Dynamic Characteristics of CANDU Reactors

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Summary:

Refinement of concepts important to today's CANDU power plant. Approximations used for separating those variables important for accident analysis, normal operation, fuel management, and structural changes. Effects of heterogeneity -- Lattice cell effects and neutron cross-section averaging.

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1. Neutron Chain Reaction

Figure 1 depicts the neutron cycle in a thermal reactor. (Fast reactors are somewhat different because fission occurs mostly at high energies, and neutrons either are absorbed or leak out before they slow down to thermal energy.)

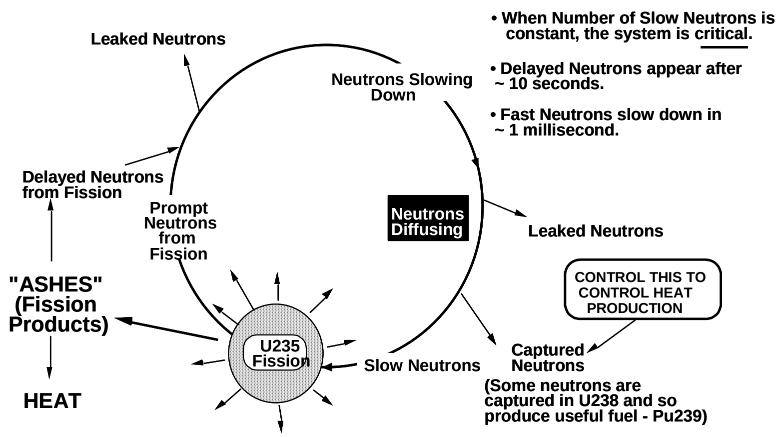

Figure 1 – The Neutron Cycle in a Thermal Reactor

In normal operation, a thermal reactor is very near to a “critical” state, wherein the neutron population is independent of time (at least on the time scale of seconds-to-hours.) The population of neutrons determines the fission rate and, therefore the amount of power produced. To increase power, we temporarily reduce the number of neutrons captured in control materials – the neutron population increases exponentially with time. When the desired power level is reached the control materials capture rate is again increased to keep the total neutron population constant. (Q – approximately, what reactivity increase is required to change the power level at a rate of 0.1 percent per second?).

2. Operating Domain (safe operating envelope)

Automatic control systems are designed to control the reactor at constant power within a multi-parameter domain, as shown in Figure 2 for a typical CANDU power reactor. Assume that the reactor has been designed to operate under “Design Center” conditions. During operation some state variables change and others are manipulated, causing the reactor’s state vector (operating trajectory) to “wander like a worm” in the n-dimensional space as time progresses. (The limitations of a two-dimensional sheet of paper are obvious in this Figure.)

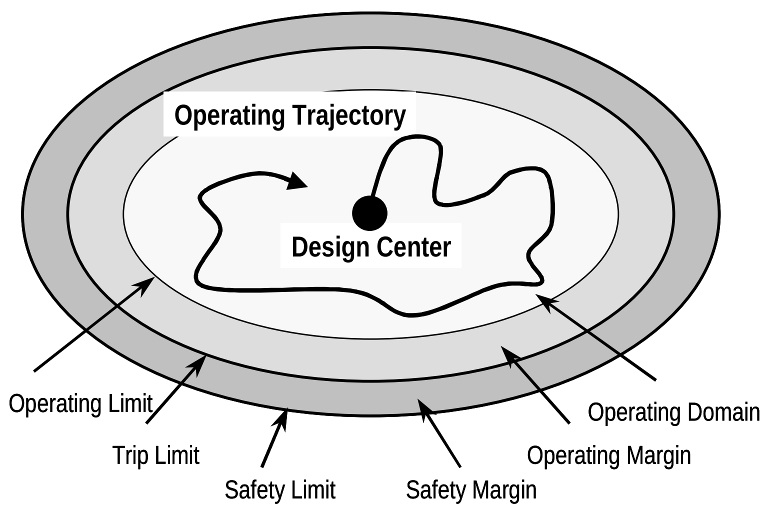

Figure 2 – Safe Operating Domain of Control Systems

The control system always directs the state vector toward Design Center, as shown in Figure 2. Perturbations such as fuelling, power changes, temperature changes, and so on tend to move the state vector toward the "Operating Limit". At this boundary, the normal control system acts immediately to return the state vector toward Design Center. If, for any reason, (e.g. a large perturbation) the normal control system cannot do this, the reactor state vector may continue moving toward the "Trip Limit" at which special safety systems respond automatically. These systems act quickly, and solely, to reduce reactor power and/or increase heat removal from the core. They are cocked or "poised" at all times for this purpose.

[Note: Several complex events and decisions led up to the disastrous Chernobyl Unit 4 reactor accident in April 1986. However, in the end, the accident was initiated by a combination of poor design and poor operation. Poor design of safety shutdown rods allowed the operators to reach an operating state wherein the first movement of safety rods (intended, of course, to shut down the reactor) actually led to a reactivity increase and consequent power increase. Poor operation was indicated by the fact that the operators deliberately put the reactor into an unsafe state prior to the accident.]

Startup, power level transients, steady-state operation, and shutdown transient states are included implicitly in the n-dimensional state space represented by the simple diagram in Figure 2. Add a few more state variables and we can describe subcritical and maintenance states. The concept is the same - the outer boundary of the Operating Domain in the Figure can be compared with (for example) an aircraft 'stall' warning. Protective actions are taken if the aircraft state reaches the boundary of the operating domain (or operating envelope) so as to prevent equipment damage. There is no comparison between aircraft and a reactor's automatic safety systems – as such these would be equivalent to a system that automatically <u>landed</u> the plane in case of trouble. The task equivalent to "landing" is relatively simple in the case of a reactor – shut off the chain reaction, close up the containment, and continue cooling the fuel. Aircraft do not have this ultimate protective barrier.

Another dimension is added to the operating domain when considerations of fuel depletion are important. Referring to Figure 1, it is clear that action must be taken to maintain a constant neutron population as the concentration of uranium-235 slowly decreases, and as the concentration of neutron-capturing fission products increases. In the short term the reactor control system acts to keep the state vector (Figure 2) inside the operating envelope. However, the control system eventually will reach the limit of its range and new fuel must be added. In reactors such as CANDU this fuel changing is done daily at full power. This leads to an operating advantage for this type of reactor because the reactor neutronic parameters reach an approximate equilibrium after a year or so of operation and remain constant after that time. Another advantage of this scheme is that it becomes unnecessary to change fuel with the reactor shut down – when it is more difficult to measure the degree to which the core state is below a 'critical' state.

In reactors such as the pressurized water reactor, fuel changing requires reactor shutdown. A large fraction of the fuel is changed at one time and then the reactor is restarted. This major fuel change requires that a large amount of neutron-absorbing material be added to control reactivity. As a result core properties are changed significantly by refuelling. The Beginning of Cycle state variables then change slowly over the cycle time of approximately one year, finally returning to End of Cycle conditions just prior to the next refueling.

3. Neutron migration area

Reactor size affects the spatial stability of power generation in any fission reactor. In general, large reactors are less stable than are small reactors. (We will see the mathematical expression of this fact later in this course.)

The characteristic size that is most relevant to comparison of different reactor types, at a given power output, is the size expressed in units of the mean travel distance (migration length) between production and absorption in the core. We see from Table 1 that the HWR and the FBR are much “smaller” in terms of neutron diffusion that either the PWR or the BWR. As a result, these reactors are more stable. In detail, other factors play a part in determining core stability, as will be examined later.

Table 1: Characteristic Sizes of Power Reactors

Reactor Type	Reactor Core Diameter (cm., typ.)	Neutron Migration Length (cm., typ.)	Characteristic Size (migration lengths)
PWR	370.	6.6	56
BWR	470	7.3	64
HWR	700.	20	35
FBR	175.	5	35

The optimum power output of a CANDU reactor for a given number of channels is obtained when the reactivity of approximately half of the fuel channels, located in the center of the reactor, is deliberately lowered so that the flux (and therefore the power) is constant across the so-called ‘inner zone’ of the core. This flux flattening results, of course, in decreased stability in this portion of the reactor – a small addition of reactivity in a flat-flux zone produces a relatively large flux change. This condition usually is avoided by slightly reducing the reactivity of a few fuel channels near the center of the reactor, so as to restore sufficient stability conditions at minor cost in terms of power output.

Large thermal reactors are subject to another sort of instability, this one induced by production, capture, and decay of the isotope Xenon-135. The best design strategy to eliminate resultant power fluctuations is to install an automatic control system such as that in the CANDU-HWR. Generally, it is impractical to control these fluctuations manually.

Reflectors are used in all power reactors to reduce neutron losses and to assist in flattening the power distribution. Escaping neutrons are captured in a uranium radial blanket in the case of the FBR. Production of plutonium for new fuel is greatly enhanced by this design feature. Reflectors are relatively unimportant in the LWR because of their very small migration length and the presence of a strongly absorbing steel core barrel surrounding the fuel.

The CANDU-HWR uses a heavy water radial reflector. This reflector also has an important effect on reactor dynamics because of its very long diffusion length.

The mean flight path length of a neutron between collisions (the "mean free path") gives an indication of the degree of inhomogeneity of any reactor assembly consisting of fuel elements and coolant/moderator. If this flight path is long relative to the thickness/diameter of a fuel element at all neutron energies, the reactor is said to be "quasi-homogeneous'. Most fast reactors are nearly homogeneous in this sense. The McMaster Nuclear Reactor is typical of a quasi-homogeneous thermal reactor.

In most thermal power reactors the flight path at some neutron energies is short relative to the dimension of a fuel element, or to the moderator thickness between fuel elements. These reactors must be treated as "heterogeneous", with consequent higher complexity in their analysis. The LWR and the HWR are typical heterogeneous reactors. Heterogeneity can be seen at more than one level; for example, mechanical control absorbers in a CANDU exhibit a second level of heterogeneity that must also be considered in analysis. These effects will be discussed in later chapters where specific cases arise that are important to reactor dynamic behaviour.

4. Reactor dynamics when the reactor is shut down

Many of the available theoretical treatments of reactor dynamics do not address the important field of the shut down or "subcritical" states of the reactor. This observation should be considered along with the historical observation that the majority of reactor accidents actually began from a shutdown state. One could postulate several reasons for this common omission, and the omission itself should not be blamed for all such accidents. However, if the lack of knowledge of fission reactors in their subcritical state might have been responsible even partly for one such accident, we are fully justified in studying this case.

As written, the simple neutron kinetics program referred to in the Week 1 course notes does not consider the subcritical condition. In order to do so we must add another term to the equation solved therein. The following equation set defines the point-reactor kinetic behaviour in a reactor containing an external neutron source. This equation will be derived later in this course – however, it is expected that the student already is familiar with the general form:

$$\frac{dN(t)}{dt} = \left[\frac{\rho(t)-\tilde{\beta}}{\ell(t)} + \frac{k_0 - 1}{k_0 \ell(t)}\right] N(t) + \sum_s \lambda_s \tilde{C}_s(t) + \tilde{S}(t)$$

$$\frac{d\tilde{C}_s(t)}{dt} = \frac{\tilde{\beta}_s}{\ell(t)} N(t) - \lambda_s \tilde{C}_s(t) \qquad (s = 1,6)$$

$$\rho(t) = \frac{1}{D(t)} \left\langle \phi^* \middle| F(t) - M(t) \middle| \psi(t) \right\rangle$$ - Reactivity

$$\tilde{\beta}_s = \frac{1}{D(t)}\left\langle \phi^* \middle| \beta_s F(t) \middle| \psi(t) \right\rangle$$ - Effective Delayed Neutron Fraction

F(t) and *M*(t) are, respectively, the space-and-time dependent fission neutron source and sink.

D(t) is the weighted integral of the fission neutron source (inner product).

$$\sum_s \tilde{\beta}_s = \tilde{\beta} \qquad (s = 1, 6)$$ - Delayed Neutron Fraction

$$\ell(t) = \frac{1}{D(t)}\left\langle \phi^* \middle| 1/v \middle| \psi(t) \right\rangle$$ - Prompt Neutron Lifetime

$$\tilde{C}_s(t) = \frac{1}{D(t)\ell(t)}\left\langle \phi^* \middle| C_s(t) \right\rangle$$ - Effective Neutron Precursor Concentration

$$\tilde{S}(t) = \frac{1}{D(t)\ell(t)}\left\langle \phi^* \middle| S(t) \right\rangle$$ - Effective Source Strength

The non-physical parameter k_0 is the inverse of the largest eigenvalue of the source-free reactor equation, for the known subcritical steady state.

At steady-state initial conditions, the left hand side of each of the seven governing equations is zero and by definition, $N(0) = 1.0$, and $\rho(0) = 0$. Therefore,

$$\tilde{C}_s(0) = \frac{1}{\lambda_s}\frac{\tilde{\beta}_s}{\ell} \qquad (s = 1, 6) \qquad \text{and}$$

$$N(0) = 1.0 = -\frac{\ell \tilde{S}(0)}{\left(\frac{k_0 - 1}{k_0}\right)}$$

Existence of a steady-state solution requires that $k_0 < 1.0$; that is, the reactor must be subcritical. As the reactor approaches a critical condition via any given state trajectory (for example by increasing fuel concentration or by reducing the neutron leakage) k_0 tends toward unity and $N(0)$ increases. But if the initial source is identically zero, there will be no neutrons present and so $N(0) = 0.0$ even if $k_0 > 1.0$. This (hypothetical) situation obviously is very dangerous because addition of only <u>one</u> neutron to the system may lead to a very rapid energy release. Such events in real systems (for example the recent ‘flash criticality’ accident in the fuel processing facility at Tokai in Japan and the fuel loading incident in the McMaster Nuclear Reactor) usually result from mistaken physical changes that alter the reactor state from k_0 less that unity to k_0 greater than unity.

It is obvious that one <u>necessary</u> condition for safety in a subcritical reactor is that an external neutron source be present at all times. Its strength requirement is determined by the resultant neutron flux level and the sensitivity of neutron detectors – these detectors also are essential so that any change in the state of subcriticality can be detected by the operating staff and so that automatic safety systems will have time to respond. The second part of this safety requirement is, of course, that sudden physical changes that lead to an increase in k_0 be limited in rate or in absolute magnitude.

The 'approach to critical' (or otherwise, the methods for safely starting up a reactor) will be addressed later in this course.

Referring back to Figure 1, it is obvious that the presence of an external source of neutrons means that an operating reactor will never be precisely critical. It must be kept in a state slightly below critical so as to maintain constant power. However, this aspect of operation becomes important mainly under shutdown conditions. Here again, the HWR is unique in that the gamma-neutron reaction in deuterium produces a significant neutron source under all conditions except the first startup with all-fresh fuel. Spontaneous fission of some actinide elements also produces neutrons, but in small numbers. In all cases in which there is no other large source of neutrons, a special neutron-emitting source must be added to the reactor during startup.

Consideration of reactor dynamics under shutdown conditions also must include the dynamics of high-absorption fission products, especially the Iodine-Xenon chain. Following shutdown from high power, the Xenon concentration increases so that a thermal reactor generally cannot be restarted. The eventual decay of the Xenon-135 isotope permits restart after 30-40 hours' delay. Restarting a thermal reactor containing substantial amounts of Xenon-135 requires careful manipulation of in-reactor control systems, because the initial power increase 'burns up' this residual Xenon and increases the reactivity of the core.

Another important special situation arises after reactor shutdown from high power. Fission products produce about 7 percent of the total heat during operation; after the chain reaction is shut off, this heat is still produced. The amount decays to about 1 percent of initial power after 3 hours, and continues to decrease thereafter. Nevertheless, removal of this 'decay heat' is an important consideration in fuel cooling. [Note: During the 1979 accident at Three Mile Island, the fuel was well cooled for about 2 hours following reactor shutdown. Mistakes in operation then led to removal of cooling from some fuel after that time; the fuel that was uncooled then melted (at ~2800 C) and posed a significant risk to the integrity of the pressure vessel. Fortunately, this molten fuel was re-solidified without extensive heat transfer to the vessel wall.]

5. Spatial Heterogeneity

Figures 1 to 4 of the general description given in [Link-reference to Lecture 5 of "6-Lecture Course"] show the geometry of the CANDU-6 reactor. Those Figures can be "zoomed" to view details of the reactor core design. The most notable features for our purposes are (a) the horizontal (z-direction) orientation of fuel channels, the (b) vertical (y-direction) orientation of some control devices, and the (c) horizontal (x-direction) orientation of other control devices. The fuel channels are filled with fuel plus heavy water; the control assemblies are either filled with neutron-absorbing material or heavy water. The absorbing and neutron producing properties of the fuel change as the fuel is irradiated. Spaces between these structures are filled with pure heavy water moderator.

It is obvious that the modeling of neutron migration in this complex lattice depends not only on the properties of the isotopes, but also on their location in the lattice as well as on the proximity of other isotopes in other parts of the lattice. This extremely heterogeneous geometry is simplified to some extent by the fact that most of the neutrons present are well thermalized. A typical neutron flux distribution in a CANDU lattice is shown in Figure 3. As can be seen, most of the fast neutrons produced in the fuel diffuse to the moderator region where they are slowed down to thermal energies, and then diffuse back to the fuel where they are absorbed.

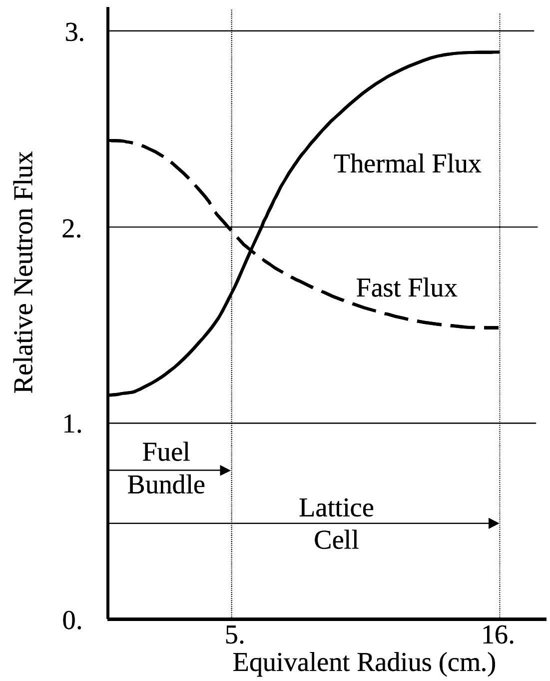

Figure 3 – Typical Flux Distribution in CANDU Lattice Cell

Properties of the core materials are normally averaged over discrete volumes and over discrete energy intervals and collected in tabular form suitable for interpolation over space and fuel irradiation so as to select appropriate values to represent the specific state of the core (fuel irradiation, temperature, near-neighbor effects) at any location. The time-dependence of these properties (due to changes in density, temperature, etc.) also is pre-calculated and stored in interpolation tables for use in dynamic analysis.

Most reactors must shut down for refueling. In these systems, extensive computer simulations are done to select the optimum new-fuel loading (enrichment, burnable poison concentrations, spatial core layout, excess reactivity requirements, etc.) for the next ‘cycle’ of about one year duration. Once the final calculation is made fuel is ordered and the decisions become very difficult to reverse. The CANDU system is fuelled at full power, at a rate of 15-20 identical fuel bundles per day (2-3 fuel channels). Pre-calculation is necessary to select the best channels for refueling, to maintain the correct local power distribution, core reactivity, etc. Neither enrichment nor burnable poison is required. The shift supervisor makes the final choice of channels to be fuelled and the fuelling system operator initiates the automatic fuelling system.

Addition of new fuel in CANDU initiates a small reactor transient because the new-fuel reactivity is higher than that of old fuel. The zone controllers compensate this reactivity addition automatically, to maintain reactor power and correct local power levels. Removal of any failed fuel elements is essentially identical to refueling. New bundles are added to replace those containing failed elements.

6. Long-term effects

Power reactors must be designed to operate for 30 years or more. During such long periods, the high-intensity neutron and gamma irradiation leads to degradation of materials. Examples are: radiolysis of water, transport of active corrosion products to inhabited areas, swelling and embrittlement of steel, production and transport of tritiated water, and creep of zirconium pressure tubes. Such processes are an intrinsic part of the “dynamics” of any power reactor’s operation even though their characteristic time scale is quite long, because the operating life of a power reactor may itself extend for 60 years or more.

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About this document

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