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Core Design Criteria - Physics

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Two types of criteria, those related to safety/licensing, & those related to the intended function of the reactor – run at a certain power level, flux level, flux in beam port, run for a certain time or for a certain energy generation for power reactor. Usually multiple objectives.

For LWR power reactors:.

Units of **cycle length** - burnup (MWD/MTU, MWD/STU, MWD/kgU, FIMA), cal days, fpd, efpd, GE uses MWD. Typical are 12, 18, 24 calendar month cycles. Typical refuel outage is 21 to 42 days. Typical forced outage rate 3 to 5%. The physical properties of the fuel degrade with burnup (more about this later), so the discharge burnup of the fuel assemblies (their burnup when they are discharged from the core) is an important parameter.

Specific power - thermal power per mass of heavy metal, kw/kgU or kw/kgHM. Typically about 40 kw/kgU in PWRs, 20 kw/kgU in BWRs.

The numerical value of the specific power equals the burnup per full power day. (Watch the units)

Power density - thermal power per volume of core, usually expressed in kw/l of core. Typically about 100 kw/l in PWRs and 50 kw/l in BWRs.

Thermal efficiency = electric power/thermal power, typically 32 - 33% in LWRs.

Heat rate = thermal BTU / KWhelectric
The pure conversion factor from BTU to KWh is 3413 BTU/KWh, so the efficiency is 3413/ (heat rate).

Note that efficiency & heat rate depend on status of secondary systems and temperature of heat sink.

Capacity factor over some time period = energy actually generated/ energy that would have been generated if unit ran at full power all the time. Thermal and electrical capacity factors can be and usually are different.

When all cycles are identical (a useful idealization known as "equilibrium"), a useful relationship between the above quantities ensues. Let:

- B_c = cycle burnup
- B_d = discharge burnup
- n = number of equal size regions (typically 2 to 4 in LWRs)
- S = specific power
- T = calendar length of every cycle
- L = capacity factor

THEN
the cycle burnup is

$$B_c = STL$$

and since batches are all in-core for n cycles,

$$B_d = nSTL$$

NOTE 1: can define capacity factor based on power operating period (startup after refueling to shutdown for next refueling, or on total cycle length. But the definition of T has to be similar.)

NOTE 2: for non-integral n, this latter relationship is still exactly true if the batch power distribution is flat (all batches have the same power), and is approximately true if batch powers are different.

SAFETY/LICENSING
[See 1 page reading: "ATOMIC ENERGY ACT TO TECH SPECS"]

Insofar as the light water reactor is concerned, the health and safety of the public is protected by maintaining the integrity of the three barriers to the release of radioactive material - the fuel cladding, the pressure vessel, and the containment. This is achieved by assuring that no "anticipated operational occurence" will destroy their integrity (some breaches are allowed for unlikely,severe accidents) and this in turn is assured by maintaining various reactor physics parameters within certain bounds.

The reactor physics part of safety and licensing is a matter of four general topics:

- power distribution in the core,
- reactivity and its control,
- delayed neutron fraction, and
- prompt neutron lifetime.

Power distribution is usually expressed in terms of ratios to the average power. A number of ratios are in common use and core power distributions are commonly displayed as maps of these ratios:

- Relative assembly or fuel rod power, the enthalpy rise hot channel factor, the maximum value of which is $F_{\Delta h}$; this is the ratio of the power in the selected or hottest fuel rod (assembly) to that in the average fuel rod (assembly) in the core.

- Relative axial power, whose maximum value is F_z; this is the ratio of the power in the selected or hottest axial slice of the core to that in the average slice. Sometimes people refer to the axial power distribution in a particular assembly or particular fuel rod.
- Relative hot spot power, Fq, the ratio of the power at the hottest point to the average power. Also known as the heat flux hot channel factor. In a core where the power distribution is separable into a radial and an axial component (rarely true in real life), $Fq = F_{\Delta h} * F_z$;
- F_{local}, the ratio of the hottest rod power in a particular assembly to the average rod power in that assembly.

Reactivity control

Reactivity = net neutron produced per neutron produced

Long term reactivity control must allow for the negative effects due to depletion of fissile isotopes (U235 in most LWRs), fission product buildup, and equilibrium xenon in LWRs (as well as Pa in reactors containing thorium), plus the positive effects of buildup of Pu in present day LWRs and of other fissile transuranics in advanced PWRs or other types of reactors.

Long term reactivity control [CE graph] – achieved by flow control and control rods in BWRs, by boron shim (=boron dissolved in cooling water) in PWRs, with the help of burnable poison in both types of reactor.

Short term reactivity control must allow for the effects of power level changes - which usually appear through resulting changes in fuel or water temperature, and (particularly in BWRs) through gross changes in water density. In BWRs, core flow is directly controllable and this directly changes the average void in the core.

Response of the reactor is related to its total reactivity. Whether or not you calculate it with a more complicated 3 dimensionsal method, this is best understood phenomenologically through the point kinetics equation:

$$dP/dt = (\rho - \beta) P/ l + \Sigma\lambda C$$

$$dC/dt = \beta P/ l - \lambda C$$

where there are actually six sets of lambdas and betas, and ρ is the sum of all the reactivity effects discussed. More recently, space-time kinetics calculations are sometimes used, but phenomenological explanations always revert to the point kinetics formalism.

The effective delayed neutron fraction is primarily a function of the fissioning species and is not really adjustable by the reactor designer (except by changing the total character of the reactor). Fast fission in U238 has a larger β than that of U235, and thermal fission in Pu239 has a much smaller value.

Similarly, the prompt neutron lifetime is not easily adjustable by the designer.

The total reactivity to be used in the point kinetics equation can be considered as the sum of individual effects starting from hot, full power, beginning of cycle:

$$\rho = \rho_{COLD,BOC} + \rho_{DOP} + (\rho_{MODTEMP} + \rho_{SOLB}) + \rho_{XE} + \rho_{SM} + \rho_{CR}$$

$$+ \rho_{UDEP} + \rho_{FP} + \rho_{PU} + \rho_{MA} + \rho_{BP} + \rho_{OTHER}$$

The cold , beginning of cycle reactivity must be sufficient to get the core up to full power and to provide the desired core lifetime. At BOC, the reactivity effects of U235 depletion (UdEP), the fission products (FP), the plutonium (PU), and the minor actinides (MA) are zero. In the cold, BOC condition, the positive reactivity is balanced by the negative reactivity of the burnable poison (BP) included in the fuel, by a high concentration of soluble boron (SOLB) in the moderator, and by the insertion of the control banks of control rods (CR).

As core power is increased, the temperature of fuel and moderator rise yielding negative reactivities for the DOP and MODTEMP; the negative xenon reactivity reaches its equilibrium value within a few days, and samarium (SM) reaches its equilibrium in about a month. The temperature changes are usually offset by withdrawal of the control rods, and the xenon and samarium by a combination of a reduction in soluble boron and further withdrawal of control rods to nearly their full out position.

As the core burns, the long term reactivity effects appear – the negative effects of U235 depletion, fission product buildup and minor actinide buildup; plus the positive effects of plutonium buildup and burnup of the burnable poison. The net effect of the long term reactivity effects is offset by a reduction in soluble boron concentration.

Short term reactivity effects are often addressed in terms of reactivity coefficients, the change in reactivity per unit change in fuel temperature or power for doppler reactivity, coolant void fraction for void coefficient, or coolant temperature for moderator temperature coefficient (MTC). Since the soluble boron is dissolved in the cooling/moderating water, its concentration changes as the water density changes. Thus the moderator temperature coefficient and void coefficient always include the effect of changes in the soluble boron. Note that these are the coefficients in LWRs; other reactor types may have other important reactivity coefficients.

To assure control of the reactor, it is desired that the total power coefficient be negative; that is, an increase in reactor power should naturally decrease the reactivity. In typical LWRs, the doppler coefficient is always negative due to increased resonance capture in U238 as the temperature increases. This provides an immediate negative reactivity response to increases in power. However, if, for example, fertile free fuel (FFF - fuel in which the fissile isotope is embedded in non-fissile material without the fertile U238) is used, then the doppler coefficient may not always be negative.

The MTC in PWRs is normally negative, but can become positive if enough soluble boron is dissolved in the coolant for the purpose of controlling long term reactivity. This is of somewhat less concern because the water temperature changes more slowly than the fuel temperature since the heat requires time to flow. In PWRs, however, the NRC does not allow positive MTC at full power, although it is sometimes allowed during the power ascension.

It should be understood that the separation of total reactivity into separate parts due to fuel temperature, moderator temperature, fission product concentration, etc is a reasonable way of understanding what is transpiring, but generally represents an inexact way of calculating exactly what is happening because all the effects are really intertwined: for example, the moderator temperature coefficient depends not only on the moderator temperature and soluble boron concentration, but also on control rod insertion, core burnup, etc. Fortunately, most of the important dependencies have been identified.

Other physics issues:

- Decay heat in the core after shutdown, and in the spent fuel;
- Neutron production in the spent fuel;
- Discharge fissile enrichment & proliferation (20 w/o enr limit, SWU to reach weapons grade) and related concerns about possible criticality of spent fuel;
- Fast fluence to the reactor vessel.
- Criticality prevention in the fresh fuel storage area, spent fuel pit, and dry storage.