11 Nuclear Reactors

To most people, nuclear weapons and reactors are probably the only radiation source that comes to mind when exposure to radiation is discussed because of the connection between nuclear reactors and nuclear weapons, the publicity accompanying the accidents at Three Mile Island and Chernobyl and fear of the unknown. Nuclear reactors may have accidents, however, since they produce a great proportion of the U.S. and world electrical power, it is important that you know something about nuclear reactors to properly assess the risks associated with them.

11.1 Basic Physics of Nuclear Reactors

A neutron will only interact with the nucleus. When a neutron collides with a nucleus, the neutron may be either absorbed or scattered. The probability a particular neutron-nucleus interaction will occur depends upon both the nucleus (or nuclide) and the energy of the neutron (Table 11-1). As seen in Figure 11-1, the absorption of a thermal or slow neutron is much more probable than the absorption of a fast neutron, but the probabilities vary greatly.

The probability of a particular reaction occurring between a neutron and a nucleus is the (microscopic) cross section, \( \sigma \), of the nucleus for the particular reaction. The microscopic cross section can be thought of as the effective area which the target nucleus presents to the neutron for a particular interaction. The larger the effective area, the larger the probability of the interaction. For example, consider a small ball thrown at a larger ball. If the larger ball has a radius, \( R \), the effective area it presents to the small ball is \( \pi R^2 \). This particular area is the geometric cross section of the larger ball. The probability that the small ball will strike the larger ball increases as the radius (and hence the geometric cross section) of the large ball increases. Although neither the neutron nor the nucleus is a ball, we can imagine the nucleus as presenting an effective area, or cross section, to the neutron. The larger this cross section, the larger the probability that the neutron will interact with the nucleus. However, it is possible for the microscopic cross section to be much larger than the geometric cross section of the nucleus. The cross section or probability of interaction is expressed in units of area (square centimeter). Because the nuclear diameter is on the order of \( 10^{-12} \) cm, a nuclear cross section is of the order of \( 10^{-24} \) cm\(^2\). The microscopic cross section is usually expressed in multiples of this area known as barns where 1 barn = \( 10^{-24} \) cm\(^2\).

Nuclear reactors produce energy by nuclear fission (Figure 11-2). In fission, after absorbing a neutron, a large nucleus (\( Z \geq 90 \)) splits into two lighter nuclei called fission fragments (or fission products). When the nucleus splits, neutrons, gamma rays, and a large quantity of energy is emitted. The amount of energy released per fission can be calculated from Einstein's mass-energy equation, \( E = mc^2 \) (i.e., energy = mass x speed of light squared). The mass, called the mass difference (i.e., the "m"'), results from the difference in masses between the fission products and the original atom and neutron.

Although several isotopes are capable of fission (e.g., \( ^{233}\text{U}, ^{235}\text{U}, ^{239}\text{Pu} \)), the primary isotope used in United States nuclear reactors is \( ^{235}\text{U} \). When a \( ^{235}\text{U} \) nucleus captures a neutron, it is transformed into an excited, \( ^{236}\text{U}^* \) nucleus. This capture may lead to one of several possible outcomes: elastic scattering, radiative capture, or fission.

- In elastic scattering, a neutron is captured by \( ^{235}\text{U} \) and re-emitted from the excited \( ^{236}\text{U}^* \) nucleus with no apparent energy loss. Elastic scattering is the most likely interaction between fast neutrons and low atomic number absorbers. Inelastic scattering is similar except the nucleus is left in an excited state (e.g., \( ^{239}\text{U}^* \)).
In radiative capture, the excited $^{236}\text{U}^*$ nucleus loses the excitation energy, $Q$, through the emission of one or more gamma photons, but emits no particles (i.e., no neutrons).

In fission, the unstable $^{236}\text{U}^*$ nucleus splits into two fission fragments (e.g., $^{139}\text{Xe}, ^{94}\text{Sr}$), two or three free neutrons and releases a large quantity of gamma-ray (called capture γ-ray) energy, $Q$.

The amount of energy released in the fission reaction producing $^{139}\text{Xe}, ^{94}\text{Sr}$, and 3 neutrons can be found by summing the masses (in units of atomic mass units, 1 amu = 931.16 MeV) of the nucleons involved on each side of the reaction and calculating the mass difference.

$$
\text{Mass Difference} = 0.193393 \text{ amu} = 180.08 \text{ MeV}
$$

11.1.a  Fission Energy

The fission example above is only one of the many possible outcomes when a $^{235}\text{U}$ nucleus absorbs a neutron. If all possible fission reactions were considered for the unstable $^{236}\text{U}^*$ (the absorption of the 'n' made the nucleus unstable), the resulting fission products would follow the equation:

$$
^{236}\text{U}^* \rightarrow \text{ff}_1 + \text{ff}_2 + 2 \text{ or } 3 \text{ } n + Q
$$

The mean number of neutrons from all the possible $^{236}\text{U}^*$ reactions is 2.5 and the average energy released is about 200 MeV. The approximate energy distribution among the fission products is shown in Table 11-2. In a reactor, most of this energy is dissipated as heat in the fuel rod assembly and this heat energy is used by the nuclear power reactors to boil water and produce electricity. Consider how this (MeV) energy is related to the more common energy unit, watts (i.e., how many fissions does it take to produce 1 watt).

$$
\text{1 watt} \times \frac{1 \text{ Joule}}{1 \text{ watt} \cdot \text{sec}} \times \frac{1 \text{ MeV}}{1.6 \times 10^{-13} \text{ Joule}} \times \frac{1 \text{ fission}}{200 \text{ MeV}} = 3.121 \times 10^{10} \frac{\text{fissions}}{\text{sec}}
$$

It takes about $3.1 \times 10^{19}$ fissions per second to produce 1 watt of thermal energy. Most power generating stations are rated in megawatts (MW), how much nuclear fuel does it require to produce 1 MW of heat energy for one day?

$$
\frac{3.121 \times 10^{10} \text{ fission}}{\text{1 watt} \cdot \text{sec}} \times \frac{1 \times 10^6 \text{ W}}{1 \text{ MW}} \times \frac{8.64 \times 10^4 \text{ sec}}{1 \text{ day}} \times \frac{235 \text{ gm U}}{6.023 \times 10^{23} \text{ atoms}} = 1.06 \frac{\text{gm U}}{\text{MW} \cdot \text{day}}
$$

From a mass standpoint nuclear power is appealing. It usually requires all of the coal taken from an average-size coal mine to provide coal for one average-sized coal-fired power plant. Table 11-3 compares the fuel requirements to produce 1 GW (1-million kilowatt) of electricity (i.e., enough electricity for a city of 560,000 people) by common energy sources.

In the discussion above, we were talking about thermal energy. For power plants, the conversion of thermal energy to electrical energy is about 33% efficient (by comparison, an automobile is probably about 20 - 25% efficient) and it requires approximately 3 MW thermal to produce 1 MW of electricity. Power plants are usually listed by electrical output, not thermal power.
11.1.b Fission Products

The atomic number of the fission fragments range from a Z = 30 (i.e., 72Zn) to a Z = 64 (i.e., 168Gd). Figure 11-3 shows the isotopic yield of fission fragments (of mass number A) per fission. The majority of fission fragments are found in a cluster of two groups around A = 92 and A = 138. All fission fragments are radioactive and, because they have a high neutron to proton ratio, decay by one or more β− emissions before becoming stable. For example, the fission fragment 90Kr emits four β− particles before becoming stable 90Zr.

Similarly, the decay of 137I to stable 137Ba results in the emission of three β−.

The fission products with the highest probability (∼7%) of occurring have mass numbers 95 and 139. Each fission fragment decays at its unique decay constant. The total activity of all fission products is the sum of the exponential decay curves for each product. Empirically, the total fission product activity, A, in the reactor t days after f fissioning events have occurred is:

\[ A = f \times (3.81 \times 10^{-6}) \times t^{-1.2} \text{ Bq} = f \times (1.03 \times 10^{-16}) \times t^{-1.2} \text{ Ci} \]

Thus, we could calculate the fission product activity resulting from 1 MW-day of electrical production. The first step is to determine how many fissions are involved.

\[ 1.06 \text{ g mU} \times \frac{6.023 \times 10^{21} \text{ Atoms}}{235 \text{ g mU}} = 2.72 \times 10^{21} \text{ fissions} \]

From the fission product activity equation, if a reactor operated for 1 day and then shut down, after 1 day the activity would be:

\[ A = (2.72 \times 10^{21} \text{ fis}) \times (1.03 \times 10^{-16}) \times (1^{-1.2}) = 280160 \text{ Ci} \text{ (0.28 MCi or 10.66 PBq)} \]

After 2 days of shut down the activity would be:

\[ A = (2.72 \times 10^{21} \text{ fis}) \times (1.03 \times 10^{-16}) \times (2^{-1.2}) = 122000 \text{ Ci} \text{ (0.12 MCi or 4.51 PBq)} \]

This high fission product activity, called the source term, is the reason safeguards are implemented and also why many people do not consider nuclear power to be a viable energy source.

<table>
<thead>
<tr>
<th>Fuel</th>
<th>σ_in</th>
<th>σ_f</th>
<th>A</th>
</tr>
</thead>
<tbody>
<tr>
<td>235U</td>
<td>578.8</td>
<td>531.1</td>
<td>2.43</td>
</tr>
<tr>
<td>239U</td>
<td>680.8</td>
<td>582.2</td>
<td>2.42</td>
</tr>
<tr>
<td>239Pu</td>
<td>1011.3</td>
<td>742.5</td>
<td>2.87</td>
</tr>
<tr>
<td>241Pu</td>
<td>1377</td>
<td>1009</td>
<td>2.93</td>
</tr>
</tbody>
</table>

11.1.d Nuclear Fuel

Fission can be produced in several heavy nuclides under different conditions. Thermal neutrons are neutrons in thermal equilibrium (∼68°F) with the nuclear material and have an average kinetic energy of a few hundredths of an eV (E ∼ 0.025 eV). At that energy, the fission cross section for 235U and 239Pu is about 500 barns. However, 238U is not fissionable by thermal neutrons. Neutrons emitted from fission initially have energies in the MeV range. Fast neutrons (i.e., E > 0.1 MeV) can fission 235U, 239U, Th, and Pa. Nuclides which are fissionable by thermal neutrons are called fissile. While there are several fissile nuclides, few are of sufficient abundance with a long enough half-life to be of practical value. The most viable fissile nuclides are 233U, 235U, 239Pu, and 241Pu. Table 11-4 gives the total (σ_in) and thermal (σ_f) neutron cross section in barns (10−24 cm²) of several important fissile isotopes and the average number of neutrons produced per fission (A). Note that the ratio of the radiative capture to fission cross-section (i.e., [σ_in - σ_f]/σ_f) is low for 233U, 235U, and 239Pu, giving a higher probability for fission. While 235U is the major nuclear fuel used in the United States, only 0.72% of natural uranium is 235U (i.e., only 1 in 139 atoms); the 238U isotope comprises 99.27% of all the natural uranium.

However, if nuclear power were based solely on 235U the world’s resources of 235U will not be sufficient to meet the future demand for nuclear fuel. The solution is to manufacture certain fissile isotopes from abundant, non fissile isotopes, a process called conversion. In conversion, non fissionable isotopes capture a neutron, are converted to different isotopes and by the process of radioactive decay a fissionable isotope is produced (Figure 11-4). Isotopes...
which are themselves not fissile, but from which fissile isotopes can be produced are said to be fertile. The two most important fertile isotopes are $^{232}\text{Th}$ and $^{238}\text{U}$. Fissionable $^{235}\text{U}$ is obtained from thorium by the absorption (i.e., radiative capture) of a neutron.

$$^{232}\text{Th} + \frac{1}{0} \text{n} \rightarrow ^{233}\text{Th} - \frac{\beta}{22.2 \text{ m}} \rightarrow ^{233}\text{Pa} - \frac{\beta}{27.4 \text{ d}} \rightarrow ^{233}\text{U}$$

Similarly, although $^{238}\text{U}$ is not readily fissionable by thermal neutrons, it is important in the production of $^{239}\text{Pu}$ by the radiative capture a neutron creating $^{239}\text{U}$ which decays to $^{239}\text{Pu}$:

$$^{238}\text{U} + \frac{1}{0} \text{n} \rightarrow ^{239}\text{U} - \frac{\beta}{23.5 \text{ m}} \rightarrow ^{239}\text{Np} - \frac{\beta}{2.35 \text{ d}} \rightarrow ^{239}\text{Pu}$$

The efficiency of the conversion process is quantified by the conversion ratio, $C$, the average number of fissile atoms produced in a reactor per fissile fuel atom consumed. When the conversion ratio is equal to one (i.e., $C = 1$), an infinite amount of fertile material can be converted starting with a given amount of fuel. When the conversion ratio is greater than one (i.e., $C > 1$), more than one fissile atom is produced for every fissile atom consumed, a special process called breeding. A measure of breeding efficiency is the doubling time, the length of time it takes the breeder to produce twice as much fissile material (i.e., double) as there was originally. For a $^{238}\text{U}$ breeder reactor the doubling time is normally on the order of 6 to 9 years.

Although the United States had embarked on an experimental breeder program during the 1970s, the political and economic climates were not conducive to continuing the program. Consequently, all U.S. power reactors use $^{235}\text{U}$ as fuel. The problem with using natural uranium as fuel is that the amount of $^{235}\text{U}$ in natural uranium is too low and the fuel must be enriched; that is the fuel has to have a greater percentage of $^{235}\text{U}$ than the natural ratio of 0.72%. In commercial reactors, the fuel is enriched to about 3% to 6% (depending upon type of moderator) while in some small reactors (e.g., submarines) it may be highly enriched (i.e., 90% $^{235}\text{U}$).

### 11.2 Neutron Cycle

To achieve a self-sustaining nuclear reaction, or criticality, the rate of neutron production must be at least equal to the rate of neutron loss. In a thermal reactor (i.e., one using thermal neutrons for fission), $^{235}\text{U}$ is the primary fuel. Normally, $^{238}\text{U}$ only fissions with neutrons having energies greater than 1 MeV (i.e., fast neutrons), and even then the probability that a fast neutron will fission $^{238}\text{U}$ is relatively low. Fast fission of $^{238}\text{U}$ accounts for only a few percent of the total fissions even though $^{238}\text{U}$ may compose 95% of a reactor's fuel.

Producing power by nuclear fission is not easy. Not all neutrons produced in the fission process are available to sustain the process. Some diffuse away from the fuel region and are lost. Some that remain in the fuel region undergo non-fission (i.e., radiative) capture by the fuel and materials used for reactor construction. Production and losses are illustrated in the neutron cycle, Figure 11-5. Upon the fissioning of $^{235}\text{U}$, the fast neutrons released may cause $^{238}\text{U}$ atoms to fission producing additional neutrons or the fast neutrons may strike the moderator. The moderator reduces the neutron's energy through elastic scattering.
to thermal or near thermal energies. At these energies they are most likely to cause further fissioning of $^{235}$U. To reduce the loss of neutrons from the system, one can increase the size of the system or reflect some of the neutrons back into the core using a reflector. The minimum quantity of fissile material necessary to maintain a chain reaction is called the critical mass. The critical mass for a given reactor depends on a wide range of factors, although for a specific reactor design it will always have a definite value.

The ratio of neutrons available for fissioning in any one generation to the number available in the preceding generation is called the effective multiplication factor, $k_{eff}$, and is calculated by:

$$k_{eff} = \frac{\text{number of fissions, one generation}}{\text{number of fissions, preceding generation}} = \frac{N_f}{N_{f-1}}$$

That is, $N_f$, the number of neutrons produced in the current generation, divided by the number of neutrons in the preceding generation, $N_{f-1}$.

For the chain reaction to be sustained in a steady state, $k_{eff}$ must be 1. In this case the system is said to be critical, and the number of neutrons available for further fissioning balances those lost through leakage or capture. When $k_{eff} < 1$, the chain reaction is not maintained and the system is sub-critical, more neutrons are lost than are being produced. For $k_{eff} > 1$, a surplus of neutrons is being produced in each generation causing more fissions than the previous generation and the system is super-critical. In reality, $k_{eff}$ depends on the supply of neutrons of proper energy to initiate fission and the availability of fissile atoms. However, $k_{eff}$ is often approximated from the four factor equation which assumes an infinite ($\infty$) system. This equation is based on neutron losses and is expressed by:

$$k_{\infty} = \eta \cdot e \cdot p \cdot f$$

The reproduction factor, $\eta$, ($\eta \geq 1$) is the average number of neutrons emitted per thermal neutron absorbed in the fuel (i.e., $\nu$ in Table 11-4, is the average number of neutrons emitted per fission as opposed to per neutron absorbed in the fuel). The average number depends on the fuel type, enrichment, etc. For each fission of $^{235}$U by thermal neutrons, about 2.5 neutrons are emitted; however most of the fuel is $^{238}$U, so for most reactors $1 < \eta < 2.1$.

The fast fission factor, $e$, is a ratio of the total number of fission neutrons produced by both fast and thermal fission divided by the number of neutrons produced by thermal fission alone. This factor accounts for the percent of neutrons which produce fissions in $^{239}$U and contribute neutrons from that reaction. In most reactors $1.0 \leq e \leq 1.29$.

The resonance escape probability, $p$, is the probability that a neutron will not be absorbed by nuclei having resonances above the thermal region before it becomes thermal. It is the fraction of fast neutrons that finally become thermalized or the probability that the neutron will escape capture and become thermal. It depends on the amount and type of moderator and the fuel type. It is desirable to have $p \approx 1$. For pure, unmoderated, natural uranium $p = 0$, so natural uranium cannot become critical unless it is moderated. Depending upon enrichment, $0.8 \leq p \leq 1$.

The thermal utilization factor, $f$, is the ratio of thermal neutrons which cause fissions in $^{235}$U to the total number of thermal neutrons which are absorbed by the system. As only 84% of the thermal neutrons absorbed by $^{235}$U cause fissions, this factor accounts for the nonproductive thermal neutrons. It is desirable that $f \approx 0.85$.

In reality, some neutrons escape a finite system, the four factor equation is modified by the addition of $L$, the non-leakage factor, with $L < 1.0$, thus $k_{eff} = k\cdot L = (\eta e p f) L$. Regardless, only $\eta$ is dependent on the fuel. The other factors; $e, p,$ and $f$, depend on composition, physical arrangement, moderator type, and homogeneous dispersion of the fuel in the moderator.

### 11.3 Reactor Design and Radiation Hazards

The need to keep $k_{eff} \geq 1$, to reduce the number of neutrons escaping from the reactor, to convert heat energy into electric energy, and to keep the radiation out of the environment dictate the inclusion of several common types of components in the basic reactor design (Figure 11-6).

#### 11.3.a Reactor Core

The core consists principally of the fuel, moderator, and structural material. Because of design and operating difficulties inherent in fuels consisting of naturally occurring uranium, enriched uranium (contains a higher percentage of $^{235}$U than the 0.72% which occurs naturally) is used. The degree of enrichment depends on the design features of the reactor or vice versa. The advantages of fuel enrichment are:

- High fuel burn-up is achieved; that is a larger percentage of the $^{235}$U nuclei are fissioned so the fuel can be more efficiently used.
A wide variety of materials (coolant, moderator, construction material, etc.) is possible in reactor design.

Higher power densities (power output per core volume) is obtained with enriched uranium (see 11.1.d).

The fuel is formed into rods or plates to improve heat removal and encased in a protective cladding to contain the fission products, prevent chemical reactions between fuel and moderator and provide structural support. Several physical factors considered for cladding materials are: low neutron capture probability, structural strength at high temperatures, good heat transfer, and non-corrodible characteristics. Commonly used cladding materials that meet these requirements are aluminum, zirconium, alloys of the two, and stainless steel.

11.3.b Moderator and Reflector

Slow neutrons have a higher probability of producing fission in $^{235}$U than do fast neutrons. Neutrons emitted from the fission of $^{235}$U have a wide spectrum of energies from 0.025 eV to approximately 7 MeV. Because the reactor needs slow neutrons, a moderator is used to slow (or moderate) the neutrons down and enhance the fission process. On the average, neutrons lose more energy per elastic collision with particles of equal mass (i.e. hydrogen nuclei) than they do in colliding with heavier particles. For example, it takes less than 20 collisions to thermalize a neutron using ordinary water as a moderator, but more than 100 collisions with graphite. For this reason, materials with low atomic weight (generally hydrogen or hydrogenous compounds) are used for moderators.

In the moderation process, some of the neutrons may be scattered at angles which project or reflect them back toward where they came from (i.e., toward the core). Thus, some moderating materials may also be suitable reflectors which serve to reduce neutron leakage. Usually such reflector materials are of low mass number may be interspersed among the fuel elements where they serve as a moderator and, when placed outside the reactor core area they can serve as a neutron reflector. An important criteria is that the material used for the reflector and moderator have a low probability for neutron capture. Most commonly used reflector and moderator materials are:

- **Water** - Because of its hydrogen content, water serves as a good moderator and reflector. Since water is also cheap, it is often used in reactors in a dual role as moderator/reflector and coolant. A disadvantage is its relatively low boiling point requiring expensive high pressure vessels and piping for high temperature operations. Also, since water has a relatively high thermal neutron absorption cross section, the fuel in a water moderated reactor must be enriched with $^{235}$U to offset the additional neutron loss. Most US reactors, both pressurized and boiling water, use water.

- **Heavy Water** - Of all the moderators in use, heavy water (D$_2$O) has the lowest neutron absorption to thermal ratio. It is a very good moderator and coolant, but high production and purity maintenance costs limit its use. The Canadians often use this type reactor.

- **Graphite** - This is easily purified, machined, and relatively inexpensive and it has a very low thermal neutron capture cross section. Its brittleness causes difficulties when it is used as a structural material and graphite reacts with oxygen at high temperatures. As contrasted to water reactors, graphite reactors still need an additional coolant to remove reactor heat.

- **Beryllium / Beryllium Oxide** - These are good moderators but are relatively expensive. Their disadvantages are that they are brittle and react with air or the moisture in air. They are also very toxic and therefore hazardous to fabricate so, they are normally only used in special applications.

11.3.c Coolant

The great quantities of heat produced in the reactor core must be removed to prevent the fuel elements from melting. In a power reactor, the core heat is used to make steam which may turn a turbine-generator to produce electricity or, in ships to turn propellers. The cooling system removes the core heat by circulating a heat absorbing material through the core. The heat generated in the fuel elements is transferred to this coolant and circulated out of the core. Desirable coolant materials should have a low probability for neutron capture, have good heat transfer capabilities, and be easy to move through the core. The most commonly used coolant materials are:
Water / Heavy Water - These are both good coolants and good moderators. The chief disadvantages are water's corrosiveness and its low boiling point. Pressurized water reactors solve the latter problem by keeping the water under high pressure (1500 psi) thereby raising its boiling point (500 °F). Boiling water reactors, although pressurized, allow the water to boil and form steam reducing the need for a steam generator (Figure 11-7) between the core and the turbine-generator.

Gas - Air was used as the coolant in early reactors. Now gas-cooled, high power reactors use gases other than air because air reacts unfavorably with structural material in the core. Helium and carbon dioxide have been successfully used as coolants. Gas cooling systems require expensive pumping arrangements unless the gas is kept under extremely high pressure, and gas systems at low pressure have relatively poor heat transfer characteristics.

Liquid Metal - Sodium and sodium-potassium have been used as coolants where neutron moderation was undesirable, such as in fast breeder reactors. These coolant systems solidify at room temperature and must first be warmed up before the reactor is started. Additionally, these coolants tend to be highly reactive with water, hence extensive containment systems are required.

11.3.d Radiation Shielding
Reactor shields may be designed for several functions. Shielding to reduce the radiation exposure to persons in the reactor building is called biological shielding. Neutrons and gamma rays emitted by the fission fragments produced in the fuel elements present the most serious shielding problems. Alpha / beta particles and the recoil fission fragments are generally absorbed by the fuel cladding and other materials used in reactor construction.

Because the probability of neutron capture/removal increases as the neutron kinetic energy decreases (i.e., becomes thermal), a shield for neutrons must necessarily first moderate (i.e., slow down) the neutrons and then remove them through capture reactions. Good neutron moderators are low density materials with a high hydrogen content. Good absorbers for gamma rays are high density materials such as lead or iron. Concrete is a good compromise for shielding against both gamma rays and neutrons from reactors. It contains both low and fairly high atomic weight materials (hydrogen and silicon). Besides good shielding properties, concrete has good structural qualities and is relatively inexpensive. Iron punchings and boron can be added to enhance gamma shielding and neutron capture, respectively. Because of these assets, concrete is the most often used shielding material in reactors.

Water has been used as a shielding material in special applications. Although the shielding properties of water are good, its use presents considerable construction difficulties (e.g., no form). Thus, in a reactor the coolant provides shielding as an added benefit of its cooling role.

11.3.e Control and Safety
The operation of a reactor can be described in terms of the multiplication factor, k_{eff}. Control rods maintain the proper k_{eff} factor for various stages of reactor operation. Control rods are made of materials which have a high capture cross section, removing them from the core region and making them unavailable for further fissioning. The neutron population is controlled by moving the rods in or out of the core region. With precise positioning of these rods, it is easy to maintain the point where k_{eff} = 1 is reached and produce a stable, critical state in the reactor.

Regulating or control rods of cadmium or boron-steel are usually positioned electrically and/or hydraulically. Control rods are classified as either coarse or shim rods. The names refer to their degree of adjustment. Coarse control rods are used for making gross adjustments, while the shim rods are used for making finer adjustments in the number of fission events. Other rods called safety or scram rods, are strategically positioned in the core. In the event of a drastic increase in k_{eff} (i.e., super-criticality), these rods are inserted in the core immediately to shut down the reactor. Control rods may also be used as safety rods.

A unique concept of reactor control is the adding of boron directly to the coolant. The concentration of boron in the coolant is varied for routine control with major reactivity changes controlled by the rods.

11.4 Reactor Classification
Reactors may be classified by many categories, such as fuel-moderator arrangement, type of coolant, reactor use, or a combination of these. The two major types of light water (i.e., H₂O) power reactors used in the U.S., pressurized and boiling water, differ primarily in temperature and pressure within the reactor core.
11.4.a Fuel-Moderator Arrangement
In a **homogeneous reactor**, the fuel and moderator are in contact and intimately mixed with each other. A **heterogeneous reactor** is one in which the fuel is lumped into rods surrounded by a moderator and coolant. Power reactors are heterogeneous reactors.

11.4.b Reactor Use
There are four basic uses for reactors: research, power production, isotope production, and breeder. Some of the salient features of each are:

- **Research** - A reactor primarily used for research, either as a prototype or proving ground for future reactor design or operated to produce neutrons for pure scientific research. A homogeneous reactor would be an example of a research reactor.
- **Power** - Heat produced in the core is removed by the coolant and put through various heat exchanger systems and it is eventually converted to electrical or mechanical energy.
- **Isotope Production** - High neutron fluxes inside the reactor may be used to produce radioisotopes or other products (e.g., colored gemstones, etc.) through neutron capture. One of the more familiar reactions is the production of $^{32}\text{P}$ by the absorption of a neutron by $^{31}\text{P}$, the only naturally occurring isotope of phosphorus (i.e., abundance = 100%).

\[
\begin{align*}
^{31}\text{P} + ^{1}\text{n} & \rightarrow ^{32}\text{P} \\
^{16}\text{O} + ^{0}\beta + ^{+}Q & \text{ (1.709 MeV)}
\end{align*}
\]

Similarly, the radiopharmaceutical most frequently used in Nuclear Medicine (cf. Chapter 13) can be produced by separating $^{99}\text{Mo}$ (abundance = 24.13%) from natural Molybdenum and bombarding the $^{98}\text{Mo}$ with neutrons to produce $^{99}\text{Mo}$. The $^{99}\text{Mo}$ is then placed in a generator (see Figure 13-2) which can be used to elute $^{99m}\text{Tc}$ for diagnostic nuclear medicine.

\[
^{98}\text{Mo} + ^{1}\text{n} \rightarrow ^{99}\text{Mo} \rightarrow ^{99m}\text{Tc} \rightarrow ^{99}\text{Tc} + ^{+}Q \text{ (1.214 MeV)}
\]

- **Breeder / Converter** - In addition to producing energy which may be used for power generation, the breeder reactor produces more fissionable material than it consumes. The reactor may be designed solely to produce fissionable material (e.g., $^{239}\text{Pu}$) which may then be processed and used at another facility.

11.4.c Coolant
Reactors may also be classified by the type of coolant employed to remove the fission energy from the reactor core and produce steam to turn the turbine-generator.

- **Boiling Water** - The system is pressurized, but controlled boiling is allowed to occur in the core. Steam is removed via a steam separator and sent to the turbine-generator or heating system.
- **Heavy Water** - Deuterium Oxide (D$_2$O) is used instead of ordinary water.
- **Pressurized Water** - The coolant system is pressurized to the extent necessary to prevent boiling in the core. Steam is produced in a secondary system (i.e. steam generator) at lower pressures.
- **Liquid Metal** - Various liquid metals are used as coolants, primarily in fast breeder reactors where no moderation of neutrons is wanted.
- **Gas** - Inert gases or air serve as the heat removal material.

Most power reactors in the United States are either pressurized or boiling water reactors. Gas cooled reactors were popular in Europe, however gases have poor heat transfer characteristics and are difficult to pump. At one time the US had investigated breeder reactors, but without fuel reprocessing, a breeding proved to be politically impossible. As mentioned, the Canadians use heavy water in their CANDU (Canadian deuterium uranium) reactor design because it can use unenriched, natural uranium fuel. Natural uranium, graphite moderated reactors were developed in the US during World War II to convert $^{238}\text{U}$ into $^{239}\text{Pu}$ for military purposes and natural-uranium fueled reactors became the starting point for the nuclear power industry. However, the construction of diffusion plants to produce enriched-uranium fuel has resulted in a reduction in natural-uranium fueled systems except in Russia and former Soviet states. There are very few graphite moderated power reactors in the US.
11.5 Power Reactors

Most thermal power reactors work in a similar manner. Fuel is burned to heat water, the water boils and the resultant steam is piped to a steam turbine to spin the turbine blades which drive an electric generator. Any left over energy in this steam is removed and the steam condensed back into water where it is pumped back into the furnace. A nuclear power reactor is nothing more than a steam-electric generating station in which the nuclear reactor takes the place of a furnace and the heat comes from the continuous fissioning of uranium atoms rather than from the burning of fossil fuel. To control the heat production, control rods made of materials which absorb neutrons, are placed among the fuel assemblies. When the control rods are pulled out of the core, more neutrons are available and the chain reaction speeds up, producing more heat. When they are inserted into the core, more neutrons are absorbed, and the chain reaction slows or stops, reducing the heat.

Most commercial nuclear reactors in the United States use water to remove the heat created by the fission process. These are called light water reactors in contrast with the reactors which use heavy water (\(\text{H}_2\text{O}\) or deuterium oxide) to remove heat. As noted in paragraph 11.3.b, the water also serves to slow down, or moderate, the neutrons. In the United States, two different light-water reactor designs are currently in use for producing steam from the heated water, pressurized and boiling water reactors.

11.5.a Pressurized Water Reactor (PWR)

In a pressurized water reactor (Figure 11-7) the heat is removed from the reactor by water flowing in a closed pressurized loop called the primary loop. This loop is kept under high pressure (e.g., 2250 psi) so that the water in the primary does not turn to steam even at temperatures of 600 °F (315 °C). The heat energy of the primary water is transferred to a second water loop (or secondary) in a heat exchanger or steam generator. This secondary loop is kept at a lower pressure, allowing the water to boil and create dry steam (i.e., steam with very little water vapor), which is used to turn the turbine-generator and thus produce electricity. Afterward, the steam is cooled and condensed back into water which is easier to pump and the water is pumped back to the heat exchanger.

The main benefit of the pressurized water reactor design is that the two separate loops of water never physically mix. While there may be some radioactivity dissolved in the primary loop, the addition of the secondary loop helps insure the risk of contamination of the environment is kept low. Additionally, because of the high temperatures used, the water can hold a large quantity of reactor heat. The major disadvantage of the PWR is that the additional loop and pressurizing equipment add significant cost to the system. Also, some efficiency is lost in the steam generator, so a PWR may be a bit less efficient than a BWR in producing electricity.

11.5.b Boiling Water Reactor (BWR)

In a boiling water reactor, Figure 11-8, the water is piped around and through the reactor core and is transformed into steam as it flows between the fuel elements removing reactor heat. The steam leaves the reactor at the top of the reactor vessel and goes directly to the turbine-generator to produce electricity. Here, too, the spent steam is condensed back to water and pumped back into the reactor vessel to continue the process.

At atmospheric pressure, water turns to steam at a temperature of 212 °F (100 °C). But, at such a low temperature, steam contains too little energy to be efficiently used in a turbine-generator. To raise the temperature and the energy content, the water in a boiling water reactor is kept at a pressure of approximately 1000 psi, instead of the
normal atmospheric pressure of about 15 psi. At 1000 psi of pressure, the water does not boil and turn to steam until it reaches a temperature of approximately 545 °F (285 °C). Thus, although the BWR is pressurized, it has fewer components and may be cheaper to construct. However, because the BWR uses only a single coolant loop with the water passing directly through the core, some contamination may be spread to the turbine and other reactor components.

11.6 UW Research Reactor
The Nuclear Engineering Department of the School of Engineering operates a small, research reactor for instruction, training and research. The original reactor was constructed and installed by the Atomic Power Equipment Department of General Electric Company. The present core is composed of TRIGA-FLIP fuel supplied by the General Atomic Company. The reactor achieved initial criticality on 26 March 1961 and its original maximum steady state power level was 10 kW. This level was increased to 250 kW on 7 December 1964 and later up_graded to the present maximum steady state power level of 1,000 kW on 14 November, 1967. Although the reactor is capable of a steady state power of 1 MW and is normally operated at this level for approximately 7 hours twice each week, it is capable of producing pulses of approximately 1000 MW. When operating at a steady state power of 1 MW, the reactor produces approximately \(3.2 \times 10^{13}\) thermal neutrons per cm\(^2\) per second (fluence) and the 1000 MW pulse produces a neutron fluence approximately 1000-times as intense. Because this type of reactor does not produce electricity, it is also commonly called a non-power reactor (NPR).

11.6.a UW Research Reactor Components
Figure 11-9 shows a cutaway view of the UW Reactor. It is a heterogeneous pool-type reactor and is approximately 8 x 12 x 27½ feet deep. The actual core is only about 15 x 17 x 15 inches deep. The fuel is uranium enriched in \(^{235}\)U to 70%. There are 91 fuel elements in the core, each element containing an average of 123 grams of \(^{235}\)U, and there is approximately 24.6 lb. (11.2 kg) of \(^{235}\)U in the core. Light water acts as both coolant and moderator as well as being a biological shield. The core is reflected on two sides by graphite and on two sides by water. The water-reflected areas are used as locations to perform irradiations. Control is accomplished by three vertical safety blades and there is also one vertical regulating blade for fine adjustment. The safety blades provide a shutdown margin of about 5\% \(k_{\text{eff}}\).

The Reactor Laboratory has facilities to permit use of radiations from the reactor in experimental work without unduly endangering personnel. These facilities include three hydraulic irradiation facilities known as “whales”, four beam ports, one thermal column, and a pneumatic transfer system known as a “rabbit”.

The Hydraulic Irradiation Facility (whale) is composed of three aluminum pipes of 2-7/16” internal diameter that extend from approximately 18” below the pool surface to grid box positions on the periphery of the core. These pipes draw sample bottles made of polyethylene down and position them approximately at the center line of the fuel. Two sample containers can be loaded in each tube. The addition of a second sample bottle, however,
causes the natural rotation of the first bottle to stop. Thermal Neutron Fluxes in these positions are approximately \(1 \times 10^{13}\) neutrons per cm\(^2\) per second.

The **Thermal Column** is a graphite-filled horizontal penetration through the biological shield which provides neutrons in the thermal energy range (about 0.025 eV) for irradiation experiments. The thermal column, which is about 8 feet long, is filled with about 6 feet of graphite. A small experimental air chamber (40" x 40" x 24") between the face of the graphite and the thermal column door has conduits for service connections (air, water, electricity) to the biological shield face. Personnel in the building are protected against gamma radiation from the column by a dense concrete door which closes the column at the biological shield. The door moves on tracks set into the concrete floor perpendicular to the shield face.

There are four, 6-inch **Beam Ports** which penetrate the shield and provide fluxes of both fast and thermal neutrons for experimental use. The ports are air filled tubes, welded shut at the core ends and provided with water-tight covers on the outer ends. The portions of the ports within the pool are made of aluminum, while the portions within the shield are steel. A shutter assembly, made of lead encased in aluminum, is opened for irradiations by a lifting device. When closed, the shutter shields against gamma rays from the shut down core, allowing experiments to be loaded and unloaded without excessive radiation exposure to personnel. Shielding plugs are installed in the outer end of each port. The plugs, made of dense concrete in aluminum casings, have spiral conduits for passage of instrument leads.

A **Pneumatic Tube (rabbit)** system conveys samples from a basement room to an irradiation position beside the core. The rabbits used in the system will convey samples up to 1¼-inch diameter and 5½-inch long. The system operates as a closed loop with helium cover gas preventing generation of \(^{41}\)Ar activity.

All samples to be inserted in the reactor are encapsulated according to specific reactor procedures. All liquids, gases, and solids in dust or powdered forms must be double encapsulated so that they can not break open and contaminate the facility. Solid materials not subject to flaking need only a plastic bag encapsulation for insertion into the beam port or thermal column.

### 11.6.b UW Research Reactor Uses

The UW Nuclear Reactor Laboratory was established as a teaching laboratory for the Nuclear Engineering and Engineering Physics Department. However, the capabilities of the laboratory are available for use by others. In addition to instruction for students from the department, students from a number of other educational institutions use the facilities under the sponsorship of the U.S. Department of Energy Reactor Sharing Program. Such use has ranged from individual laboratory sessions to semester-long laboratories on selected topics (reactor operation characteristics, neutron activation analysis, and radiation safety instrumentation).

A Research Reactor Training program was developed, and has been used by utilities from several states as part of training programs for operators, senior operators, and shift technical advisors.

Research support is provided for industry as well as other educational institutions. The primary activities in the past have involved sample analyses by neutron activation analysis. Other support has been production of radioactive isotopes and irradiation of materials for various effects. Additional services have been provided in instrument calibration and evaluation, decontamination studies, and determination of effectiveness of filtration. In addition, a neutron radiography capability has been added. This facility was designed to provide real-time imaging of operating systems, and is available for use by others outside the department.

Currently, one of the biggest research uses of the reactor by non-engineering researchers involves neutron activation. Neutron activation analysis is a physical method of analysis of materials for elemental composition. A sample is exposed to neutrons, resulting in activation of many of the constituent elements. Specific radiations emitted by the activation products are detected to determine the amount of the elements present in the sample. The UW reactor practices instrumental neutron activation analysis (INAA), a technique in which gamma ray emissions are detected. Beta and positron emitters may also be detected to determine elements, but this requires radiochemical procedures to separate elements with similar emissions. In most cases, gamma ray energies and half-lives are distinctive enough that elements may be determined without chemical separations or special sample preparation. Samples of relatively uniform volume and mass are sealed inside polyethylene vials. Standards (materials with known elemental composition) are similarly prepared. An irradiation time is selected, depending upon elements to be detected. Likewise, decay time (from end of irradiation to start of counting) and counting time are selected for the particular analysis desired. Standards and samples are counted on high purity germanium gamma ray spectrometers. Spectra are analyzed and results are computed by comparison with the standards (computer programs will calculate results based on element nuclear characteristics, but standardization is preferred).
Even though this is a small reactor, the department employs many of the same security and safety devices which power reactors employ. Thus, while the reactor personnel gladly give tours of their facility, they first insure that persons in the tour group receive a thorough briefing and they measure radiation exposures throughout the facility to insure that no visitor will be exposed to radiation levels in excess of 0.5 mrem/hr (remember background radiation of the US population averages 300 mrem per year and 100 mrem per year is considered to be an allowable exposure rate for areas accessible to members of the general public).

11.7 Review Questions - Fill-in or select the correct response
1. Nuclear reactors produce energy by nuclear ________________.
2. All fission fragments are ________________ and usually decay by ________________ emission.
3. To achieve a self-sustaining nuclear reaction, or criticality, the rate of neutron production must be greater than or equal to the rate of neutron loss. **true / false**
4. The major difference between a pressurized water reactor (PWR) and a boiling water reactor (BWR) is the steam generator and secondary loop. **true / false**
5. The UW research reactor is used to generate the electrical power for the campus. **true / false**
6. Nuclei produced by neutron absorption in a large nucleus are called ________________ fragments.
7. Criticality is achieved when neutron ________________ is at a rate at least equal to neutron ________________.
8. Natural uranium needs to be enriched, that is, contain a higher percentage of (235U or 238U) before it is a viable fuel. **true / false**
9. ________________ and ________________ present the greatest shielding challenges in a reactor.
10. Water is a good shielding material for neutrons because of its ________________ content.
11. The ________________ is the material which slows down fast neutrons and allows them time to diffuse until they are captured by the ________________.
12. The isotope of uranium capable of being fissioned by thermal neutrons is ________________.
   The nuclear process in which some of the neutrons produced from fission further the reaction is called a nuclear ________________ reaction.
13. The non-fissile isotopes 232Th and 238U are called ________________ because they may be used to produce fissile isotopes (233U and 239Pu) through a process called ________________.
14. The thermal neutron fission cross section (σf) for 235U is ________________ barns.
15. ________________ is the process in which more than one fissile atom is produced for every fissile atom consumed.

11.8 References
Klimov, A., *Nuclear Physics and Nuclear Reactors*, Mir, Moscow, 1975