FOUR FACTOR FORMULA

Multiplication Factor, \( k \)

\[
k = \frac{\text{number of neutrons in current generation}}{\text{number of neutrons in previous generation}}
\]

\[
k_{\text{eff}} = k_{\infty} \xi_{\text{th}} \xi_{\text{fast}} \quad k_{\infty} = \eta f \epsilon \phi
\]

\[
k < 1 \quad (\rho < 0) \quad \text{subcritical} \quad B_m^2 < B_g^2
\]

\[
k = 1 \quad (\rho = 0) \quad \text{critical} \quad B_m^2 = B_g^2
\]

\[
k > 1 \quad (\rho > 0) \quad \text{supercritical} \quad B_m^2 > B_g^2
\]

Reproduction Factor, \( \eta \)

\[
\eta = \frac{\text{Number of neutrons produced by fission}}{\text{Number of neutrons absorbed by fuel}} = \nu \frac{\sigma_f^{\text{Fuel}}}{\sigma_a^{\text{Fuel}}} = \frac{\nu}{1 + \alpha}
\]

where \( \nu \) is the number of neutrons produced per fission, and \( \alpha \) is the capture-to-fission ratio, \( \alpha = \sigma_c / \sigma_f \).

Thermal Utilization, \( f \)

\[
f = \frac{\text{thermal neutrons absorbed by fuel}}{\text{total thermal neutrons absorbed}} = \frac{\sum_a^{\text{Fuel}}}{\sum_a^{\text{Total}}} = \frac{\sum_a^{\text{Fuel}}}{\sum_a^{\text{Fuel}} + \sum_a^{H_2O} + \sum_a^{\text{Steel}} + \cdots}
\]

Fast Fission Factor, \( \epsilon \)

\[
\epsilon = \frac{\text{total fission neutrons from thermal and fast fission}}{\text{fission neutrons from thermal fission}}
\]

Resonance Escape Probability, \( \phi \)

\[
\phi = \frac{\text{number of neutrons slowing to thermal energy}}{\text{total number of fast neutrons available for slowing}}
\]

Non-leakage Probabilities, \( \xi \)

\[
\xi_{\text{NL}} = \frac{\text{absorption}}{\text{production}} = \frac{\text{absorption}}{\text{absorption} + \text{leakage}} = \xi_{\text{th}} \xi_{\text{fast}}
\]

\[
\xi_{\text{fast}} = e^{-B_g^2 \tau} \quad \xi_{\text{th}} = \frac{\sum_a}{\sum_a + DB_g^2} = \frac{1}{1 + L_{\text{th}}^2 B_g^2}
\]

where \( D \) is the diffusion coefficient, \( B_g^2 \) is the geometric buckling, and \( L \) is the diffusion length.

Conversion Ratio (Breeding Ratio)

\[
CR = \frac{\text{Average rate of fissile atom (Pu – 239) production}}{\text{Average rate of fissile atom (U – 235) consumption}}
\]

\[
= \frac{\text{No. of neutrons absorbed in U – 238}}{\text{No. of neutrons absorbed in U – 235}} \approx \frac{\sum_a^{U-238}}{\sum_a^{U-235}}
\]

\[
= \eta_{U-235} \epsilon (1 - \phi) \xi_{\text{fast}} + \frac{\sum_a^{U-238}}{\sum_a^{U-235}}
\]