem little larger than the reprocessing and construction costs required of other energy systems. (For example, about 10% of each barrel of oil is used to run the refinery processing the oil.)

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CHAPTER 2. COMMERCIAL NUCLEAR REACTORS

In this chapter we will discuss and, with the help of numerous figures, show detail on the water- and gas-cooled thermal reactors mentioned briefly in the last chapter: the PWR, BWR, CANDU, and HTGR. These are often called "converter" reactors, a reference to the fact that, while running, they convert fertile material to fissile, which can then contribute to the chain reaction in that same or (assuming fuel reprocessing) another reactor. A reactor whose conversion ratio exceeded 1, indicating that it produced more fissile than it consumed, would be called a "breeder"-see the next chapter.

These four reactors all depend for their initial operation on fission of ^{235}U . As indicated in Table 1-1, the percentage enrichment varies considerably. However, in each case, the uranium is still present in the same chemical form, uranium dioxide (UO_2).^{*} Use of this compound (which in solid form is a ceramic), rather than metallic uranium, reduces the density of uranium atoms in the fuel and also reduces the heat conductivity of the fuel, both disadvantages from the point of view of reactors. However, the dioxide has the advantage of greater stability against radiation damage and chemical corrosion; it also retains a large portion of the gaseous fission products which in other environments might more likely escape. Its high melting point (2800°C) allows high-temperature operation, overcoming to some extent its low heat conductivity. The other nuclear materials with possible importance in reactors are plutonium and thorium, both of which are expected to be utilized in the form of the dioxide.^{*}

For water-cooled reactors, the UO_2 is pressed into "pellets", cylinders about 0.5 in. or less in diameter and about 0.6 in. in height. These pellets are sintered (heated to high temperature), ground to the proper dimensions, then sealed, along with a helium atmosphere, in a cladding material (see Fig. 2-1). This cladding is typically zircaloy, an alloy of zirconium. The fuel pin (or rod) is welded shut to prevent escape of radioactivity to the coolant. These pins, each more than 12 ft long for LWR's, are assembled into bundles, the operational unit for handling, refueling, etc.

Should plutonium be recycled into water reactors, it would be handled in much the same way. (In the United States, it is proposed for LWR's that

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^{*} See, however, the section on HTGR's.

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Fig. 2-1. CUTAWAY VIEW OF OXIDE FUEL FOR COMMERCIAL LWR POWER PLANT (from WASH-1250).

the PuO_2 be finely mixed with the UO_2 before a fuel pellet is formed.)

As discussed below, HTGR's have a substantially different form of fuel than water-cooled reactors.

2.1. Pressurized-Water Reactor

The core of a pressurized-water reactor (PWR) consists of a large number of square fuel assemblies or bundles. Each of these (see Fig. 2-2) contains about 200 fuel pins closely held together in a matrix with no outer sheath. A full-sized (i.e., 1000 MWe) PWR may contain nearly 40,000 fuel pins, containing almost 200,000 lb of UO_2 . The power generation density in the core is about 98 kW/liter. Some of the fuel assemblies also have provision for passage of control rods through the bundle. These rods are manipulated by drives at the top of the reactor (Fig. 2-3). Additional control is available by addition of neutron absorbers (such as boric acid) to the coolant. The reactor vessel itself may be 40 ft high by 14 ft in diameter, made of steel 8 in. or more thick. The top head is removable for refueling.

The coolant enters the reactor vessel near the top of the core and, constrained by a "core barrel" between the vessel and the core, flows to the bottom of the vessel, then up through the core itself and out to the steam generators from which it is recirculated by large pumps. Maintaining the pressure at about 2250 psi prevents the formation of steam in this "primary" system. Instead, steam is raised in a "secondary" system by allowing heat to flow from the high pressure primary coolant to the lower pressure secondary fluid, through the walls of large numbers of tubes through which the primary coolant circulates in the steam generator. After passing through separators to remove water droplets, the steam leaves the steam generator for use in producing electricity. After condensation, it returns to the steam generator (see schematic, Fig. 2-4). A number of parameters for a PWR system are summarized in Table 2-1.

A large PWR has four external circuits indicated schematically in Fig. 2-5, each with its own steam generator and pump. Since maintenance of the pressure near the design value is crucial (to avoid the formation of steam, on the one hand, and rupture of the primary circuit, on the other), a PWR system also includes a "pressurizer", as shown in the figure, connected to the "hot" leg of one of the steam generator circuits. This device is filled partially with water and partially with steam; it initiates condensation or vaporization as needed to keep the pressure within specified operating limits.

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Fig. 2-2. PWR FUEL ASSEMBLY (from WASH-1250).

XBL 766-8011

Fig. 2-3. CUTAWAY VIEW OF INTERNALS OF A TYPICAL PWR REACTOR (from WASH-1250).

XBL 766-8010

Fig. 2-4. SCHEMATIC FLOW DIAGRAM OF PWR HEAT REMOVAL
(From WASH-1250).

Table 2-1. PWR (representative characteristics).

core thermal power	3411 MW(th)
plant efficiency	32%
plant electrical output	1100 MW(e)
core diameter	134 in.
core (or fuel rod) active length	144 in.
core weight	276,000 lb
core power density	98 kW/liter
cladding material	zircaloy-4
cladding diameter (OD)	0.422 in.
cladding thickness	0.024 in.
fuel material	UO ₂
pellet diameter	0.37 in.
pellet height	0.6 in.
assembly array	15x15 rods, open structure
number of assemblies	193
total number of fuel rods	39,372
control rod type	B4C or Ag-In-Cd in cylindrical rod inserted in assembly
number of control rod assemblies	60 (may vary considerably)
number of control rods per control assembly	20 (may vary considerably)
total amount of fuel (UO ₂)	217,000 lb
fuel power density	38 MW/metric ton of fuel
fuel/coolant ratio	1/4.1
coolant	water (liquid phase)
total coolant flow rate	136x10 ⁶ lb/hr
core coolant velocity	15.5 ft/sec
coolant pressure	2250 psi
coolant temperature (inlet at full power)	552°F
coolant temperature (outlet at full power)	617°F
nominal clad temperature	657°F
nominal fuel central temperature	4140°F
radial peaking factor (variation in power density)	1.5
axial peaking factor	1.7
design fuel burnup	32,000 MWd/metric ton of fuel; varies
fresh fuel assay	3.2% ²³⁵ U (less in initial load)
spent fuel assay (design)	0.9% ²³⁵ U
refueling sequence	one-third of the fuel per year

XBL 766-8002

Fig. 2-5. SCHEMATIC ARRANGEMENT OF PWR NSSS (from WASH-1250).

All of these components of the "primary" system--the reactor vessel, steam generators, pumps, and pressurizer--are in the containment building (see Fig. 2-6). This structure is steel-lined, reinforced concrete, designed to withstand the overpressures expected if all the primary coolant were released in an accident. Sprays or other means are available for condensing steam (thereby reducing pressure) and for removing any radioactive material released into the containment atmosphere.

Various features are available for limitation of abnormalities. These include introduction of control to turn off the chain reaction, and systems for continuation of core cooling thereafter. In case much of the primary coolant is lost through a pipe-break or other leak, one of a number of redundant emergency core cooling systems (ECCS) becomes operative. The first is an accumulator which automatically injects water into the system (at or near the reactor vessel) if the pressure becomes low enough--as it would in a major loss of coolant accident (LOCA)--to cause check valves to open. In addition, there are independently-powered systems for actively pumping stored water into the primary system. These systems rely on redundancy of components and flow paths to assure their availability in emergencies. This is also true of other devices. For example, each of the four primary coolant pumps has a capacity large enough to provide by itself for removal of decay heat after shutdown of the reactor.

The actual behavior of a reactor system under emergency conditions is complicated and not completely understood. Our knowledge of transient reactor response relies on computer simulations and on associated "separate effects" tests of basic reactor components and scaled-down "system" tests. The computer "codes" attempt to model the extremely complex transient behavior of a reactor incorporating, for example, data from component testing. System tests are intended to check the ability of these codes to predict actual accident conditions. An example of results from these codes is given in Fig. 2-7, where calculated clad temperature behavior is shown after a postulated loss of coolant using two models, one making a best estimate of temperature, the other making assumptions that are conservative, resulting in higher temperatures.

XBL 766-8009

Fig. 2-6. PWR PRIMARY COOLANT SYSTEM AND CONTAINMENT (from WASH-1250).

COMPARISON OF PWR FUEL CLAD TEMPERATURES DURING
A POSTULATED LOSS-OF-COOLANT EVENT

XBL 766-8008

Fig. 2-7. PWR FUEL CLAD TEMPERATURES DURING POSTULATED LOSS-OF-COOLANT ACCIDENT as calculated using "realistic" and "conservative" assumption models (from WASH-1250).

2.2. Boiling-Water Reactor

The BWR is conceptually different than the PWR (or CANDU) in that steam is actually allowed to form in the core. The pressure is maintained at about 1000 psi, at which pressure water boils at 545°F. Formation of steam changes the density of the water and thus of the moderator. As a result, neutrons will be better moderated (and will be more likely to induce fission) in regions where there is less steam, such as at the bottom of the reactor core, where the water enters.

The other important result of allowing the coolant to boil is that no secondary system is necessary for producing steam to drive the turbogenerators. As indicated in the schematic drawing of a BWR vessel (Fig. 2-8), the fluid exiting from the core passes through steam separators just above the core. About 13%, by weight, of this fluid is steam, the remainder being liquid that is recirculated. The steam from the separators exits the reactor vessel after passing through dryers (see Fig. 2-9) which remove most of the remaining liquid. After passing through the turbogenerator, the steam is condensed and returned to the reactor as feedwater, thus completing the cycle. As a result, the reactor vessel (and contents) is basically the entire steam supply system, in contrast to the PWR NSSS shown in Fig. 2-5.

The core of a BWR is not much different than that of a PWR, except for its size. The fuel rods have a slightly larger diameter, and a typical assembly (Fig. 2-10) is a square array of 49 (or 64) rods in a channel that is only open at the top and bottom. A large BWR has around 40,000 fuel rods, containing a total of about 200 tons of UO_2 .

Water is forced to circulate upward through the core by the action of jet pumps at the wall of the reactor vessel. Some water is actually withdrawn from the vessel by pumps which recirculate it for use in driving the jet pumps.

Aside from the recirculation rate (which affects the formation of steam and, therefore, the reactivity), the reactor is typically controlled by elements made of two plates which cross at a line along the space where the corners of four fuel assemblies met. By pushing the element up (using a rod which, in the case of a BWR, passes through the bottom of the reactor vessel), neutron absorbing plates are thus inserted between each pair of these four assemblies.

XBL 766-8007

Fig. 2-8. SCHEMATIC ARRANGEMENT OF BWR NSSS (from WASH-1250).

XBL 766-8006

Fig. 2-9. CUTAWAY VIEW OF INTERNALS OF TYPICAL BWR REACTOR
(from WASH-1250).

XBL 766-8005

Fig. 2-10. BWR FUEL ASSEMBLY (from WASH-1250).

The reactor vessel is larger than that of a PWR of comparable power, both because of lower power density (51 kW/liter) and because more equipment is contained in the vessel. A 1000 MWe BWR vessel is about 72 ft high and 21 ft in diameter. The walls are 6 to 7 in.-thick steel. Many BWR parameters are given in Table 2-2.

The reactor vessel and primary piping are contained in a steel pressure vessel (the "dry well" of Fig. 2-11) which constitutes the reactor's primary containment. This is connected through very large piping to a series of "downcomer" pipes which open into a pool of water in the large torus shown in the figure.* Should the water level in the reactor become too low, or under other abnormal conditions, valves close the steam lines which pass through the dry well to the turbogenerator. Any steam released from the reactor would force fluid down into the suppression pool, where steam would be condensed, thus relieving the pressure in the dry well. The latter is designed to withstand the transient pressures to which it would be subjected under accident conditions.

There are two types of ECCS available. As soon as the water level drops below a preset minimum, a high-pressure injection system driven by steam turbines is activated. Backing this up are low-pressure electrically-driven core-spray and coolant-injection systems, which would become operative after failure of the high-pressure system to cope and subsequent depressurization (through pressure relief valves) into the dry well and downcomer arrangement. The low-pressure systems are sized to handle the reactor decay heat without damage to the core.

The steel dry well, and the reinforced concrete structure immediately surrounding it, are enclosed by a secondary containment building (Fig. 2-12). Gas exhausting from this building passes through multiple filtration systems for trapping volatile radioactive species. In addition to an altered suppression pool arrangement, as mentioned above, more recent BWR systems have another leak-tight containment structure between the primary containment and the reactor building.

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* In more current models, a suppression pool is still used, but it does not take the form of the downcomer-torus arrangement.

Table 2-2. BWR (representative characteristics).

core thermal power	3293 MW(th)
plant efficiency	33%
plant electrical output	1100 MW(e)
core diameter	187 in.
core (or fuel rod) active length	144 in.
core weight (fuel assemblies)	500,000 lb
core power density	51 kW/liter
cladding material	zircaloy-2
cladding diameter (OD)	0.563 in.
cladding thickness	0.032 in.
fuel material	UO ₂
pellet diameter	0.487 in.
pellet height	0.5 in.
assembly array	7×7, with wall enclosing array
number of assemblies	764
total number of fuel rods	37,436
control rod type	"cruciform" control rods inserted from bottom between sets of four assemblies, supplemented in initial fuel load by temporary control curtains
number of control rods	185 (essentially one between each set of four assemblies)
total amount of fuel (UO ₂)	370,000 lb
fuel/coolant ratio	1/2.4 (cold)
coolant	water (two-phase)
total coolant flow rate	105×10 ⁶ lb/hr
coolant pressure	1020 psi
coolant temperature (steam system design)	562°F
feed water temperature	376°F
average coolant exit quality (% steam weight)	13%
average clad temperature	560°F
maximum fuel central temperature	4400°F
average volumetric fuel temperature	1210°F
axial peaking factor	1.5

Table 2-2. Continued

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design fuel burnup	27,500 MWd/metric ton of fuel
fresh fuel assay	2.7% ²³⁵ U
spent fuel assay	0.8% ²³⁵ U
refueling sequence	one-fourth of the fuel per year
vessel wall thickness	6-5/16 in.
vessel material	carbon steel, clad with minimum 1/8 in. stainless steel
vessel diameter	20 ft, 11 in.
vessel height	72 ft
vessel weight (including head)	1,500,000 lb

XBL 766-8004

Fig. 2-11. SCHEMATIC ARRANGEMENT OF BWR REACTOR AND PRIMARY CONTAINMENT (from WASH-1250).

XBL 766-8003

Fig. 2-12. BWR SECONDARY CONTAINMENT BUILDING (from WASH-1250).

2.3. Pressurized Heavy-Water Reactor (CANDU)

We will here describe the reactor currently available in Canada (and marketed elsewhere). The term "CANDU" applies specifically to the fact that this reactor is a deuterium-moderated uranium-fueled device, but its usage has broadened to an entire class of reactors being considered by the developer, Atomic Energy of Canada Limited (AECL), all of which have the same basic moderator and heat-transfer design that is depicted in Fig. 2-13. That figure actually describes the pressurized heavy-water design that is presently available. Instead of moderating neutrons with "light" water, containing hydrogen with an atomic mass of 1, this design uses "heavy" water (containing deuterium, mass 2), which absorbs fewer neutrons than light water does.

The fuel of a CANDU is very similar to that of an LWR in that fuel pins containing uranium dioxide pellets are arranged into bundles (Fig. 2-14) which can be inserted into the reactor fuel channels. Each of these channels is a pressure tube, two of which are shown schematically in Fig. 2-13, containing a single line of fuel bundles arranged end-to-end. The core actually consists of a large number of these tubes, all of which pass through a lattice of tubes which are part of a "Calandria" which contains the moderator. The pressurized coolant is pumped through the pressure tubes and boilers (steam generators) much like the coolant of a PWR, except that the CANDU coolant is sealed off from the moderating fluid, which is maintained at a much lower pressure and temperature than the coolant. (The lower pressure removes the necessity of fabricating a single large pressure vessel.) At the boilers, steam is raised in a light water system which drives the turbogenerators. Both the moderator and coolant fluids of current CANDU's are heavy water; a possible variant uses light water as the coolant (such as England's "steam generating heavy water reactor").

One result of the individual pressure tube structure of the CANDU is that the reactor can be refueled without shutdown. A single channel can be opened and partially refueled by two refueling machines which attach at both ends of the channel. The fact that the reactor is refueled only a little at a time amounts to continuous refueling, as opposed to the abrupt once-yearly operation in United States reactors. The latter schedule causes abrupt increases in the fissile load, thus increasing the neutron multiplication factor unless substantial amounts of control are used to rob the system of

XBL 766-8039

Fig. 2-13. CANDU SCHEMATIC FLOW DIAGRAM (from the AECL).

END VIEW

- 1 ZIRCALOY BEARING PADS
- 2 ZIRCALOY FUEL SHEATH
- 3 ZIRCALOY END SUPPORT PLATE
- 4 URANIUM DIOXIDE PELLETS
- 5 INTER ELEMENT SPACERS
- 6 PRESSURE TUBE

XBL 766-8038

Fig. 2-14. CANDU FUEL BUNDLE (from the AECL).

neutrons. This required control decreases as the amount of excess fissile load decreases. A reactor such as the CANDU does not need this much control (since the fissile load remains essentially constant), thus leaving more generated neutrons available for useful purposes such as conversion of fertile material to fissile. Decreased control thus improves conversion even beyond that afforded by use of heavy, rather than light, water as the moderator.

A basic feature of the CANDU, permitted by the use of a moderator that absorbs few neutrons, is use of natural uranium as the fuel. Uranium fed into the CANDU contains only 0.7% ^{235}U and this is reduced to about 0.2% by the time a bundle is removed. (The irradiated bundle also contains plutonium, as a result of neutron capture by ^{238}U .) This obviates the need for costly enrichment of uranium to the 2 to 4% ^{235}U content needed by LWR's. (However, an equivalent cost is production of heavy water by separation from ordinary water, of which 0.016% is heavy.) The lower fuel enrichment means that excess fissile material is not available in the fuel. As a result, the design must be based on a minimum of control and on an associated continuous refueling procedure, as just described.

The pressure maintained in the primary coolant system, again by a pressurizer as in a PWR, is ≈ 1500 psi, and the coolant is allowed to rise to 590°F , below the boiling point for that pressure. The moderator, on the other hand, is maintained at approximately atmospheric pressure, under a cover of helium gas. Operating temperatures are in the range 110° - 160°F . As a result, the calandria, which has a basically cylindrical shape, 25 ft in length, with an outer diameter of about 25 ft, is constructed of much thinner steel than the pressure vessels for light-water reactors. (The outer cylindrical wall is about 1-1/8 in. thick, and the ends of the cylinder about 2 in. thick, both of stainless steel. The tubes through which the fuel pressure tubes pass are made of zircaloy.)

Control of the reactor is maintained by a dual system composed of a number of variable neutron absorbers (actually compartments into which a specified amount of light water can be pumped) and a number of adjusting rods (typically consisting of an absorber such as stainless steel, although these have sometimes been booster rods, consisting of highly enriched fuel). Shutdown capability exists in the form of gravity operated absorbing rods, backed up by a system for injecting poisons into the moderator. It is also possible to drop the moderator out of the calandria into a storage tank.

- 1 MAIN STEAM SUPPLY PIPING
- 2 BOILERS
- 3 MAIN PRIMARY SYSTEM PUMPS
- 4 FEEDERS
- 5 CALANDRIA ASSEMBLY
- 6 FUEL CHANNEL ASSEMBLY
- 7 FUELLING MACHINE BRIDGE
- 8 MODERATOR CIRCULATION SYSTEM

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CANDU PHW REACTOR

Fig. 2-15. CANDU NSSS (from the AECL).

1 MAIN STEAM SUPPLY PIPING	9 FUELLING MACHINE
2 BOILERS	10 FUELLING MACHINE DOOR
3 MAIN PRIMARY SYSTEM PUMPS	11 CATENARY
4 CALANDRIA ASSEMBLY	12 MODERATOR CIRCULATION SYSTEM
5 FEEDERS	13 PIPE BRIDGE
6 FUEL CHANNEL ASSEMBLY	14 SERVICE BUILDING
7 DOUSING WATER SUPPLY	
8 CRANE RAILS.	

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Fig. 2-16. CANDU REACTOR BUILDING (from the AECL).

Table 2-3. CANDU (representative characteristics).

core thermal power	2140 MW(th)
plant efficiency	28%
plant electrical output	600 MWe
core diameter	248 in.
core length	234 in.
core weight (fuel bundles)	240,000 lb
core power density	12 kW/liter
cladding material	zircaloy
cladding diameter (OD)	0.515 in.
cladding thickness	0.016 in.
fuel material	UO ₂
pellet diameter	0.478 in.
fuel bundle array	37 rods, arranged in concentric circles
array diameter (OD)	4 in.
total number of bundles	4560
total number of fuel rods	168,720
total amount of fuel (UO ₂)	210,000 lb
control rod types	variable neutron absorbers (light water compartments), adjustable absorbers (such as stainless steel); shutdown by absorbing rods or poison injection
number of control rods or compartments	from 4 to 21 of each type of absorber
coolant	heavy water (liquid, plus some gas, phase), >95% D ₂ O
total coolant flow rate	60×10 ⁶ lb/hr
coolant pressure (entrance to channel)	1602 psi
coolant pressure (exit of channel)	1493 psi
coolant temperature (entrance)	512°F
coolant temperature (exit)	594°F
average coolant exit quality	3%
moderator	heavy water, 99.75% D ₂ O (molecular ratio)
moderator pressure	approximately atmospheric
moderator temperature (entrance)	110°F
moderator temperature (exit)	160°F
total heavy water inventory	1.02×10 ⁶ lb
maximum fuel temperature	3832°F
maximum clad temperature	684°F
axial peaking factor	1.5
radial peaking factor	1.2

Table 2-3. Continued.

fuel residence time	470 full-power days
design fuel burnup	7000 MWd/metric ton
fresh fuel assay	0.71% ²³⁵ U
spent fuel assay	0.2-0.3% ²³⁵ U
refueling sequence	on-line, essentially continuous, refueling
calandria outer diameter	25 ft
calandria length	25 ft
calandria wall thickness (stainless steel)	1-1/8 in. side walls, 2 in. ends
number of calandria tubes (zircaloy)	380
lattice array	square with 11.2 in. pitch

The CANDU has an additional cooling system (also heavy water) of sufficient capacity to cool the reactor after shutdown. Connected to this shutdown system is an emergency-core-cooling supply of light water which may be injected should the basic system fail.

The calandria is actually in a vault that is filled with light water, which serves as shielding and helps to maintain the calandria at a constant temperature. The vault in which the calandria sits is itself in a sealed containment building (which includes the elements of the NSSS) and in some designs can be maintained at a negative pressure with respect to the external surroundings. A spray system exists within the containment building. The NSSS is shown in Fig. 2-15, the reactor building in Fig. 2-16. A summary table for the CANDU is given as Table 2-3.

2.4. High-Temperature Gas-Cooled Reactor

The HTGR core consists of a massive pile of hexagonal graphite blocks, each containing fueled regions, the whole structure being cooled by pressurized helium gas. The graphite serves as the neutron moderator. The fuel itself consists of highly-enriched uranium as the fissile material and thorium 232 as the fertile. These materials, in the form of the dioxide or carbide, are present as small fuel kernels with ceramic coatings,* so that both the fuel and its "cladding" are ceramic. As indicated in Fig. 2-17, the initial fuel will be composed of two particle types, one containing uranium enriched to 93% ^{235}U , the other containing the thorium. As the reactor runs, fissile ^{233}U would build up in the latter particles. The fuel particles are bound into fuel rods (with graphite as the binder) which are incorporated into the basic block or element. These elements are stacked as indicated in Fig. 2-18. A basic refueling region consists of a central stack, which has two vertical control rod channels, and the six adjacent stacks (without such channels). The central stack has an additional channel into which boron carbide balls can be poured as a reserve shutdown system. All of the fuel elements have holes through which the coolant flows.

The core and the other NSSS components are contained in a "prestressed-concrete reactor vessel" (PCRv) which has steel-lined cavities for each component (Fig. 2-19). The reactor is fueled through penetrations above the central stack of each refueling region; this same penetration also serves for the drive for the control rod pair penetrating that stack. There are

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*The thorium particles are coated with pyrolytic carbon, the ^{235}U particles with pyrolytic carbon and silicon carbide.

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Fig. 2-17. HTGR COATED FUEL PARTICLES, ROD, AND ELEMENT
(from General Atomic).

REACTOR CORE ARRANGEMENT

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Fig. 2-18. HTGR FUEL ELEMENT ARRANGEMENT (from General Atomic).

XBL 766-8001

Fig. 2-19. HTGR PRESTRESSED CONCRETE REACTOR VESSEL
ARRANGEMENT (from General Atomic).

additional penetrations for piping and removal plugs for servicing (and even replacement) of steam generators, circulators, etc. The vessel is prestressed with vertical steel tendons and with circumferential wire wrapping. The PCRV and its contents are extremely massive (100,000,000 lb); and indeed the core itself is more massive, by about one order of magnitude, than the core of an LWR. The PCRV is surrounded by a containment building (Fig. 2-20) much like that of a PWR.

The helium gas, at a pressure of 700 psi, is pumped downward through the core, and exits with a temperature of 1370°F. The gas then passes into one of the 4 or 6 (depending on the plant size) heat exchangers, raising steam as usual. Above each steam generator is a circulator which pumps the helium. The higher reactor operating temperature, which is permitted by the gaseous form of the coolant and the good high-temperature characteristics of the core, provides steam which can be converted to electrical energy with an efficiency of 39%, substantially greater than in a water-cooled reactor. Moreover, the potential exists, with development of helium-driven turbogenerators, to raise this efficiency even more. Table 2-4 summarizes many HTGR features.

The accident response of an HTGR is noticeably different than that of a water-cooled reactor, principally because of the mass of the core. Were shutdown to occur with complete loss of cooling, no damage to the fuel would occur for about half an hour (as compared with the roughly 1 min of an LWR). The fuel particle, due to its ceramic coating, is much less susceptible to melting than its counterpart, a metal-clad fuel rod. And, due to the heat capacity of the core, elevated temperatures take much longer to reach. Moreover, the core's structural strength is provided by graphite, whose strength improves as the temperature rises.

To make a complete loss of coolant incredible, flow restrictors are incorporated around PCRV penetrations to reduce helium loss should the vessel integrity be violated there. As a result, helium is always presumed to be in the system. Should all the primary cooling loops become unavailable, difficult to envision since they are largely independent, the two or three loops of the auxiliary cooling system (see Fig. 2-19) are activated. These are sized to be sufficient for safe shutdown.

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Fig. 2-20. HTGR PLANT ARRANGEMENT (from General Atomic).

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Table 2-4. HTGR (representative characteristics).

core thermal power	2900 MW(th)
plant efficiency	39%
plant electrical output	1160 MW(e)
core diameter	27.8 ft
core active height	20.8 ft
core power density	8.4 kW/liter
number of core stacks (columns)	493
number of fuel elements per column	8
number of fuel elements	3944
element geometry	hexagonal shape, 31 in. high, 14 in. across flats
control rod type	pairs of control rods in central stack of each refueling region (set of seven stacks)
number of control rods	73 pairs
reserve shutdown system	spheres of boron carbide in carbon
form of fuel	fissile and fertile materials in different fuel particles, ^{235}U as UC_2 , Thorium-bred ^{233}U in other particle type. Types have different coatings to facilitate separation.
average (maximum) fuel temperature	1450°F (2750°F)
average moderator temperature	1320°F
coolant	helium gas
coolant flow rate	10.4×10^6 lb/hr
coolant pressure	700 psi
coolant temperature (inlet)	636°F
coolant temperature (outlet)	1366°F
fuel exposure	98,000 MWd/metric ton of fuel
fresh fuel assay (fissile particles)	93% ^{235}U (in initial loading)
spent fuel assay (fissile particles)	30% ^{235}U (from initial loading)
refueling sequence	one-fourth of the fuel per year
weight of core and innards	6×10^6 lb
weight of PCRV (empty)	90×10^6 lb

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An Evaluation of High-Temperature Gas-Cooled Reactors, USAEC Report WASH-1085, Dec. 1969--a general discussion of high-temperature gas-cooled reactors.

Gas-Cooled Reactors: HTGR's and GCFRB's, CONF-740501. Topical Conference, Gatlinberg, May 7, 1974--good collection of papers on gas-cooled reactors that are otherwise hard to find.

The specific references in which most of the materials on specific reactor types may be found are:

For the PWR and BWR:

WASH-1250 plus numerous Safety Analysis Reports submitted in support of applications for licensing of construction of individual reactors. In particular, the materials in Tables 2-1 and 2-2 came principally from those reports.

For the CANDU:

CANDU Nuclear Power Station, a publication of Atomic Energy of Canada Limited.

For the HTGR:

Safety Design Bases of the HTGR, General Atomic Report GA-A12618 and other reports from the General Atomic Company.

CHAPTER 3. FUTURE DIRECTIONS FOR FISSION POWER

3.1. Uranium Utilization and Thermal Reactors3.1.1. Fuel resources and growth of nuclear generating capacity.

Uranium resources, like many available to us, are not unlimited. Although a gram of ^{235}U , "burned" in a reactor, yields more energy than a ton of coal, reasonably assured U. S. deposits of uranium are only enough to supply, until the year 2000, reactors as they are presently being built and operated. By that year, ERDA (the Energy Research and Development Administration) estimates, there will be roughly 1000 nuclear power plants operating, and the cumulative uranium requirements for initial cores and for refueling will amount to 2 or 3 million tons of U_3O_8 . This is comparable to the total amount of uranium that is presently thought to be available at reasonable economic and environmental cost. The question is, then, after the year 2000, how do we supply (1) initial cores for new plants starting up and (2) the continuing fuel needs of the accumulated nuclear power system? Even if we agreed not to build any new fission power plants after that year, fuel requirements for those already built would substantially exceed the uranium now thought to be available. The standard solution to this problem is that late in this century we should begin building "breeder" reactors which not only replace their own fuel, but produce an excess to be used in other reactors.

Before discussing breeders themselves, let us examine in slightly more detail the premises on which the need for them is based. An ordinary reactor, such as described in the last chapter, requires that--during its operating lifetime--several thousand tons of U_3O_8 be supplied to the nuclear fuel cycle in order to create enough slightly enriched fuel to keep it running. The details of precisely how much fuel might be needed are discussed in the next section (see especially Table 3-2). Were we to use a figure of 5000 tons of U_3O_8 per reactor, it is easy to see how the resulting cumulative lifetime requirement of 5 million tons for a 1000-reactor system would strain our resources, since ERDA estimates them to be 3.5 million tons. A breakdown of this estimate is given in Table 3-1. Resources are categorized by "forward costs", the estimated actual cost to produce the U_3O_8 concentrate, and by state of discovery. The latter categorization has led to speculation that the estimates may be substantially in error. Note that only a small portion of the estimate is "reserves", i.e., developed or fully discovered resources. The meaning of the "potential" categories may be had from ERDA:*

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*From ERDA-1535.

"For its study of resources, ERDA subdivided potential resources into three categories: probable, possible, and speculative, to reflect the varying nature of the estimates depending upon the specific situation. Probable resources are those contained within favorable trends largely delineated by drilling data within known productive uranium districts. In this situation, favorable geologic characteristics of a formation are known from drilling or outcrop data, and quantitative estimates of potential resources are made by considering the size of the favorable areas and by comparing the geologic characteristics with those present in the areas with ore deposits.

"Possible potential resources include those situations that are outside of identified mineral trends but which are in formations and geologic provinces that have been productive. Speculative resources are those estimated to occur in formations or geologic provinces which have not been productive but which, based on the evaluation of available geologic data, are considered to be favorable for the occurrence of uranium deposits. There is inherent uncertainty in these estimates, much more so for the speculative than the probable potential."

The Energy Research and Development Administration has been criticized for concentrating its attention on areas where high-grade ore is known to exist, thereby neglecting the possible resources in other parts of the United States. There are, however, those who think ERDA's extrapolations to be too optimistic, constituting a dangerously shaky base on which to found a large nuclear power system.

(In addition to the type of ores considered above, various areas of the United States contain mineral deposits with lower concentrations of uranium, i.e., 0.01% or less. Working such deposits for uranium would require the processing of extremely large amounts of material, comparable--for equivalent energy content--to the mining of coal, except that the mechanical and chemical processing involved would be much more extensive than for coal. The economic and environmental costs involved make utilization of such low-grade resources extremely unattractive, so that they are typically not considered in resource estimates.)

ESTIMATED U. S. URANIUM RESERVES AND POTENTIAL RESOURCES, December 31, 1975

Tons U ₃ O ₈					
	Potential				
	<u>Reserves</u>	<u>Probable</u>	<u>Possible</u>	<u>Speculative</u>	<u>Total</u>
\$10	315,000	440,000	420,000	145,000	1,320,000
\$10-15 Increment	<u>105,000</u>	<u>215,000</u>	<u>255,000</u>	<u>145,000</u>	<u>720,000</u>
\$15	420,000	655,000	675,000	290,000	2,040,000
\$15-30 Increment	<u>180,000</u>	<u>405,000</u>	<u>595,000</u>	<u>300,000</u>	<u>1,480,000</u>
\$30	600,000	1,060,000	1,270,000	590,000	3,520,000
By Product*					
1975-2000	90,000	-	-	-	90,000
2000-2020	<u>150,000</u>				<u>150,000</u>
	840,000	1,060,000	1,270,000	590,000	3,760,000

* By-product of phosphate and copper production.

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Table 3-1
(from ERDA-1535)

Given a resources estimate, one still needs a view of how rapidly nuclear electric-generating capacity will develop to foresee how severely resources will be pressed. Over the period during which the civilian nuclear power industry has developed, the view of the Atomic Energy Commission (and now ERDA) on this matter has been changing. For most of the present decade, the AEC's estimate of the number of reactors operating by the year 2000 has been somewhat more than 1000, although in the last 2 years, the AEC has--in its projections--included at least one possible case with fewer than 1000 reactors. For example, in the AEC document, WASH-1139(74), dated February 1974, several cases are discussed, projecting from 850 to 1400 1000-MWe nuclear plants in the year 2000. Since 1974, these projections will have decreased even further.

To see the implications in terms of resources, consider Fig. 3-1, from a review of the LMFBR program, which shows commitments of U_{308} to reactors built by the year 2000 for the 850-reactor case (WASH-1139(74)). The reactors considered are primarily LWR's of the type currently being built, with 15% HTGR's of the standard type (see previous chapter and next section), so that the projection is basically an extension of the present program of building LWR's. The figure shows both cumulative demand and lifetime demand of the reactors built by any given time. Results are shown presuming plutonium recycle and for two assumptions of the ^{235}U content of enrichment tails. Looking at lifetime demand, we see that estimated resources would eventually be consumed by the 850-reactor system. The situation would thus be quite sensitive to differences between the actual size of the nuclear generating system (and actual uranium resources) and the above estimates, unless an alternative type of reactor becomes available sometime this century. Moreover, continued growth of the electrical generating system after the year 2000 would require some alternatives, even if the estimates are correct.

The total electric-generating capacity available in 1975 approached 500,000 MWe, of which only a small portion was nuclear.* The presumption of ERDA and the utilities has been that nuclear plants will assume an increasingly important role until--by 2000--they constitute 50 or 60% of capacity. In the last decades, generating capacity has grown much faster

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* In 1975, less than 10% of capacity was nuclear; however, in some regions--such as the Chicago area--the figure was as high as 30%.

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Fig. 3-1. CUMULATIVE AND LIFETIME U_3O_8 DEMAND for postulated U. S. nuclear power system of 850 gigawatts by 2000 (from ERDA-1).

(at a rate of 6 or 7% growth per year) than total energy use (3 or 4% annually). A continuation of the 6% rate leads to a total capacity of 2100 gigawatts (1000 MW) by 2000, and about 1200 of this would be expected to be nuclear. However, more recent expectations lower both numbers by projecting a growth rate of 4 or 5%. A 4.5% rate would yield a total capacity of 1400 gigawatts, in which case it is unlikely that even the low case of 850 reactors would be achieved. Any slowing of the nuclear building program can have an impact on uranium resource lifetime. As discussed in the next section, introduction of more "advanced" converter reactors could increase this impact markedly.

We have so far neglected a number of important steps between identification of resources and their actual utilization in reactors. Effectively we have presumed that development of mines and mills, and construction of UF_6 conversion facilities, enrichment plants, and fuel fabrication facilities will proceed rapidly enough to supply the projected thermal reactor system with fuel. Moreover, the above projections assumed availability of fuel reprocessing plants and the licensing of plutonium recycle. Some uncertainty exists as to the efficiency with which enrichment plants will be operated, i.e., how much ^{235}U will be left in the tails. (The separative work associated with, say, leaving only 0.2% ^{235}U in the tails costs a great deal more and consumes more energy than for leaving 0.3% ^{235}U in the tails.) It would appear that a justifiable assumption is that facilities will come on line and decisions will be made as are necessary to supply the reactor system that exists at any given time. Presumably development, construction, and licensing difficulties are comparable to those of other advanced systems, and the effect of any slowdown would be to stretch out the time scale, effectively reducing the rate of energy growth.

3.1.2. High conversion thermal reactors.

It is possible to build thermal reactors, of the type discussed in the last chapter, which have a higher conversion ratio than those reactors now achieve. Since this is the ratio of fissile material produced to fissile consumed, achievement of a ratio of 1 on a sustained basis would mean that no more fissile material would have to be supplied from external sources to keep a given reactor running. An average of only slightly more than two fission neutrons are released per thermal neutron absorbed by fissile material (see Appendix B). Committing one neutron to subsequent absorption by the fissile material, thereby continuing the chain reaction, implies that great

care must be taken if--on the average--another neutron is to be captured by a fertile nucleus to replace the fissile nucleus that was destroyed.

In current light-water reactors, a ratio of only about 0.6 is achieved for two fundamental reasons: The water itself absorbs a significant portion of the neutrons from fission, and the nuclear fuel itself cannot practically sustain a conversion ratio of 1 because the ^{235}U - ^{238}U - ^{239}Pu combination is not very good neutronically. By the latter we mean that ^{235}U and ^{239}Pu produce such a small excess of neutrons (per neutron absorbed) above the two required for breakeven, that any small unavoidable losses will depress the conversion ratio below 1. In fact, the value of eta (η , see Appendix B) for ^{239}Pu is actually less than two for a large range of energies that would be substantially populated as neutrons are thermalized, so that a spectrum-averaged value is typically 1.8 or 1.9, well below the absolute minimum for breakeven. (Remember that this entire discussion refers to thermal reactors.)

Were ^{232}Th used as the fertile material, the situation would be quite different. First, ^{232}Th has a higher neutron absorption cross section than does ^{238}U (Appendix B), so that more conversion takes place in the former. As a result, more ^{233}U is produced than ^{239}Pu in equivalent circumstances. (This means, of course, that more fissile material must be supplied initially to overcome the greater absorption of the ^{232}Th .) Furthermore, ^{233}U is a much better thermal fuel than ^{239}Pu , and is even superior to ^{235}U . (This is the reverse of the situation in today's LWR's, where the produced fissile species, ^{239}Pu , is poorer than the initial fissile type, ^{235}U .) The reason for ^{233}U 's superiority, as depicted in Fig. 3-2, is that--for each neutron absorbed by ^{233}U --noticeably more than 2 neutrons are produced. Eta (η) typically has a value of 2.2 or more in a thermal spectrum. The comparable number for ^{235}U is very close to 2, and we noted above that ^{239}Pu is even worse.

The best possible situation has ^{233}U and ^{232}Th as the fuel, with ^{232}Th being converted to replace the ^{233}U that is consumed. Since ^{233}U does not occur naturally (due to its "short" half-life), this condition must be reached by beginning with ^{235}U as the fissile fuel, in the presence of which ^{233}U is produced. Since there is little doubt that reactors using ^{233}U - ^{232}Th can be built to achieve a conversion ratio of 1, what has to be supplied from external sources is the fissile material to produce enough ^{233}U .

Some inkling of the potential value of the thorium cycle can be obtained from Table 3-2, which gives the fuel needs of a variety of reactor types under

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Fig. 3-2. NEUTRON YIELD PER NEUTRON ABSORBED--typical light-water reactor values for ^{239}Pu and ^{233}U fuel (from ERDA-1541).

REACTOR CHARACTERISTICS⁽¹⁾

	BWR	PWR	STD HTGR	HIGH GAIN HTGR	CANDU (THOR)	CANDU	ATR
Thermal efficiency, %	34	33	39	39	30	30	31
Specific power, MW _{th} /MT	28	38	82	69	25.6	22	13
Initial core (and blanket*) average							
Irradiation level, MWD _{th} /MTU	17000	22600	54500	33500	18500	6900	13700
Fresh Fuel Assay, Wt% ²³⁵ U	2.03	2.26	93.15	93.15	93.15	0.711	1.2
Spent Fuel Assay, Wt% ²³⁵ U	0.86	0.74	~60	~60	~60	0.31	0.33
Fissile Pu recovered, kg/MTU ⁽²⁾	4.8	5.8	--	--	--	1.7	4.3
Feed Required, ST U ₃ O ₈ /MWe ⁽³⁾							
at 0.2% tails	0.494	0.422	0.367	0.338	0.520	0.199	0.516
at 0.3% tails	0.581	0.498	0.456	0.419	0.646	0.199	0.577
*Feed Required, ST ThO ₂ ⁽³⁾	--	--	42.5	52.8	160	--	--
Separative Work Required, SWU/MWe ⁽³⁾							
at 0.2% tails	239	222	366	336	519	--	--
at 0.3% tails	185	174	311	286	441	--	--
Replacement Loadings (Annual rate at steady state; 75% capacity factor)							
Irradiation Level, MWD _{th} /MTU	27500	32600	95000	48800	27000	9600	20000
Fresh Fuel Assay, Wt% ²³⁵ U	2.73	3.21	93.15	93.15	93.15	0.711	0.711
Spent Fuel Assay, Wt% ²³⁵ U	0.84	0.90	30	30	--	0.15	0.2
Fissile Pu Recovered, kg/MTU ⁽²⁾	5.9	7.0	--	--	--	2.3	--
Feed Required, ST U ₃ O ₈ /MWe ⁽³⁾							
at 0.2% tails	0.144	0.154	0.085	0.045	0.020	0.125	--
at 0.3% tails	0.179	0.191	0.106	0.055	0.024	0.125	--
*Feed Required, ST ThO ₂ /MWe ⁽³⁾	--	--	8.7	19.7	41.5	--	--
Separative Work Required, SWU/MWe							
at 0.2% tails	105	117	85	45	19	--	--
at 0.3% tails	84	94	73	38	16	--	--
Replacement Loadings (Annual rate with Pu recycle ⁽⁵⁾ , 75% capacity factor)							
Fissile Pu Recycled, kg/MWe	0.163	0.167					0.3
Fissile Pu Recovered, kg/MTU ⁽²⁾⁽⁶⁾	8.1	9.5					5.5
*Fissile ²³⁵ U Recovered, kg/MTU ⁽²⁾	--	--					--
Feed Required, ST U ₃ O ₈ /MWe ⁽³⁾⁽⁴⁾							
at 0.2% tails	0.121	0.129					
at 0.3% tails	0.148	0.158					
Separative Work Required, SWU/MWe ⁽³⁾							
at 0.2% tails	82	93					
at 0.3% tails	66	75					
*Lifetime ⁽⁷⁾ Commitment Required, 40-year life, ST U ₃ O ₈ /1900 MWe							
40 (Replacement requirement) + Initial Core and Blanket							
Without Pu recycle							
at 0.2% tails	5130	5330				4160	
at 0.3% tails	6340	6550				4160	
With Pu recycle							
at 0.2% tails	4020	4100					2600
at 0.3% tails	4900	5050					2660
With Thorium and ²³³ U recycle							
at 0.2% tails			2980	1920	1440		
at 0.3% tails			3700	2400	1780		

* Information requested not previously in Table 12, WASH 1139(74), page 24

- (1) MW_{th} is thermal megawatts; MWe is net electrical megawatts; MWD_{th} is thermal megawatt days; MTU is metric tonnes (thousand of kilograms) of uranium; and ST U₃O₈ is short tons of U₃O₈ yellowcake from an ore processing mill. One SWU is equivalent to one kg of separative work.
- (2) After losses.
- (3) For replacement loadings, the required feed and separative work are net, in that they allow for the use of uranium recovered from spent fuel. Allowance is made for fabrication and reprocessing losses.
- (4) Includes natural uranium to be spiked with plutonium; 0.0087 ST U₃O₈/MWe for BWR and 0.0067 for PWR.
- (5) Plutonium available for recycle ratchets up each pass because not all of the plutonium charged is burned. Therefore, more plutonium is recovered from mixed oxide fuel than from standard uranium fuel, and this increment increases with each cycle (5-6 years per cycle) requiring several passes to reach steady state. The data shown represent conditions for the 1980's when most reactors will be discharging fuel which has only seen one recycle pass.
- (6) Average for all fuel discharged with full recycle of self-generated plutonium. For mixed oxide fuel (natural U spiked with self-generated plutonium) the spent fuel from BWRs contains 15.1 kg Pu per MTU and from PWRs, 18.7.
- (7) Lifetime commitments assume operations at 40% Capacity Factor (CF) for the first year, 65% CF for the next two years, followed by 12 years at 75% CF. Thereafter, CF drops two points per year, reaching 25% in the last (40th) year.

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Table 3-2.
(from ERDA-1)
REACTOR CHARACTERISTICS

a number of conditions. The results for the BWR and PWR correspond to presently constructed plants, as do those for the ordinary CANDU described in the last chapter. The quantity that we will concern ourselves with, for simplicity, is the lifetime uranium requirements. Note that the LWR results depend on whether or not fuel is reprocessed, in particular whether plutonium is recycled, and on the percentage ^{235}U left in the enrichment tails. Neither of these factors affects the standard CANDU because as currently operated reprocessing is presumed not to occur, and only natural uranium is used as the fuel. CANDU without reprocessing still compares well with the best that LWR's can do (see Table 3-2). This is due to the higher conversion ratio possible when heavy water is used as moderator-coolant.

The "standard" HTGR is the only reactor for which current designs depend on the thorium cycle. As discussed in the last chapter, HTGR fuel particles initially contain ^{235}U and ^{232}Th . Eventually, noticeable concentrations of "bred" ^{233}U will reside in the thorium particles. After reprocessing, this ^{233}U is recycled back to the reactor, as a result of which the lifetime feed of natural uranium is less than 3000 tons of U_3O_8 as compared with the more than 4000 tons for the current water-cooled reactors. The lesser figure is also due to other factors, such as the use of carbon and helium as moderator and coolant, respectively.

However, better results can be obtained even for the water-cooled reactors by introducing a thorium cycle. Preliminary studies* indicate that in LWR's, a lower lifetime requirement can be achieved simply by replacing most of the ^{238}U in the present fuel loads by ^{232}Th . This initial use of highly-enriched uranium together with ^{232}Th , followed by recycle of bred ^{233}U , makes this concept similar to the HTGR scheme just described. However, the fuel form would be very different, and the continued use of light water would inevitably cost neutrons (and hence conversion ratio). The net effect, though, is still a reduction by about 15% (to about 3500 tons of U_3O_8). However, this calculation involves no redesign of the core, and thus took no account of altered cooling, control, and other characteristics. An effort to optimize the core neutronically and in these other respects would certainly cause greater improvement in the uranium requirement. As discussed

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* EPRI Report NP-2.

in Section 3.2, such redesign is the basis for the light-water breeder reactor, and could presumably be incorporated with some success in present LWR power plants.

However, the CANDU and HTGR are ultimately more susceptible to improvement because of their inherently better moderator systems (from the neutronic point of view). Table 3-2 shows results for a "high-gain" HTGR which has a modified fuel-moderator ratio and a more frequent refueling schedule to achieve better conversion ratio (see Section 1.2.3). The decrease in U_3O_8 requirement is quite substantial, a reduction of one-third. The "Thorium CANDU", also shown, involves a greater change, i.e., from natural uranium to ^{233}U - ^{232}Th fuel. Since Canada intends to build none of the enrichment plants that would be necessary to secure relatively pure ^{235}U , plutonium is necessary as an intermediary. Present CANDU's produce ^{239}Pu and--as in LWR's--it contributes to the heat produced by the reactor. Rather than leave this plutonium in the spent fuel, as is now done, this fuel would be reprocessed to obtain Pu , which would then be used as feed to an initially plutonium-thorium core. By adding enough plutonium, sufficient ^{233}U can be bred to fuel the reactor at a conversion ratio of 1, so that the reactor is self-sustaining thereafter. One thousand, three hundred and twenty (1320) tons of U_3O_8 is the figure quoted by AECL as the total feed to the reactor system to eventually produce the ^{233}U for one self-sustaining reactor (and the number may even be less if the plutonium stage could be avoided by beginning with highly enriched uranium). This last condition is much desired; the 1320 tons not only supplies the reactor for its lifetime, but also the reactor replacing it at its decommissioning. (Note that Table 3-2, from the LMFBR program review, attributes a somewhat larger requirement to the thorium CANDU.)

To see the effect of these higher-conversion-ratio reactors on uranium resources, it is useful to make estimates along the lines used by ERDA in Fig. 3-1. Let us take the most conservative and simplest point of view and estimate total commitment of uranium for a reactor system using, as above, the lifetime requirements associated with various reactor types, and taking no credit for possible introduction of a breeder reactor at a later date. We calculate the total uranium commitment for the following conditions:

1. One thousand reactors by the year 2000, using the same proportions of LWR's and HTGR's as from the LMFBR program review.* For lifetime requirements, we assume the reactors are run as presently envisioned (4070 tons for LWR, 2980 for HTGR).
2. Same as case 1, but conscious choice of high-gain reactors, with only 2000 tons committed to each reactor built after 1985. (See, however, last paragraph of this section.)
3. Only 700 reactors by 2000, with increasing emphasis on achieving higher gain. We assume an effort to retrofit pre-1985 reactors to the thorium cycle, so that lifetime requirements for those reactors average 3500 tons, and that only 2000 ton reactors are built after 1985, the requirement being even further reduced to 1500 for those built after 1990.

Growth of the reactor system for each of these cases is displayed in Fig. 3-3. Figure 3-4 shows the lifetime commitment for the accumulated reactor system as a function of year.

The difference between case 3 and case 1 is striking. Case 1 is comparable to what has been presumed by ERDA; case 3 gives an indication of what could be done by a choice of high-gain thermal systems instead of low-efficiency thermal systems and a fast breeder reactor. Note that case 3 still assumes a very substantial reliance on nuclear fission power and would require no radical change in either the size or direction of energy growth.

What causes the differences between these cases? Case 2 has the same number of reactors as case 1, so that the reduced U_3O_8 requirement is due entirely to the greater efficiency of high-gain reactors. Case 3 presumes somewhat fewer reactors than case 2, and also somewhat more efficient ones; however, a quick check shows that the difference in uranium commitment relative to case 2 is primarily due to the smaller number of reactors in case 3.

What is important to note is that the lifetime commitment associated with case 3 in the year 2000 is less than half of that of case 1. Such a reactor system, even if it grows for some time beyond 2000, could be supported by United States uranium deposits. After that time, if the need for nuclear generating capacity is still growing, the breeder reactors of the next section could be introduced.

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* ERDA document ERDA-1.

Fig. 3-3. PROJECTED NUMBERS OF 1000 MWe NUCLEAR POWER PLANTS VS YEAR

Fig. 3-4. PROJECTED LIFETIME URANIUM COMMITMENTS VS YEAR

The proposed timetable given above for introduction of high-gain thermal reactors will seem unduly optimistic. It is hard to imagine the nuclear industry gearing up to produce such reactors in quantity by 1985, particularly since General Atomic, the vendor of HTGR's in this country, withdrew from the market in late 1975. However, the industry is pushing to establish standardized designs and site evaluations in order to compress construction timetables to about 5 years. With this, it is possible to envision numbers of high-gain reactors coming on line in the late 1980's, provided a commitment were soon made in that direction. This stretching of the date for introducing high-efficiency systems to near 1990 does little violence to the comparison made above, largely because the schedule for construction of conventional LWR's is being pushed back in any case.

3.2. Breeder Reactors

Alternatively, if current practice for thermal reactors and rapid growth of nuclear capacity continue, breeders will be necessary by the end of this century unless uranium resources are substantially underestimated. The basic idea of a breeder is to effectively increase conversion enough so that the conversion ratio exceeds 1. To be more precise, the actual quantity of interest for breeders is the "breeding ratio", the ratio of the number of fissile atoms produced to the number of the same kind that are consumed. This definition is more precise than that of conversion ratio (where we do not distinguish between fissile types) and is appropriate for breeders where basically the same material is bred as is supplied. Exceeding 1 implies production of enough fissile material both to replace that consumed in the reactor and to fuel additional reactors.

For an expanding nuclear system, an important number is the "doubling time", the period required to double the amount of fissile that was initially committed. It is the time after which a second reactor could be fueled from the excess productivity of the first.* Assuming that the electrical energy produced is simply proportional to the total number of fissioned nuclei, it is

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* We have not defined doubling time precisely enough. The most basic clarification we can make is to distinguish the doubling time associated only with the fuel inventory of the reactor itself and that associated with the fuel not only in the reactor, but also in the fuel cycle, i.e., in storage, reprocessing, fabrication, etc. The practical doubling time of a breeder reactor system is the time it takes the reactor and its support facilities to yield enough excess fissile material to supply an additional reactor and its support facilities.

easy to see how the doubling time is related to the breeding ratio. Defining the breeding gain as the excess of the breeding ratio over 1, the amount of excess fissile material produced per day for any given power level will be proportional to this gain assuming other factors--such as the electrical energy per fission and the ratio of absorption to fission cross sections--are fixed. The doubling time is simply the ratio of the reactor (or fuel cycle) inventory to this amount of excess material, so that doubling time is proportional to inventory divided by breeding gain.

The distinction between the high conversion (or "near" breeder) reactors of the last section and an actual breeder can be very narrow. A working distinction might reserve the term "breeder" for reactors with a doubling time comparable to the doubling time for the size of the nuclear generating network. If this capacity approaches a no-growth situation, then reactors with conversion ratios that are just barely 1 would practically be breeders. This could include the thorium CANDU or the light-water breeder (below).

In practical terms, there are two classes of breeder: "thermal" breeders, which "tune up" the conversion ratio by using a thorium cycle to achieve breakeven or better, as discussed in the last section, and "fast" breeders, which take care not to moderate the neutrons, thereby taking advantage of the increased value of η (the number of neutrons produced per neutron absorbed in the fuel) in a fast spectrum (see Table B-4). The latter number exceeds 2 by an amount which suggests that breeding is easily available. However, difficulties arise because the fission cross sections are much smaller than for thermal neutrons, so that a higher concentration of fissile is necessary to make use of the neutrons. The result is that the core of a fast breeder is typically very compact, with fissile enrichments of 15% and greater; around this "seed" region is a "blanket" of fertile fuel to utilize any neutrons that escape for fissile production. Details for both the liquid metal- and gas-cooled fast breeder reactors are given below. The liquid metal fast reactor (LMFBR) is now the mainstay of every breeder development program in existence, including that of the United States. Before discussing fast breeders, we first describe the light-water breeder, since it is a straightforward extension of the ideas of the last section. The molten-salt breeder reactor is also a thermal reactor, but its concept is radically different; because it is no longer being developed, we leave it until the end of this chapter.

In general, in this section, discussion of overall reactor systems will not be as detailed as in the last chapter. The purpose is to point out the fundamental aspects of each reactor type.

3.2.1. Light-water breeder reactor.

As now being developed by ERDA, but on a rather low-level basis, the light-water breeder reactor (LWBR) is basically a pressurized water reactor in which the core runs on an optimized ^{233}U - ^{232}Th cycle, much as discussed in the section on high-conversion thermal reactors, except that the moderator and coolant are still light water. One might wonder how such a substantial neutronic disadvantage can be overcome. The novel feature of the LWBR, aside from its dependence on the thorium cycle, is the lack of neutron poisons as control. Although boron poisons are held in reserve as a backup shutdown system, they are not used under normal operating conditions. The reactor is controlled by moving fuel assemblies with higher-than-average enrichment in and out of the core. This is equivalent to changing the core configuration, and hence the neutron multiplication factor, and avoids the neutron loss caused by control poisons. Presumably, withdrawal of the high-enrichment (i.e., about 6%) fuel assemblies causes a greater fraction of neutrons to be absorbed by thorium, which is more predominant in the stationary assemblies. Thus the movable assemblies control the ratio of fissions to conversions, and can be used to maintain a multiplication factor of 1, as required for constant-power operation (see Appendix D).

Figure 3-5 shows a test-PWR reactor vessel as modified for testing an LWBR core. (The reactor vessel is part of the Shippingport plant, designed for producing much less power than the full-scale 1000 MWe plants of which we have been speaking.) Since an LWBR is simply a modification of a PWR system, one should envision such a reactor vessel replacing the vessel in an ordinary PWR system (Fig. 2-5). The modifications required on the vessel are primarily a new core, vessel head, and control system. (Adapting a BWR to this concept would be different because of the different water conditions and because the control system consists of plates manipulated from the bottom--see chapter 2.) In Fig. 3-5, note the large mechanisms for moving the "seed" (i.e., higher enrichment) regions; these mechanisms raise seed regions into the core to increase the multiplication factor.

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Fig. 3-5. LWBR CORE IN THE SHIPPINGPORT VESSEL
(from ERDA-1541).

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Fig. 3-6. LWR CORE CROSS SECTION for the Shippingport reactor (from ERDA-1541).

The fuel assemblies themselves are not wholly dissimilar from those of an LWR, consisting of dioxides formed into cylindrical ceramic pellets sealed into cladding which--in this case--is a zirconium alloy which absorbs particularly few neutrons. Fuel rods will contain different mixtures of oxides, depending on the core region shown in Fig. 3-6. The "seed" and "blanket" regions initially contain a mixture of $^{233}\text{UO}_2$ and ThO_2 , the seed region having up to 6% uranium and the blanket regions about 3%. The reflector region (analogous to the so-called "blanket" region of a fast breeder) initially contains all thorium, part of which is converted to uranium as the reactor operates. It is claimed that this configuration will breed, although quite slowly, with a ratio exceeding 1 by no more than a few percent.

Much as in the case of thorium CANDU, getting to the high conversion or breeding situation (where the core contains only ^{233}U and ^{232}Th) requires a "prebreeder" phase in which the ^{233}U is produced. In the case of the LWBR, this phase could use a combination of thorium plus uranium highly enriched in ^{235}U to breed the ^{233}U . Achieving the breeder stage would require a total investment of roughly 2000 tons of U_3O_8 per reactor, after which--as noted in Section 3.1.2--the bred ^{233}U can fuel an LWBR essentially forever.

As mentioned above, this concept can be thought of as a PWR with a modified core design. One might then think of retrofitting existing PWR systems with such cores in order to upgrade their uranium utilization characteristics. This is, in fact, the scheme in using the Shippingport reactor as a test of the LWBR. However, there it was found necessary to reduce the operating power level of the system because the core heat transfer characteristics are significantly changed, due to the altered fuel-coolant configuration. Such a reduction in the power level of commercial reactors would meet overwhelming resistance because of the high capital cost of those systems. So realistically, the LWBR could probably be implemented commercially only in new power plants. It may, however, be possible to introduce thorium fuel cycle to existing LWR's in a less thorough manner. As mentioned in a previous section, simply introducing it--with no modification of the core design--seems to reduce lifetime uranium requirements.

Optimization of the core to the nuclear characteristics of the thorium cycle, but within the restrictions that power level be maintained and that existing criteria for light-water reactor safety remain applicable, would

further reduce requirements, and such a possibility is being investigated.*

3.2.2. Liquid metal fast breeder reactor.

The alternative to slowing down neutrons to take advantage of the large fission cross sections at low neutron energies is to enrich the fuel enough that little moderation occurs prior to absorption by the fuel and to make use of the fact that at high neutron energies, for every neutron captured by ^{239}Pu , about 2.7 neutrons are produced. This exceeds the basic number 2 by a sufficiently large amount to succeed at breeding even if an appreciable fraction of the neutrons are captured unprofitably. The basic nuclear configuration for all fast breeders is similar. The core is a compact arrangement of fuel assemblies that are similar to those used in light water reactors except that the fuel material is more highly enriched, the fuel pins typically have a smaller diameter, and the cladding is stainless steel rather than zircaloy (see Fig. 3-7).

Instead of using ^{235}U or ^{233}U as the fissile material, the present choice is to load the core with ^{239}Pu ; this is the inevitable product of the current generation of light water reactors, and--because of its high value of η (neutrons produced per neutron absorbed) in a fast spectrum--it is much more profitably used in the breeder than in thermal reactors. (^{235}U , on the other hand probably does not yield enough neutrons for a practical breeder; this is not true of ^{233}U , but the latter nuclide is not presently being produced in any large quantity.) The fertile nuclide now chosen is ^{238}U , at least in the LMFBFR (see next section for the alternative). This is expected to produce more than enough ^{239}Pu to replace that which is consumed in the core. The "core" itself would be a mixture of oxides of plutonium and uranium. Surrounding this core would be a blanket of uranium. In both cases, the uranium would be almost entirely ^{238}U . Fission would occur primarily in the core region and conversion would occur in both regions. Regular reprocessing would be needed to recover the bred fissile material.

The blanket assemblies would have rods with the same composition throughout. Core assemblies, on the other hand, would have fissile loads along the portion of the rods that constitutes the core, but only fertile material at the top and bottom of the rods. In this way, the entire core is surrounded with blanket regions.

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* By contractors to the Electric Power Research Institute.

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Fig. 3-7. LMFBR FUEL ROD AND ASSEMBLY (from ERDA-1535).

Power generation in the core will be quite intensive, compared with thermal reactors. As a result, the coolant (now a minimal moderator) must have good heat transfer properties. In the present case, the LMFBR, this has led to the choice of sodium as the coolant. A metal that is liquid over a large range of temperature, sodium can successfully cool the very compact core. Additional advantages of great importance are that it can be used at essentially atmospheric pressure thereby making design easier, and that it can operate at high enough temperature to permit a higher plant efficiency (see Appendix C) than water-cooled reactors can achieve. Sodium, however, is highly reactive chemically. At reactor operating temperatures, it burns if exposed to air; moreover, it reacts violently with water. Stringent efforts must be made to prevent any breaks or corrosion that lead to sodium leakage.

In all LMFBR designs, the sodium that cools the core is not used to raise the steam that drives the turbogenerators. Instead, there is an intermediate sodium loop which avoids the possibility of releasing radioactive sodium during any steam generator problem. This requires use of an intermediate heat exchanger as an interface between the primary and secondary sodium loops. It has the effect of more effectively isolating the primary sodium and thus the sodium-filled reactor from any water. It does not, however, eliminate the difficulty of designing steam generators which effectively separate sodium and water.

Two basic types of LMFBR are being considered. That most favored by other countries is the "pot" type, in which not only the core, but also a number of other components are contained in the reactor vessel. (An example of this is given in Fig. 3-8, which show the French Phenix reactor.) The vessel is filled with sodium at roughly atmospheric pressure, in which are immersed the core, refueling machines, the primary coolant pump, and the intermediate heat exchanger, so that the entire primary sodium loop is contained in the same vessel. This assembly markedly reduces the amount of external piping. The alternative scheme, called the "loop" arrangement, is anticipated in the United States and is more similar to conventional LWR systems in that individual components of the heat transfer system are connected by pipes, and the reactor vessel only contains the core and associated equipment. Such a system is employed in the Clinch River breeder reactor design, as shown schematically in Fig. 3-9. (Figure 3-10

Net electric power	233 MWe
Thermal power	563 MWth
Date of criticality	4/1973
Fuel material	UO_2 - PuO_2
Enrichment (volume fissile/total volume)	
Central core	19.2 %
Outer core	27.1 %
Blanket	Natural or depleted U
Cladding material	SS 316
Initial critical mass	735 kg ^{239}Pu equivalent
Maximum expected burnup	50,000 MWd/te
Overall breeding ratio	1.16

- A. Control rod mechanism
Mécanismes des barres de contrôle
- B. Fuel transfer arm control
Mécanisme de commande du bras de transfert
- C. Intermediate heat exchanger (6)
Echangeur intermédiaire (6)
- D. Primary circulation pumps (3)
Pompe primaires (3)
- E. Fuel transfer arm
Bras de transfert
- F. Rotating plug
Bouchon tournant
- G. Surrounding neutron shielding
Protection neutronique latérale
- H. Top neutron shielding
Protection neutronique supérieure
- J. Core
Coeur
- K. 21 Suspension points
21 suspentes de cuves
- L. Inner containment
Cuve principale
- M. Double envelope vessel
Double enveloppe calorifugée
- N. Primary containment
Cuve d'enceinte primaire
- O. Conical support collar for reactor
Virole conique support de platelage

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Fig. 3-8. SCHEMATIC REACTOR VESSEL FOR PHENIX, the French demonstration LMFBF; Phenix employs a "pot" arrangement.

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Fig. 3-9. SCHEMATIC PRIMARY COOLANT LOOP FOR THE CLINCH RIVER BREEDER REACTOR, the U. S. demonstration LMFBR (from ERDA-1535).

XBL 766-8024

Fig. 3-10. SCHEMATIC OF TYPICAL LMFBR REACTOR for "loop" arrangement (from ERDA-1535).

shows the type of reactor vessel typically employed in the loop arrangement.) In either arrangement of the primary system, the vessels containing primary components are surrounded by guard vessels, so that any rupture of the primary system does not lead to a large loss of sodium. Finally, in either, the secondary sodium proceeds from the intermediate heat exchanger to a steam generator, which produces steam for driving the turbines.

In any LMFBR, an effort is made to minimize the amount of time the reactor is shut down for refueling. In either the pool or loop arrangement, use is often made of a rotating plug on the reactor vessel closure head. On this plug will be mounted an in-vessel fuel transfer machine, as well as the control rod drives, which are disengaged from the core when the plug rotates, enabling the transfer machine to move fuel from the core to an in-vessel holding point, or vice versa. In the pool arrangement, this holding point may amount to an actual storage drum, and fuel stored therein may be allowed to decay prior to exchange (using an external machine) for fuel outside the vessel. This exchange may be accomplished while the reactor is still running. In the loop arrangement, decay storage must be outside the reactor vessel.

An important question for fast breeders is that of fuel lifetime. In thermal reactors, only a small percentage of the fuel atoms fission before the fuel is removed for storage and/or reprocessing. In fact, since the fissile load is typically less than 4% of the fuel, and conversion ratios are small, it is clear that the "burnup"* is only 2 or 3%. At burnups more than this, the fuel material and the cladding may suffer enough damage to prevent continued use of the fuel. (Moreover, in thermal reactors, one would wish to reprocess to remove neutron poisons.) But the breeder has a fissile loading of 15% or more and the power density (and, correspondingly, the per volume rate at which nuclei fission) is much greater than in water-cooled reactors. For the fuel assemblies to have a sensibly long lifetime in the reactor, the burnup which they can tolerate must be higher, around 10%. The United States has 15% as the maximum burnup that fuel assemblies should stand, corresponding to a thermal energy generation of 150,000 megawatt-days per metric ton of fuel (MWd/Te). The anticipated

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* Burnup is the percentage of heavy metal fissioned (including all fuel materials, both fissile and fertile). It may also be given in terms of the amount of energy released per unit mass.

figure for LWR's is 20,000 to 30,000 MWd/Te for the average assembly. (Average and maximum burnup may differ by up to a factor of two in a given reactor.) The particular consideration with such high burnup is that the stainless steel cladding may begin to swell and distort, thus damaging the fuel rods and even blocking the coolant flow. The relatively compact LMFBR core implies a greater neutron flux through the core structure than typically occurs in thermal reactors, giving rise to the possibility of swelling and alteration of the core configuration over a period of time. Restraints are typically included in the LMFBR core design in order to prevent such changes.

In general, the possibility of changes in the core configuration are of more concern in a fast reactor than in thermal reactors because the fuel in the former case has a substantially higher percentage of fissile material. Changes in core geometry could more easily result in significant changes in the multiplication factor, leading to concern about the possibility of a core disruptive accident.* This concern is aggravated by the fact that development of a void in the coolant (due to, say, boiling) reduces the absorption and moderation of neutrons, both reductions leading to increases in the multiplication factor in a fast reactor. (In a thermal reactor, a void and the resulting decrease in moderation reduces the multiplication factor.) Care must, therefore, be taken that the design prevents significant changes in core geometry and that intrinsic neutronic feedback mechanisms are sufficient to prevent coolant voids, at least those which could propagate to affect large portions of the core. The latter mechanisms are basically intended to provide a decrease in the multiplication factor (or reactivity--see Appendix D) as the temperature rises. An important contributor to this negative temperature coefficient for fast reactors is the "Doppler" effect (discussed in Appendix D) which arises because--as the temperature rises--the effectiveness of resonance capture (without fission) in removing neutrons from the system is increased, so that the fission rate is decreased. Finally, concern is often expressed that the prompt neutron lifetime (the time it takes for a given neutron to be absorbed, thereby producing the next generation neutron) is much shorter for fast reactors than for thermal; this time is

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In case a significant portion of the core should ever melt, the design must prevent the high-fissile-load fuel from reassembling (for example, at the bottom of the reactor vessel) into a critical configuration.

COMPARISON OF DESIGN PARAMETERS (LMFBR)

Design Parameter	FFTF	CRBRP	Commercial and Commercial Prototype
Overall Plant			
Thermal Power, MWt	400	975	3,800
Net Electric Power, MWe	NA	350	1,500
Overall Plant Efficiency, Percent	NA	35.9	39.5
Net Plant Heat Rate, Btu/kwhr	NA	9,507	8,650
Plant Capacity Factor	NA	.75	.85
Number of Primary Loops	3	3	3
Containment Diameter/depth below operating floor/height above operating floor, ft.	135/80/109	186/85/172	168/86/163
Reactor			
Fuel Material	Oxide	Oxide	Oxide
Cladding Material	SS 316	SS 316	Low Swelling/SS 316
Fuel Rod Diameter, in.	.23	.23	.27
Fuel Rod Pitch/Diameter Ratio	1.26	1.26	1.24
Number Fuel Rods/Assembly	217	217	271
Number Core Assemblies	76	198	318
Number Blanket Assemblies	None	150	234
Number Control/Safety Assemblies	6/3	15/4	27/4
Core Height/Diameter, Ft.	3.0/4.0	3.0/6.2	4.0/10.2
Maximum Cladding Wall Temperature, °F	1,170	1,215	1,200
Linear Power Rating, Peak/Ave, kw/ft.	14/7.6	14.5/7	16/11
Peak Fuel Burnup, MWD/T	80,000	150,000	150,000
Fuel Volume Fraction	.329	.325	.342
Breeding Ratio	NA	1.2	1.25
Doubling Time, yrs.	NA	23	12/15
Refueling Type	Offset Arm	Straight Pull	Offset Arm
Vessel Diameter/length, ft.	20.7/43.8	20.7/53.8	23.5/59.5
Primary Heat Transport System			
Reactor Outlet Temperature, °F	1,050	995	1,000
Reactor Inlet Temperature, °F	685	730	725
System Flow Rate, Total 10 ⁶ lb/hr.	18.3	41.5	136.8
Pump Flow Rate at Pump Temperature, gpm	14,500	33,700	122,400
Pump Developed Heat at Design Flow, ft. Na	500	450	500
Pump Location	Hot Leg	Hot Leg	Cold Leg
Intermediate Heat Transport System			
Hot Leg Temperature, °F	950	936	935
Cold Leg Temperature, °F	600	651	650
System Flow Rate, Total 10 ⁶ lb/hr.	19.3	38.3	131.4
Pump Flow Rate at Pump Temperature	14,500	29,500	101,100
Pump Developed Heat at Design Flow, ft. Na	400	410	475
Pump Location	Cold Leg	Cold Leg	Cold Leg

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Table 3-3
(from ERDA-1)

about 5×10^{-7} sec, roughly one-thousandth the comparable lifetime of a thermal-reactor neutron, which spends a relatively large amount of time slowing down. As indicated in Appendix D, this would be of greatest concern if the multiplication factor exceeded 1 by so much that the reactor was critical on prompt (as opposed to delayed) neutrons alone, a condition to be avoided in any type of reactor system. In general, the intrinsic safety aspects of fast reactors differ from those of thermal reactors. As indicated above, some of the differences work to the disadvantage of the designer under some conditions. However, one must note that other features--such as the good heat transfer properties of sodium, and the fact that it can be utilized at low pressure and over a large temperature range--work to the designer's advantage. The brief discussion here cannot adequately describe the complex interplay of various factors important to reactor safety.

The British, French, and Russians already have prototype LMFBR's operating, each with outputs of 250 to 350 MWe. The analogous United States reactor, the Clinch River breeder reactor, is scheduled for completion in the mid 1980's. This will be preceded by the "fast flux test reactor", an important component of the breeder fuel testing program, and--presumably--followed by a prototype commercial scale reactor. Some important parameters for these reactors are given in Table 3-3.

3.2.3. Gas-cooled fast breeder reactor.

The GCFR (Fig. 3-11) has many similarities to the HTGR, except that the core is radically different; the core (Fig. 3-12), in fact, has great similarity to that of the LMFBR, as indicated in Table 3-4. The most substantial difference in cores, more a matter of choice than necessity, is that the blanket material here is ThO_2 rather than UO_2 . As a result, the GCFR primarily breeds ^{233}U instead of ^{239}Pu . This ^{233}U could then be utilized very well in a high conversion thermal reactor (such as the HTGR). As seen in Table 3-4, the demonstration GCFR is claimed to have a substantially larger breeding ratio than the LMFBR demonstration plant. This is attributed, at least partially, to the fact that helium--unlike sodium--does not absorb or moderate neutrons.

The high power density typical of fast reactor cores requires that special steps be taken to insure that the helium coolant of the GCFR successfully cool the core. Designs are incorporated to guarantee that no large amount of the coolant ever escapes from the prestressed concrete

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Fig. 3-11. GCFR PRESTRESSED CONCRETE REACTOR VESSEL ARRANGEMENT (from General Atomic).

XBL 766-8032

Fig. 3-12. GCFR CORE (from General Atomic).

Table 3-4

COMPARISON OF GCFR AND LMFBR DEMONSTRATION PLANTS CORE CHARACTERISTICS

	GCFR	LMFBR ^a
Thermal rating, MW(t)	830	975
Electrical rating, MW(e) net	300	350
Net plant efficiency, %	36.0	36.0
Fuel type	Mixed oxide	Mixed oxide
Primary coolant	Helium	Sodium
Secondary coolant	Steam/water	Steam/water
Intermediate coolant	None	None
Core inlet temperature, °F	613	720
Core outlet temperature, °F	1022	980
Fuel elements		
Rod diameter, in.	0.286	0.23
Cladding thickness, in.	0.019	0.015
Cladding and duct material	C.W. 316 S.S.	C.W. 316 S.S.
Core height, in.	39.2	36.0
Number of rods per element	270	217
Number of fuel and control elements	118	198
Average power density, kwt/liter	240	350
Peak fuel burnup, MWd/T	100,000	80,000
Peak linear power, kW/ft	12.5	15.5
Maximum cladding temperature, mid-wall, °F	1268	1215
Residence time, FPD	750	367 (peak) 506 (avg)
Peak total flux, n/cm ² -sec	6 x 10 ¹⁵	8 x 10 ¹⁵
Fission-gas pressure control	Venting through pressure equal- ization system	Fuel-rod plenum spaces
Radial blanket elements		
Fertile material	ThO ₂ ^b	UO ₂
Number	147	150
Rods per assembly	126	61
Breeding ratio	1.4	1.15

^aR. V. Vijuk, "Review of Core Fuel and Radial Blanket Designs for the LMFBR Demonstration Plant," Meeting at HEDL, October 24-25, 1973.

^bUO₂ is an alternative material.

reactor vessel. The surface of the fuel pins is roughened so that heat transfer characteristics between the cladding and the coolant are improved.

The fact that helium is used as a coolant obviously eliminates the difficulty of handling sodium. An intermediate cooling loop is no longer necessary. Moreover, the helium, because it is essentially transparent to neutrons, cannot cause changes in reactivity (as the sodium could if a void developed in the coolant). The fact that helium does not moderate neutrons means that the core can be constructed in a slightly less compact fashion to accommodate any radiation-induced swelling of the fuel assemblies.

(On the other hand, a sodium-cooled reactor has a large amount of sodium surrounding the core, which can absorb heat in case pumping capability is lost for some reason. Natural convection of sodium can even be relied upon to help to some extent. In the GCFR, however, forced cooling must always be provided.)

3.2.4. Molten salt breeder reactor.

Relative to the other reactors we have discussed, the molten salt breeder reactor (MSBR) is the least developed. Test reactors have run and, until 1977, there has been a place for the MSBR in the AEC or ERDA budget. However, at the present time, no government agency or commercial vendor is seriously developing the concept. We discuss it here because the concept is often mentioned in the context of advanced systems, and it may offer the highest breeding ratio in a thermal system.

The concept of a fluid reacting material is radically different from the other systems which we have discussed. In such a reactor, the "core" is simply that point in the primary loop where a critical mass assembles. This point occurs, not surprisingly, in the reactor vessel. Fluid is withdrawn from this vessel for purposes of extracting heat for generating power and for reprocessing purposes, i.e., to remove from the system neutron poisons that would reduce the breeding ratio. A basic conception of the MSBR is shown in Fig. 3-13. The reactor fluid is fissile uranium and fertile thorium as tetrafluorides, dissolved in a carrier salt of lithium and beryllium fluoride. At the indicated heat exchangers, heat is transferred from the primary fluid to a sodium fluoroborate fluid which runs through the steam generators. Molten salt enters at the bottom of the reactor vessel and is pumped upward in a once-through arrangement. The internals of the vessel are defined by graphite moderator elements. These direct the flow patterns and also determine the fuel-moderator composition throughout the vessel. In

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Fig. 3-13. SCHEMATIC MSBR SYSTEM (from ERDA-1535).

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one concept, the central portion was 19% salt (as opposed to graphite), the next 17%, and the outer 44%; the salt was, measured in moles, 68% LiF, 20% BeF₂, 12% ThF₄, and 1/3% ²³³UF₄. The resulting breeding ratio was estimated to be 1.05.

A basic requirement of this reactor is that methods for extracting various chemicals to prevent deterioration of the breeding ratio be found. It is particularly important to prevent buildup of protactinium 233, the intermediate step in the decay of ²³³Th (formed by ²³²Th + neutron) to ²³³U. If ²³³Pa, which has a 27-day half-life, captures a neutron, not only is a neutron wasted, but a ²³³U nucleus that would have been formed is not. This double loss must be avoided by rapid processing directly from the salt. The proposed solution is a relatively delicate liquid-bismuth extraction process. Basically, liquid bismuth comes into contact with the reactor salt, and relatively different chemical potentials should cause a favorable balance in the passage of uranium, protactinium, and fission products into the bismuth fluid (a delicate matter indeed).

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MISCELLANEOUS UNITS, CONVERSIONS, AND EQUIVALENCES

Energy-Related:

1 MeV = 1 million electron volts (eV) = 1.52×10^{-16} Btu = 1.60×10^{-13} watt-sec
or joule

1 Btu = 1 British thermal unit = 1.05×10^3 joules

1 kilowatt-hour (kWh) = 3413 Btu = 3.6×10^6 joules

1 megawatt-day (MWd) = 8.2×10^7 Btu = 8.6×10^{10} joules

Typical fission energy yield = 200 MeV = 3.0×10^{-14} Btu $\approx$ 0.1% of ^{235}U mass*

1 gigawatt = 1000 megawatts, 1 megawatt = 1000 kilowatts, 1 kilowatt = 1000 watts

1 watt = 1 joule/sec

Mass-Related:

1 atomic mass unit (amu) = $\frac{1}{12} \times ^{12}\text{C}$ mass = 1.66×10^{-24} gm = 931.44 MeV*

neutron mass = 1.0087 amu = 1.67×10^{-24} gm = 939.5 MeV*

proton mass = 1.0073 amu = 1.67×10^{-24} gm = 938.2 MeV*

electron (beta particle) mass = $\frac{1}{1836}$ proton mass = 0.511 MeV*

^{235}U mass = 235.04 amu

1 metric ton (Te) = 2205 lb = 1000 kg

1 pound (lb) = 454 gm = 0.454 kg

1 ton = 2000 lb = 907 kg

Temperature Equivalence:

Centigrade temperature: C

Kelvin temperature: $K = C + 273$

Fahrenheit temperature: $F = \frac{9}{5} C + 32$

Rankine temperature: $R = \frac{9}{5} C + 492 = F + 460$

Other:

1 inch (in.) = 2.54 cm

1 foot (ft) = 30.48 cm = 0.3048 m

1 cubic foot = 28.3 liters = 2.83×10^4 cm³ = 0.0283 m³

1 gallon = 3.78 liters

atmospheric pressure $\approx$ 14.7 psi = 14.7 lb/in.²

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*Mass-energy equivalence.

GLOSSARY

- absorption, neutron--any reaction in which a free neutron is absorbed by a nucleus, including capture and fission
- blanket--see core
- breeding--production of fissile material (usually in excess of that consumed) from fertile material
- breeding ratio--ratio of number of fissile atoms produced to number of the same kind that are consumed
- burnup--the percentage of heavy metal (i.e., fuel) atoms fissioned or the thermal energy produced per mass of fuel
- capacity factor--ratio of average plant power output to rated output
- capture, neutron--a reaction in which a nucleus absorbs a neutron (and may emit a gamma ray, but does not fission)
- chain reaction, nuclear--the sequence of reactions in which neutrons, the products of one fission reaction, induce subsequent fission reactions
- condenser--heat exchanger in which steam is transformed into (liquid) water by removing heat (and transferring it to a cooling river, pond, etc.)
- containment--a structure designed to contain the products (primarily radioactive) of any abnormality
- control--neutron absorbing mechanisms used for maintenance of the multiplication factor (and hence of the chain reaction) at the desired level
- control rods--neutron absorbing rods which constitute an important form of control for any commercial reactor
- conversion--process whereby fertile material is changed to fissile material
- conversion ratio--the ratio of fissile atoms produced to fissile consumed
- coolant--the fluid which removes the nuclear-generated heat from the core (in an LWR the coolant is also the moderator)
- core--the region of a reactor containing the nuclear fuel. It is in this region that the nuclear reactions occur, with the exception of those caused outside the core by escaping radiation or radioactivity. Whereas "core" is usually used to refer to all the fuel assemblies, it often refers more particularly--in the case of breeders--to those assemblies with substantial fissile loading. However, this may also be called the "seed" region, as opposed to the "blanket", which is intended principally to contain fertile material in which fissile material can be generated by neutrons escaping from the core or seed region.
- critical mass--the minimum mass for a barely critical chain reacting system of specified composition, assuming criticality is possible for that composition
- critical size--the minimum size to just yield criticality for a specified core composition and shape, assuming criticality is possible for that composition
- criticality--a condition where the number of neutrons (or fissions) in a chain reacting system is the same from one generation to the next
- cross section--a measure of the probability of interaction between two particles for specified initial conditions and results

decay, radioactive--the process by which a nucleus of one type transforms into another, accompanied by emission of radiation, example: $^{239}\text{U} \rightarrow ^{239}\text{Np} + \beta$

decay heat--the heat produced by radioactive decay of materials which are primarily remnants of the chain reaction

delayed neutrons--neutrons emitted after radioactive decay, and hence which appear later (thereby making control easier) than the reaction which produced the decaying nucleus. The fraction of neutrons which are delayed is given by β .

doubling time--the time during which a reactor system produces excess fuel equal to the system inventory

emergency core cooling system--any engineered system for cooling the core in the event of failure of the basic cooling system. May include core sprays, injectors, etc.

enrichment--the percentage of fuel atoms that are fissile or the process of increasing this percentage

eta(η)--the number of neutrons produced per neutron absorbed in a specified fuel material

fast neutrons--neutrons which have not been thermalized, i.e., with energy comparable to their energy immediately after production from fission

feed water--water (usually from a condenser) supplied to replenish the water inventory of any component (such as a steam generator)

fertile nucleus--one which may capture a neutron to form a product which eventually decays to become a fissile nucleus

fissile load--the percentage of fuel atoms in the reactor that are initially fissile or the total initial mass of fissile material in the reactor

fissile nucleus--one which can be induced to fission by a slow neutron

fission--the splitting of a heavy nucleus to form two lighter "fission fragments", as well as less massive particles, such as neutrons

fissionable nucleus--a nucleus which can be induced to fission by a neutron (of unspecified energy)

fuel--basic chain reacting material, including both fissile and fertile materials

fuel pellet, rod (pin), cladding (coating), assembly (bundle, element)--basic form in which fuel is present in solid-fueled reactors

half-life--the time during which half of the nuclei in a sample of a particular radioactive nuclide will decay

heat rate--inverse of plant efficiency, usually given in units of Btu/kWh

"high gain"--refers loosely to reactors with high conversion ratios

hot (cold) leg--the portion of a coolant circuit (usually primary) through which the coolant exits (enters) the site at which it is heated (usually the reactor)

inventory, fuel--the fuel (or amount thereof) contained in a reactor and, possibly, associated storage, reprocessing, etc.

linear power--usually refers to power generated per unit length of fuel rod

moderator--any material chosen to slow neutrons by elastic collisions

multiplication factor--the ratio of neutron (or fission) number in succeeding generations of a chain reaction

neutron poison--any non-fuel material which absorbs neutrons. Some poisons are used for control

nuclear steam supply system (NSSS)--the basic reactor and support equipment, plus any associated components necessary to produce the steam that drives the turbines

nuclide--a single nuclear type, specified by proton and neutron numbers (the sum of which is the mass number, A)

plant efficiency--ratio of net electrical energy produced to total thermal energy produced

power density--the power generated per unit volume (usually of the core)

pressurizer--maintains the primary coolant circuit of a PWR within specified limits

primary coolant system (or loop)--the entire circuit through which the fluid which actually cools the core passes, including all piping, vessels, and components, such as the reactor vessel, coolant pump, steam generators, etc.

prompt neutrons--neutrons which appear immediately after the fission (or other) reaction responsible for their production

pump (circulator)--a device which forcibly circulates a fluid, the most important example being the coolant

quality--the percentage, by weight, of the cooling water that is present as steam

reaction, nuclear--an interaction between two (or more) nuclei, nuclear particles, or radiation, possibly causing transformations of nuclear type, examples:
 ${}^1_0\text{n} + {}^1_1\text{H} \rightarrow {}^1_1\text{H} + {}^1_0\text{n}$ (elastic scattering, the basic mechanism for neutron moderation), ${}^{238}_{92}\text{U} + {}^1_0\text{n} \rightarrow {}^{239}_{92}\text{U} + \gamma$ (capture)

reactor--the core and its immediate container in which the chain reaction takes place

reactor vessel--the container of the nuclear core or critical assembly; may be a steel pressure vessel, a prestressed concrete reactor vessel (PCRV), a low-pressure vessel (such as a calandria or sodium pot), etc.

reflector--material placed around the core (and blanket, if any) to reflect escaping neutrons back into the core; the reflector, therefore, decreases neutron loss and critical size

scram--the rapid shutdown (via introduction of neutron absorbers) of the chain reaction

slow neutrons--neutrons at approximately thermal energy

steam generator (boiler)--a heat exchanger in which steam is produced by heat transferred from the primary coolant or a secondary fluid

thermal energy--can refer either to heat or to the average energy of particles in thermal equilibrium with (i.e., having the same temperature as) their surroundings; at room temperature, thermal energy is about 1/40 eV

turbogenerator--device in which steam (or some other gas) is allowed to expand, thereby doing work which drives an electrical generator

APPENDIX A. REACTIONS, CROSS SECTIONS, AND MODERATION

The most basic process occurring in a nuclear reactor, maintenance of the chain reaction, is nuclear, as opposed to "atomic" or "chemical" in nature. It depends on interactions between atomic nuclei or "parts" thereof for its continuation. The energy made available from fission events is due to the differences in nuclear potential energy (or, let us say, mass) of the initial and final particles. Moreover, in present-day reactors, the basic initiator of fission events, the neutron, is slowed down by nuclear collisions, thereby increasing its probability of inducing fission.

The most fundamental physical law to which a nuclear reactor "owes" the possibility of its existence is the conservation of energy. This must be thought of in its most general terms, i.e., one must include the fact that an amount of mass has an energy equivalence, $E = mc^2$, where c is the speed of light. Prior to this century, it was well understood that in a collision between two or more particles, if the outgoing particles differed from the incoming, energy conservation still worked, provided one took account of the modification of the internal energies of the particles. Only in this century has it been recognized that this modification would manifest itself as a change in mass. The result is that energy conservation may often be seen as the statement that the change in the kinetic energy of the system is balanced by its change in mass. As an example, take the following reaction:

(where the 200 MeV could manifest itself as gamma rays, neutrinos, kinetic energy and, perhaps, production of some electrons). Energy conservation is implicit in this as: the mass-energy of the initial particles equals that of the final particles + 200 MeV.

Granting energy conservation, much more could be said about a reaction such as this. Other quantities, such as linear momentum and angular momentum, would be conserved. More importantly for our purposes, we can describe in some precise way the probability of various outcomes of some initial condition.

Consider an interaction between two particles and assume, for the moment, that both particles survive the "collision", but that their state of motion may be altered. The total probability of such interaction is measured by

the elastic cross section σ_e , where "elastic" means the particles themselves are not changed. (Historically, it meant that no kinetic energy was lost in the collision, but as discussed above, this must be true if the particles are unchanged and no new particles are produced.) Such a scattering event may be specified in more detail. One basic specification is the direction in which one of the particles is scattered, relative to its original direction. This angular dependence must be considered in some detail to arrive at some of the results on neutron moderation given below.

The normalization for cross section is a very natural one, consistent with its interpretation as an effective collision area. Consider an artificial case in which we have N targets, each with an area σ , and a pointlike projectile which interacts with a target if (and only if) it strikes one of those areas σ . If we direct our projectile at an area A throughout which the N targets are dispersed, the probability of interaction is clearly the ratio of $N\sigma$ to A . For more general kinds of interactions, the "cross section" σ is defined so that the interaction probability is still $N\sigma/A$. (If, then, we have a current of projectiles i , the rate at which interactions occur is $r = i\sigma N/A$.) This same definition applies to other possible outcomes of an interaction than elastic scattering, such as:

- (1) The two interacting objects may effectively "stick" together as in neutron capture, where a nucleus absorbs a neutron, giving off a gamma ray in the process. The cross section corresponding to capture is σ_c .
- (2) Two or more different particles may emerge from the interaction. The most important example of this, for reactors, is fission, where the cross section is labelled as σ_f . Another possibility in this class would be $n + \text{nucleus} \rightarrow n + n + \text{nucleus}$.

For a single initial state, the sum of cross sections for all outcomes other than elastic scattering is the absorption cross section, which for a neutron-nucleus interaction is made up of many parts, including the two given above: $\sigma_a = \sigma_f + \sigma_c + \dots$. The balance between σ_f and σ_c is an important matter for any nuclear fuel (see Appendix B).

These cross sections have an important dependence on energy. For example, the cross section for fission of ^{235}U is given schematically in Fig. A-1. The relatively large cross section at low energy is a reason for "moderating" neutrons down from the MeV-range energies they have on emergence from fission events.

This moderation is achieved by including (as is done in "thermal" reactors) a large amount of low-mass material with which neutrons can collide, thereby reducing their energy. The most effective material for accomplishing this is hydrogen, because its nucleus (one proton) has essentially the same mass as a neutron. It can, therefore, accept a large portion of the neutron energy as recoil, whereas a heavy (that is to say, massive) nucleus can only accept a small amount of energy in an elastic collision. This can be seen by considering head-on elastic collisions (essentially "billiard-ball" collisions) for two types of target. If the target has the same mass as the projectile neutron, in a classical head-on collision the neutron will stop, giving up all its energy to the target, which then proceeds with the same velocity the neutron originally had. At the other extreme, if the target is much heavier than the neutron, the neutron simply bounces backward, and its new energy is only slightly less than its old. Although the exact details depend on the neutron energy and angular considerations, it can be shown that to thermalize neutrons many more collisions are necessary with heavy targets than with light. To be more precise, we may express the result in terms of logarithms, i.e., that the average decrease (per collision) in the natural log of the neutron energy is,

$$\xi = (\ln E_{\text{initial}} - \ln E_{\text{final}})_{\text{ave}} \approx 1 + \frac{(A - 1)^2}{2A} \ln \frac{A - 1}{A + 1} ,$$

where A is the nucleon number of the target nucleus. For $A = 1$ (a hydrogen nucleus, i.e., a proton), ξ is actually 1; for $A > 1$, it is approximately $2/(A + 2/3)$. In an average (not necessarily head-on) collision with a proton, a neutron would be reduced to $1/2.72^{\text{th}}$ of its initial energy. More to the point, to reduce the neutron energy the eight orders of magnitude from, say, 2 MeV (the typical energy of a fission neutron) to thermal energy (1/40 eV) requires

$$\frac{\ln(2 \times 10^6) - \ln(1/40)}{1} = 18$$

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Fig. A-1. SCHEMATIC FISSION CROSS SECTION FOR ^{235}U .

collisions with protons. The corresponding numbers for some other interesting elements are deuterium 25, helium 43, beryllium 86, carbon 114, uranium 2172.

A more useful measure of the effectiveness of a moderating material would fold in the cross section for the collision and the number density of the scattering centers in the material. The "slowing down power" is such a quantity and it is $S = \sum_i N_i \sigma_{ie} \xi_i$, where i corresponds to the i^{th} type of atom in the moderating material, N is the number per unit volume, σ_e the elastic cross section, and ξ the average logarithmic decrease. S is an average logarithmic decrease per unit length that the neutron traverses.

The slowing down power still does not take account of the possibility of absorption of the neutron by the moderator; a moderator that robs the system of many neutrons can be a great disadvantage. A useful quantity is the "moderating ratio" $S/\sum_i N_i \sigma_{ia}$, where σ_{ia} is the absorption cross section of the i^{th} constituent. In a sense, this is the ratio of the slowing down power to the absorptive power.

These last two quantities are strongly energy dependent, since they involve cross sections which depend on energy. However, suitably weighting them, they can be useful. Some results given in Table A-1 are illuminating. There it is seen that heavy water, because it rarely absorbs neutrons, has a much larger moderating ratio than other materials. (Hence the ability of the CANDU to sustain a chain reaction for long periods of time with only natural uranium as a fuel.)

Table A-1. Slowing down properties of moderators.*

Moderator	Slowing Down Power (cm ⁻¹)	Moderating Ratio
Water	1.28	58
Heavy Water	0.18	21,000
Helium**	10 ⁻⁵	45
Beryllium	0.16	130
Graphite	0.065	200

* From Glasstone and Sesonske (see Chapter 1 bibliography).

** At atmospheric pressure and temperature.

APPENDIX B. CHARACTERISTICS OF NUCLEAR MATERIALS

The essential constituent of a nuclear reactor is the fissionable material contained in its core. The energy available from the reactor is basically determined by how many fissions* occur in the course of the chain reaction; continuation of the chain reaction depends on the neutrons which become available, principally as a result of fission.

Of the fissionable nuclides in Table B-1, those whose neutron-induced fission threshold is zero (^{235}U , ^{239}Pu , ^{233}U) are referred to as "fissile", while the others are "fertile", since neutron capture onto them can lead, via the conversion sequences indicated there, to a fissile nuclide. However, even the fertile nuclides in the table are subject to "fast fission", i.e., fission induced by neutrons of energy above the indicated non-zero threshold. Note that the "fissile" materials are those which can serve as the nuclear fuel in a reactor where the neutrons produced are very rapidly "thermalized" to low energies (see Appendix A).

All of the above materials are radioactive. For the fissionable nuclides listed, this is not relevant to their behavior in a reactor, since their half-lives are so long (see Table B-2), but can affect externalities, such as the danger they present to humans. However, the radioactive character of intermediate nuclides in a conversion sequence can be of great importance, since they lead to more useful (i.e., fissile) nuclides. For example, protactinium 233, one of the intermediate stages in the thorium cycle conversion, has a long enough half-life that the probability that it captures a neutron before decaying to ^{233}U may cause some difficulty in achieving high conversion ratios. Table B-2 contains information on the most important nuclides occurring in reactors.

Having noted that the neutrons generated by fission can either produce more fissions (by interacting with a fissionable material) or produce fissile material (by being captured by a fertile material), we present the basic information about neutron production, and fission and capture probabilities. Table B-3 gives the cross-section for thermal (read "slow") neutron-induced fission and capture on the indicated nuclides. For the fissile materials,

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* At an average of about 200 MeV per fission (or 77 million Btu per gram of fissioned ^{235}U or 22 thousand kWh (thermal) per gram of ^{235}U). Most of the energy is released as kinetic energy of the fission fragments, which typically stop in the fuel itself, thus heating up the fuel material. Of the energy carried off by other particles or radiation ($n, \alpha, \beta, \gamma, \nu$) due to fission or subsequent decays, only the energy associated with the neutrinos--a few MeV, on the average--escapes from the reactor.

Table B-1. Fissionable nuclides.

Nuclide	Fission Threshold (Neutron Energy in MeV)	Average Energy Available from Thermal Fission (MeV)
Fertile ^{232}Th	1.4	---
Fissile ^{233}U	0	198
Fissile ^{235}U	0	202
Fertile ^{238}U	0.6	---
Fissile ^{239}Pu	0	210

Conversion Sequences (half-lives are given in Table B-2):

"Plutonium Cycle" $^{238}\text{U} + n \rightarrow ^{239}\text{U} \xrightarrow{\beta} ^{239}\text{Np} \xrightarrow{\beta} ^{239}\text{Pu}$

"Thorium Cycle" $^{232}\text{Th} + n \rightarrow ^{233}\text{Th} \xrightarrow{\beta} ^{233}\text{Pa} \xrightarrow{\beta} ^{233}\text{U}$

(Both cycles involve uranium.)

Table B-2. Radioactivity of nuclear materials.

Nuclide	Activity	Half-Life*
$^{232}\text{Th}^{**}$	α	1.41×10^{10} yr
$^{238}\text{U}^{**}$	α	4.51×10^9 yr
$^{235}\text{U}^{**}$	α	7.1×10^8 yr
^{233}Th	β	22.2 min
^{233}Pa	β	27.0 days
^{233}U	α	1.62×10^5 yr
^{239}U	β	23.5 min
^{239}Np	β	2.35 days
^{239}Pu	α	2.44×10^4 yr
^{232}U	α [+ γ rays]	72 yr

* Lederer et al, Table of Isotopes (6th Edition), Wiley, NY, 1967.

** Occur in substantial quantities in nature.

^{233}U , ^{235}U , ^{239}Pu , a small capture/fission ratio is an advantage because neutrons captured onto them are lost. The fact that ^{232}Th has a higher capture cross-section than ^{238}U accounts in part for a higher conversion ratio in the thorium than in the plutonium cycle because of the greater probability that the thorium cycle conversion sequence (see above) is initiated.

Another primary consideration is the number of neutrons produced per fission, ν , given as the first column of Table B-4. These neutrons are the currency of the reactor's economy. However, since--in the neutron economy--we want to know what return we get on a neutron absorbed (including those lost to capture) we can correct ν , using the $\frac{\text{capture}}{\text{fission}}$ ratio, to obtain η , the number of neutrons liberated per neutron absorbed thermally. We also give the equivalent number for a fast (unmoderated) neutron spectrum. For thermal reactors, note that--of these fissile nuclides-- ^{233}U liberates the most neutrons per neutron absorption. Since thorium is also relatively easily converted to ^{233}U , this accounts roughly for the advantage of the thorium cycle over the plutonium.

For the fast breeder reactor, ^{235}U is clearly the poor cousin of either ^{233}U or ^{239}Pu , since it yields just barely more than two neutrons per neutron absorbed, and a greater excess is required, considering losses, if we are to (1) continue the chain reaction (one neutron), (2) produce a new fissile nucleus (a second neutron), and (3) have an excess left over to breed extra fissile materials.

Actually, although the numbers in Table B-4 do indicate qualitatively the differences between fissile fuels, they are not enough to yield conversion ratios because

(1) they only indicate results from fission due to thermal and fast neutrons, neglecting neutrons of intermediate energy. Thus, although the thermal values for η given above are roughly correct for $E_n = 0.01$ to 1.0 eV for ^{233}U and ^{235}U , η is noticeably less than 2.0 for most of this region in the case of ^{239}Pu ,*

(2) they do not account for the relative importance of neutron capture by fertile material as opposed to the reactor structure or coolant.

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*Thermal as used here is 1/40 eV.

A more complete representation of the behavior of η with energy is given in Fig. B-1. Note that although we have only spoken of "thermal" reactors and "fast" reactors, ^{233}U has a behavior that would actually enable it to function well at intermediate energies.

Table B-3. Thermal neutron cross-sections.*

Nuclide	Fission (barns)	Capture (barns)	$\frac{\text{Capture}}{\text{fission}}$ ratio
^{232}Th	-	7.4	-
^{233}U	527	54	0.102
^{235}U	577	106	0.184
^{238}U	-	2.7	-
^{239}Pu	742	287	0.387

* See Appendix A for discussion of cross section.

Table B-4. Neutrons liberated.

Nuclide	ν , Per Thermal Neutron Induced Fission	η , Per Thermal Neutron Absorbed	η , Per Fast Neutron Absorbed
^{233}U	2.50	2.27	2.60
^{235}U	2.43	2.06	2.18
^{239}Pu	2.90	2.10	2.74

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Fig. B-1. NEUTRON YIELD PER NEUTRON ABSORBED (η) VS ENERGY OF ABSORBED NEUTRON (from ERDA-1541).

APPENDIX C. THERMAL EFFICIENCY AND COOLING

Although a large light-water reactor generates heat at the rate of about 3000 megawatts, the electrical output of the plant is only 1000 MWe, representing a conversion efficiency of 33%. The fact that this is so much less than 1 reflects primarily a basic limitation to the efficiency of "heat engines", devices which convert thermal energy to mechanical energy.

A heat engine can be idealized as withdrawing heat from a large reservoir at a fixed temperature T_1 and, operating in cyclic fashion, converting a portion of that energy to work, while rejecting the remainder to a second large reservoir at a temperature T_2 , lower than T_1 (see Fig. C-1). For a device operating over a range of temperatures, a combination of these idealized devices may be used to represent the system. The efficiency of our engine is given by the ratio W/Q_1 , where Q_1 is the amount of heat withdrawn from the high-temperature reservoir and W is the work performed by the engine. By conservation of energy, $W = Q_1 - Q_2$, where Q_2 is the heat rejected to the low-temperature reservoir. These comments assume, clearly, that the internal energy of the heat engine itself does not change, at least from one cycle to the next. (Any cyclic device satisfies this requirement, since, at the end of each cycle, it has returned to its state at the end of the previous cycle.)

figure C-1

It is a basic question in thermodynamics to inquire what is the best possible efficiency for an engine operating between temperatures T_1 and T_2 . One is led to the answer by considering a "Carnot" engine, one which uses a working fluid on a cycle with four parts. Expressed in terms of a gas confined by a piston, (1) the gas is allowed to absorb heat Q_1 at a constant temperature T_1 (the gas will expand, doing work on the piston), (2) it is allowed to expand further, but with no heat transfer (i.e., adiabatically), until its temperature decreases to T_2 , (3) the gas then gives up heat Q_2 at a constant temperature T_2 until the volume decreases just enough that, (4) the gas is compressed adiabatically until its temperature rises to T_1 and it

returns to its original volume. In the first two stages, the gas does work on the piston, in the second two, the piston does work on the gas. The difference is the net work $W = Q_1 - Q_2$ and it can be shown that for an "ideal" gas, $W = (1 - T_2/T_1) Q_1$.^{*}

It can further be shown that no heat engine can achieve a better efficiency than the Carnot result, $W/Q_1 = 1 - T_2/T_1$. Note that this efficiency approaches 1 only as T_2 approaches absolute zero or T_1 becomes very large. A reactor which is generating heat in a core whose temperature is 600°F (1060°R) and which ultimately rejects heat to a 70°F (530°R) river has a Carnot efficiency of $1 - \frac{530}{1060} = 0.5$. The only way--in principle--to improve this is to lower the temperature of the cooling river or raise the temperature at which the reactor operates. But on earth the rejection temperature can never be much lower than 500°R (300°K). And it is hard to imagine operating a water-cooled reactor at higher than 600°F, where the pressures required are on the order of 2000 psi. The situation can be improved by using a gaseous coolant--or even metal--at higher temperature. Steam is not appropriate because at high temperatures it is very corrosive of core materials. Helium or CO₂ are the usual gaseous alternatives--see HTGR, Chapter 2.

There are two obvious reasons for which the Carnot efficiency is not achieved. One is the rather general fact that actual operating systems include turbulence, friction, and temperature differentials across system components that preclude adherence to any idealized cycle. A second is directly pertinent to systems that include the boiling of water, since the boiling and condensation that occur in a steam cycle are reasonably well described as constant pressure processes (isobars), rather than the constant temperature (isothermal) processes required for the Carnot cycle. The steam cycle, therefore, is usually approximated by the Rankine cycle, which is described as follows (in terms of water taken from the condenser after the turbine):

- (1) the water from the condenser, which is at a relatively low pressure, is subjected to adiabatic compression to reach the pressure of the boiler;
- (2) in the boiler, which operates at a relatively fixed pressure, the water

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^{*}For this result to hold, the temperatures T_1 and T_2 must be given on an "absolute" temperature scale, one where 0° specifies absolute zero. The absolute scale corresponding to the Celsius degree is the Kelvin scale (on which water freezes at 273°K); corresponding to the Fahrenheit degree is the Rankine scale (0°F = 460°R).

is heated isobarically to produce steam, which is heated even above the boiling temperature; (3) the steam is expanded adiabatically (in an expansion chamber, the turbine); and (4) the low pressure steam from the chamber undergoes isobaric isothermal condensation to form water which then begins the cycle again. This cycle cannot achieve the theoretical efficiency available for the Carnot cycle.

For both of the reasons above, the achievable efficiency is reduced below that suggested by the expression $1 - T_2/T_1$. On implementation, the efficiencies available in water-cooled reactor power plants are 30 to 33%, less than from fossil-fired generating plants, where the fuel is burned at higher temperatures. These efficiencies are often expressed in an alternative form, the "heat rate" (actually the reciprocal of the efficiency), which gives the amount of thermal energy required per electrical energy produced, usually in units of Btu/kWh. (One can then easily show that typical heat rates are about 9000 to 11,000 Btu/kWh; an efficiency of 1 would give 3413 Btu/kWh.)

APPENDIX D. CRITICALITY AND CONTROL

Chapter 1, Appendix A, and Appendix B discuss some of the basic notions of reactor physics. However, the idea of criticality has only been suggested. It is clear that for maintenance of a given power level, an essentially steady-state condition, as many neutrons must be "born" as are absorbed. The relative size of the rates at which neutrons appear and are absorbed is expressed in terms of the "multiplication factor" k for the reactor. k may be defined as the ratio of the number of fissions in one generation to the number in the previous generation. (It may also be defined as the ratio of the number of neutrons in succeeding generations.) For a steady-state condition, k must equal 1, in which case the reactor is said to be "critical". $k > 1$ implies a growing fission rate (a "super-critical" condition), $k < 1$ a decreasing rate.

As discussed in Chapter 1, the short neutron "lifetime", order of 10^{-4} sec in a thermal and even less in a fast reactor, would make reactors quite unstable were it not for the fact that a small fraction of the neutrons from any generation are "delayed". These neutrons are not produced directly in fission events. Instead they occur when the decay of a fission product (or possibly a fission product daughter) leaves some nucleus in a state that is unstable to neutron decay. Though this neutron decay, if it occurs, will proceed promptly, it will have been delayed by the time it took the parent to beta decay. A reactor at steady power is subcritical on prompt neutrons, but just critical when the delayed neutrons are included. The speed of any change in the power level is determined essentially by the generation time of the delayed neutrons, provided the reactor remains subcritical on prompt neutrons. Conversely, if the reactor goes "prompt critical", the advantage offered by the presence of delayed neutrons is lost.

The decay of any radioactive species can be characterized by a "half-life" $t_{1/2}$, the time it takes for half of the nuclei in a sample to decay. (Alternatively, one can express the same information in terms of the decay constant: the rate of disintegrations in a sample will be proportional to the number of atoms remaining, N . Therefore $\frac{dN}{dt} = -\lambda N$, λ being the "decay constant". It is straightforward to show that if N_0 atoms are present at time $t = 0$, the number at any time thereafter is $N = N_0 e^{-\lambda t}$; and further $\lambda t_{1/2} = \ln 2 = 0.693$.) The time dependence of appearance of delayed neutrons makes it clear that they have a number of

of different precursors (fission products leading to these neutrons) with a range of half-lives. Historically, these neutrons have been fit into six relatively well-defined groups, corresponding to half-lives from 0.2 to 56 sec. Measurements of delayed neutrons from fission of various nuclides has shown, though, that a given group has a half-life that varies slightly from one nuclide to another. Table D-1 gives the measured half-lives of the six groups for thermal-neutron fission of ^{233}U , ^{235}U and ^{239}Pu . The observed variation in half-life indicates that the groups are due to more than one precursor, with slightly differing half-lives, and that fission of each possible nuclide populates the precursors differently. Most, but not all, of the precursors have been identified.

Table D-1 also gives the neutron yield for each group. The most important single number, for each nuclide, is the total delayed fraction β , the ratio of delayed neutron yield to total neutron yield from fission. For ^{235}U , β is 0.0065; it is smaller for ^{233}U and ^{239}Pu . For both uranium cases, the fraction is somewhat larger for fast-neutron-induced fission. ^{239}Pu is unchanged. Delayed neutrons can also arise from fertile materials. In the case of ^{239}Pu -fueled fast reactors, this contribution to the total delayed fraction is quite important, raising the total β significantly higher than the 0.2% from ^{239}Pu alone. In any case, the delayed-neutron fraction is sufficiently large to prevent the very rapid changes in neutron population suggested by the prompt neutron lifetime (see Chapter 1). As a result, rather slowly-acting means of control, insertion of rods, addition of chemicals, etc., are sufficient to maintain a constant or slowly changing power level. This is particularly true since, in a power reactor, k is never permitted to be much different from 1.

In passing, we should note that there is a situation where the delayed neutrons would have no practical effect, and that is a condition of "prompt criticality". As is suggested by the name, this indicates that the multiplication factor exceeds 1 without counting the delayed neutrons. There is then no fundamental restraint on a rapid increase in the number of neutrons.

Although we have avoided the concept so far, "reactivity" is a useful term in this context. The reactivity ρ of a reactor is defined to be the ratio $(k - 1)/k$ and is a quantity which occurs regularly in equations describing the neutron population. A reactor that is just critical, then, has $\rho = 0$. Prompt criticality occurs when $\rho = \beta$. It is, therefore, important that ρ be kept less than β . We will return to this question below.

Table D-1. Delayed neutron data for thermal fission in ^{233}U , ^{235}U , and ^{239}Pu .*

Group	^{233}U		^{235}U		^{239}Pu	
	Half-Life (sec)	Yield (Neutrons per Fission)	Half-Life (sec)	Yield (Neutrons per Fission)	Half-Life (sec)	Yield (Neutrons per Fission)
1	55.00	0.00057	55.72	0.00052	54.28	0.00021
2	20.57	0.00197	22.72	0.00346	23.04	0.00182
3	5.00	0.00166	6.22	0.00310	5.60	0.00129
4	2.13	0.00184	2.30	0.00624	2.13	0.00199
5	0.615	0.00034	0.610	0.00182	0.618	0.00052
6	0.277	0.00022	0.230	0.00066	0.257	0.00027
Total Yield:		0.0066		0.0158		0.0061
Total Delayed Fraction (β):		0.0026		0.0065		0.0021

* Adapted from Lamarsh (see Chapter 1 for reference).

It is worth considering in slightly greater detail what contributes to the multiplication factor. Appendix B discusses one very important aspect of this question, i.e., η , the number of neutrons liberated per neutron absorbed by the fuel. However, there are other possible ends to a neutron, even a thermal neutron; and a fission-produced neutron is far from thermal--there are many hazards twixt one fission event and the next. So let us follow what happens when n thermal neutrons are absorbed by the fuel. For the sake of argument, let us assume that the fuel is low-enrichment uranium. The concepts will be more generally applicable.

First, the η of Appendix B was only defined for a pure material. We want to apply it to whatever we call "fuel". This will include fissile and fertile material, and may include other constituents such as the cladding. ν is the number of neutrons liberated per fission. η is, then, just ν multiplied by the ratio of the probability of fission to the total probability of neutron absorption in the fuel. If we specify the first probability by $N_{\text{fissile}}\sigma_f$, the second is $\sum_i N_i\sigma_{ia}$, where N_i is the number density for the i^{th} constituent of the fuel and σ_{ia} is the corresponding cross section for neutron absorption. (See Appendix A for discussion of cross sections.) We calculate η for the thermal neutrons available and obtain

$$\eta = \nu \frac{N_{\text{fissile}}\sigma_f}{\sum_i N_i\sigma_{ia}}$$

And the absorption of n neutrons by the fuel yields $n\eta$ fission neutrons.

Before the $n\eta$ fast neutrons can be thermalized, some of them will be absorbed by the fuel, in which case they may induce fission, thus increasing the number of neutrons. Most of such events will be induced in ^{238}U , provided the neutron energy is still above ^{238}U 's fission threshold. The number of neutrons slowing down past this threshold is then $n\eta\epsilon$, where ϵ is the "fast-fission factor", the meaning of which is obvious. η depends on the fuel, but ϵ depends on all aspects of the core. Its definition is more operational than fundamental. (It has a value within a few percent of 1 for thermal reactors.)

In continuing to slow down to thermal energies, some neutrons will be absorbed (without--for the most part--causing fission), particularly by sharp resonances. (An example of such resonances, but for fission rather than

absorption, is given in Fig. A-1.) The portion of neutrons escaping such an end is p , the "resonance escape probability", so that the total number of neutrons reaching thermal energies is $n\eta p$. Of these, only a fraction f will be absorbed by the fuel, so that--whereas n thermal neutrons of the previous generation were absorbed by fuel-- $n\eta p f$ of the present generation are so absorbed. Therefore, the multiplication factor $k = \eta p f$. (We have avoided the question of finite reactor size and consequent leakage of neutrons from the system. To be precise, we would specify f to be the ratio of thermal neutrons absorbed in the fuel to those absorbed by all reactor materials, and $\eta p f$ would really be the multiplication factor for an infinite system. k for a finite system would then be obtained by multiplying $\eta p f$ by a "non-leakage probability".)

The resonance escape probability is a very complicated quantity, and it is in fact temperature dependent. One such dependency is called the "Doppler effect", a name suggested by the fact that the fuel nuclei are in thermal motion; these thermal energies clearly depend on the temperature. Consider the effect of an absorption resonance on the spectrum of neutron energies. The resonance robs neutrons from an energy region comparable in width to that of the resonance. If the neutron resonance is broad, the number of neutrons absorbed by the resonance could be obtained by integrating (over the resonance) the product of neutron flux* times absorption cross section, where we could use for the flux the expected value in the absence of the resonance. We can do this because the flux at any given value of energy is only slightly modified if the resonance is broad. If, however, we sharpen the resonance considerably, but let the resonance keep the same strength--i.e., integral of the cross section--the resonance will significantly decrease the flux at the resonance energy, so that fewer neutrons will be absorbed, since the integral of flux x cross section will be less. Thus a very sharp resonance masks itself. There are many very sharp resonances through which the neutrons must pass. The thermal motion of the fuel nuclei,

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* The neutron flux at a point in space is basically the product of the number of neutrons (of the energy being considered) per unit volume and the speed of these neutrons. If all these neutrons were moving in the same direction, the result is simply the rate at which neutrons pass through a unit area perpendicular to this direction; this is consistent with the ordinary usage of the word "flux".

however, broadens the resonances, decreasing this masking effect, and thus effectively decreasing the total number of neutrons. An increase in temperature, therefore, decreases k , assuming that the temperature change does not affect the fission rate via other mechanisms. Hence the Doppler effect may constitute a negative "reactivity insertion" with increasing temperature (usually measured in terms of a "Doppler coefficient"), and is an important intrinsic safety characteristic, particularly for fast reactors.

Loss of coolant in a water-cooled reactor also decreases k because the neutrons are less effectively moderated. A void in the coolant of the fast reactor may increase k . Such reactivity insertions are to be avoided. Often, reactivity is measured in terms of "dollars" where one dollar corresponds to β , the delayed-neutron fraction. For example, individual control rods are often "worth" slightly less than one dollar, indicating that a single control rod could not be responsible for a catastrophic reactivity insertion. The effects of consuming the fissile fuel, producing or burning neutron poisons, change in the coolant density, etc., can all add or subtract reactivity and, therefore, are liable to measurement in dollars.

One of the most important fission product poisons is ^{135}Xe , formed from the beta decay of ^{135}I . ^{135}Xe is a very strong neutron absorber, so that in a strong neutron flux it is destroyed primarily by neutrons. At any specified reactor power level, the amount of ^{135}Xe will be established by a balance between xenon production and its removal by neutron absorption and by beta decay of the xenon. If the power level is decreased suddenly, ^{135}Xe will continue to be formed by beta decay of its parent, and its concentration will increase for a time because of decreased neutron absorption. If an attempt is then made to bring the reactor back up to power, the effect of elevated levels of ^{135}Xe on the neutron population will have to be overcome. Until the previous balance is re-established, excessive numbers of neutrons will be lost to xenon, so that--to keep a multiplication factor of 1--excess reactivity will have to have been built into the reactor. For "excess reactivity" read "extra fissile". For normal reactor operation, this excess reactivity must be masked by control, one more reason for its incorporation into the reactor system. One alternative is to wait out the xenon transient, i.e., wait until the ^{135}Xe decays away; this can result in a substantial shut-down period, because ^{135}I and ^{135}Xe have half-lives of 7 and 9 hr, respectively, and, therefore, the concentration of xenon increases rather slowly before

finally decreasing. Another alternative is to build in only enough excess reactivity to overcome xenon provided the reactor is returned to power soon after the power decrease, and thus before xenon has grown in very much.

^{135}Xe is also a noticeable steady-state poison; at typical neutron fluxes, it may absorb a few percent as many neutrons as are absorbed by ^{235}U . This usually exceeds the total absorption by all the other fission products.

APPENDIX E. THE NUCLEAR FUEL CYCLE

Production of nuclear fuel and its disposition after use involve a number of important steps, which together constitute the "fuel cycle" depicted in Fig. E-1. The processes at the "front" end of the cycle, from uranium acquisition to fuel fabrication, are reasonably well established on a production basis, a necessary feature of the commercialization of nuclear power. Establishment of many of these facilities, from uranium mines through enrichment plants, were required by military programs. With the sale of numerous nuclear plants to utilities, both reactor vendors and other businesses have established commercial fuel-fabrication plants, aided by experience with making fuel for military and research reactors.

The incentives for operation of other parts of the fuel cycle, while not as overwhelming as the need for producing fuel, are still of great interest. One is to dispose of fuel which, after residence in a reactor for a period up to several years, contains a large amount of radioactivity. The second is to remove from the fuel assemblies nuclear materials which are still useful, particularly any remaining fissile material. Some experience has been gained from reprocessing of fuels from AEC (now ERDA) programs and from operation, for a time, of the Nuclear Fuel Services plant in New York. However, no commercial reprocessing plant is presently operating; nor has the mode for disposal of high-level radioactive wastes been chosen. As a result, the fuel cycle is incomplete.

The fuel cycle which we describe below is for the light-water reactor system. Its basic structure, however, would be similar for any system, provided obvious modifications are made (see last section of this appendix).

Fuel Cycle Operations

The highest grade uranium deposits in the United States occur primarily in the Colorado plateau, including parts of Arizona, Colorado, New Mexico, and Utah. Ores from these deposits usually contain less than 1% uranium, so that substantial processing is required, largely in a mill near the mine site, to obtain any relatively pure compound of uranium. Although it is possible to achieve some separation of uranium by physical methods, the basic scheme is to crush the ore, then to leach out the uranium by mixing the crushed ore with an acidic or alkaline solution (sulphuric acid is often used). The uranium that goes into solution is then extracted by a sequence of chemical operations (such as ion exchange or solvent extraction, followed

Fig. E-1. NUCLEAR FUEL CYCLE

by precipitation and drying) which leave it in the form of U_3O_8 concentrate with a purity of about 70%. This product of the milling operation is sealed in drums for transport to the UF_6 conversion plant. An important remnant of the milling operation is a "tailings" pile (see below).

It is next necessary to purify the U_3O_8 and produce uranium hexafluoride (UF_6), the compound which is used in the enrichment plant. It is particularly important to remove any impurities that would reduce the quality of the uranium as a nuclear fuel, especially those isotopes with large neutron-absorption cross sections. The purification is achieved by an extraction process, the product of which is oxidized to UO_3 . This is converted by treatment with hydrogen, hydrogen fluoride, and fluorine to UF_6 , a solid at room temperatures, which is shipped to gaseous diffusion plants for enrichment.

At the diffusion plant, use is made of the fact that UF_6 containing ^{235}U (the fissile isotope) has a lower molecular weight than that containing ^{238}U . At thermal equilibrium, the lighter molecules will have the same average energy as the heavier, and, therefore, will be somewhat faster. When heated to a gaseous form, the lighter molecules, because of their greater speed, strike any containing walls more often, and, therefore, will diffuse through a porous barrier slightly faster than the heavier. The enrichment plant consists of a large number of such barriers in a "cascade" where, at a given stage, the enriched product proceeds to the next higher stage, while the depleted remnant recycles to the next lower. In principle, a cascade can be designed long enough to yield a product arbitrarily pure in ^{235}U . In practice, the highest commercial grade of uranium is 93% ^{235}U , which would be used in the HTGR. LWR's require 2 to 4% enrichment. Natural uranium contains 0.7% ^{235}U .

UF_6 with the desired enrichment is shipped to fuel fabrication plants. A typical process line would then expose the UF_6 to ammonium hydroxide, yielding ammonium diuranate which is dried, then reduced (using hydrogen) to UO_2 . This is made into a powder which (for water-cooled reactors) is pressed into a pellet, then sintered (heated) to form a ceramic. This pellet is ground, finally, to cylindrical shape, then incorporated into fuel pins (Chapter 2) and assemblies.

In U. S. LWR's, fuel remains in the reactor for about 3 years of operation, a portion of the fuel being replaced each year. On removal, the fuel is stored in water pools, which provide cooling and shielding, for several months; during this time, the fuel's radioactivity decreases substantially, due to decay of short-lived species. The fuel, cooled and shielded by special shipping casks, would then be transferred to a fuel reprocessing plant, were such now available.

At the reprocessing plant, hardware is removed from the assemblies, the pins are chopped up and placed in nitric acid, which dissolves the fuel. The remaining cladding hulls are relegated to waste storage. Uranium and plutonium are removed from this acidic solution by solvent extraction, leaving the high-level wastes, consisting primarily of fission products (in acid solution). The plutonium may then be separated by another solvent extraction, and the uranium purified by processes similar to those at the UF_6 conversion plant (appropriate, since one of the products most desired would be UF_6 for return to the enrichment plant).

A full-scale commercial reprocessing plant is now nearing completion at Barnwell, SC. This plant has a through-put of 1500 metric tons of heavy metal (uranium and products) per year, equal to the output of about 50 LWR's. A major question in licensing of this plant will be uncertainty in handling of plutonium, particularly to prevent its dispersion and/or illegal acquisition. In a complete LWR cycle, the uranium would be returned to enrichment plants, the plutonium to fuel fabrication facilities (see Fig. E-1).

The high level liquid wastes are an acidic, highly radioactive solution which can be consolidated somewhat by evaporation, but ultimately must be reduced to a solid. Methods are available for "calcining" these wastes to a dry, granular solid which can be stored on an interim basis in stainless steel bins or capsules. But the ultimate disposal that is most favored is a (silicate) glass which would be deposited in geologic formations. The details of this are not settled, nor has there been any need due to the lack of any commercial reprocessing. Moreover, this method has not been fully demonstrated and particularly not to the satisfaction of groups critical of nuclear power.

Environmental Releases

The most significant releases of radioactivity from routine fuel cycle operations occur at the mining and milling site, the nuclear power plant itself, and the fuel reprocessing plant. A flowsheet for the LWR fuel cycle is shown in Fig. E-2.

The releases from the mine site have a different origin than from other facilities, being due to daughters of ^{238}U already present in the ore when it was mined. In particular, radium 226--to which ^{238}U decays in several steps--decays by alpha emission to radon 222, a gas which becomes available to human ingestion (in spite of the fact that it is noble) because it decays to ^{218}Po , which is so chemically active that it immediately attaches itself to particulates in the air, assuming the radon has escaped to the atmosphere. Daughters of ^{218}Po may then remain attached. In any event, they are available for collection by human respiratory membranes on breathing, and the various decays in the sequence can thereby expose humans to radioactivity. This very source is one component of natural radioactive background, and the population group which shows the effect (high lung cancer rates) of breathing elevated levels of radon and its daughters are uranium miners. Other groups may also suffer from elevated levels because of the presence of the tailings pile at the mill site. This makes a substantial quantity of ^{238}U -daughters available to the open atmosphere, a fact that is significant when tailings have been used for landfill under homes and buildings, or for other purposes.

Radioactivity released from reactor and reprocessing sites is generally produced in the course of the nuclear chain reaction or subsequently. The most significant routine releases by far are the gaseous species tritium (^3H), produced by fission or formed by neutron capture on deuterium, and krypton (^{85}Kr) or xenon (^{135}Xe), noble gas fission products. The tritium becomes incorporated in water, and is thus subject to routine release from an LWR. The noble gases are to some extent released to the coolant, from which they could routinely escape to the atmosphere, but more significantly, ^{85}Kr would ordinarily be released at the reprocessing plant unless the steps necessary to contain a noble gas were taken. The ^{85}Kr would cause the largest routine exposure to humans, presuming it were all released. (But for a 1000-reactor system operating worldwide, it would still only be 1% of natural background.)

Fig. E-2. FUEL CYCLE MATERIAL AND ENVIRONMENTAL RELEASE FLOWSHEET for typical 1000 MWe light-water reactor nuclear power plant (from "Comprehensive Standards . . .").

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It should be noted that all of the routine releases above are of gaseous products or isotopes (tritium) which are chemically identical to a major volatile constituent of the reactor system. Accidental releases, however, can involve a broader range of materials simply because accident conditions may make other escape routes available. Barriers against release concentrate on two general classes of material: the fuel material itself (uranium and plutonium) and by-products of running nuclear reactors. (This is not an unambiguous distinction since plutonium and some isotopes of uranium are produced in reactors, and portions of the material produced would end up in the waste stream at the reprocessing plant.)

Fuel materials occur throughout the processes at the front end of the fuel cycle and continue to the reprocessing plant. For the plutonium recycle case, plutonium occurs from fabrication to reprocessing. (For a thorium-based cycle, ^{233}U would replace plutonium in this part of the cycle.) The fabrication process, the reactor, transportation, and reprocessing must all be regarded as possible sites for plutonium releases, and probabilities must be assigned to each release sequence.

Control of the large amount of radioactivity generated in the reactor is the purpose of the reprocessing and waste-management program. The most carefully-analyzed sites for accidents making these materials available to the general environment are the reactor itself, a point where much energy is available to initiate releases, the fuel reprocessing plant, where the radioactivity is freed from the fuel assemblies, and the disposal site, where the wastes must be contained for long periods of time.

Modifications for "Advanced" Reactor Systems

Some obvious changes to the fuel cycle of Figs. E-1 and E-2 would be necessary were converter reactors based on thorium or breeder reactors to come into use. As one example, consider a converter where the initial feed is ^{235}U plus thorium as the fertile material. Eventually, most of the fissile feed is ^{233}U generated from thorium, but this uranium does not require enrichment (since it would not be "diluted" by ^{238}U). Thus ^{233}U effectively replaces plutonium in the fuel cycle flowsheet; but any ^{235}U that is supplied has to be highly enriched (although this might not be necessary in some schemes). Moreover, an auxiliary mining and milling branch must be established to supply thorium to the fabrication plant, and thorium recovered from reprocessing would be recycled (probably after holdup to

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allow certain species to decay). Additional changes in handling of recycled material would be necessary because of penetrating radiation associated with a daughter of the ^{232}U that is produced in small amounts in the reactor.

A breeder reactor system as presently envisioned would differ from the LWR fuel cycle in one striking way: because the fissile material is primarily plutonium, no enrichment of uranium feed is necessary. In fact, the ^{235}U -depleted enrichment tails would be used. Effectively, then, all of the breeder fuel would be LWR by-products or material recycled in the breeder fuel cycle.

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