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FISSION PRODUCT DISTRIBUTION IN ORR FUEL ELEMENTS

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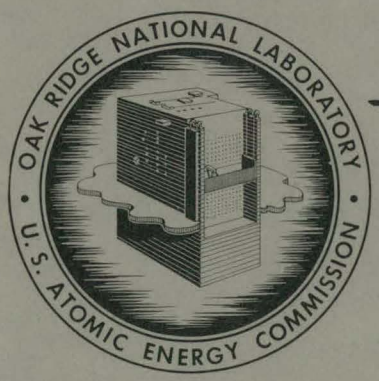

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ABSTRACT

The gamma rays emitted by ORR fuel elements and by the fuel section of shim rods are measured as a function of position along the elements with a very small graphite ionization chamber.

Comparison of the fuel burnup calculated from the gamma measurements and by the flux-time method shows good agreement. This means that the gamma-ray distribution measurements could be a good method of determining the U^{235} consumption in fuel elements.

Distributions of the macroscopic absorption cross section and the infinite multiplication factor along fuel elements are computed from the gamma dose rate distribution.

The limited usefulness of the shim rod fuel section is discussed in the last section.

EXPERIMENTAL SETUP

To measure the fission product activity along a fuel element without distortion and with a good resolution, two small graphite ionization chambers were built. These chambers were calibrated against a Victoreen model 131 integrating ionization chamber. Their geometrical characteristics and sensitivities are given in Table 1.

Table 1. Size and Sensitivity of the Graphite Ionization Chambers

Chamber Designation	Size of Sensitive Volume		Sensitivity* ($r \cdot amp^{-1} \cdot hr^{-1}$)
	Diameter	Length	
			$\times 10^{13}$
GICS	0.2 in.	0.2 in.	8.26 ± 0.4
GICL	0.2 in.	0.4 in.	4.49 ± 0.2

*Corrected to 22°C and 760 mm Hg.

Figure 1 shows that, with these small chambers, distortion of the dose rate distribution is negligible. Curve 1 was measured with an ionization chamber 1 in. in diameter and 2.5 in. long. Curves 2 and 3 were measured with GICL and GICS respectively. The fuel element used was OR-121, whose irradiation history is given in Table 2. These curves show that, in order to measure the fission product distribution in a fuel element, it is absolutely necessary to work with a chamber as small as possible and to keep the chamber in contact with the fuel element.

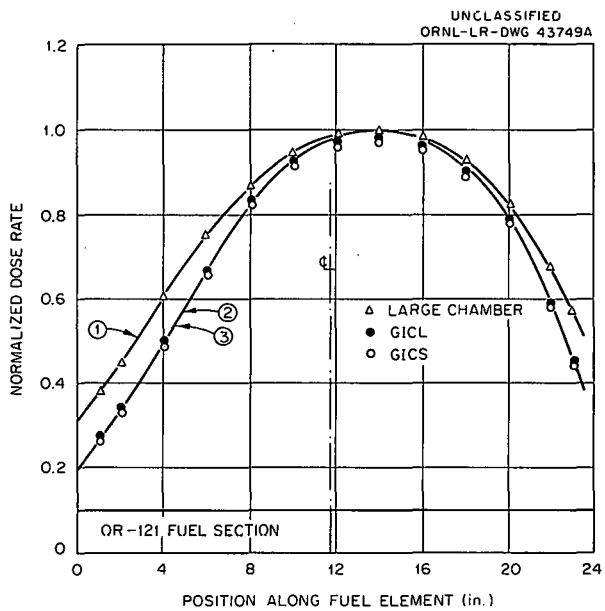

Fig. 1. Distortion in the Dose Rate Distribution Due to Chamber Size.

MEASUREMENTS OF THE FISSION PRODUCT DISTRIBUTION ALONG FUEL ELEMENTS

The fission product activity was measured for the following elements: OR-158, OR-159, OR-150, and OR-121. Table 2 summarizes the irradiation history of these elements. The cooling time given

Table 2. Irradiation History of the Measured Fuel Elements

Element No.	Irradiation Time (days)	Power (Mw)	Position in Reactor	Cooling Time (days)	U^{235} Consumption (g)
OR-158	20	16	C-7	23	16
OR-159	20	16	E-7	23	18
OR-150	10	16	E-7	45	9
	20	16	D-7	23	18
OR-121	14	20	D-1	137	4
	21	20	D-3	113	25
	11	20	B-5	84	10
	14	16	C-6	63	11
	4	16	D-2	53	2
	10	16	D-5	25	7

in Table 2 is the elapsed time between the end of a given irradiation and the measurement. The U^{235} consumption is obtained from the reactor power, the irradiation time, and the reactor position factor. The normalized fission product distributions for these four elements are shown in Figs. 2–5. These distribution curves are normalized by dividing all the measured values along the curve by the value of the maximum. This operation facilitates comparisons of curves. It is possible to see from these figures that the fission product distribution is not symmetrical about the horizontal center plane of the element. This is not surprising, because the reactor flux is not symmetrical about its central plane and some of the elements that have received more than one irradiation were not

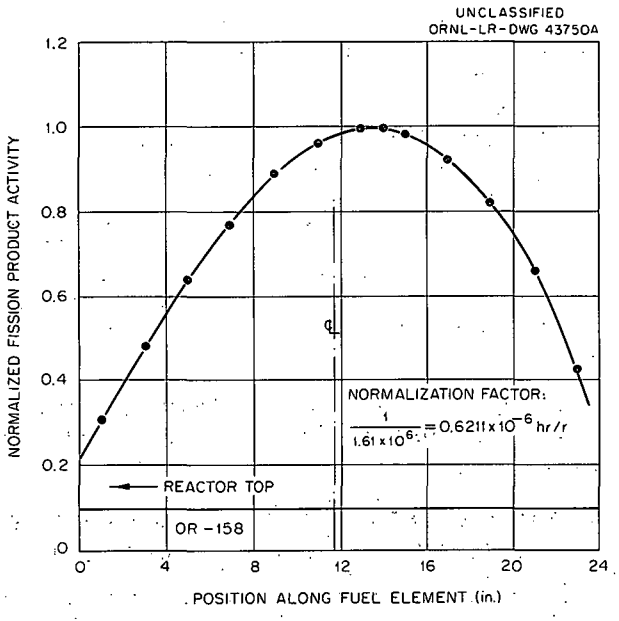

Fig. 2. Fission Product Distribution in Fuel Element OR-158.

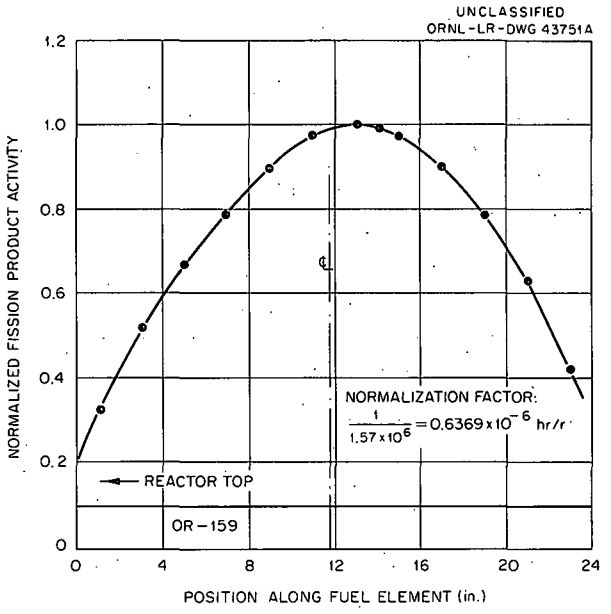

Fig. 3. Fission Product Distribution in Fuel Element OR-159.

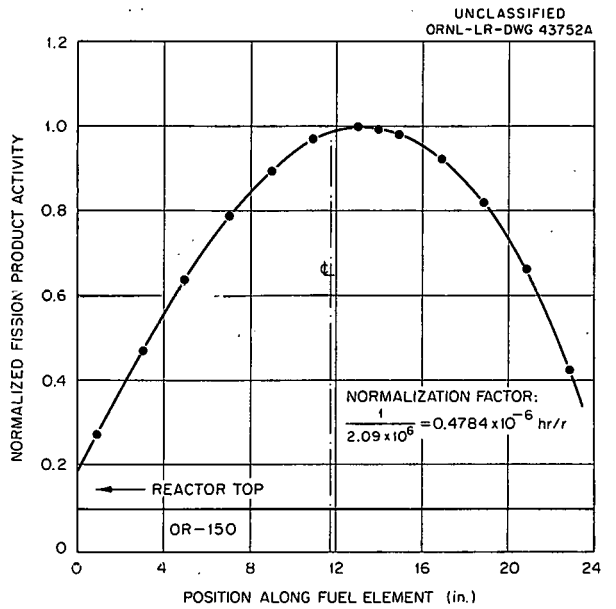

Fig. 4. Fission Product Distribution in Fuel Element OR-150.

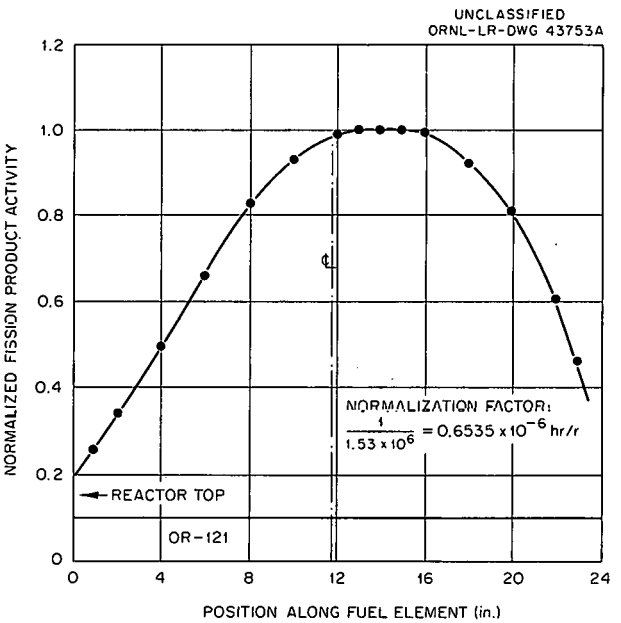

Fig. 5. Fission Product Distribution in Fuel Element OR-121.

inverted. ORR fuel elements are built with identical end boxes at the top and bottom. Thus, it is possible to insert them in the core with either end as the top. Fuel elements should, if possible, be inverted from one irradiation to the other. This not only prolongs their life (this effect is not very important, however, being smaller than 5%) but also helps keep the maximum flux as close as possible to the reactor horizontal center plane (which is important in getting as great a flux as possible in the experimental facilities).

Fission product distribution measurements can be used to check the correctness of the calculated values for U^{235} consumption. The value of the integral over the distribution curve is proportional to the total amount of fission products accumulated in the fuel element. This is, in turn, proportional to the total amount of U^{235} consumed.

Table 3 shows the value of these integrals over the normalized distributions; the third column contains the value in the second column multiplied by the inverse normalization factor.

Table 3. Integrals over Fission Product Distributions

Element No.	Integral of Normalized Distribution	Integral of Distribution (in.·r·hr^{-1})
		$\times 10^7$
OR-158	17.6	2.84
OR-159	17.7	2.78
OR-150	17.6	3.69
OR-121	17.6	2.69

It is interesting to notice that the values of the integrals over the normalized distributions are, for the elements measured up to now and within the experimental errors, independent of the number of irradiations given to the element and of the core location where these irradiations took place. This means that the shape of the U^{235} consumption distribution along a given element is independent of the initial U^{235} concentration or, in other words, that the thermal flux distribution is always inversely proportional to the U^{235} concentration. This interesting effect should be examined with care, because it is suspected that it will not hold if the irradiation is made in the vicinity of a moving absorber, that is, a control rod. Further investigations should clear this point.

It is now possible to check whether the U^{235} consumption calculated in the conventional manner

is right. A relation between the amount of U^{235} burned and the value of the integral over the fission product activity distribution can be obtained from the fuel elements (OR-158 and OR-159) that had had only one irradiation. For 23 days of cooling time between the end of the irradiation and the time of the activity measurement, 1 g of U^{235} consumption corresponds to 1.65×10^6 in.·r·hr^{-1}. Table 4 represents the contribution of each irradiation to the total integral. Cooling times were corrected to 23 days in order to be able to use the relation gained from elements OR-158 and OR-159. Fission product decay data were taken from Perkins and King.[1]

These results show that the fission product activity measurement represents a good method to determine experimentally the U^{235} consumption in fuel elements. This method gives results that correspond within 12% with those in Table 2, and could certainly be made quite accurate. Calibration could be done by measuring the neutron flux in a given new element at the beginning and at the end of an irradiation in order to compute accurately the amount of U^{235} burned. Then gamma activity measurements at different times after the end of irradiation will give the relation between grams of U^{235} burned and gamma activity as a function of cooling time.

The main advantage of this method is that, knowing the distribution of U^{235} and fission products in the fuel element, it is possible to compute the variation of the infinite multiplication factor and the macroscopic absorption cross section along the element. The knowledge of these quantities can be very helpful if some flux deformation (flattening or pushing the flux toward an experiment) is to be done. Figure 6 shows the variation of the infinite multiplication factor and the macroscopic absorption cross section along fuel element OR-121.

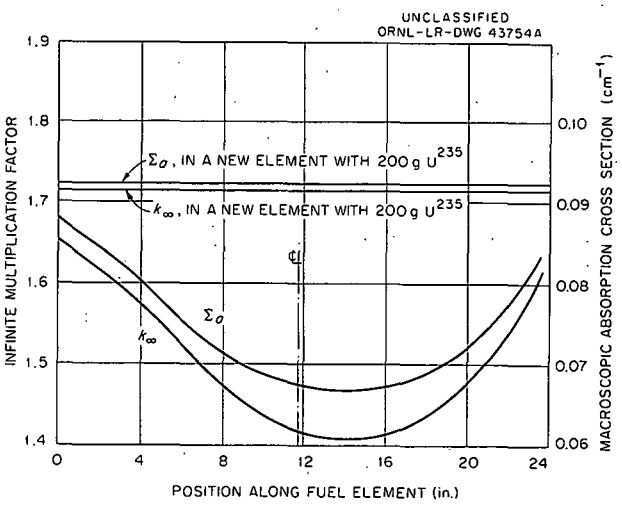

Fig. 6. Infinite Multiplication Factor and Macroscopic Absorption Cross-Section Variation Along Fuel Element OR-121.

[1] J. F. Perkins and R. W. King, *Nuclear Sci. and Eng.* 3, 726–46 (1958).

Table 4. Contribution from Each Irradiation to the Total Amount of Fission Products

Element No.	U^{235} Consumption (g)	Activity at Total Cooling Time / Activity at 23 Days	Contribution to Integral (in.·r·hr^{-1})
OR-150	9	0.50	7.45×10^6
	18	1	2.97×10^7
		Total	3.72×10^7
OR-121	4	0.134	0.88×10^6
	25	0.192	7.92×10^6
	10	0.225	3.72×10^6
	11	0.337	6.13×10^6
	2	0.340	1.12×10^7
	7	0.912	1.05×10^7
		Total	3.03×10^7

These curves were obtained by the following calculation. The key for the symbols used here is given in the "Nomenclature."

The fission product distribution is described by the following equation:[2]

$$N_{FP}(z) = \frac{2N_{25}^0}{1+\alpha_{25}} \left[1 - e^{-\sigma_{25}\phi(z)t} \right] . \quad (1)$$

By using Eq. (1) and the measured fission product activity, and taking advantage of the fact that the integral of the normalized fission product activity distribution is a constant regardless of initial and irradiation conditions, the value of $\phi(z)t$ can be calculated as a function of the position along a fuel element. Knowing $\phi(z)t$ it is possible to compute $\Sigma_a(z)$ and $k_\infty(z)$ from the following relations:

$$\Sigma_a(z) = \Sigma_a^{H_2O+Al} + N_{25}^0 \sigma_{25} e^{-\sigma_{25}\phi(z)t} + \\ + N_{FP}\sigma_{FP} + N_{Sm}\sigma_{Sm} , \quad (2)$$

$$k_\infty(z) = \frac{\nu N_{25}^0 \sigma_f e^{-\sigma_{25}\phi(z)t}}{\Sigma_a(z)} , \quad (3)$$

with N_{Sm}, the atomic density of samarium at saturation, given by[3]

$$N_{Sm} = 3.21 \times 10^{-4} N_{25}^0 (1 - e^{-\sigma_{25}\phi(z)t}) . \quad (4)$$

It should be noted that the values obtained for $\Sigma_a(z)$ and $k_\infty(z)$ are right only if xenon is allowed to decay. They do not represent the condition of a core during operation but can be used when loading the reactor with elements that have cooled at least three days.

The following cross-section values (Maxwell-Boltzmann distribution at 20°C) were used:

$\sigma_{25} = 593$ barns	$\sigma_{H_2O} = 0.586$ barn
$\sigma_f = 502$ barns	$\sigma_{FP} = 38$ barns
$\sigma_{Al} = 0.204$ barn	$\sigma_{Sm} = 5.3 \times 10^4$ barns

[2]M. Benedict and T. H. Pigford, *Nuclear Chemical Engineering*, p 94, McGraw-Hill, New York, 1957.

[3]S. Glasstone and M. C. Edlund, *The Elements of Nuclear Reactor Theory*, p 338–39, Van Nostrand, Princeton, N. J., 1952.

MEASUREMENT OF THE FISSION PRODUCT DISTRIBUTION ALONG SHIM RODS

The fission product activity was measured along two shim rods, OR-2-S and OR-6-S, by the same method that was described in the preceding section. The results were found to be almost the same for the two shim rods, so only rod OR-2-S will be discussed here.

Shim rod OR-2-S was in the reactor in position D-6 from November 4, 1958, to February 23, 1959, and in B-6 from February 28, 1959, to June 16, 1959. The total amount of U^{235} burned, as computed in the conventional way, was 85 g. The fission product distribution measured is plotted in Fig. 7. Most of the fission products are accumulated in the upper half of the rod. This is due to the fact that this section is always in the reactor, whereas the lower section is out of the reactor at the beginning of an irradiation cycle and slowly moves inward during the cycle.

Using the same methods as described earlier (see preceding section), it is now possible to obtain the variation of the macroscopic absorption and the infinite multiplication along the shim rod.

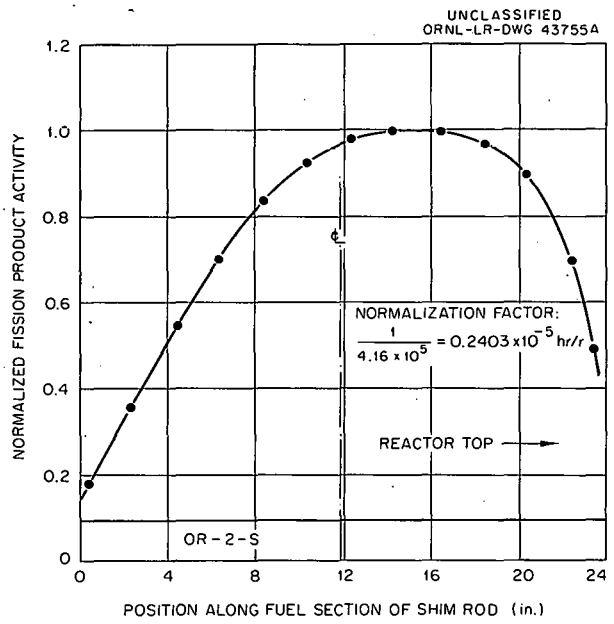

Fig. 7. Fission Product Distribution Along the Fuel Section of Shim Rod OR-2-S.

These results are shown in Fig. 8. It is striking to see that, at the minimum, k_∞ is only equal to 0.4; and that even the average of k_∞ over the length of the element is smaller than 1. The consequence of this is that the fuel section of this shim rod does not add appreciably to its control capacity.

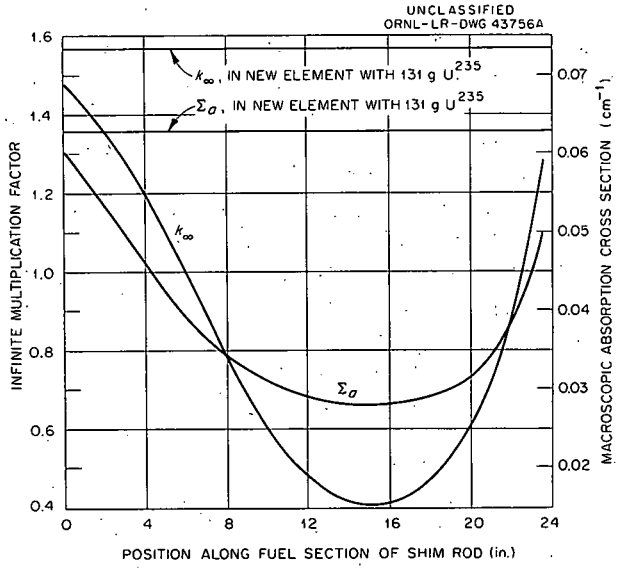

Fig. 8. Variation of Infinite Multiplication Factor and Macroscopic Absorption Cross Section Along the Fuel Section of Shim Rod OR-2-S.

In order to obtain an idea of what maximum total burnup should be allowed in the fuel section of ORR shim rods, the U^{235} concentration, the macroscopic absorption cross section, and the infinite multiplication were calculated as functions of total burnup. These quantities are plotted against position along the fuel section in Figs. 9–11. It can be seen from these results that the shim rods are kept in operation beyond such a high burnup that their fuel sections are almost completely depleted, thus adding nothing to the control capacity. At the same time the absorption cross section will decrease, as can be seen in Fig. 10, thus allowing the thermal flux to increase in the fuel section of the shim rod. This may

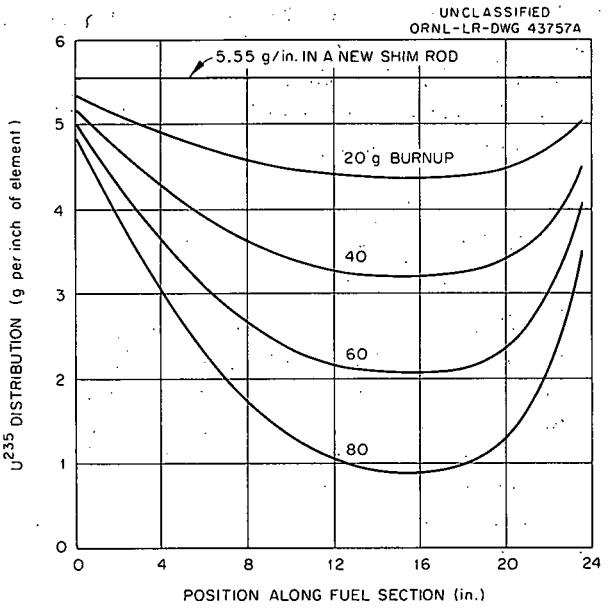

Fig. 9. Concentration of U^{235} Along the Fuel Section of an ORR Shim Rod as a Function of Total Burnup.

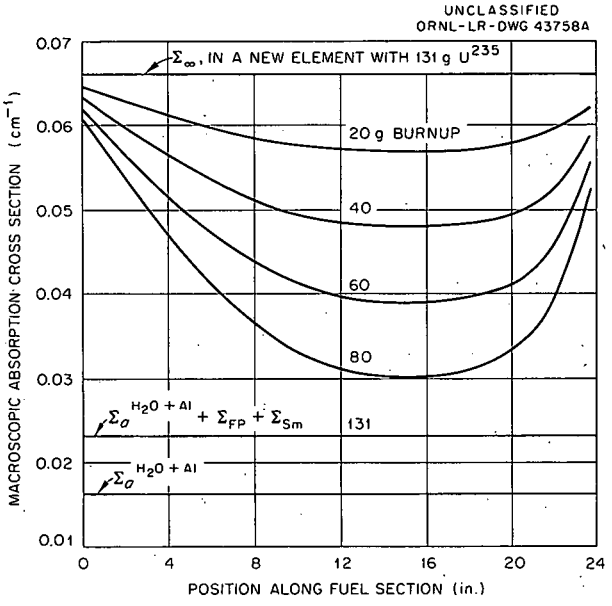

Fig. 10. Macroscopic Absorption Cross Section Along the Fuel Section of an ORR Shim Rod as a Function of Total Burnup.

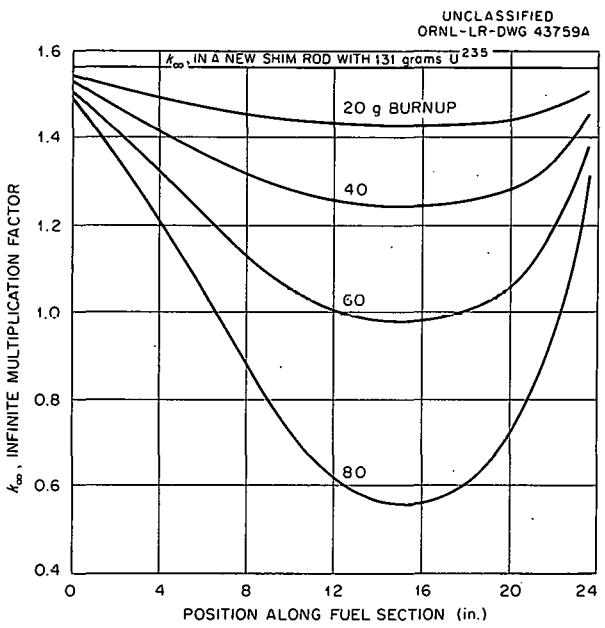

Fig. 11. Infinite Multiplication Factor Along the Fuel Section of an ORR Shim Rod as a Function of Total Burnup.

produce hot spots in the adjacent elements. Operating experience shows that the reactor is still behaving well under these conditions. This means that if the shim rods are not to be changed every two or three cycles they could stay in the reactor for a very long period of time. The fuel section of the control rods could be used as a long-time irradiation facility to study the behavior of different kinds of fuel.

The most convenient solution consists, certainly, in replacing the fuel section with an aluminum section. Aluminum, having a lower absorption cross section and a lower slowing-down power than water, will add to the control capacity and at the same time help to reduce flux peaking. The other advantage of the aluminum follower is that the control worth of the shim rods will stay constant.

NOMENCLATURE

k_∞	Infinite multiplication factor
s	Ionization chamber sensitivity
t	Irradiation time
z	Position along a fuel element
N	Atomic density
N^0	Atomic density at the beginning of an irradiation
α	Capture-to-fission ratio
ν	Average number of neutrons emitted by one fission
σ	Microscopic absorption cross section
σ_f	Microscopic fission cross section
Σ_a	Macroscopic absorption cross section
ϕ	Thermal neutron flux

The subscripts are used in accordance with ref 2.

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