Impact of spallation neutrons on criticality

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Neutron spectrum in an accelerator driven sub critical system is different to the neutron spectra of thermal and fast reactors. The spallation neutron spectrum falls over a nuclear fuel in an ADS. Thus, for an arbitrary nuclear fuel, production and absorption cross sections of neutrons would vary with the incident neutron energy and the criticality coefficient, $k_{\text{eff}}$ is expected to behave differently to a thermal or the fast reactor. For developing methods of estimation of $k_{\text{eff}}$ in an ADS, let us assume that there is a cylindrical spallation target of dimension $d \times L = 20 \times 50$ cm$^2$ enclosed in side a fuel blanket of thickness $X$. The energy spectra of spallation neutrons produced in collision of 1GeV proton with a thick lead target is simulated using the CASCADE code [1] and the standard neutron spectra [2] of the thermal and a fast reactor have been used to estimate the spectrum average cross sections of different reactions occurring in different fuel elements such as $^{232}$Th and $^{235}$U of an arbitrarily assumed fuel system of an ADS. Spallation neutron spectrum spread up to several hundred MeV, may cause several reactions such as $^{232}$Th (n,3n) ($E_{\text{th}} = 11.61\text{MeV}$), $^{232}$Th (n,3np) ($E_{\text{th}} = 18.82\text{MeV}$) and $^{232}$Th (n,8n) ($E_{\text{th}} = 43.55\text{MeV}$) etc. that were not possible in a critical reactor. The main problem arising because of spallation neutron spectrum is that in the energy range $E_n > 20\text{MeV}$ the experimental data is scarce and in this situation one has to depend on the calculations from the models or a deterministic code and a combination of a deterministic code and a Monte Carlo simulation code. Thus, one can estimate $k_{\text{eff}}$ from the knowledge of cross sections of all the production channels such as, i) single neutron type – (n,n’), (n,np), (n,nd), (n,nt), (n,nα), (n,nh) and (n,n2p) ii) multiple neutron type – (n, xn) where x = 2, 3, ..., (n, f), (n,2nh), (n,2nd), (n,2nt), (n,2npd), (n,2n2p), (n,3np), (n,3nd), (n,3nh), (n,3nt) and (n,3nα) and iii) neutron removal type – (n,γ), (n, p), (n, d), (n, t) and (n,α). The (n, f) reaction contributes in both production and the removal channels. Considering all the aforesaid reaction channels we have calculated [3] the $k_{\text{eff}}$ for the three neutron spectra by using the following formulas and further details of the procedure will be published elsewhere. Let us assume $I_0$ is the incident neutron intensity falling on a material then the intensity of surviving neutrons after passing through $x$ distance, $I_x = I_0 \exp (-x \Sigma_t)$, here, $\Sigma_t$ is the total macroscopic cross section of a neutron in the given material, Considering $I_0 = 1$ for a single neutron intensity, then the removal term, R of the neutron by way of all removal processes, (n, γ), (n, p) and (n,2n) etc. may be written as,

$$R = (1-\exp (-x \Sigma_t)) \left[ \Sigma(n, \gamma) + \Sigma(n, p) + \Sigma(n, d) + \Sigma(n, nd) + \Sigma(n, nt) \right] / \Sigma_{\text{tot}} \tag{1}$$

and the remaining fraction, 1-R = L may be assumed to leak out from the fuel system. Similarly, a production term, P may be written as follows,

$$P = (1-\exp(x \Sigma_t)) \left[ 2\Sigma(n, 2n) + 3\Sigma(n, 3n) + 4\Sigma(n, 4n) + 9\Sigma(n, 9n) + <v>\Sigma(n, f) + 2\Sigma(n, 2np) + 3\Sigma(n, 3n\alpha) + ... \right] / \Sigma_{\text{tot}} \tag{2}$$

Here $<v>$ is fission neutrons of a fuel element. In the estimation of $P$, $R$ and $L$ contribution of the (n, n’) channel is not included because of the obvious reasons. Thus, $k_{\text{eff}} = P / (R+L)$ \tag{3}
The elementary cross sections of different reactions up to 250 MeV neutron energy are calculated using the TALYS-1.0 code and for the energy 250 MeV the values of the cross sections are considered constant at the last value corresponding to 250 MeV. This is a fair approximation because n-flux at $E_n > 250$ MeV is very small.

From the data given in figs 1a) and b) we can infer that for the spallation spectrum $k_{\text{eff}}$ is dominant in case of $^{232}$Th while it is below the values corresponding to the thermal & fast spectra in case of $^{235}$U. This shows that (n,xn) reaction play dominant role in case of fertile Th-fuel because of the presence of high energy neutrons compared to the fissile $^{235}$U fuel. This kind of behavior was pointed out earlier in ref. [4] in a detailed study of (n,xn) reactions in fertile and fissile fuels.

We have compared results of our calculations with that estimated by Cullen et al. [5] and a good agreement is seen in the two calculations.

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References


