PLUTONIUM RECYCLE IN HEAVY WATER-POWER REACTORS

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The performance of a heavy-water reactor using U\textsuperscript{238} with an equilibrium concentration of U\textsuperscript{239}, Pu\textsuperscript{240}, Pu\textsuperscript{241} nuclei and small additions of U\textsuperscript{235} and with natural and depleted uranium make-up is discussed. A portion of the spent fuel discharged from the reactor is cleaned up of fission fragments and recycled to the reactor. Another (smaller) portion is withdrawn from the cycle after plutonium has been extracted and is replaced in the core by natural uranium.

It is good practice to use only a small portion of the available natural uranium in thermal power reactors with no provisions for fuel regeneration in the cycle. If the use of natural uranium in these reactors could be successfully stepped up 5 to 10 times, and the fuel consumed per electric power output reduced in the same portion, there could be an appreciable expansion in the fuel resources available to thermal nuclear power reactors and a consequent increase in nuclear power station ratings.

Approximate estimates indicate that natural and depleted uranium can be used in high-level heavy-water reactors burning U\textsuperscript{238} with an equilibrium concentration of plutonium and U\textsuperscript{235} additions, by raising fuel burnup to 30 kg fission fragments or more per ton of natural uranium expended in the fuel cycle. This conclusion applies in part to other types of thermal reactors burning natural and enriched uranium.

The concentrations of plutonium isotopes in the fuel of thermal power reactors using natural or enriched uranium with no fuel recycling are far from equilibrium concentrations, and in this case the amount of electric power generated will be primarily at the expense of the U\textsuperscript{235} used up.

Consider the operation of a continuously reloaded thermal reactor in steady-state operation at a steady-state equilibrium (or near-equilibrium) concentration of Pu\textsuperscript{239}, Pu\textsuperscript{240}, Pu\textsuperscript{241} and concentrations of U\textsuperscript{235} and of fission fragments in the U\textsuperscript{238} maintained at constant levels. In order to achieve this mode of operation, this part of the fuel has to be discharged from the reactor as soon as the critical concentration of fission fragments is attained in a portion of the fuel, and it must be replaced by fresh fuel. Most of the discharged fuel must be cleaned up to get rid of fission fragments and recycled with the plutonium back to the core, and the remainder withdrawn from the cycle as spent fuel after the plutonium has been extracted (U\textsuperscript{238} with minimal U\textsuperscript{235} content), and replaced by natural uranium or by depleted uranium, so calculated as to maintain the specified average design concentration of U\textsuperscript{235} in the reactor. The plutonium extracted from this second portion of the fuel is either recycled to the reactor or withdrawn from the cycle, depending on whether the reactor is only on power service or whether it is also being used to breed plutonium for fast power reactors. Chemical reprocessing thus reduces in practice to purifying the spent fuel of fission fragments.

It might appear that fuel reprocessing connected with the maintenance of a constant quantity of fission fragments and constant concentration of U\textsuperscript{235} and plutonium in the U\textsuperscript{238} necessitates a sizable increase in the capacity of spent fuel reprocessing plants and of fuel element manufacture. But this is not the case, since given the same residence time of uranium in the reactor and the same burnup, the amount of fuel discharged from the reactor and directed to chemical reprocessing will be the same whether the reactor is burning natural uranium or fuel with equilibrium plutonium concentration.

Equilibrium concentrations of Pu\textsuperscript{239}, Pu\textsuperscript{240}, Pu\textsuperscript{241}, Pu\textsuperscript{242}, and U\textsuperscript{236} in U\textsuperscript{238} with the U\textsuperscript{235} concentration maintained constant can be found from equations describing the variation in the number of uranium and plutonium atoms in transient reactor operation [1,2] if we assume the number of U\textsuperscript{238} and neptunium atoms constant (Q\textsubscript{NP} = const),
ignore the isotope $^{239}$U (whose lifetime is very short) and assume $\rho_{AV} = \text{const}$, but \[ \frac{d\rho_0}{dz} = 0; \frac{d\rho_1}{dz} = 0; \frac{d\rho_2}{dz} = 0; \frac{d\rho_3}{dz} = 0; \quad (1) \]

where $\rho_i$ are the concentrations of $^{235}$U, $^{241}$Pu, $^{241}$Pu, $^{239}$Pu, and $^{235}$U. The unit is $\rho_0$, the concentration of $^{235}$U in natural uranium; $z_0$ is the effective time ($dz = n \nu \sigma_0 dt$) in natural uranium operation, and is related to the time $z$ (referable to $^{235}$U operation with equilibrium concentrations of plutonium and constant $^{235}$U level) by the formula

\[ z = z_0 \frac{\sigma_0'Q_0E_0}{\sigma_0Q_0E_5' + \sigma_0'Q_0E_0 + \sigma_1'Q_1E_1}, \quad (2) \]

where $E_i$ are the energies liberated in the fission of $^{235}$U, $^{239}$Pu, and $^{241}$Pu.

The number of fission neutrons generated per single neutron absorbed in the fuel may be determined by the formula

\[ \eta = \frac{\nu_5Q_5 + \nu_9Q_9 + \nu_1Q_1}{c_5 + \sigma_0Q_0 + \sigma_5Q_5 + \sigma_9Q_9 + \sigma_1Q_1 + \sigma_2Q_2 + \sigma_6Q_6}, \quad (3) \]

where $\nu_i$ are the numbers of secondary neutrons per neutron absorbed in $^{235}$U, $^{239}$Pu, $^{241}$Pu ($\nu_5 = 2.06; \nu_9 = 2.00; \nu_1 = 2.26$)*.

The diagram shows a plot of $k/\theta = \eta \eta \rho_1 (z) \rho_0 \rho_0$ as a function of $\rho_{AV}$, the average $^{235}$U content in the fuel when a continuously reloaded heavy-water reactor is operating in steady state with an equilibrium concentration of plutonium isotopes and a constant specified average $^{235}$U content plus reactor poisons. Plots of the equilibrium concentrations $\rho_0$, $\rho_5$, and $\rho_1$ are similar. The curve of $k/\theta$ is defined for $\rho_0 = 1$: in burnup estimates, the resonance absorption in the poisons is taken into account by appropriately increasing the thermal absorption, i.e. by decreasing $\theta$.

The following constants were used in estimates of the $\rho_i$ values: $\sigma_8 = 2.75; c_8 = 0.573; \sigma_9 = 667; \sigma_{1f} = 555; \sigma_9 = 1235; \sigma_{1f} = 846; \sigma_{1f} = 389; \sigma_9 = 730; \sigma_1 = 1100; \sigma_2 = 36; \sigma_6 = 5.5; \sigma_8 = 1480; \sigma_8 = 1.85; \sigma_{1f} = 1.27; \sigma_{1f} = 0.495; \sigma_1 = 2.22; \sigma_2 = 0.045; \sigma_6 = 0.008$.

The following resonance integrals (in barns) were also used: $I_9 = 2600; I_{1f} = 1620; I_5 = 980; I_5 = 556; I_{1f} = 400; I_9 = 400$ ($E_P = 0.22$ eV). The moderating power of heavy water was assigned the value $I_9/I_5 = 5.54$.

The $^{242}$Pu and $^{236}$U content in the fuel was taken into account partially in the calculations, since the low $^{235}$U and $^{241}$Pu content in the fuel mean that equilibrium concentrations of $^{242}$Pu and $^{236}$U can be achieved in no sooner than 40 years of continuous reactor operation (without counting decay time and fuel reprocessing time). $^{241}$Am and $^{242}$Pu and all their isotopes of higher mass number were left out of account, since it is not clear at present just how $^{241}$Am is to be separated from the fuel in reprocessing.

The curves on the diagram were plotted for a gas-cooled heavy-water reactor burning natural uranium in fuel elements 10.5 cm in diameter, with uranium cross section filling factor $\varepsilon = 0.25$, resonance-escape probability $\varphi = 0.9$ in $^{235}$U, fast multiplication factor $\mu = 1.027$. These figures are not optimized from the standpoint of reactor operation on the proposed fuel cycle. In this case, it appears feasible to increase fast multiplication by raising the uranium content in the channels and by decreasing the specific power.

Neutron absorption and fission in moderation ($\mu \times \lambda$) are taken into account with the aid of formulas derived in [2]: resonance absorption in $^{235}$U is handled by the formula $\varphi_0 = 1/e^{\varphi_0}$, in poisons by using the factor $\varphi_0$.

The relative consumption of natural or depleted uranium employed to raise the $^{235}$U concentration in the recycled fuel, over a reactor run $x$, can be found from the condition

*Values of $\nu_5$, $\nu_9$, and $\nu_1$ correspond to neutron temperature $425^{\circ}$K (Byulleten' informats. tsentr po yadernym dannym. No. 1. Moscow, Atom press, 1964, p. 285).