Reactor Kinetics and Operation

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SUBCRITICAL MULTIPLICATION

Subcritical multiplication is the phenomenon that accounts for the changes in neutron flux that takes place in a subcritical reactor due to reactivity changes. It is important to understand subcritical multiplication in order to understand reactor response to changes in conditions.

EO 1.1 **DEFINE the following terms:**

a. Subcritical multiplication
b. Subcritical multiplication factor

EO 1.2 **Given a neutron source strength and a subcritical system of known k_{eff}, CALCULATE the steady-state neutron level.**

EO 1.3 **Given an initial count rate and k_{eff}, CALCULATE the final count rate that will result from the addition of a known amount of reactivity.**

EO 1.4 **Given count rates vs. the parameter being adjusted, ESTIMATE the value of the parameter at which the reactor will become critical through the use of a 1/M plot.**

Subcritical Multiplication Factor

When a reactor is in a shutdown condition, neutrons are still present to interact with the fuel. These source neutrons are produced by a variety of methods that were discussed in Module 2. If neutrons and fissionable material are present in the reactor, fission will take place. Therefore, a reactor will always be producing a small number of fissions even when it is shutdown.

Consider a reactor in which k_{eff} is 0.6. If 100 neutrons are suddenly introduced into the reactor, these 100 neutrons that start the current generation will produce 60 neutrons (100 x 0.6) from fission to start the next generation. The 60 neutrons that start the second generation will produce 36 neutrons (60 x 0.6) to start the third generation. The number of neutrons produced by fission in subsequent generations due to the introduction of 100 source neutrons into the reactor is shown below.

Generation	1st	2nd	3rd	4th	5th	6th	7th	8th	9th	10th	11th	12th
Neutrons	100	60	36	22	13	8	5	3	2	1	0	0

Because the reactor is subcritical, neutrons introduced in the reactor will have a decreasing effect on each subsequent generation. The addition of source neutrons to the reactor containing fissionable material has the effect of maintaining a much higher stable neutron level due to the fissions occurring than the neutron level that would result from the source neutrons alone. The effects of adding source neutrons at a rate of 100 neutrons per generation to a reactor with a k_{eff} of 0.6 are shown below.

Generation	1st	2nd	3rd	4th	5th	6th	7th	8th	9th	10th	11th	12th
	100	60	36	22	13	8	5	3	2	1	0	0
		100	60	36	22	13	8	5	3	2	1	0
			100	60	36	22	13	8	5	3	2	1
				100	60	36	22	13	8	5	3	2
					100	60	36	22	13	8	5	3
						100	60	36	22	13	8	5
							100	60	36	22	13	8
								100	60	36	22	13
									100	60	36	22
										100	60	36
											100	60
												100
Total n	100	160	196	218	231	239	244	247	249	250	250	...

A neutron source strength of 100 neutrons per generation will result in 250 neutrons per generation being produced from a combination of sources and fission in a shutdown reactor with a k_{eff} of 0.6. If the value of k_{eff} were higher, the source neutrons would produce a greater number of fission neutrons and their effects would be felt for a larger number of subsequent generations after their addition to the reactor.

The effect of fissions in the fuel increasing the effective source strength of a reactor with a k_{eff} of less than one is *subcritical multiplication*. For a given value of k_{eff} there exists a *subcritical multiplication factor* (M) that relates the source level to the steady-state neutron level of the core. If the value of k_{eff} is known, the amount that the neutron source strength will be multiplied (M) can easily be determined by Equation (4-1).

$$M = \frac{1}{1 - k_{eff}} \tag{4-1}$$

Example:

Calculate the subcritical multiplication factors for the following values of k_{eff}.

1) $k_{eff} = 0.6$
2) $k_{eff} = 0.986$

Solution:

1)

$$M = \frac{1}{1 - k_{eff}}$$

$$= \frac{1}{1 - 0.6}$$

$$= 2.5$$

2)

$$M = \frac{1}{1 - k_{eff}}$$

$$= \frac{1}{1 - 0.986}$$

$$= 71.4$$

The example above illustrates that the subcritical multiplication factor will increase as positive reactivity is added to a shutdown reactor, increasing the value of k_{eff}. If the source strength of this reactor were 1000 neutrons/sec, the neutron level would increase from 2500 neutrons/second at a k_{eff} of 0.6 to a neutron level of 71,400 neutrons/sec at a k_{eff} of 0.986.

Effect of Reactivity Changes on Subcritical Multiplication

In a subcritical reactor, the neutron level is related to the source strength by Equation (4-2).

$$N = (S)(M) \tag{4-2}$$

where:

N = neutron level
S = neutron source strength
M = subcritical multiplication factor

If the term M in Equation (4-2) is replaced by the expression 1/1-k_{eff} from Equation (4-1), the following expression results.

$$N = S \left(\frac{1}{1 - k_{eff}} \right) \tag{4-3}$$

Example:

A reactor contains a neutron source that produces 110,000 neutrons per second. The reactor has a k_{eff} of 0.986. Calculate the stable total neutron production rate in the reactor.

Solution:

The neutron production rate is calculated using Equation (4-3).

$$N = S \left(\frac{1}{1 - k_{eff}} \right)$$

$$= 110{,}000 \ \frac{\text{neutrons}}{\text{second}} \left(\frac{1}{1 - 0.986} \right)$$

$$= 7.86 \ \text{x} \ 10^6 \ \frac{\text{neutrons}}{\text{second}}$$

To this point it has been necessary to know the neutron source strength of the reactor in order to use the concept of subcritical multiplication. In most reactors the actual strength of the neutron sources is difficult, if not impossible, to determine. Even though the actual source strength may not be known, it is still possible to relate the change in reactivity to a change in neutron level.

Consider a reactor at two different times when k_{eff} is two different values, k_1 and k_2. The neutron level at each time can be determined based on the neutron source strength and the subcritical multiplication factor using Equation (4-3).

$$N_1 = S \left(\frac{1}{1 - k_1} \right) \qquad\qquad N_2 = S \left(\frac{1}{1 - k_2} \right)$$

The equation for N_1 can be divided by the equation for N_2.

$$\frac{N_1}{N_2} = \frac{S\left(\frac{1}{1 - k_1}\right)}{S\left(\frac{1}{1 - k_2}\right)}$$

$$\frac{N_1}{N_2} = \frac{1 - k_2}{1 - k_1}$$

Because the source strength appears in both the numerator and denominator, it cancels out of the equation. Therefore, the neutron level at any time can be determined based on the neutron level present at any other time provided the values of k_{eff} or reactivity for both times are known.

The neutron level in a shutdown reactor is typically monitored using instruments that measure the neutron leakage out of the reactor. The neutron leakage is proportional to the neutron level in the reactor. Typical units for displaying the instrument reading are counts per second (cps). Because the instrument count rate is proportional to the neutron level, the above equation can be restated as shown in Equation (4-4).

$$\frac{CR_1}{CR_2} = \frac{1 - k_2}{1 - k_1} \tag{4-4}$$

where:

CR_1 = count rate at time 1
CR_2 = count rate at time 2
k_1 = k_{eff} at time 1
k_2 = k_{eff} at time 2

Equation (4-4) is very useful during the shutdown operation of a reactor. Before adding positive reactivity to a reactor, it is possible to predict the effect the reactivity addition will have on the neutron level.

Example:

A reactor that has a reactivity of -1000 pcm has a count rate of 42 counts per second (cps) on the neutron monitoring instrumentation. Calculate what the neutron level should be after a positive reactivity insertion of 500 pcm from the withdrawal of control rods.

Solution:

Step 1: Determine the initial value of k_{eff} for the core.

$$k_1 = \frac{1}{1 - \rho_1}$$

$$= \frac{1}{1 - (-0.01000)}$$

$$= 0.9901$$

Step 2: Determine the final value of k_{eff} for the core. The final value of reactivity will be -500 pcm (-1000 + 500).

$$k_2 = \frac{1}{1 - \rho_2}$$

$$= \frac{1}{1 - (-0.00500)}$$

$$= 0.9950$$

Step 3: Use Equation (4-4) to determine the final count rate.

$$\frac{CR_1}{CR_2} = \frac{1 - k_2}{1 - k_1}$$

$$CR_2 = CR_1 \left(\frac{1 - k_1}{1 - k_2}\right)$$

$$= 42 \text{ cps} \left(\frac{1 - 0.9901}{1 - 0.9950}\right)$$

$$= 83 \text{ cps}$$

Notice from this example that the count rate doubled as the reactivity was halved (e.g., reactivity was changed from -1000 pcm to -500 pcm).

Use of 1/M Plots

Because the subcritical multiplication factor is related to the value of k_{eff}, it is possible to monitor the approach to criticality through the use of the subcritical multiplication factor. As positive reactivity is added to a subcritical reactor, k_{eff} will get nearer to one. As k_{eff} gets nearer to one, the subcritical multiplication factor (M) gets larger. The closer the reactor is to criticality, the faster M will increase for equal step insertions of positive reactivity. When the reactor becomes critical, M will be infinitely large. For this reason, monitoring and plotting M during an approach to criticality is impractical because there is no value of M at which the reactor clearly becomes critical.

Instead of plotting M directly, its inverse (1/M) is plotted on a graph of 1/M versus rod height.

$$M = \frac{1}{1 - k_{eff}}$$

$$\frac{1}{M} = 1 - k_{eff}$$

As control rods are withdrawn and k_{eff} approaches one and M approaches infinity, 1/M approaches zero. For a critical reactor, 1/M is equal to zero. A true 1/M plot requires knowledge of the neutron source strength. Because the actual source strength is usually unknown, a reference count rate is substituted, and the calculation of the factor 1/M is through the use of Equation (4-5).

$$\frac{1}{M} = \frac{CR_o}{CR} \tag{4-5}$$

where:

1/M	=	inverse multiplication factor
CR_o	=	reference count rate
CR	=	current count rate

In practice, the reference count rate used is the count rate prior to the beginning of the reactivity change. The startup procedures for many reactors include instructions to insert positive reactivity in incremental steps with delays between the reactivity insertions to allow time for subcritical multiplication to increase the steady-state neutron population to a new, higher level and allow more accurate plotting of 1/M. The neutron population will typically reach its new steady-state value within 1-2 minutes, but the closer the reactor is to criticality, the longer the time will be to stabilize the neutron population.

Example:

Given the following rod withdrawal data, construct a 1/M plot and estimate the rod position when criticality would occur. The initial count rate on the nuclear instrumentation prior to rod withdrawal is 50 cps.

Rod Withdrawal (inches)	Count Rate (cps)
2	55
4	67
6	86
8	120
10	192
12	500

Solution:

Step 1: Calculate 1/M for each of the rod positions using equation (4-5). The reference count rate is 50 cps at a rod position of zero.

Rod Withdrawal (inches)	Count Rate (cps)	CR_o/CR
0	50	1
2	55	0.909
4	67	0.746
6	86	0.581
8	120	0.417
10	192	0.260
12	500	0.100

Step 2: Plotting these values, as shown in Figure 1, and extrapolating to a 1/M value of 0 reveals that the reactor will go critical at approximately 13 inches of rod withdrawal.

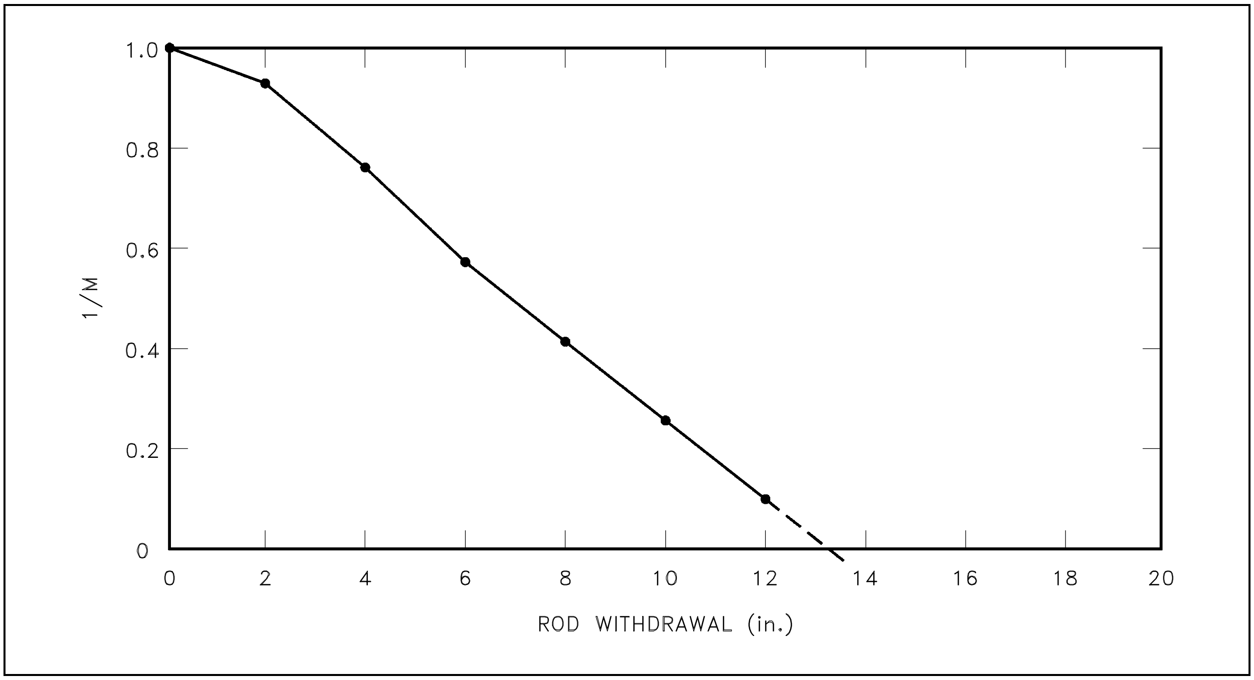

Figure 1 1/M Plot vs. Rod Withdrawal

Summary

The important information in this chapter is summarized below.

Subcritical Multiplication Summary

- Subcritical multiplication is the effect of fissions in the fuel increasing the effective source strength of a reactor with a k_{eff} less than one.
- Subcritical multiplication factor is the factor that relates the source level to the steady-state neutron level of the core.
- The steady-state neutron level of a subcritical reactor can be calculated based on the source strength and k_{eff} using Equation (4-3).

$$N = S \left(\frac{1}{1 - k_{eff}} \right)$$

- The count rate expected in a subcritical reactor following a change in reactivity can be calculated based on the initial count rate, initial k_{eff}, and amount of reactivity addition using Equation (4-4).

$$\frac{CR_1}{CR_2} = \frac{1 - k_2}{1 - k_1}$$

- 1/M plots can be used to predict the point of criticality.

REACTOR KINETICS

The response of neutron flux and reactor power to changes in reactivity is much different in a critical reactor than in a subcritical reactor. The reliance of the chain reaction on delayed neutrons makes the rate of change of reactor power controllable.

EO 2.1 **DEFINE the following terms:**

- **a. Reactor period**
- **b. Doubling time**
- **c. Reactor startup rate**

EO 2.2 **DESCRIBE the relationship between the delayed neutron fraction, average delayed neutron fraction, and effective delayed neutron fraction.**

EO 2.3 **WRITE the period equation and IDENTIFY each symbol.**

EO 2.4 **Given the reactivity of the core and values for the effective average delayed neutron fraction and decay constant, CALCULATE the reactor period and the startup rate.**

EO 2.5 **Given the initial power level and either the doubling or halving time, CALCULATE the power at any later time.**

EO 2.6 **Given the initial power level and the reactor period, CALCULATE the power at any later time.**

EO 2.7 **EXPLAIN what is meant by the terms prompt drop and prompt jump.**

EO 2.8 **DEFINE the term prompt critical.**

EO 2.9 **DESCRIBE reactor behavior during the prompt critical condition.**

EO 2.10 **EXPLAIN the use of measuring reactivity in units of dollars.**

Reactor Period (τ)

The *reactor period* is defined as the time required for reactor power to change by a factor of "e," where "e" is the base of the natural logarithm and is equal to about 2.718. The reactor period is usually expressed in units of seconds. From the definition of reactor period, it is possible to develop the relationship between reactor power and reactor period that is expressed by Equation (4-6).

$$P = P_o \, e^{t/\tau} \qquad (4\text{-}6)$$

where:

P = transient reactor power
P_o = initial reactor power
τ = reactor period (seconds)
t = time during the reactor transient (seconds)

The smaller the value of τ, the more rapid the change in reactor power. If the reactor period is positive, reactor power is increasing. If the reactor period is negative, reactor power is decreasing.

There are numerous equations used to express reactor period, but Equation (4-7) shown below, or portions of it, will be useful in most situations. The first term in Equation (4-7) is the prompt term and the second term is the delayed term.

$$\tau = \frac{\ell^*}{\rho} + \frac{\bar{\beta}_{eff} - \rho}{\lambda_{eff}\,\rho + \dot{\rho}} \qquad (4\text{-}7)$$

where:
ℓ^* = prompt generation lifetime
$\bar{\beta}_{eff}$ = effective delayed neutron fraction
ρ = reactivity
λ_{eff} = effective delayed neutron precursor decay constant
$\dot{\rho}$ = rate of change of reactivity

Effective Delayed Neutron Fraction

Recall that β, the *delayed neutron fraction*, is the fraction of all fission neutrons that are born as delayed neutrons. The value of β depends upon the actual nuclear fuel used. As discussed in Module 1, the delayed neutron precursors for a given type of fuel are grouped on the basis of half-life. The following table lists the fractional neutron yields for each delayed neutron group of three common types of fuel.

TABLE 1
Delayed Neutron Fractions for Various Fuels

Group	Half-Life (sec)	Uranium-235	Uranium-238	Plutonium-239
1	55.6	0.00021	0.0002	0.00021
2	22.7	0.00141	0.0022	0.00182
3	6.22	0.00127	0.0025	0.00129
4	2.30	0.00255	0.0061	0.00199
5	0.61	0.00074	0.0035	0.00052
6	0.23	0.00027	0.0012	0.00027
TOTAL	-	0.00650	0.0157	0.00200

The term $\bar{\beta}$ (pronounced beta-bar) is the *average delayed neutron fraction*. The value of $\bar{\beta}$ is the weighted average of the total delayed neutron fractions of the individual types of fuel. Each total delayed neutron fraction value for each type of fuel is weighted by the percent of total neutrons that the fuel contributes through fission. If the percentage of fissions occurring in the different types of fuel in a reactor changes over the life of the core, the average delayed neutron fraction will also change. For a light water reactor using low enriched fuel, the average delayed neutron fraction can change from 0.0070 to 0.0055 as uranium-235 is burned out and plutonium-239 is produced from uranium-238.

Delayed neutrons do not have the same properties as prompt neutrons released directly from fission. The average energy of prompt neutrons is about 2 MeV. This is much greater than the average energy of delayed neutrons (about 0.5 MeV). The fact that delayed neutrons are born at lower energies has two significant impacts on the way they proceed through the neutron life cycle. First, delayed neutrons have a much lower probability of causing fast fissions than prompt neutrons because their average energy is less than the minimum required for fast fission to occur. Second, delayed neutrons have a lower probability of leaking out of the core while they are at fast energies, because they are born at lower energies and subsequently travel a shorter distance as fast neutrons. These two considerations (lower fast fission factor and higher fast non-leakage probability for delayed neutrons) are taken into account by a term called the *importance factor* (I). The importance factor relates the average delayed neutron fraction to the effective delayed neutron fraction.

The *effective delayed neutron fraction* $\left(\bar{\beta}_{eff}\right)$ is defined as the fraction of neutrons at thermal energies which were born delayed. The effective delayed neutron fraction is the product of the average delayed neutron fraction and the importance factor.

$$\bar{\beta}_{eff} = \bar{\beta}\ I$$

where:

$\bar{\beta}_{eff}$	=	effective delayed neutron fraction
$\bar{\beta}$	=	average delayed neutron fraction
I	=	importance factor

In a small reactor with highly enriched fuel, the increase in fast non-leakage probability will dominate the decrease in the fast fission factor, and the importance factor will be greater than one. In a large reactor with low enriched fuel, the decrease in the fast fission factor will dominate the increase in the fast non-leakage probability and the importance factor will be less than one (about 0.97 for a commercial PWR).

Effective Delayed Neutron Precursor Decay Constant

Another new term has been introduced in the reactor period (τ) equation. That term is λ_{eff} (pronounced lambda effective), the *effective delayed neutron precursor decay constant*. The decay rate for a given delayed neutron precursor can be expressed as the product of precursor concentration and the decay constant (λ) of that precursor. The decay constant of a precursor is simply the fraction of an initial number of the precursor atoms that decays in a given unit time. A decay constant of 0.1 sec^{-1}, for example, implies that one-tenth, or ten percent, of a sample of precursor atoms decays within one second. The value for the effective delayed neutron precursor decay constant, λ_{eff}, varies depending upon the balance existing between the concentrations of the precursor groups and the nuclide(s) being used as the fuel.

If the reactor is operating at a constant power, all the precursor groups reach an equilibrium value. During an up-power transient, however, the shorter-lived precursors decaying at any given instant were born at a higher power level (or flux level) than the longer-lived precursors decaying at the same instant. There is, therefore, proportionately more of the shorter-lived and fewer of the longer-lived precursors decaying at that given instant than there are at constant power. The value of λ_{eff} is closer to that of the shorter-lived precursors.

During a down-power transient the longer-lived precursors become more significant. The longer-lived precursors decaying at a given instant were born at a higher power level (or flux level) than the shorter-lived precursors decaying at that instant. Therefore, proportionately more of the longer-lived precursors are decaying at that instant, and the value of λ_{eff} approaches the values of the longer-lived precursors.

Approximate values for λ_{eff} are 0.08 sec^{-1} for steady-state operation, 0.1 sec^{-1} for a power increase, and 0.05 sec^{-1} for a power decrease. The exact values will depend upon the materials used for fuel and the value of the reactivity of the reactor core.

Returning now to Equation (4-7) for reactor period.

$$\tau = \underbrace{\frac{\ell^*}{\rho}}_{\text{(prompt term)}} + \underbrace{\frac{\bar{\beta}_{eff} - \rho}{\lambda_{eff}\,\rho + \dot{\rho}}}_{\text{(delayed term)}}$$

If the positive reactivity added is less than the value of $\bar{\beta}_{eff}$, the emission of prompt fission neutrons alone is not sufficient to overcome losses to non-fission absorption and leakage. If delayed neutrons were not being produced, the neutron population would decrease as long as the reactivity of the core has a value less than the effective delayed neutron fraction. The positive reactivity insertion is followed immediately by a small immediate power increase called the *prompt jump*. This power increase occurs because the rate of production of prompt neutrons changes abruptly as the reactivity is added. Recall from an earlier module that the generation time for prompt neutrons is on the order of 10^{-13} seconds. The effect can be seen in Figure 2. After the prompt jump, the rate of change of power cannot increase any more rapidly than the built-in time delay the precursor half-lives allow. Therefore, the power rise is controllable, and the reactor can be operated safely.

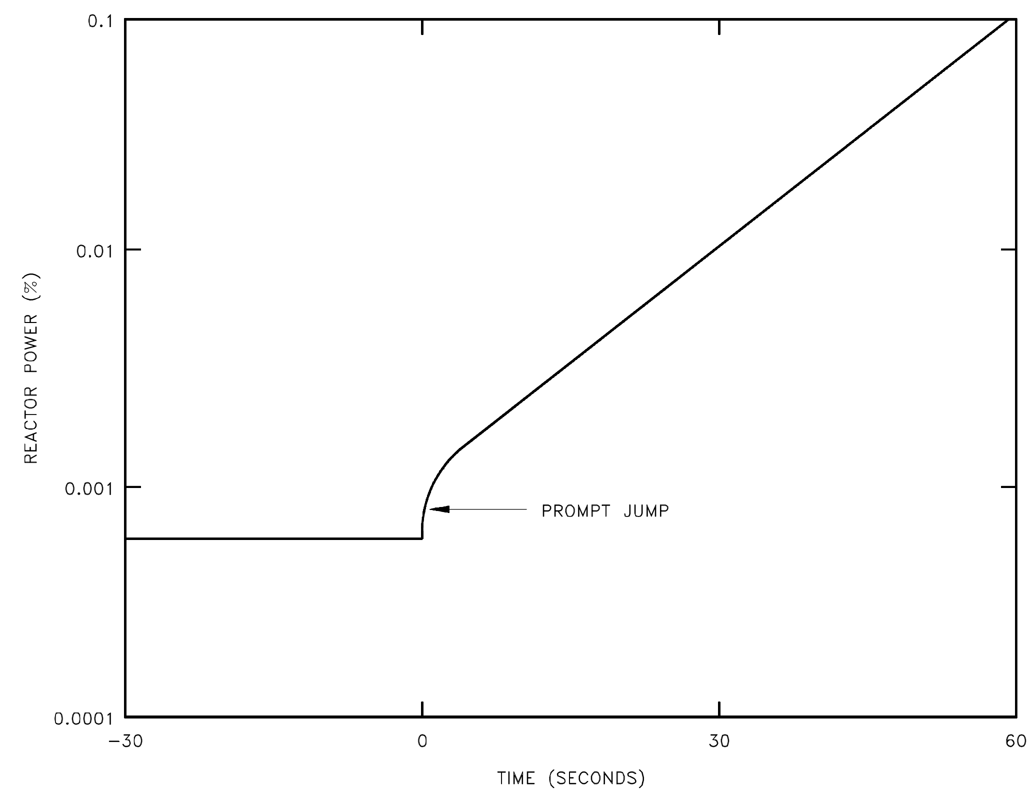

Figure 2 Reactor Power Response to Positive Reactivity Addition

Conversely, in the case where negative reactivity is added to the core there will be a prompt drop in reactor power. The *prompt drop* is the small immediate decrease in reactor power caused by the negative reactivity addition. The prompt drop is illustrated in Figure 3. After the prompt drop, the rate of change of power slows and approaches the rate determined by the delayed term of Equation (4-7).

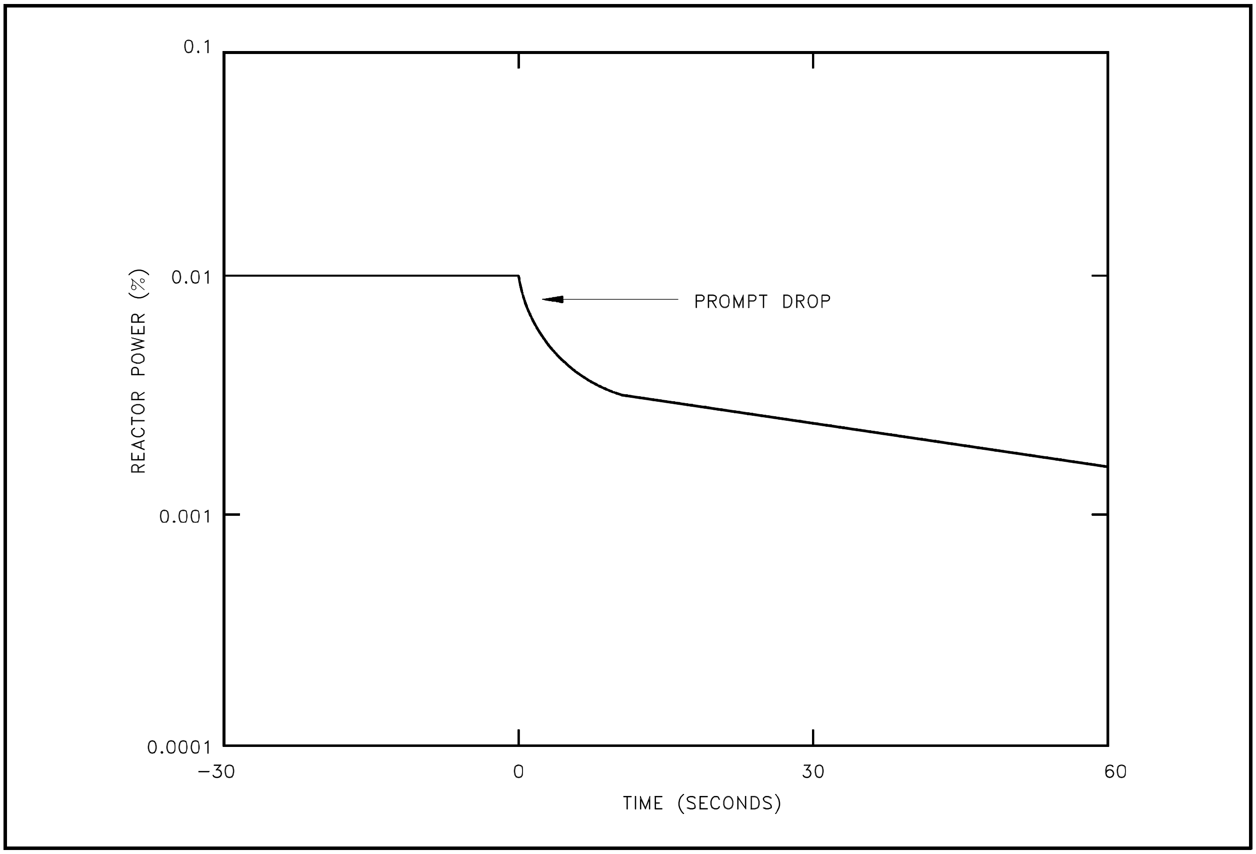

Figure 3 Reactor Power Response to Negative Reactivity Addition

Prompt Criticality

It can be readily seen from Equation (4-7) that if the amount of positive reactivity added equals the value of $\bar{\beta}_{eff}$, the reactor period equation becomes the following.

$$\tau = \frac{\ell^*}{\rho}$$

In this case, the production of prompt neutrons alone is enough to balance neutron losses and increase the neutron population. The condition where the reactor is critical on prompt neutrons, and the neutron population increases as rapidly as the prompt neutron generation lifetime allows is known as *prompt critical*. The prompt critical condition does not signal a dramatic change in neutron behavior. The reactor period changes in a regular manner between reactivities above and below this reference. Prompt critical is, however, a convenient condition for marking the transition from delayed neutron to prompt neutron time scales. A reactor whose reactivity even approaches prompt critical is likely to suffer damage due to the rapid rise in power to a very high level. For example, a reactor which has gone prompt critical could experience a several thousand percent power increase in less than one second.

Because the prompt critical condition is so important, a specific unit of reactivity has been defined that relates to it. The unit of reactivity is the dollar ($), where one dollar of reactivity is equivalent to the effective delayed neutron fraction $\left(\bar{\beta}_{eff}\right)$. A reactivity unit related to the dollar is the cent, where one cent is one-hundredth of a dollar. If the reactivity of the core is one dollar, the reactor is prompt critical. Because the effective delayed neutron fraction is dependent upon the nuclides used as fuel, the value of the dollar is also dependent on the nuclides used as fuel.

Stable Period Equation

For normal reactor operating conditions, the value of positive reactivity in the reactor is never permitted to approach the effective delayed neutron fraction, and the reactor period equation is normally written as follows.

$$\tau = \frac{\bar{\beta}_{eff} - \rho}{\lambda_{eff}\,\rho + \dot{\rho}} \qquad (4\text{-}8)$$

Equation (4-8) is referred to as the *transient period equation* since it incorporates the $\dot{\rho}$ term to account for the changing amount of reactivity in the core. The ℓ^*/ρ term (prompt period) is normally negligible with respect to the remainder of the equation and is often not included.

For conditions when the amount of reactivity in the core is constant ($\dot{\rho} = 0$), and the reactor period is unchanging, Equation (4-8) can be simplified further to Equation (4-9) which is known as the stable period equation.

$$\tau = \frac{\bar{\beta}_{eff} - \rho}{\lambda_{eff}\,\rho} \qquad (4\text{-}9)$$

Reactor Startup Rate (SUR)

The *reactor startup rate* (SUR) is defined as the number of factors of ten that power changes in one minute. The units of SUR are powers of ten per minute, or decades per minute (DPM). Equation (4-10) shows the relationship between reactor power and startup rate.

$$P = P_o \, 10^{SUR\,(t)} \tag{4-10}$$

where:

SUR = reactor startup rate (DPM)
t = time during reactor transient (minutes)

The relationship between reactor period and startup rate can be developed by considering Equations (4-6) and (4-10).

$$P = P_o \, e^{t/\tau} \quad \text{and} \quad P = P_o \, 10^{SUR\,(t)}$$

$$\frac{P}{P_o} = e^{t/\tau} = 10^{SUR\,(t)}$$

Changing the base of the exponential term on the right side to "e" ($10 = e^{2.303}$) and solving the result yields the following.

$$e^{t\,(sec)/\tau} = e^{2.303\;SUR\;(t\;(min))}$$

$$\frac{t\;(sec)}{\tau} = 2.303\;SUR(t\;(min))$$

$$\frac{60}{\tau} = 2.303\;SUR$$

$$SUR = \frac{26.06}{\tau} \tag{4-11}$$

Doubling Time

Sometimes it is useful to discuss the rate of change of reactor power in terms similar to those used in radioactive decay calculations. *Doubling or halving time* are terms that relate to the amount of time it takes reactor power to double or be reduced to one-half the initial power level. If the stable reactor period is known, doubling time can be determined as follows.

Doubling time (DT) = τ (ln 2)

where:

τ = stable reactor period

ln 2 = natural logarithm of 2

When the doubling time is known, the power level change from P_o is given by the following equation.

$$P = P_o \, 2^{t/DT} \tag{4-12}$$

where:

t = time interval of transient

DT = doubling time

The following example problems reinforce the concepts of period and startup rate.

Example 1:

A reactor has a λ_{eff} of 0.10 sec^{-1} and an effective delayed neutron fraction of 0.0070. If k_{eff} is equal to 1.0025, what is the stable reactor period and the SUR?

Solution:

Step 1: First solve for reactivity using Equation (3-5).

$$\rho = \frac{k_{eff} - 1}{k_{eff}}$$

$$= \frac{1.0025 - 1}{1.0025}$$

$$= 0.00249 \ \Delta k/k$$

Step 2: Use this value of reactivity in Equation (4-9) to calculate reactor period.

$$\tau = \frac{\bar{\beta}_{eff} - \rho}{\lambda_{eff} \, \rho}$$

$$= \frac{0.0070 - 0.00249}{\left(0.10 \ sec^{-1}\right) 0.00249}$$

$$= 18.1 \ sec$$

Step 3: The startup rate can be calculated from the reactor period using Equation (4-11).

$$SUR = \frac{26.06}{\tau}$$

$$= \frac{26.06}{18.1 \text{ sec}}$$

$$= 1.44 \text{ DPM}$$

Example 2:

130 pcm of negative reactivity is added to a reactor that is initially critical at a power of 100 watts. λ_{eff} for the reactor is 0.05 sec^{-1} and the effective delayed neutron fraction is 0.0068. Calculate the steady state period and startup rate. Also calculate the power level 2 minutes after the reactivity insertion.

Solution:

Step 1: Use Equation (4-9) to calculate the reactor period.

$$\tau = \frac{\bar{\beta}_{eff} - \rho}{\lambda_{eff} \rho}$$

$$= \frac{0.0068 - (-0.00130)}{\left(0.05 \text{ sec}^{-1}\right)(-0.00130)}$$

$$= -124.6 \text{ sec}$$

Step 2: The startup rate can be calculated from the reactor period using Equation (4-11).

$$SUR = \frac{26.06}{\tau}$$

$$= \frac{26.06}{-124.6 \text{ sec}}$$

$$= -0.2091 \text{ DPM}$$

Step 3: Use either Equation (4-1) or Equation (4-10) to calculate the reactor power two minutes after the reactivity insertion.

$$P = P_o e^{t/\tau}$$

$$= (100 \text{ W}) e^{(120 \text{ s}/-124.6 \text{ s})}$$

$$= 38.2 \text{ W}$$

$$P = P_o 10^{SUR (t)}$$

$$= (100 \text{ W}) 10^{(-0.2091 \text{ DPM})(2 \text{ min})}$$

$$= 38.2 \text{ W}$$

Example 3:

A reactor has a power level of 1000 watts and a doubling time of 2 minutes. What is the reactor power level 10 minutes later?

Solution:

Use Equation (4-12) to calculate the final power level.

$$P = P_o\ (2)^{t/DT}$$

$$= (1{,}000\ \mathrm{W})\ (2)^{10\ \mathrm{min}/2\ \mathrm{min}}$$

$$= 32{,}000\ \mathrm{W}$$

Summary

The important information in this chapter is summarized below.

Reactor Kinetics Summary

- Reactor period is the time required for reactor power to change by a factor of e (2.718).
- Doubling time is the time required for reactor power to double.
- Reactor startup rate is the number of factors of ten that reactor power changes in one minute.
- The delayed neutron fraction (β) is the fraction of all fission neutrons that are born as delayed neutrons for a particular type of fuel (that is, uranium-235 and plutonium-239).
- The average delayed neutron fraction ($\bar{\beta}$) is the weighted average of the total delayed neutron fractions of the different types of fuel used in a particular reactor.
- The effective delayed neutron fraction ($\bar{\beta}_{eff}$) is the average delayed neutron fraction multiplied by an Importance Factor which accounts for the fact that delayed neutrons are born at lower average energies than fast neutrons.
- The reactor period equation is stated below.

$$\tau = \underbrace{\frac{\ell^*}{\rho}}_{\left(\substack{\text{prompt}\\ \text{term}}\right)} + \underbrace{\frac{\bar{\beta}_{eff} - \rho}{\lambda_{eff}\,\rho + \dot{\rho}}}_{\left(\substack{\text{delayed}\\ \text{term}}\right)}$$

where:

τ = reactor period
ℓ^* = prompt generation lifetime
$\bar{\beta}_{eff}$ = effective delayed neutron fraction
ρ = reactivity
λ_{eff} = effective delayed neutron precursor decay constant
$\dot{\rho}$ = rate of change of reactivity

Reactor Kinetics Summary (Cont.)

- Equations (4-9) and (4-11) can be used to calculate the stable reactor period and startup rate.

$$\tau = \frac{\bar{\beta}_{eff} - \rho}{\lambda_{eff}\ \rho} \qquad SUR = \frac{26.06}{\tau}$$

- The concept of doubling time can be used in a similar manner to reactor period to calculate changes in reactor power using Equation (4-12).

$$P = P_o\ 2^{t/DT}$$

- The reactor period or the startup rate can be used to determine the reactor power using Equations (4-6) and (4-10).

$$P = P_o\ e^{t/\tau} \qquad P = P_o\ 10^{SUR\ (t)}$$

- Prompt jump is the small, immediate power increase that follows a positive reactivity insertion related to an increase in the prompt neutron population.

- Prompt drop is the small, immediate power decrease that follows a negative reactivity insertion related to a decrease in the prompt neutron population.

- Prompt critical is the condition when the reactor is critical on prompt neutrons alone.

- When a reactor is prompt critical, the neutron population, and hence power, can increase as quickly as the prompt neutron generation time allows.

- Measuring reactivity in units of dollars is useful when determining if a reactor is prompt critical. A reactor that contains one dollar of positive reactivity is prompt critical since one dollar of reactivity is equivalent to λ_{eff}.

REACTOR OPERATION

It is important to understand the principles that determine how a reactor responds during all modes of operation. Special measures must be taken during the startup of a reactor to ensure that expected responses are occurring. During power operation, control of the flux shape is necessary to ensure operation within limits and maximum core performance. Even when a reactor is shut down, the fact that the fission products created by the fission process continue to generate heat results in a need to monitor support systems to ensure adequate cooling of the core.

EO 3.1 **EXPLAIN why a startup neutron source may be required for a reactor.**

EO 3.2 **LIST four variables typically involved in a reactivity balance.**

EO 3.3 **EXPLAIN how a reactivity balance may be used to predict the conditions under which the reactor will become critical.**

EO 3.4 **LIST three methods used to shape or flatten the core power distribution.**

EO 3.5 **DESCRIBE the concept of power tilt.**

EO 3.6 **DEFINE the term shutdown margin.**

EO 3.7 **EXPLAIN the rationale behind the one stuck rod criterion.**

EO 3.8 **IDENTIFY five changes that will occur during and after a reactor shutdown that will affect the reactivity of the core.**

EO 3.9 **EXPLAIN why decay heat is present following reactor operation.**

EO 3.10 **LIST three variables that will affect the amount of decay heat present following reactor shutdown.**

EO 3.11 **ESTIMATE the approximate amount of decay heat that will exist one hour after a shutdown from steady state conditions.**

Startup

When a reactor is started up with unirradiated fuel, or on those occasions when the reactor is restarted following a long shutdown period, the source neutron population will be very low. In some reactors, the neutron population is frequently low enough that it cannot be detected by the nuclear instrumentation during the approach to criticality. Installed neutron sources, such as those discussed in Module 2, are frequently used to provide a safe, easily monitored reactor startup. The neutron source, together with the subcritical multiplication process, provides a sufficiently large neutron population to allow monitoring by the nuclear instruments throughout the startup procedure. Without the installed source, it may be possible to withdraw the control rods to the point of criticality, and then continue withdrawal without detecting criticality because the reactor goes critical below the indicating range. Continued withdrawal of control rods at this point could cause reactor power to rise at an uncontrollable rate before neutron level first becomes visible on the nuclear instruments.

An alternative to using a startup source is to limit the rate of rod withdrawal, or require waiting periods between rod withdrawal increments. By waiting between rod withdrawal increments, the neutron population is allowed to increase through subcritical multiplication. Subcritical multiplication is the process where source neutrons are used to sustain the chain reaction in a reactor with a multiplication factor (k_{eff}) of less than one. The chain reaction is not "self-sustaining," but if the neutron source is of sufficient magnitude, it compensates for the neutrons lost through absorption and leakage. This process can result in a constant, or increasing, neutron population even though k_{eff} is less than one.

Estimated Critical Position

In the first chapter of this module, 1/M plots were discussed. These plots were useful for monitoring the approach to criticality and predicting when criticality will occur based on indications received while the startup is actually in progress. Before the reactor startup is initiated, the operator calculates an estimate of the amount of rod withdrawal that will be necessary to achieve criticality. This process provides an added margin of safety because a large discrepancy between actual and estimated critical rod positions would indicate that the core was not performing as designed. Depending upon a reactor's design or age, the buildup of xenon within the first several hours following a reactor shutdown may introduce enough negative reactivity to cause the reactor to remain shutdown even with the control rods fully withdrawn. In this situation it is important to be able to predict whether criticality can be achieved, and if criticality cannot be achieved, the startup should not be attempted.

For a given set of conditions (such as time since shutdown, temperature, pressure, fuel burnup, samarium and xenon poisoning) there is only one position of the control rods (and boron concentrations for a reactor with chemical shim) that results in criticality, using the normal rod withdrawal sequence. Identification of these conditions allows accurate calculation of control rod position at criticality. The calculation of an *estimated critical position* (ECP) is simply a mathematical procedure that takes into account all of the changes in factors that significantly affect reactivity that have occurred between the time of reactor shutdown and the time that the reactor is brought critical again.

For most reactor designs, the only factors that change significantly after the reactor is shut down are the average reactor temperature and the concentration of fission product poisons. The reactivities normally considered when calculating an ECP include the following.

Basic Reactivity of the Core-	The reactivity associated with the critical control rod position for a xenon-free core at normal operating temperature. This reactivity varies with the age of the core (amount of fuel burnup).
Direct Xenon Reactivity -	The reactivity related to the xenon that was actually present in the core at the time it was shutdown. This reactivity is corrected to allow for xenon decay.
Indirect Xenon Reactivity -	The reactivity related to the xenon produced by the decay of iodine that was present in the core at the time of shutdown.
Temperature Reactivity -	The reactivity related to the difference between the actual reactor temperature during startup and the normal operating temperature.

To arrive at an ECP of the control rods, the basic reactivity, direct and indirect xenon reactivity, and temperature reactivity are combined algebraically to determine the amount of positive control rod reactivity that must be added by withdrawing control rods to attain criticality. A graph of control rod worth versus rod position is used to determine the estimated critical position.

Core Power Distribution

In order to ensure predictable temperatures and uniform depletion of the fuel installed in a reactor, numerous measures are taken to provide an even distribution of flux throughout the power producing section of the reactor. This shaping, or flattening, of the neutron flux is normally achieved through the use of *reflectors* that affect the flux profile across the core, or by the installation of poisons to suppress the neutron flux where desired. The last method, although effective at shaping the flux, is the least desirable since it reduces neutron economy by absorbing the neutrons.

A reactor core is frequently surrounded by a "reflecting" material to reduce the ratio of peak flux to the flux at the edge of the core fuel area. Reflector materials are normally not fissionable, have a high scattering cross section, and have a low absorption cross section. Essentially, for thermal reactors a good moderator is a good reflector. Water, heavy water, beryllium, zirconium, or graphite are commonly used as reflectors. In fast reactor systems, reflectors are not composed of moderating materials because it is desired to keep neutron energy high. The reflector functions by scattering some of the neutrons, which would have leaked from a bare (unreflected) core, back into the fuel to produce additional fissions.

Figure 4 shows the general effect of reflection in the thermal reactor system where core power is proportional to the thermal flux. Notice that a reflector can raise the power density of the core periphery and thus increase the core average power level without changing the peak power. As illustrated in Figure 4, the thermal flux in the reflector may actually be higher than that in the outermost fuel since there are very few absorptions in the reflector.

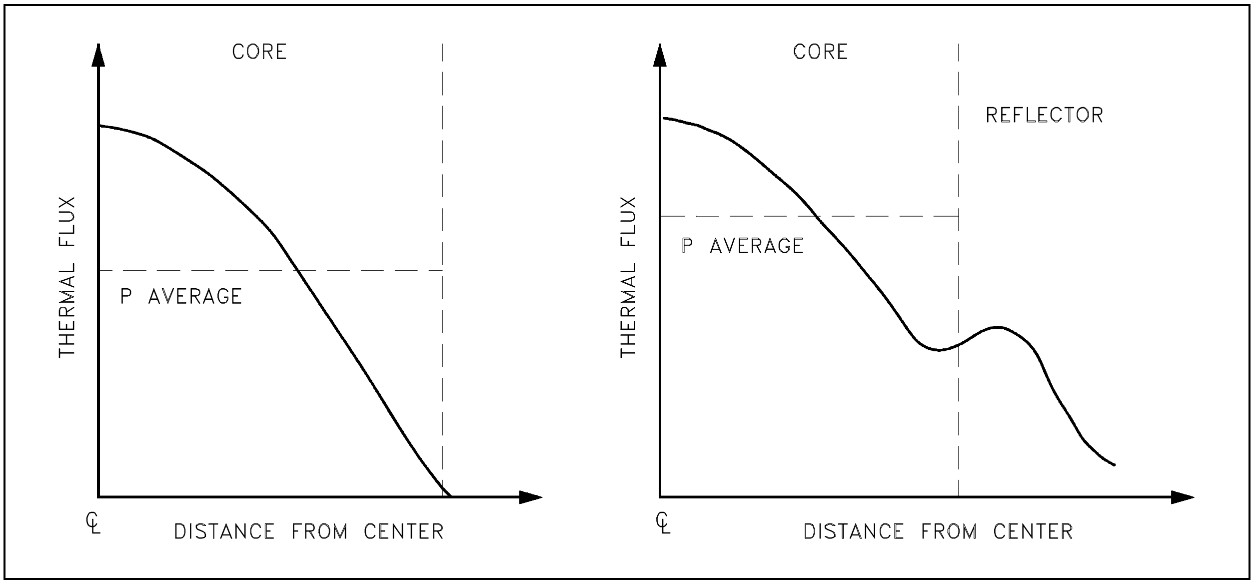

Figure 4 Neutron Radial Flux Shapes for Bare and Reflected Cores

Varying the fuel enrichment or fuel concentrations in the core radially, axially, or both, can readily be used to control power distribution. The simplified example illustrated in Figure 5 shows the effect of using a higher enrichment in the outer regions of the core. Varying fuel concentrations or poison loading for flux shaping is frequently referred to as zoning. In the example illustrated the large central peak is reduced, but the average power level remains the same.

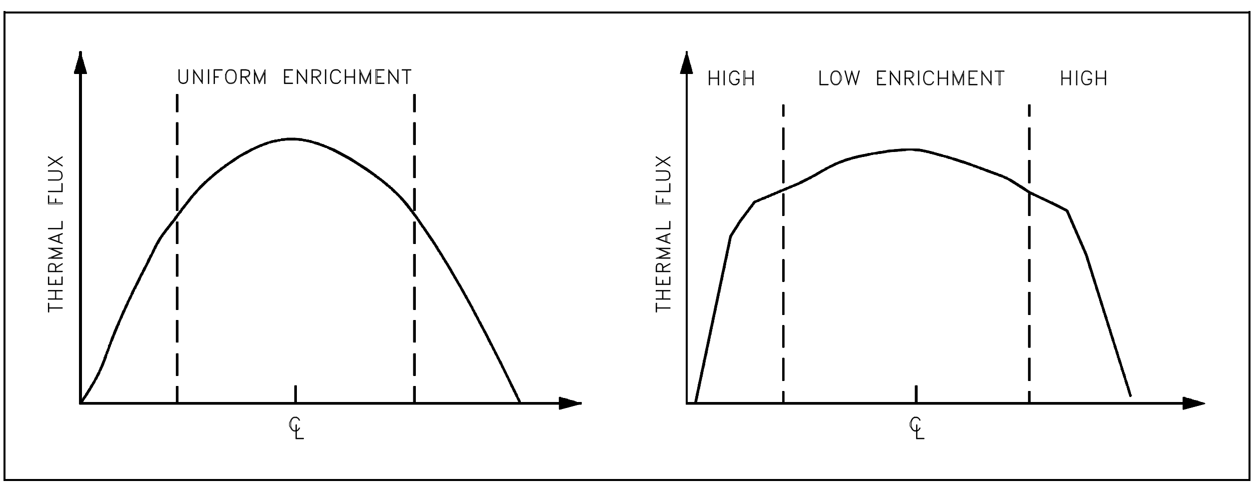

Figure 5 Effect of Non-Uniform Enrichment on Radial Flux Shape

The previous examples discuss changes in radial power distribution. Large variations also exist in axial power distribution. Figure 6(A) illustrates the power distribution that may exist for a reactor with a cylindrical geometry. The control rods in this reactor are inserted from the top, and the effect of inserting control rods further is shown in Figure 6(B). The thermal flux is largely suppressed in the vicinity of the control rods, and the majority of the power is generated low in the core. This flux profile can be flattened by the use of axial fuel and/or poison zoning.

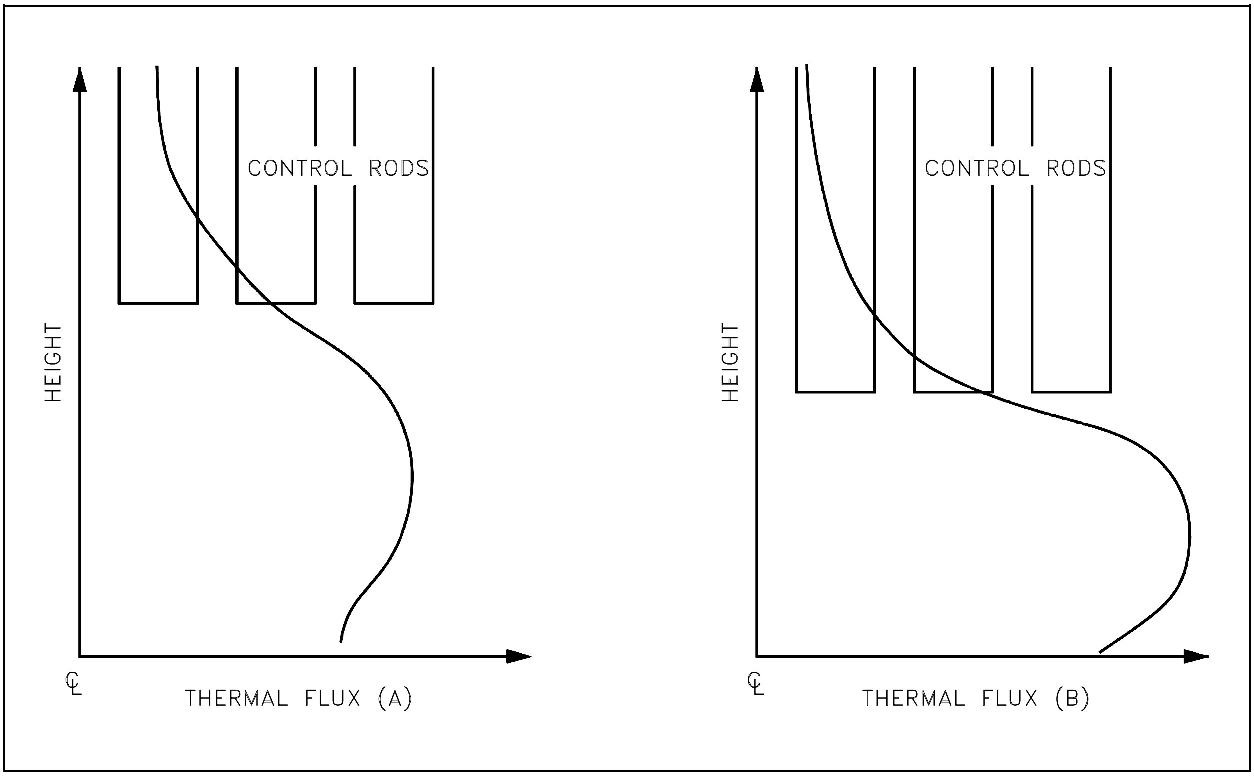

Figure 6 Effect of Control Rod Position on Axial Flux Distribution

Power Tilt

A *power tilt*, or flux tilt, is a specific type of core power distribution problem. It is a non-symmetrical variation of core power in one quadrant of the core relative to the others. The power in one portion might be suppressed by over-insertion of control rods in that portion of the core, which, for a constant overall power level, results in a relatively higher flux in the remainder of the core. This situation can lead to xenon oscillations, which were previously discussed.

Shutdown Margin

Shutdown margin is the instantaneous amount of reactivity by which a reactor is subcritical or would be subcritical from its present condition assuming all control rods are fully inserted except for the single rod with the highest integral worth, which is assumed to be fully withdrawn. Shutdown margin is required to exist at all times, even when the reactor is critical. It is important that there be enough negative reactivity capable of being inserted by the control rods to ensure complete shutdown at all times during the core lifetime. A shutdown margin in the range of one to five percent reactivity is typically required.

The stuck rod criterion refers to the fact that the shutdown margin does not take credit for the insertion of the highest worth control rod. The application of the stuck rod criterion ensures that the failure of a single control rod will not prevent the control rod system from shutting down the reactor.

Operation

During reactor operation, numerous parameters such as temperature, pressure, power level, and flow are continuously monitored and controlled to ensure safe and stable operation of the reactor. The specific effects of variations in these parameters vary greatly depending upon reactor design, but generally the effects for thermal reactors are as follows.

Temperature

The most significant effect of a variation in temperature upon reactor operation is the addition of positive or negative reactivity. As previously discussed, reactors are generally designed with negative temperature coefficients of reactivity (moderator and fuel temperature coefficients) as a self-limiting safety feature. A rise in reactor temperature results in the addition of negative reactivity. If the rise in temperature is caused by an increase in reactor power, the negative reactivity addition slows, and eventually turns the increase in reactor power. This is a highly desirable effect because it provides a negative feedback in the event of an undesired power excursion.

Negative temperature coefficients can also be utilized in water cooled and moderated power reactors to allow reactor power to automatically follow energy demands that are placed upon the system. For example, consider a reactor operating at a stable power level with the heat produced being transferred to a heat exchanger for use in an external closed cycle system. If the energy demand in the external system increases, more energy is removed from reactor system causing the temperature of the reactor coolant to decrease. As the reactor temperature decreases, positive reactivity is added and a corresponding increase in reactor power level results.

As reactor power increases to a level above the level of the new energy demand, the temperature of the moderator and fuel increases, adding negative reactivity and decreasing reactor power level to near the new level required to maintain system temperature. Some slight oscillations above and below the new power level occur before steady state conditions are achieved. The final result is that the average temperature of the reactor system is essentially the same as the initial temperature, and the reactor is operating at the new higher required power level. The same inherent stability can be observed as the energy demand on the system is decreased.

If the secondary system providing cooling to the reactor heat exchanger is operated as an open system with once-through cooling, the above discussion is not applicable. In these reactors, the temperature of the reactor is proportional to the power level, and it is impossible for the reactor to be at a higher power level and the same temperature.

Pressure

The pressure applied to the reactor system can also affect reactor operation by causing changes in reactivity. The reactivity changes result from changes in the density of the moderator in response to the pressure changes. For example, as the system pressure rises, the moderator density increases and results in greater moderation, less neutron leakage, and therefore the insertion of positive reactivity. A reduction in system pressure results in the addition of negative reactivity. Typically, in pressurized water reactors (PWR), the magnitude of this effect is considerably less than that of a change in temperature. In two-phase systems such as boiling water reactors (BWR), however, the effects of pressure changes are more noticeable because there is a greater change in moderator density for a given change in system pressure.

Power Level

A change in reactor power level can result in a change in reactivity if the power level change results in a change in system temperature.

The power level at which the reactor is producing enough energy to make up for the energy lost to ambient is commonly referred to as the *point of adding heat*. If a reactor is operating well below the point of adding heat, then variations in power level produce no measurable variations in temperature. At power levels above the point of adding heat, temperature varies with power level, and the reactivity changes will follow the convention previously described for temperature variations.

The inherent stability and power turning ability of a negative temperature coefficient are ineffective below the point of adding heat. If a power excursion is initiated from a very low power level, power will continue to rise unchecked until the point of adding heat is reached, and the subsequent temperature rise adds negative reactivity to slow, and turn, the rise of reactor power. In this region, reactor safety is provided by automatic reactor shutdown systems and operator action.

Flow

At low reactor power levels, changing the flow rate of the coolant through the reactor does not result in a measurable reactivity change because fuel and moderator temperatures and the fraction of steam voids occurring in the core are not changed appreciably.

When the flow rate is varied, however, the change in temperature that occurs across the core (outlet versus inlet temperature) will vary inversely with the flow rate. At higher power levels, on liquid cooled systems, increasing flow will lower fuel and coolant temperatures slightly, resulting in a small positive reactivity insertion. A positive reactivity addition also occurs when flow is increased in a two-phase (steam-water) cooled system. Increasing the flow rate decreases the fraction of steam voids in the coolant and results in a positive reactivity addition. This property of the moderator in a two-phase system is used extensively in commercial BWRs. Normal power variations required to follow load changes on BWRs are achieved by varying the coolant/moderator flow rate.

Core Burnup

As a reactor is operated, atoms of fuel are constantly consumed, resulting in the slow depletion of the fuel frequently referred to as core burnup. There are several major effects of this fuel depletion. The first, and most obvious, effect of the fuel burnup is that the control rods must be withdrawn or chemical shim concentration reduced to compensate for the negative reactivity effect of this burnup.

Some reactor designs incorporate the use of supplemental burnable poisons in addition to the control rods to compensate for the reactivity associated with excess fuel in a new core. These fixed burnable poisons burn out at a rate that approximates the burnout of the fuel and they reduce the amount of control rod movement necessary to compensate for fuel depletion early in core life.

As control rods are withdrawn to compensate for fuel depletion, the effective size of the reactor is increased. By increasing the effective size of the reactor, the probability that a neutron slows down and is absorbed while it is still in the reactor is also increased. Therefore, neutron leakage decreases as the effective reactor size is increased. The magnitude of the moderator negative temperature coefficient is determined in part by the change in neutron leakage that occurs as the result of a change in moderator temperature. Since the fraction of neutrons leaking out is less with the larger core, a given temperature change will have less of an effect on the leakage. Therefore, the magnitude of the moderator negative temperature coefficient decreases with fuel burnup.

There is also another effect that is a consideration only on reactors that use dissolved boron in the moderator (chemical shim). As the fuel is burned up, the dissolved boron in the moderator is slowly removed (concentration diluted) to compensate for the negative reactivity effects of fuel burnup. This action results in a larger (more negative) moderator temperature coefficient of reactivity in a reactor using chemical shim. This is due to the fact that when water density is decreased by rising moderator temperature in a reactor with a negative temperature coefficient, it results in a negative reactivity addition because some moderator is forced out of the core. With a coolant containing dissolved poison, this density decrease also results in some poison being forced out of the core, which is a positive reactivity addition, thereby reducing the magnitude of the negative reactivity added by the temperature increase. Because as fuel burnup increases the concentration of boron is slowly lowered, the positive reactivity added by the above poison removal process is lessened, and this results in a larger negative temperature coefficient of reactivity.

The following effect of fuel burnup is most predominant in a reactor with a large concentration of uranium-238. As the fission process occurs in a thermal reactor with low or medium enrichment, there is some conversion of uranium-238 into plutonium-239. Near the end of core life in certain reactors, the power contribution from the fission of plutonium-239 may be comparable to that from the fission of uranium-235. The value of the delayed neutron fraction (β) for uranium-235 is 0.0064 and for plutonium-239 is 0.0021. Consequently, as core burnup progresses, the effective delayed neutron fraction for the fuel decreases appreciably. It follows then that the amount of reactivity insertion needed to produce a given reactor period decreases with burnup of the fuel.

Shutdown

A reactor is considered to be shut down when it is subcritical and sufficient shutdown reactivity exists so there is no immediate probability of regaining criticality. Shutdown is normally accomplished by insertion of some (or all) of the control rods, or by introduction of soluble neutron poison into the reactor coolant.

The rate at which the reactor fission rate decays immediately following shutdown is similar for all reactors provided a large amount of negative reactivity is inserted. After a large negative reactivity addition the neutron level undergoes a rapid decrease of about two decades (prompt drop) until it is at the level of production of delayed neutrons. Then the neutron level slowly drops off as the delayed neutron precursors decay, and in a short while only the longest-lived precursor remains in any significant amount. This precursor determines the final rate of decrease in reactor power until the neutron flux reaches the steady state level corresponding to the subcritical multiplication of the neutron source.

The half-life of the longest lived delayed neutron precursor results in a reactor period of around -80 seconds or a startup rate of -1/3 DPM for most reactors after a reactor shutdown. One noticeable exception to this is a heavy water reactor. In a heavy water reactor, the photo-neutron source is extremely large after shutdown due to the amount of deuterium in the moderator and the large number of high energy gammas from short-lived fission product decay. The photo-neutron source is large enough to have a significant impact on neutron population immediately after shutdown. The photo-neutron source has the result of flux levels decreasing more slowly so that a heavy water reactor will have a significantly larger negative reactor period after a shutdown.

Throughout the process of reactor shutdown the nuclear instrumentation is closely monitored to observe that reactor neutron population is decreasing as expected, and that the instrumentation is functioning properly to provide continuous indication of neutron population. Instrumentation is observed for proper overlap between ranges, comparable indication between multiple instrument channels, and proper decay rate of neutron population.

A distinction should be made between indicated reactor power level after shutdown and the actual thermal power level. The indicated reactor power level is the power produced directly from fission in the reactor core, but the actual thermal power drops more slowly due to decay heat production as previously discussed. Decay heat, although approximately 5 to 6% of the steady state reactor power prior to shutdown, diminishes to less than 1% of the pre-shutdown power level after about one hour.

After a reactor is shutdown, provisions are provided for the removal of decay heat. If the reactor is to be shut down for only a short time, operating temperature is normally maintained. If the shutdown period will be lengthy or involves functions requiring cooldown of the reactor, the reactor temperature can be lowered by a number of methods. The methods for actually conducting cooldown of the reactor vary depending on plant design, but in all cases limitations are imposed on the maximum rate at which the reactor systems may be cooled. These limits are provided to reduce the stress applied to system materials, thereby reducing the possibility of stress induced failure.

Although a reactor is shut down, it must be continuously monitored to ensure the safety of the reactor. Automatic monitoring systems are employed to continuously collect and assess the data provided by remote sensors. It is ultimately the operator who must ensure the safety of the reactor.

Decay Heat

About 7 percent of the 200 MeV produced by an average fission is released at some time after the instant of fission. This energy comes from the decay of the fission products. When a reactor is shut down, fission essentially ceases, but decay energy is still being produced. The energy produced after shutdown is referred to as decay heat. The amount of decay heat production after shutdown is directly influenced by the power history of the reactor prior to shutdown. A reactor operated at full power for 3 to 4 days prior to shutdown has much higher decay heat generation than a reactor operated at low power for the same period. The decay heat produced by a reactor shutdown from full power is initially equivalent to about 5 to 6% of the thermal rating of the reactor. This decay heat generation rate diminishes to less than 1% approximately one hour after shutdown. However, even at these low levels, the amount of heat generated requires the continued removal of heat for an appreciable time after shutdown. Decay heat is a long-term consideration and impacts spent fuel handling, reprocessing, waste management, and reactor safety.

Summary

The important information in this chapter is summarized below.

Reactor Operation Summary

- An installed neutron source, together with the subcritical multiplication process, may be needed to increase the neutron population to a level where it can be monitored throughout the startup procedure.

- Reactivity balances, such as Estimated Critical Position calculations, typically consider the basic reactivity of the core and the reactivity effects of temperature, direct xenon, and indirect xenon.

- A reactivity balance called an Estimated Critical Position is used to predict the position of the control rods at which criticality will be achieved during a startup. To arrive at an ECP of the control rods, the basic reactivity, direct and indirect xenon reactivity, and temperature reactivity are added together to determine the amount of positive reactivity that must be added by withdrawing control rods to attain criticality. A graph of control rod worth versus rod position is used to determine the estimated critical position.

Reactor Operation Summary (Cont.)

- Three methods are used to shape or flatten the core power distribution.
  - Use of reflectors
  - Installation of neutron poisons
  - Axial or radial variation of fuel enrichment

- Power tilt is a non-symmetrical variation of core power in one quadrant of the core relative to the other quadrants.

- Shutdown margin is the instantaneous amount of reactivity by which a reactor is subcritical or would be subcritical from its present condition assuming all control rods are fully inserted except for the single rod with the highest integral worth, which is assumed to be fully withdrawn.

- The stuck rod criterion is applied to the shutdown margin to ensure that the failure of a single control rod will not prevent the control rod system from shutting down the reactor.

- Several factors may change during and after the shutdown of the reactor that affect the reactivity of the core.

  - Control rod position
  - Soluble neutron poison concentration
  - Temperature of the fuel and coolant
  - Xenon
  - Samarium

- Decay heat is always present following reactor operation due to energy resulting from the decay of fission products.

- The amount of decay heat present in the reactor is dependent on three factors.
  - The pre-shutdown power level
  - How long the reactor operated
  - The amount of time since reactor shutdown

- Decay heat immediately after shutdown is approximately 5-6% of the pre-shutdown power level. Decay heat will decrease to approximately 1% of the pre-shutdown power level within one hour of reactor shutdown.