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FISSION PRODUCT POISONING DATA

N. J. Pattenden

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N. J. Pattenden

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Oak Ridge, Tennessee
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FISSION PRODUCT POISONING DATA

N. J. Pattenden*

ABSTRACT

Recent measurements of the neutron cross sections and yields of fission products are summarized and compared with previous data. The effects of resonance and thermal absorption are compared. Proposals are made for further work to enable the evaluation of fission product poisoning of reactors to be more complete.

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*On temporary assignment from A.E.R.E., Harwell, England

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List of Symbols not Defined in the Text

ϕ - neutron flux.

E - neutron energy.

N - number of nuclei.

σ - neutron absorption cross section.

σ_F - neutron fission cross section.

y - fission product yield.

λ { - decay constant.
- ratio of the slowing down flux per log e energy interval to the integrated Maxwellian flux.

t - time.

k - Boltzmann's constant.

T - temperature.

1. INTRODUCTION

To estimate the poisoning due to fission products in a reactor as a function of neutron flux and integrated irradiation, in general one needs to know the half-lives, yields and cross sections of all the fission product nuclides. The reactor neutron energy distribution extends from fission energies down to thermal energies (about 10^6 to 10^{-2} eV), hence, in principle the cross sections are required over this range. If the yields vary with neutron energy (and there is evidence that those in the valley of the mass-yield curve do), then it might also be necessary to measure them as a function of energy. This represents a formidable quantity of nuclear data. Fortunately, the practical requirements are less rigorous, and reasonable criteria of poisoning significance may be set to limit the experimental work necessary, which depend on the properties of the fission products themselves and on the particular reactor specification.

2. POISONING DEFINITION

We define the poisoning as the rate of absorption of neutrons, both thermal and eipthermal, by a fission product nuclide, per atom of fissile nuclide initially present. For simplification, the latter is assumed to remain constant during operation.

The rate of absorption by nuclide i = $\phi N_i \sigma_i$.

If we ignore formation by neutron capture (second order production), and assume that the mass chain goes straight to nuclide i without loss through neutron capture, then the rate of growth of i per fissile atom is

$$\frac{dN_i}{dt} = y_i \phi \sigma_F - N_i(\lambda_i + \phi \sigma_i).$$

Saturation is reached when $y_i \phi \sigma_F = N_i(\lambda_i + \phi \sigma_i)$,

$$\text{i.e.,} \qquad N_{is} = \frac{y_i \phi \sigma_F}{\lambda_i + \phi \sigma_i}.$$

In general $$N_i = \left(1 - e^{-(\lambda_i + \phi \sigma_i)t}\right) \frac{y_i \phi \sigma_F}{\lambda_i + \phi \sigma_i}.$$

Hence, poisoning $= \dfrac{y_i \phi^2 \sigma_i \sigma_F}{\lambda_i + \phi \sigma_i} \left(1 - e^{-(\lambda_i + \phi \sigma_i)t}\right)$.

At saturation, poisoning $= \dfrac{y_i \phi^2 \sigma_i \sigma_F}{\lambda_i + \phi \sigma_i}$.

For a stable poison or one with $\phi \sigma_i \gg \lambda_i$,

$$\text{poisoning} \doteq y_i \phi \sigma_F \left(1 - e^{-\phi \sigma_i t}\right).$$

At saturation, poisoning $= y_i(\phi \sigma_F)$.

If $\phi \sigma_i t$ is small, poisoning $\doteq y_i \sigma_i \phi t(\phi \sigma_F)$.

Thus the poisoning of a saturated fission product is proportional to its yield, but the poisoning of a fission product far from saturation is proportional to the product of its yield and cross section and irradiation. For example, Table I shows the poisoning of some fission products for a U^{235}-fuelled reactor at a flux of 10^{14} neutrons/cm^2/sec.

3. HIGH AND LOW CROSS SECTION FISSION PRODUCTS

The above discussion shows that because of the wide range of cross sections involved, different fission products may have very different poisoning

effects on present day reactors. Of the known thermal cross sections, only six (Xe^{135}, Sm^{149}, Sm^{151}, Eu^{155}, Cd^{113}, and Gd^{157}) are above about 10,000 barns, and the rest are below about 500 barns. Hence, we may usefully divide fission products into high and low cross section groups. For small irradiations the first group only need be considered, but for large irradiations (greater than about 10^{20} neutrons/cm^2), their poisoning saturates at a value of about 0.078 ($\phi\sigma_F$). In the case of the second group, their poisoning is proportional to irradiation until very large irradiations are reached. At 10^{21} neutrons/cm^2 the known cross sections of the second group have a poisoning effect of about 0.04 ($\phi\sigma_F$), increasing nearly linearly with irradiation.

4. EFFECTIVE CROSS SECTIONS IN REACTOR SPECTRA

Thermal reactors have a neutron flux distribution which is approximately represented by

$$\phi(E) = M(E) + \lambda F(E),$$

where M(E) represents the normalized Maxwellian flux distribution, usually written

$$M(E)dE = E/(kT)^2 \exp(-E/kT)dE.$$

F(E) represents the slowing down flux, proportional to E^{-1} per unit energy interval at energy E. λ is the ratio of the slowing down flux per log e energy interval to the integrated Maxwellian flux.

To determine an effective absorption cross section in such a spectrum, it is necessary to know details of the variation of cross section with energy in the Maxwellian region, but not in the slowing down region, where an integral cross section is sufficient. The latter may be obtained from known resonance

parameters or from a resonance integral experiment. The effective cross section may then be derived for any thermal reactor knowing the temperature and the hardness of spectrum (λ). (Fission products are assumed to be at infinite dilution in the reactor.)

Thermal reactors are now being designed to operate in excess of 1000°K, and no doubt this temperature will be increased in the future. Hence, in our data requirements, we should consider the Maxwellian region to extend to about 1 eV.

The notation of Westcott[1] at Chalk River, and Campbell and Freemantle[2] at Harwell, is convenient for this purpose. A nuclide is considered to have an effective cross section in a thermal reactor spectrum, defined by $\sigma_{eff} = \sigma_{2200}\,(g + rs)$, where σ_{2200} is the absorption cross section at 2200 m/sec, and g and s characterize the departure of the cross section from a 1/v dependence in the Maxwellian and slowing down regions respectively. Thus the thermal cross section is

$$\sigma_M = g\,\sigma_{2200} = \frac{\int_0^\infty M(E)\,\sigma(E)dE}{\int_0^\infty M(E)\left(\frac{0.0253}{E}\right)^{\frac{1}{2}} dE} .$$

The epi-thermal cross section is

$$\sigma_{epi} = s\,\sigma_{2200} = \frac{\sigma_{r.i.}}{\int_0^\infty M(E)\left(\frac{0.0253}{E}\right)^{\frac{1}{2}} dE}$$

and $\sigma_{r.i.}$ is the resonance integral in excess of a 1/v integral,

$$\sigma_{r.i.} = \int_{\mu kT}^{\infty} \left[\sigma(E) - \sigma_M \left(\frac{0.0253}{E} \right)^{\frac{1}{2}} \right] \frac{dE}{E} .$$

Note that $\int_0^{\infty} M(E) \left(\frac{0.0253}{E} \right)^{\frac{1}{2}} dE = \left(\frac{0.0253\ \pi}{4\ kT} \right)^{\frac{1}{2}}$.

r depends on the ratio of the slowing down to the thermal flux, defined as

$$r = \frac{\lambda}{1 + 4\lambda/(\mu\pi)^{\frac{1}{2}}}$$

where the slowing down flux is assumed to have a low energy limit of μkT. Westcott takes μ to be 5, and Campbell and Freemantle use an arbitrary smooth function equivalent to a μ of 2.81.

For a cross section with a 1/v dependence g = 1 and s = 0, otherwise g and s are functions of neutron temperature T. s is effectively independent of μ unless there happens to be a resonance in the cross section in the neighborhood of μkT (unfortunately this is often the case with high cross section nuclides).

Some points which may limit the value of the convention are (i) the uncertainty in μ for a particular reactor, (ii) the well-thermalized assumption may not apply, (iii) the r value changes through the fuel and from fuel to moderator, (iv) some recent work by Poole at Harwell indicates that a Maxwellian distribution cannot accurately be applied to a reactor thermal spectrum. Consequently, the convention will not be so accurate as an integration for a particular spectrum and cross section.

The value of r is close to λ when λ is small, and is not likely to exceed 0.1 in most practical cases. Table II shows some new (i.e., 1958 Geneva Con-

ference or after) cross section data on fission product nuclides, presented where possible in the Westcott convention for a given temperature and r value. The task of compilation is made more difficult because cross sections are reported in many different forms, and sometimes details of the neutron energies or temperatures to which they refer are not stated. The σ_{2200} column gives absorption cross sections at 2200 m/sec, measured with a spectrometer at this velocity. The σ_{2200} g column gives thermal integral cross sections and also spectrometer measurements where g has been determined from a cross section curve. The $\sigma_{eff} = \sigma_{2200}$ (g + rs) gives reactor spectrum integral cross sections and also spectrometer measurements where g and s have been determined from a cross section curve. The references are to 1958 Geneva Conference papers.

The integral measurements reported by Bidinosti et al (Paper 201) on Cs^{133} and Sm^{149} were made on naturally occurring and fission product material; the observed cross sections from both sources were the same within experimental error. Hence, it seems we may eliminate the possibility of long-lived isomers affecting these cross sections.

Bidinosti et al also report a comparison of relative yield measurements from high and low flux irradiations, to obtain an indication of a high cross section in an unstable member of a mass chain. Mass chains from 88-96 and 133-152 were covered, except for masses 138, 139 and 141. An unexplained effect was observed at mass 91, and cross sections of a few hundred barns were indicated for Ce^{141} and Pr^{143}.

5. EFFECT OF RESONANCE ABSORPTION

The absorption due to resonances in the fission product cross sections has an important effect on reactor poisoning. If an appreciable part of a resonance wing extends into the thermal region (the resonance may be at positive or negative energy) the cross section will depart from a 1/v dependence, and the effective cross section will change with neutron temperature. In calculating this effect, if very high temperatures are involved, it may be necessary to use a cross section curve with the resonance Doppler broadened for the appropriate neutron temperature, rather than the true resonance shape. This effect increases the calculated effective cross section of Sm^{149} by up to about 2% for temperatures in the 500 to 750°K region.

The poisoning due to resonances in the slowing down region depends on the hardness of the spectrum. In the region above 10^2 eV, reasonable values of average total fission product cross sections may be calculated using statistical distributions of nuclear properties. This has been performed by Businaro et al[3], Greebler et al[4] and Gordeev and Pupko (Paper 2223). For each nuclide the average cross section was calculated over a range of energies knowing its radiation width, strength function ($\overline{\Gamma}_n^{\,o}/D$) for $\ell = 0$ and $\ell = 1$ interactions, mean level spacing (which must be adjusted for even-even and magic nulcides), and neutron width distribution, which was taken to be of Porter-Thomas form. Gordeev and Pupko estimate the total known fission product resonance integral per fission in the range 10^2 to 10^6 eV to be 23 barns, excluding the high thermal cross section group and products with half-lives of less than 100 days. It is probable that these estimates could now be improved because of the increased weight of

data on the $\ell = 1$ strength functions, and in particular on the strength function peak in the mass 90 region.

In the region below 10^2 eV, individual levels are important and statistical theories cannot be applied. Gordeev and Pupko have summed the resonance integrals estimated from individual parameters for all known resonances in fission products in the range 0.625 to 10^2 eV, and obtained a total resonance integral in excess of a 1/v dependence of 144 barns per fission, excluding the high cross section group, products with half-lives of less than 100 days, and I^{129} and Pm^{147}, which were not known to them. They plot a curve (shown in Fig. 1) of the total fission product cross section between 0.5 and 10^2 eV, which they derive by obtaining the accumulated resonance integrals from 10^2 eV downwards and differentiating this with respect to log E. In the lower energy region, they assume the total fission product cross section has a 1/v dependence, and that its value at 2200 m/sec is 50 barns per fission.

The most important known resonance integrals of the low (thermal) cross section fission products are listed in Table III. The discrepancies between this table and that given by Gordeev and Pupko are principally the values for Pm^{147}, Xe^{133}, Xe^{131} and Ru^{101}. The sum of the values in the table is 170 barns per fission, and the total excess resonance integral of Gordeev and Pupko from 0.625 eV to 10^6 eV is 167 barns per fission. The two figures are not entirely comparable since not all the values in the table are integrals over a wide energy range. The value of 170 barns per fission must be considered only a lower limit since it is reasonable to assume that resonances which are at present unknown exist in some fission product nuclides. The upper limit will

almost always be conjectural unless it can be set by a total fission product poisoning experiment.

The fission product cross section curve and resonance integral values of Gordeev and Pupko have been used to derive values of the Westcott g and s factors as a function of neutron temperature, which are given in Table IV. From these it can be seen that (for example) for a reactor spectrum with an r-value of 0.1 and a neutron temperature of 300°K, the resonance effect puts up the poisoning due to the low cross section products by at least a third. Hence the effect is of considerable significance. The unknown resonance integrals may well constitute the largest source of error in present poisoning estimates.

6. FISSION YIELDS

To estimate poisoning, a knowledge of the yields is required as much as of the cross sections. It seems that these are not known with sufficient accuracy at least in the cases of U^{233} and Pu^{239}. Tables V, VI and VII show data on yields presented at the 1958 Geneva Conference and compared with values given in a recent compilation by Katcoff[5]. Although most of the experimental errors stated are between 3 and 5%, in the case of U^{233} there exist discrepancies of (for example) 19% for mass 133, 24% for mass 143, and 37% for mass 147. In the case of Pu^{239}, there exist discrepancies of (for example) 32% for mass 133, 32% for mass 135, 40% for mass 143, 43% for mass 149 and 48% for mass 151.

The errors on the yields are compared with the errors on the thermal cross sections for the most important known low cross section poisons in

Table VIII. The estimated yield errors are very approximate values obtained from inspection of the previous tables. Where no error is given, it is probably safe to assume about 5% for U^{235} and 10-15% for U^{233} and Pu^{239}.

7. RECOMMENDATIONS

Recommendations for further cross section studies have already been made by Pattenden and Rose[6], and some progress has been made along these lines. An ideal fission product programme would proceed something as follows:

An integral cross section survey (using activation or mass spectrometric techniques) would be made of all fission products of yield greater than about 0.1 per cent and half-life greater than about 80 days, using fission product sample material. The purpose of this would be twofold: (1) to check that long-lived isomeric states were not causing the "stable" fission products to have different cross sections from the natural nuclides, (2) to search for any unknown high cross sections. If a high cross section were found, then a sufficient quantity of that material would be chemically, and if necessary, isotopically, separated for spectrometer and pile oscillator measurements to be performed.

It is clear that for U^{235} the yield measurements are relatively satisfactory, but for U^{233} and Pu^{239} the accuracy of poisoning calculations is limited by uncertainties in yields rather than cross sections. Therefore an important part of any poisoning data programme should be to obtain an improvement in the yield measurements. However, in total poisoning calculations, the errors on yields may tend to cancel one another, whereas in the case of cross sections, the errors on known cross sections may be less important than the unknown cross sections, particularly in the resonance region.

REFERENCES

1958 Geneva Conference Papers:

11. N. J. Pattenden.

14. H. Rose, W. A. Cooper and R. B. Tattersall.

187. K. Fritze, C. C. McCullen and H. G. Thode.

201. D. R. Bidinosti, H. R. Fickel and R. H. Tomlinson.

203. T. A. Eastwood, A. P. Baerg, C. B. Bigham, F. Brown, M. J. Cabell, W. E. Grummitt, J. C. Roy, L. P. Roy and R. P. Schuman.

644. L. R. Bunney, E. M. Scadden, J. O. Abriam, and N. E. Ballou.

673. J. A. Harvey, R. C. Block, G. G. Slaughter, W. J. Martin and G. W. Parker.

2040. M. P. Anikina, P. M. Aron, V. K. Gorshkov, R. N. Ivanov, L. M. Krizhansky, G. M. Kukavadze, A. N. Mourin, I. A. Reformatsky and B. V. Ershler.

2223. I. V. Gordeev and V. Y. Pupko.

Other References:

1. C. H. Westcott, A.E.C.L. CRRP-680 (1957) and CRRP-787 (1958).
2. C. G. Campbell and R. G. Freemantle, A.E.R.E. RP/R 2031 (1956).
3. U. Businaro, S. Gallone and D. Morgan, Jour. Nuc. Energy 4, 319 (1957).
4. P. Greebler, H. Hurwitz and M. L. Storm, Nuc. Sci. and Eng. 2, 334 (1957).
5. S. Katcoff, Nucl. 16, 78 (1958).
6. N. J. Pattenden and H. Rose, A.E.R.E. NP and R/R 2651 (1958).

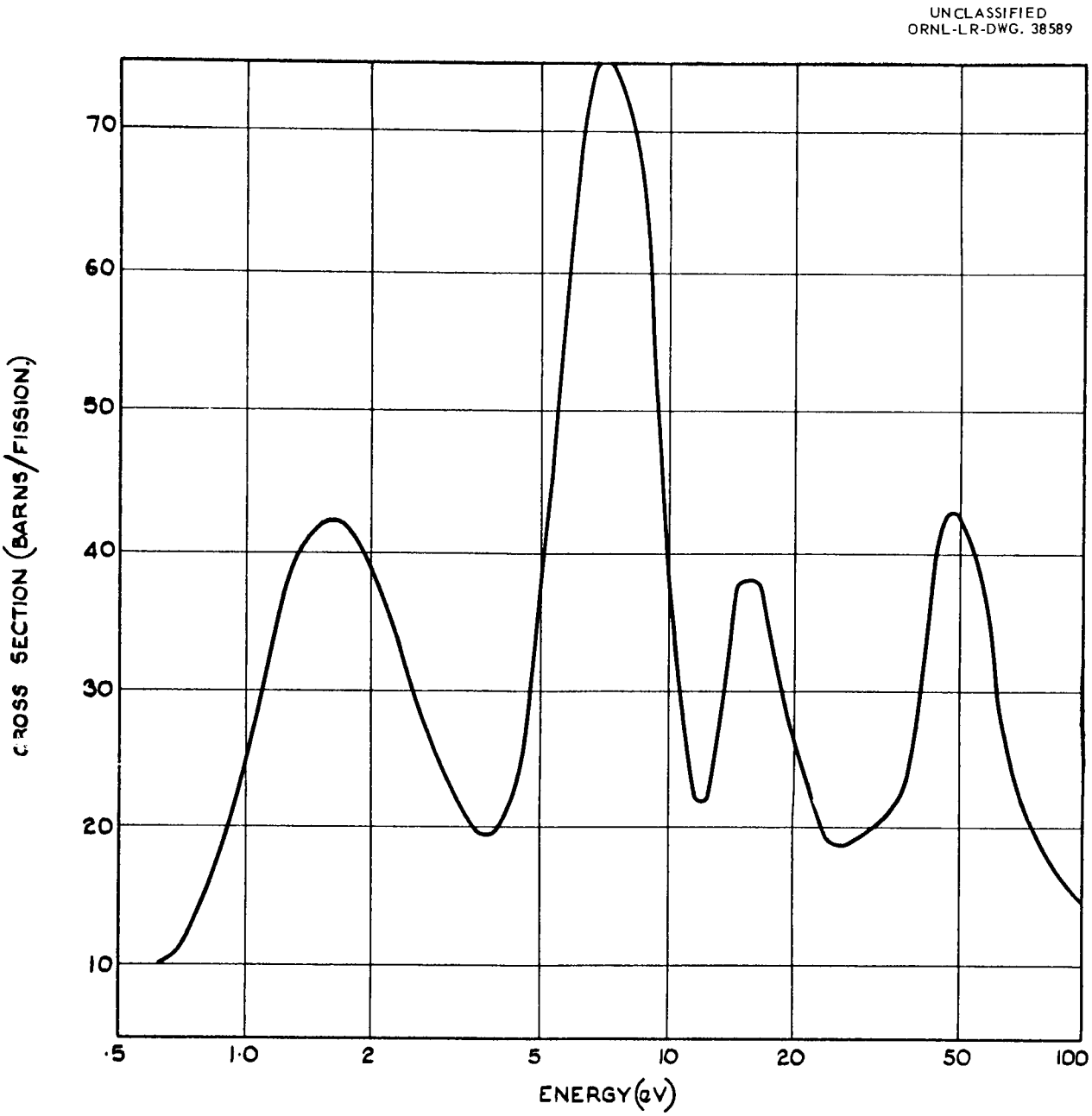

Fig. 1. Mean absorption cross section of U^{235} fission products, from Gordeev and Pupko (Paper 2223).

Table I. Poisoning of U^{235} Reactor At Flux of 10^{14} Neutrons/cm^2/sec

Fission Product	Yield (%)	σ_{th} (barns)	λ (sec^{-1})	Time to Reach 99% Saturation (sec)	Poisoning After 10^7 sec (n/sec/fissile atom)
Xe^{135}	6.3	2.8×10^6	2.1×10^{-5}	1.5×10^4	3.4×10^{-9}
Sm^{149}	1.3	68,500	0	6.7×10^5	6.6×10^{-10}
Pm^{147}	2.6	180	8.5×10^{-9}	1.8×10^8	2.4×10^{-10}
Nd^{143}	6.0	339	0	1.3×10^8	1.0×10^{-9}

Table II. Recent Fission Product Cross Section Measurements

Reference	Fission Product	σ_{2200} (barns)	σ_{2200} g (barns)	$\sigma_{eff} = \sigma_{2200}$ (g + rs) (barns)	Remarks
P/203	Rb^{88}			1.0 ± 0.3	$r\sqrt{(T/T_0)} = 0.02$
P/203	Sr^{88}			$(5.8 \pm 0.4) \times 10^{-3}$	$r\sqrt{(T/T_0)} = 0.02$
P/203	Sr^{89}			0.42 ± 0.04	$r\sqrt{(T/T_0)} = 0.02$
P/203	Y^{91}			<10	$r\sqrt{(T/T_0)} = 0.02$
P/11	Tc^{99}	24.8 ± 2	25 ± 2	28.8 ± 3	$r = 0.02$, $T = 300°K$
P/14	Rh^{103}		146 ± 5		
P/14	Cd^{113}		$26,100 \pm 300$		
P/203	I^{129}		27 ± 2		R.I. = 36 ± 4 barns; $r\sqrt{(T/T_0)} = 0.02$
ORNL	I^{129}	32 ± 5			
P/14	Cs^{133}		28 ± 1		
P/201	Cs^{133}			40	R.I. = 320 barns; $r = 0.026$. Assumes $\sigma_{2200} = 29$ barns, $T = 393°K$
P/203	Cs^{133}			29.8 ± 1.2	
P/203	Cs^{135}		8.7 ± 1.5	10.3 ± 0.5	
P/14	La^{139}		8.9 ± 0.3		
P/203	Ba^{140}			<20	R.I. = 185 ± 25 barns; $r\sqrt{(T/T_0)} = 0.02$
P/14	Pr^{141}		11.5 ± 0.3		
P/203	Pr^{143*}		88 ± 5		$r\sqrt{(T/T_0)} = 0.02$
P/11	Nd^{143*}	340 ± 20	339 ± 20	340 ± 20	$r = 0.02$, $T = 300°K$
P/14	Nd^{143}		340 ± 20		
P/11	Nd^{145}	51 ± 3	54 ± 3	62 ± 5	$r = 0.02$, $T = 300°K$

*Values re-estimated.

Table II (continued)

Reference	Fission Product	σ_{2200} (barns)	$\sigma_{2200}\,g$ (barns)	$\sigma_{eff} = \sigma_{2200}$ (g + rs) (barns)	Remarks
P/201	Pm^{147}			200 ± 50	
P/673	Pm^{147}	180			
P/2040	Pm^{147}			90 ± 20	
P/11	Sm^{149}	39,900 ± 600	68,600 ± 1,000	68,500 ± 1,000	r = 0.02, T = 300°K
P/14	Sm^{149}		68,200 ± 700		
P/201	Sm^{149}			84,000	r = 0.010–0.018, T = 393–410°K
P/2040	Sm^{149}			49,000 ± 8,000	
P/201	Sm^{151}			12,800 ± 400	
P/2040	Sm^{151}			8,000 ± 1,500	
Harwell	Sm^{151}	12,800 ± 1,500			
P/11	Sm^{152}	200 ± 6	194 ± 6	238 ± 10	r = 0.02, T = 300°K
P/14	Sm^{152}		224 ± 7		
P/11	Eu^{153}	448 ± 16	458 ± 16	498 ± 20	r = 0.02, T = 300°K
P/11	Gd^{155}	66,000 ± 2,000	56,000 ± 1,700	55,000 ± 2,000	r = 0.02, T = 300°K
P/14	Gd^{155}		49,800 ± 600		
P/11	Gd^{157}	264,000 ± 5,000	226,000 ± 4,500	221,000 ± 5,000	r = 0.02, T = 300°K
P/14	Gd^{157}		213,000 ± 2,000		

TABLE III

Resonance Integrals of Low Cross Section Fission Products

Fission Product	Yield for U^{235} (%) y	Excess Resonance Integral (barns) $\sigma_{r.i.}$	$y \times \sigma_{r.i.}$
Pm^{147}	2.7	~ 2000 (a)	54
Cs^{133}	6.6	450 ± 50 (b)	30
Xe^{131}	2.9	~ 690 (a)	20
Rh^{103}	3.0	600 (c)	18
Sm^{152}	0.73*	1960 ± 100 (d)	14.3
Nd^{145}	4.0	~ 350 (a)	14
Tc^{99}	6.0	~ 170 (a)	10
Mo^{95}	6.3	100 (c)	6.3
Eu^{153}	0.15	~ 1800 (a)	2.7
Nd^{143}	6.0	~ 15 (a)	0.9

*Includes second order product.

(a) Estimated from parameters.

(b) R. B. Tattersall, H. Rose, S. K. Pattenden and D. Jowitt, A.E.R.E. TNCC/UK 33 (1958) (unpublished).

(c) R. L. Macklin and H. Pomerance, 1st. Geneva Conference, Vol. 5, 96 (1956).

(d) H. Rose, W. A. Cooper and R. B. Tattersall, 2nd. Geneva Conference, Paper 14 (1958).

TABLE IV

Effective Cross Section of Total Fission Products

Derived from Gordeev and Pupko (Paper 2233)

σ_{2200} = 50 barns/fission

Temperature (°K)	g	s
300	1.00	3.43
500	1.00	4.52
750	1.00	5.64
1000	1.00	6.59
1250	1.02	7.32
1500	1.02	8.06
1600	1.03	8.35
1700	1.03	8.62
1800	1.04	8.90
1900	1.05	9.08
2000	1.06	9.36

Table V. U^{233} Thermal Fission Yields (%)

Product	Paper 201 ± 3%	Paper 2040 Isotopic Dilution Method	Paper 2040 Integral Mass Spectrographic Method	Katcoff Compilation
Sr^{88}		5.30 ± 0.30		6.5
Sr^{90}		5.80 ± 0.40		
Cs^{133}	5.6	5.20 ± 0.30	5.50 ± 0.13	6.18
Xe^{135}	6.0			
Cs^{137}		5.80 ± 0.30	6.16 ± 0.14	7.16
La^{139}			5.91 ± 0.23	
Ce^{140}		5.45 ± 0.50	6.16 ± 0.24	5.6
Pr^{141}			5.57 ± 0.19	
Ce^{142}		5.50 ± 0.50	6.06 ± 0.24	5.6
Nd^{143}	6.45	5.00 ± 0.30	5.19 ± 0.17	5.2
Nd^{144}		3.80 ± 0.40	3.84 ± 0.15	4.0
Nd^{145}		2.82 ± 0.25	2.88 ± 0.08	3.0
Nd^{146}		2.20 ± 0.15	2.24 ± 0.07	2.3
Pm^{147}	2.1		1.53 ± 0.06	
Nd^{148}		1.03 ± 0.10	1.07 ± 0.04	1.15
Sm^{149}	0.8	0.66 ± 0.13	0.70 ± 0.03	0.61
Nd^{150}		0.51 ± 0.08	0.49 ± 0.02	0.48
Sm^{151}	0.3		0.33 ± 0.03	0.26
Sm^{152}			0.21 ± 0.02	0.17
Eu^{153}			0.13 ± 0.02	

Table VI. U^{235} Thermal Fission Yields (%)

Product	Paper 201 ± 3%	Paper 2040	Katcoff Compilation
Cs^{133}	6.59		6.59
Xe^{135}	6.41		6.41
Ce^{140}		6.30 ± 0.30	6.44
Pr^{141}		5.60 ± 0.30	6.4
Ce^{142}		5.80 ± 0.20	5.95
Nd^{143}	5.80	5.80 ± 0.20	5.98
Nd^{144}		5.60 ± 0.30	5.67
Nd^{145}		4.00 ± 0.10	3.95
Nd^{146}		3.20	3.07
Pm^{147}	2.38	2.90 ± 0.40	2.38
Nd^{148}		1.70 ± 0.10	1.70
Sm^{149}	1.13	1.50 ± 0.30	1.13
Nd^{150}		0.70 ± 0.10	0.67
Sm^{151}	0.45		0.45

Table VII. Pu^{239} Thermal Fission Yields (%)

Product	Paper 187	Paper 201 ±3%	Paper 644	Paper 2040	Katcoff Compilation
Kr^{83}	0.29 ± 0.01				
Kr^{84}	0.47 ± 0.02				
Kr^{85}	0.55 ± 0.03				
Kr^{86}	0.75 ± 0.02				
Sr^{88}				1.39 ± 0.04	
Sr^{90}				2.31 ± 0.05	
I^{128}	$(2.1 \pm 0.1) \times 10^{-4}$*				
I^{129}	$(5.5 \pm 0.4) \times 10^{-4}$*				
Xe^{131}	3.79 ± 0.11				2.87
Xe^{132}	5.29 ± 0.16				4.02
Cs^{133}	6.95 ± 0.21	6.92		5.26 ± 0.13	5.27
Cs^{134}	7.48 ± 0.22				
Xe^{135}		7.27			
Cs^{135}				6.95 ± 0.19	5.53
Xe^{136}	6.70 ± 0.70				5.06
Cs^{137}				6.50 ± 0.16	5.24
Ba^{138}				6.26 ± 0.15	
Ce^{140}				5.52 ± 0.14	5.68
Pr^{141}				6.02 ± 0.18	
Ce^{142}				6.66 ± 0.17	6.69
Nd^{143}		4.49		6.10 ± 0.15	6.31
Nd^{144}				5.50 ± 0.17	5.29
Ce^{144}			3.6		5.29
Nd^{145}				4.20 ± 0.11	4.24
Nd^{146}				3.53 ± 0.09	3.53
Pm^{147}		2.07		2.58 ± 0.05	
Nd^{147}			2.2		
Nd^{148}				2.30 ± 0.05	2.28
Sm^{149}		1.32		1.68 ± 0.02	1.89
Pm^{149}			1.4		
Nd^{150}				1.35 ± 0.02	1.38
Sm^{151}		0.79		1.01 ± 0.02	1.17
Sm^{152}				0.75 ± 0.015	0.83
Sm^{153}			0.33		0.41
Sm^{154}				0.36 ± 0.009	0.32
Eu^{156}			0.10		0.12
Gd^{159}			0.021		
Tb^{161}			0.0039		
Dy^{166}			0.000068		

*Independent yields.

Table VIII. Important Low Cross Section Poisons

PRODUCT	$g\,\sigma_{2200} \times$ yield (U^{235}) barns/fission	ERROR ON $g\,\sigma_{2200}$ (%)	ESTIMATED ERROR ON YIELDS %		
			U^{233}	U^{235}	Pu^{239}
Nd^{143}	20	5.9	15	3.5	20
Rh^{103}	4.4	2.9			
Xe^{131}	3.5	8.0			20
Nd^{145}	2.2	5.7	5	3	5
Cs^{133}	1.85	3.6	10	3	10
Sm^{152}	1.63	2.9	10		5
Pm^{147}	1.7	~50	15	6	10
Tc^{99}	1.5	8			
Sm^{150}	1.21	5	5	10	5
Kr^{83}	1.19	5			5
Mo^{95}	0.84	10			
Pr^{141}	0.70	2.6	5	10	5
Eu^{153}	0.67	3.5	15		10
La^{139}	0.57	3.4	5		
I^{129}	0.30	16			
Nd^{146}	0.28	6.5	3	3	3
Nd^{144}	0.25	11	4	5	4
Xe^{134}	0.2	~100			3

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