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Neutron Cross Sections For ^{Ta} In The Nine - Energy - Group Format

J.W. Zwick

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ABSTRACT

Neutron microscopic cross sections for tantalum are presented in the nine-group format.

INTRODUCTION

Because of its properties, tantalum may become an important material in fast-reactor technology. Neutron cross sections of tantalum are therefore required for use in fast-reactor calculations.

At GE-NMPO, neutron cross sections of tantalum are available for use in two standard group formats: the 19-level Nuclear Data Tape (NDT) structure, and the 25-group Reactor Nuclear Data Tape (RNDT) format.*† The latter cross sections are group-averaged with a flat flux in lethargy. Microscopic neutron cross sections of tantalum in a nine-energy group structure for use in fast-reactor transport-theory calculations are presented herein. The cross sections are given in barns.

NINE-GROUP NEUTRON CROSS SECTIONS FOR TANTALUM

Table 1 shows the boundary energies and corresponding lethargies for the nine-group neutron cross section format. The microscopic neutron cross sections for tantalum, shown in Table 2, were weighted within each of the top five groups with the U^{235} fission spectrum, and within each of the remaining four groups with a flat flux in lethargy. The cross sections listed in Table 3 were flat-flux-weighted within each of the nine energy groups and are included for comparison.

The cross sections given in Tables 2 and 3 do not include the transport correction and, therefore, do not contain the anisotropy effect of elastically scattered neutrons. The tantalum cross sections, averaged in the same manner as those in Table 2 and incorporating the transport correction, are listed in Table 4.‡

The cross sections, in the 9-group format, consist of: $\nu\sigma_f$, fission production cross section; σ_{coll} , collision cross section; $\sigma_s(n \rightarrow n')$, zeroth-moment scattering transfer cross section matrix, which provides for scattering within each group and for energy-degrading scattering from each group to a maximum of five lower groups.

In Tables 2 and 3, the collision cross section in group n is given by:

$$\begin{aligned}\sigma_{coll}(n) &= \sum_{n'} \sigma_s(n \rightarrow n') + \sigma_a(n) \\ \sigma_s(n) &= \sigma_{se}(n) + \sigma_{si}(n) = \sum_{n'} \sigma_s(n \rightarrow n') \\ \sigma_{se}(n) &= \sum_{n'} \sigma_{se}(n \rightarrow n') \\ \sigma_{si}(n) &= \sum_{n'} \sigma_{si}(n \rightarrow n')\end{aligned}$$

The collision cross section in Table 4 is given by the expression

$$\sigma_{coll}(n) = \sigma_{se}(n) [1 - \bar{\mu}_e(n)] + \sigma_{si}(n) + \sigma_a(n).$$

Here, $\sigma_s(n)$, $\sigma_{se}(n)$, $\sigma_{si}(n)$ and $\sigma_a(n)$ are, respectively, the average total microscopic scattering cross section, the average elastic scattering cross section, the average inelastic scattering cross section and the average absorption cross section in energy-group, n ;

*J. W. Zwick, "Neutron Cross Sections for Rhodium and Tantalum," GE-NMPO, XDC 61-4-35, May 1961.

†J. W. Zwick and T. J. Kostigen, "Twenty-Five Group Reactor Nuclear Data Tape Neutron Cross Sections," GE-NMPO, APEX-709, July 1961.

‡The transport correction was incorporated by C. S. Robertson.

$\sigma_s(n \rightarrow n')$, $\sigma_{se}(n \rightarrow n')$ and $\sigma_{si}(n \rightarrow n')$ are, respectively, the cross sections for neutron total scattering, elastic scattering and inelastic scattering from group n to group n' , while $\bar{\mu}_e(n)$ is the average cosine of the elastic scattering angle in the laboratory coordinate system.

The methods used in processing of the tantalum cross sections and the sources of the cross section data are described in the references previously cited.

The nine-group tantalum cross sections listed in Table 4 will be used in fast-reactor calculations with transport theory Program S.*

TABLE 1
NINE-GROUP STRUCTURE BOUNDARY ENERGIES AND
CORRESPONDING LETHARGIES

Group Index	Group Boundary		Group Width	
	Energies, ev	Lethargies	Energies, ev	Lethargies
1	1.0×10^7	0.0	7.0×10^6	1.204
2	3.0×10^6	1.204	1.6×10^6	0.762
3	1.4×10^6	1.966	5.0×10^5	0.442
4	9.0×10^5	2.408	5.0×10^5	0.811
5	4.0×10^5	3.219	3.0×10^5	1.386
6	1.0×10^5	4.605	8.3×10^4	1.772
7	1.7×10^4	6.377	1.3646×10^4	1.623
8	3.354×10^3	8.0	2.9×10^3	2.0
9	454.0	10.0	392.56	2.0
	61.44	12.0		

TABLE 2
GROUP-AVERAGED MICROSCOPIC NEUTRON CROSS SECTIONS OF TANTALUM^a

Group	N	$\nu\sigma_f$	σ_{coll}	$\sigma_S(N - K \rightarrow N)$						σ_a
				K						
				0	1	2	3	4	5	
1	0.0	6.12721	3.35467	0.0	0.0	0.0	0.0	0.0	0.03736	
2	0.0	7.27719	4.48387	0.34006	0.0	0.0	0.0	0.0	0.07802	
3	0.0	7.61320	5.05354	0.35571	0.48243	0.0	0.0	0.0	0.13133	
4	0.0	7.79033	6.29601	0.93165	1.06552	1.07592	0.0	0.0	0.19415	
5	0.0	8.12451	7.63760	1.04156	1.25002	1.12294	0.74034	0.0	0.28608	
6	0.0	9.46774	8.47399	0.19321	0.24656	0.23388	0.16489	0.09043	0.69980	
7	0.0	13.23700	10.46959	0.29395	0.00749	0.01073	0.00923	0.00624	2.44101	
8	0.0	22.34117	13.86360	0.32640	0.0	0.00023	0.00031	0.00026	8.27219	
9	0.0	04.85310	33.27990	0.20538	0.0	0.0	0.0	0.00001	31.57320	

^aGroups 1-5 are U²³⁵ fission spectrum weighted; groups 6-9 are flat flux (in lethargy) weighted.

*B. H. Duane, "Neutron and Photon Transport-Plane-Cylinder-Sphere GE-ANPD Program S Variational Optimum Formulation," GE-ANPD, XDC 59-9-118, January 1959.

TABLE 3
GROUP-AVERAGED MICROSCOPIC NEUTRON CROSS SECTIONS OF TANTALUM^a

GROUP 1: HEAVY METALS: MICROSCOPIC NEUTRON CROSS SECTIONS OF CARBON										
Group	$\sigma_s(N - K \rightarrow K)$									
	N	$\nu\sigma_f$	σ_{coll}	K						σ_a
				0	1	2	3	4	5	
1	0.0	5.64877	3.05576	0.0	0.0	0.0	0.0	0.0	0.02989	
2	0.0	7.27412	4.47651	0.38608	0.0	0.0	0.0	0.0	0.07795	
3	0.0	7.62004	5.03561	0.36127	0.51227	0.0	0.0	0.0	0.13268	
4	0.0	7.80372	6.30617	0.95469	1.06540	0.98099	0.0	0.0	0.20078	
5	0.0	8.23067	7.66310	1.04140	1.25231	1.12195	0.61318	0.0	0.29552	
6	0.0	9.46774	8.47399	0.26573	0.24417	0.23508	0.16473	0.07060	0.69980	
7	0.0	13.23700	10.46959	0.29395	0.00606	0.01074	0.00930	0.00631	2.44101	
8	0.0	22.34117	13.86360	0.32640	0.0	0.00026	0.00045	0.00037	8.27219	
9	0.0	64.85310	33.27990	0.20538	0.0	0.0	0.0	0.00001	31.57320	

^aFlat flux (in lethargy) weighting used in all nine groups.

TABLE 4
GROUP-AVERAGED MICROSCOPIC NEUTRON CROSS SECTIONS OF TANTALUM^a

Group	$\sigma_S(N - K \rightarrow N)$									σ_a
	N	$\nu\sigma_f$	σ_{coll}	K						
				0	1	2	3	4	5	
1	0.0	3.44392	0.73490	0.0	0.0	0.0	0.0	0.0	0.0	0.03736
2	0.0	4.13579	1.39913	0.27654	0.0	0.0	0.0	0.0	0.0	0.07802
3	0.0	5.11288	2.64005	0.29905	0.48243	0.0	0.0	0.0	0.0	0.13132
4	0.0	6.08123	4.60473	0.84482	1.06552	1.07592	0.0	0.0	0.0	0.19415
5	0.0	6.88085	6.40246	1.02474	1.25332	1.12294	0.74634	0.0	0.0	0.28608
6	0.0	9.22613	8.24048	0.18459	0.24656	0.23388	0.16489	0.09043	0.0	0.69980
7	0.0	13.19357	10.42747	0.28585	0.00749	0.01073	0.00923	0.00624	2.44101	2.44101
8	0.0	22.28889	13.81208	0.32509	0.0	0.00023	0.00031	0.00026	8.27219	8.27219
9	0.0	64.72942	33.15622	0.20462	0.0	0.0	0.0	0.00001	31.5732	31.5732

^aGroups 1-5 are U²³⁵ fission-spectrum weighted; groups 6-9 are flat-flux (in lethargy) weighted. Transport correction included.

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