Theory of Neutron Chain Reactors." Univ. of Chicago Press, 1958.

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Multiregion Fuel Elements

The performance of metallic uranium fuels is limited by the operating temperature at which dimensional instability becomes excessive. Although uranium oxide fuels may be operated at much higher temperatures, they have the disadvantages of low uranium density and low thermal conductivity. A multiregion fuel element can be designed to take advantage of the high density and high thermal conductivity of the metal as well as of the high operating temperature of the oxide. This is accomplished by using a rod-shaped element having an outer annulus of metal and a central core of oxide. The discussion is limited to metallic uranium and uranium oxide; however, the same principles apply to various uranium alloys and refractory uranium compounds. For instance the core may be a high molybdenum alloy, or uranium carbide, instead of uranium oxide; the annulus may be an alloy of uranium instead of unalloyed uranium.

Preliminary analytical evaluation of the relative merits of the following four types of fuel elements have been made: (1) solid metallic uranium rods, (2) solid uranium oxide rods, (3) cored metallic uranium rods, (4) multiregion metallic uranium–uranium oxide rods. The evaluation concluded that the multiregion fuel element provides improved performance over other elements by permitting 20% increase in power generation, or 25% increase in burnup, or 1% increase in reactivity.

Although the potential increase in permissible burnup would make multiregion elements desirable for enriched, compact reactors, it is expected that their greatest utility would be found in natural or low enrichment fuel power reactors, where the increased reactivity is of prime importance.

The method used in the evaluation and the results obtained are discussed below.

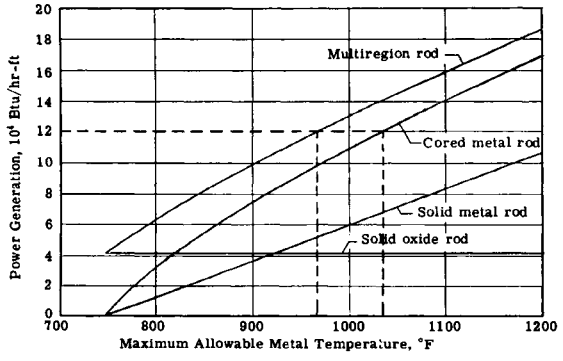

FIG. 1. Fuel element power generation. Assumptions: Equal uranium content in all elements. Equal maximum metal temperature in all elements. Equal maximum oxide temperature in all elements (4000°F). Equal surface temperature in all elements (750°F).

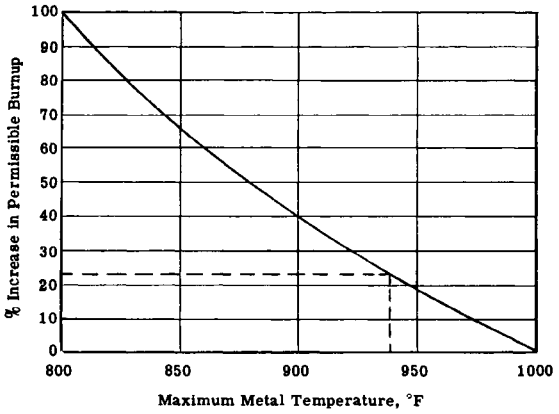

FIG. 2. Burnup as a function of temperature. Assumptions: 15% swelling at 0.5 atom % burnup. ANL swelling data are applicable. Relative burnup at 1000°F is 100%.

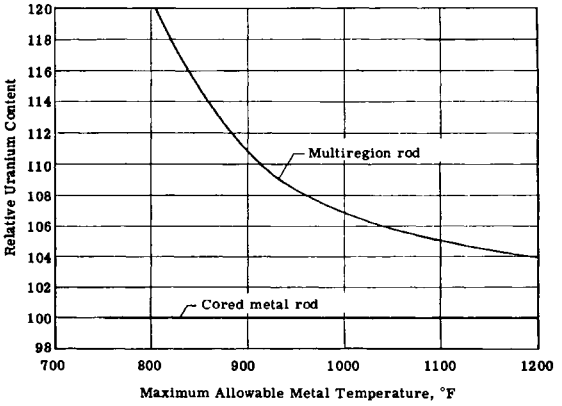

FIG. 3. Fuel element uranium content. Assumptions: Equal power generation in all elements. Equal maximum metal temperature in all elements. Equal surface temperature in all elements (750°F). Maximum oxide temperature of 4000°F.

The results of the permissible power generation evaluation are shown graphically in Fig. 1 along with the assumptions used. The data show that the multiregion element provides greater power generation per unit length than any of the other elements. It represents an improvement of nearly 20% over its nearest competitor, the cored metallic element.

The maximum permissible burnup of a fuel element is limited either by the radiation-induced swelling or by the available excess reactivity; in general, metallic elements are governed by the former while oxide elements are governed by the latter. A comparison was made between the cored metallic element and the multiregion element to determine the relative advantage of the multiregion element with respect to swelling limited burnup. To make quantitative estimates it was assumed that swelling limits the burnup, and that the amount of swelling is a function of maximum metal temperature. Using the Argonne National Laboratory experimental data (*1*) and assuming a permissible swelling of 15% volume increase at 0.5 atom per cent burnup, the allowable relative burnup as a function of temperature was determined. This is shown in Fig.

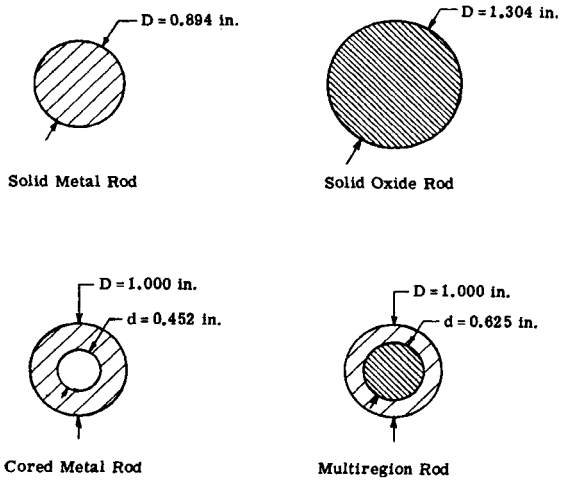

FIG. 4. Fuel element configurations for equal uranium content. Assumptions: Maximum allowable metal temperature = 1100°F. Maximum allowable oxide temperature = 4000°F. Surface temperature = 750°F.

2. From Fig. 1 it is seen that the maximum metal temperature at any given power generation rate is about 65°F lower for the multiregion element than for the cored metal element. Using the data in Fig. 2, this indicates an increase in the maximum permissible burnup of about 25%. Thus a considerable improvement is obtained in the swelling limited burnup.

Calculations have been made to determine the relative burnup of the four elements under consideration when limited by available excess reactivity. With a given total uranium content and a given enrichment—natural or 2% enriched—the burnup limits are the same for all elements except the solid oxide rod. The latter permits 5% to 25% less burnup under these conditions. Thus the multiregion element has no disadvantage with respect to reactivity-limited burnup.

The reactivity of a fuel element in a given lattice for any given enrichment will depend upon the total uranium content of the element. To determine the results of this effect on the multiregion fuel element, calculations were performed to determine the maximum permissible uranium content at a given maximum metal temperature and power generation rate. The results are illustrated in Fig. 3 in comparison with the cored metallic rod. It can be seen that the multiregion element permits from 4% to 20% greater uranium content under otherwise identical operating conditions.

In the course of previous work at United Nuclear, estimates of the effect of uranium content on reactivity were made for similar fuel elements. Then calculations were made specifically for the Sodium-Deuterium Reactor (SDR). Based on these data it is expected that a multiregion element will have between 0.5% and 2.0% greater reactivity than a cored metal element of the same outside diameter, operating at the same power generation rates and maximum metal temperatures. One could take advantage of this gain in three different ways: lower enrichment, smaller core size, or lower refueling frequency if the burnup is reactivity-limited.

It is also necessary to evaluate the effect of element geometry on reactivity to determine whether the above-cited gains are actually available. For this purpose the four fuel elements shown in Fig. 4 have been used. They have equal uranium content. Calculation of the reactivity for an infinite lattice (K_∞) of such elements was made. The multiregion element does not differ appreciably from either the cored or solid metal element and is somewhat superior to the solid oxide element. The conclusions reached above regarding the greater reactivity of the multiregion element because of its greater uranium content are therefore sustained.

REFERENCES

1. H. KITTEL AND S. PAINE, Effects of irradiation on powder compacts of uranium and some uranium alloys. ANL-5664 (June 1957).

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Thermal Flux Disadvantage Factors for Slab Geometry

It is the purpose of this note to point out that thermal flux disadvantage factors for slabs can be calculated accurately by an extremely simple method. This method is based partly on blackness theory.

In 1959, Maynard (*1*) proposed a method for the theoretical calculation of thermal disadvantage factors. This method was based *only* on blackness theory. The DB_1 approximation was reported to give the best results. However, this procedure is not simple for calculational purposes with the presently available tabulated functions.

Recently, Theys (*2*) has given a simple expression for thermal flux disadvantage factors. Theys' treatment is based on the Integral Transport Theory, and on the arguments put forward by Amouyal and Benoist. In Theys' notation, the flux disadvantage factor is given by Eq. (1).

$$\bar{\phi}_1/\bar{\phi}_0 = G + a\Sigma_0^c[\Sigma_1^{tr}(b - a) + 0.13] \qquad (1)$$

where

$$G = \phi_a/\bar{\phi}_0 .$$

Here, $\bar{\phi}_1$ and $\bar{\phi}_0$ are the average fluxes in the moderator region, and the fuel element respectively. ϕ_a is the neutron flux at the surface of the fuel slab. The subscripts 0 and 1 refer to the fuel element and the moderator, respectively. The half thickness of the fuel (absorbing) slab and of the moderator region are denoted by a and $(b - a)$. The fuel disadvantage factor is given by Theys as:

$$G = 1 + \frac{\Sigma_0^c}{\Sigma_0^T}\left[\frac{P_c}{1 - P_c} - 2a\Sigma_0^T\right] \cdot \left[1 + \alpha^*\left(\frac{\Sigma_0^S}{\Sigma_0^T}\right) + \beta^*\left(\frac{\Sigma_0^S}{\Sigma_0^T}\right)^2\right]. \qquad (2)$$