

Program EVALPLOT  
(Version 2018-1)

by

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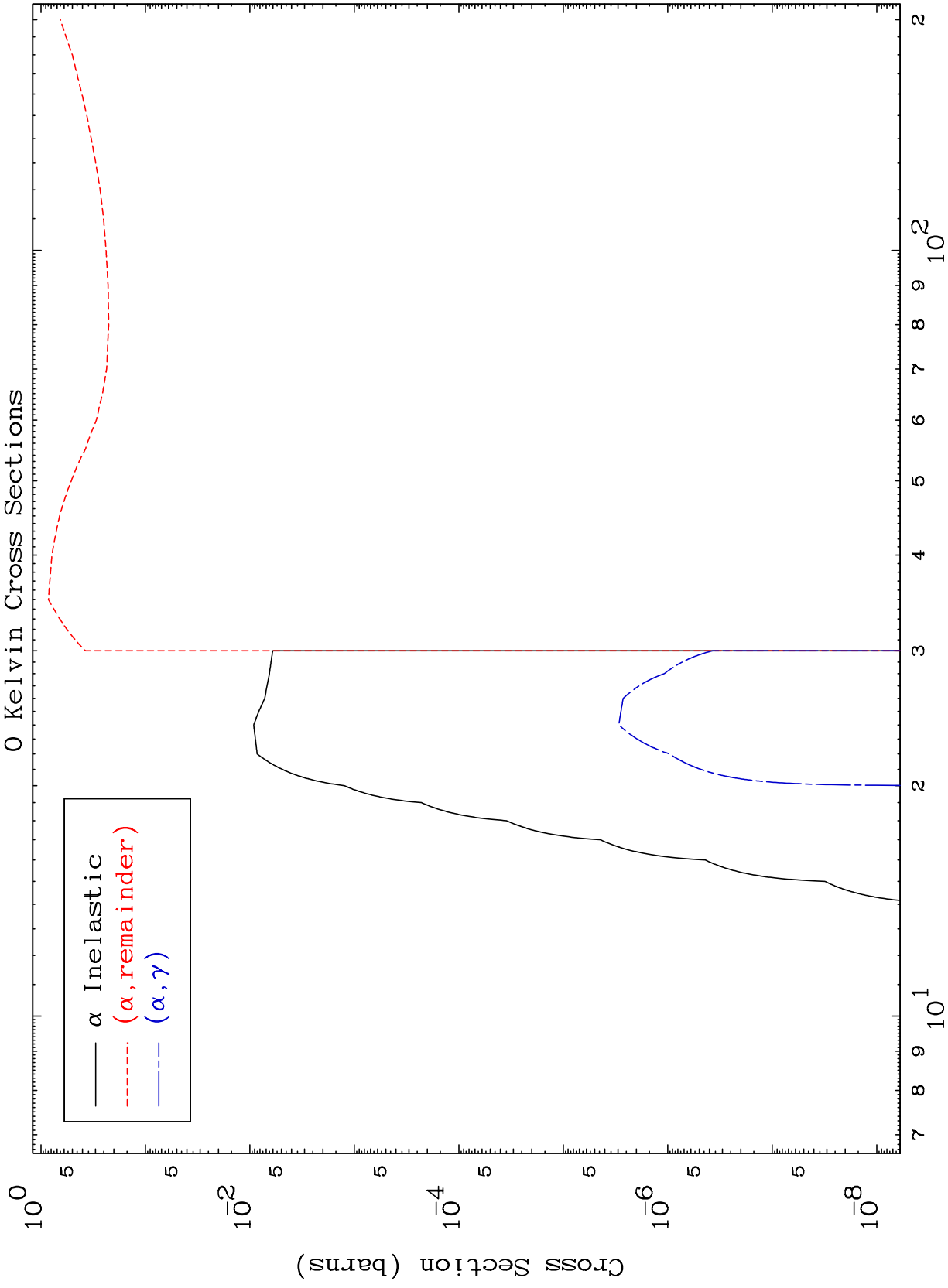
Press Mouse Button to Start

MAT 9743

0 Kelvin

$\alpha$  Major  
Cross Sections

97-Bk-246



$\alpha$  Inelastic  
( $\alpha$ , remainder)  
( $\alpha$ ,  $\gamma$ )

1

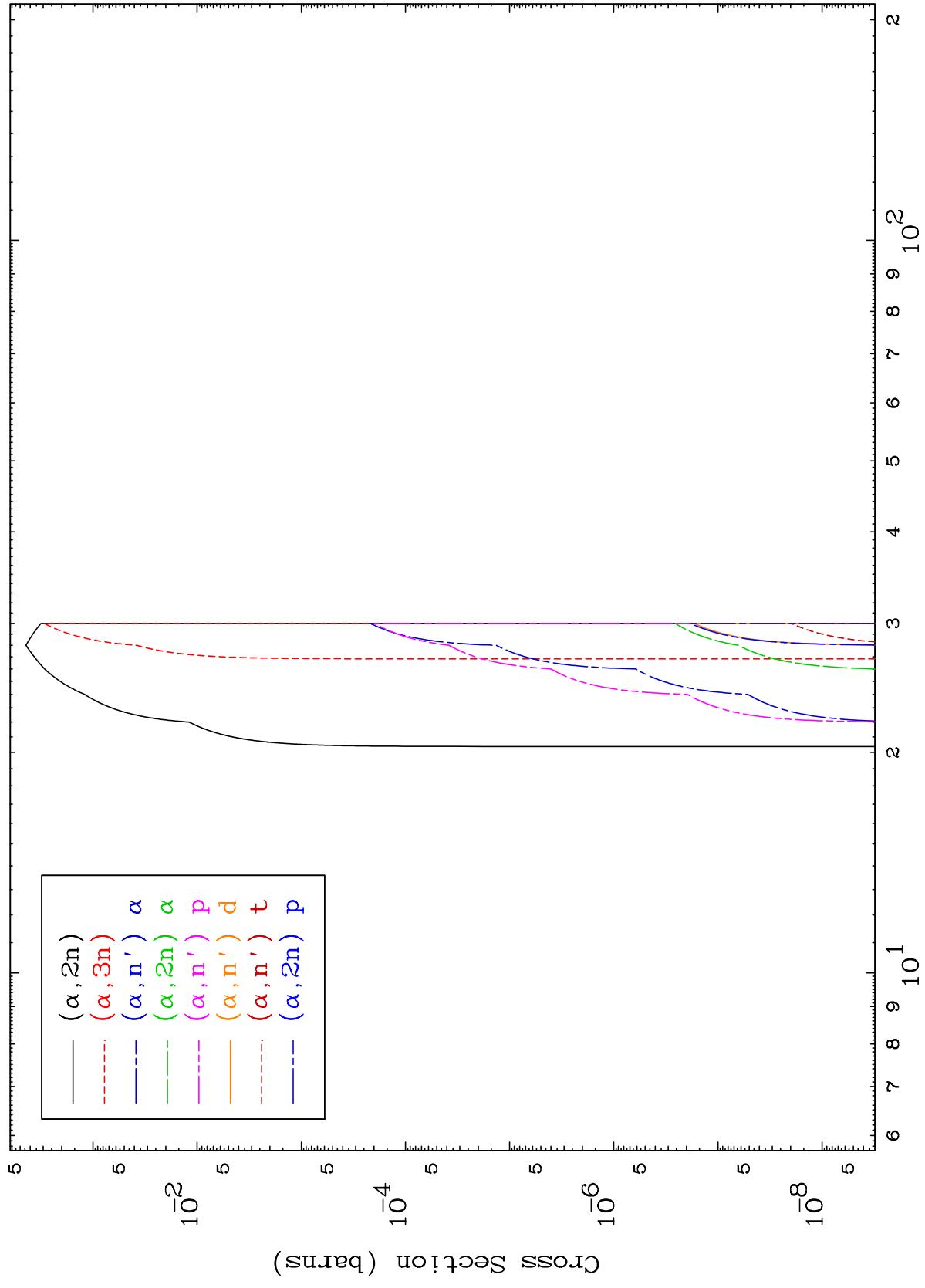
Incident Energy (MeV)

97-Bk-246

MAT 9743

$\alpha$  Neutron Production  
0 Kelvin Cross Sections

97-Bk-246



2

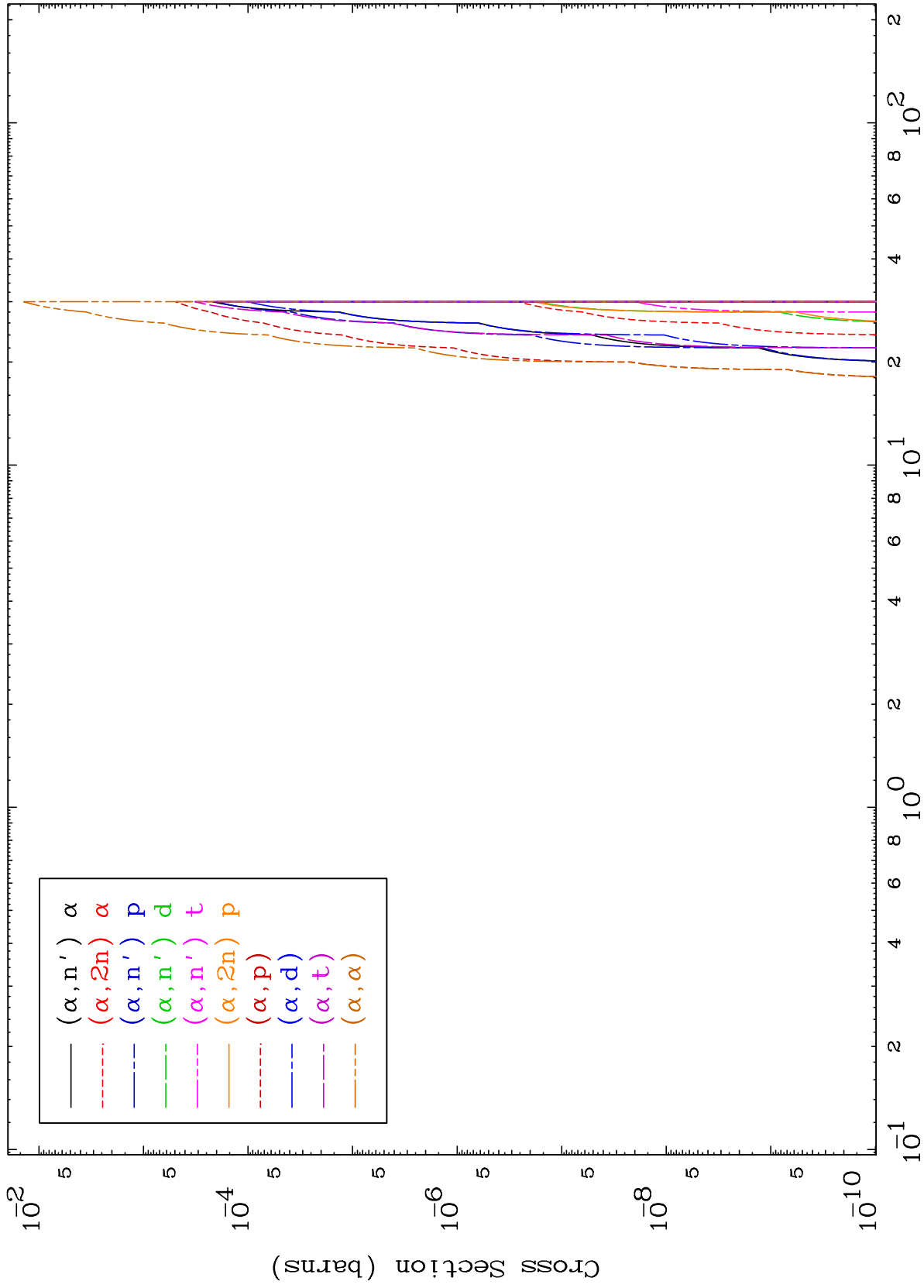
Incident Energy (MeV)

97-Bk-246

MAT 9743

$\alpha$  Charged Particle  
0 Kelvin Cross Sections

97-Bk-246



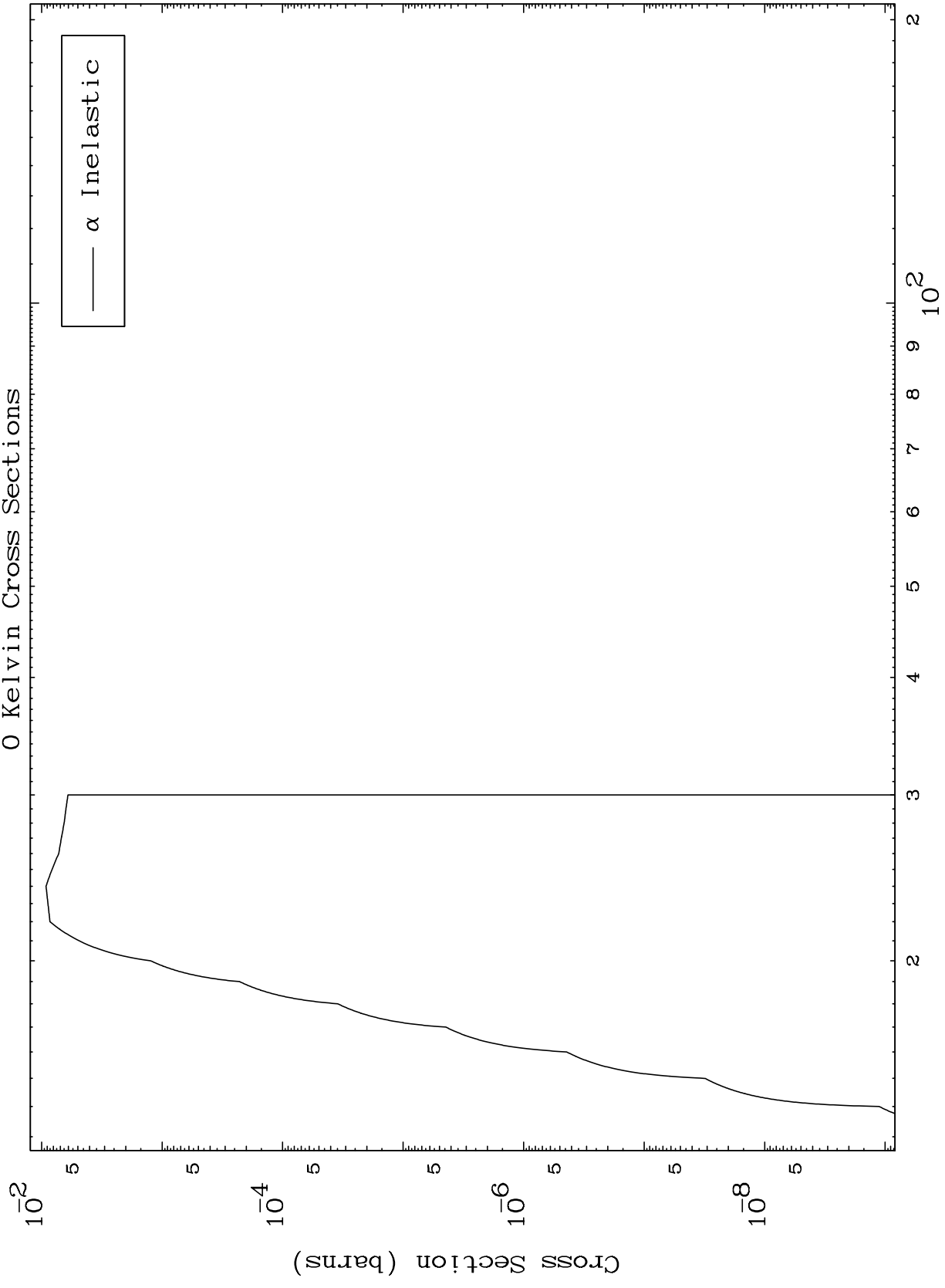
Incident Energy (MeV)

97-Bk-246

MAT 9743

( $\alpha, n'$ ) Level  
0 Kelvin Cross Sections

97-Bk-246



4

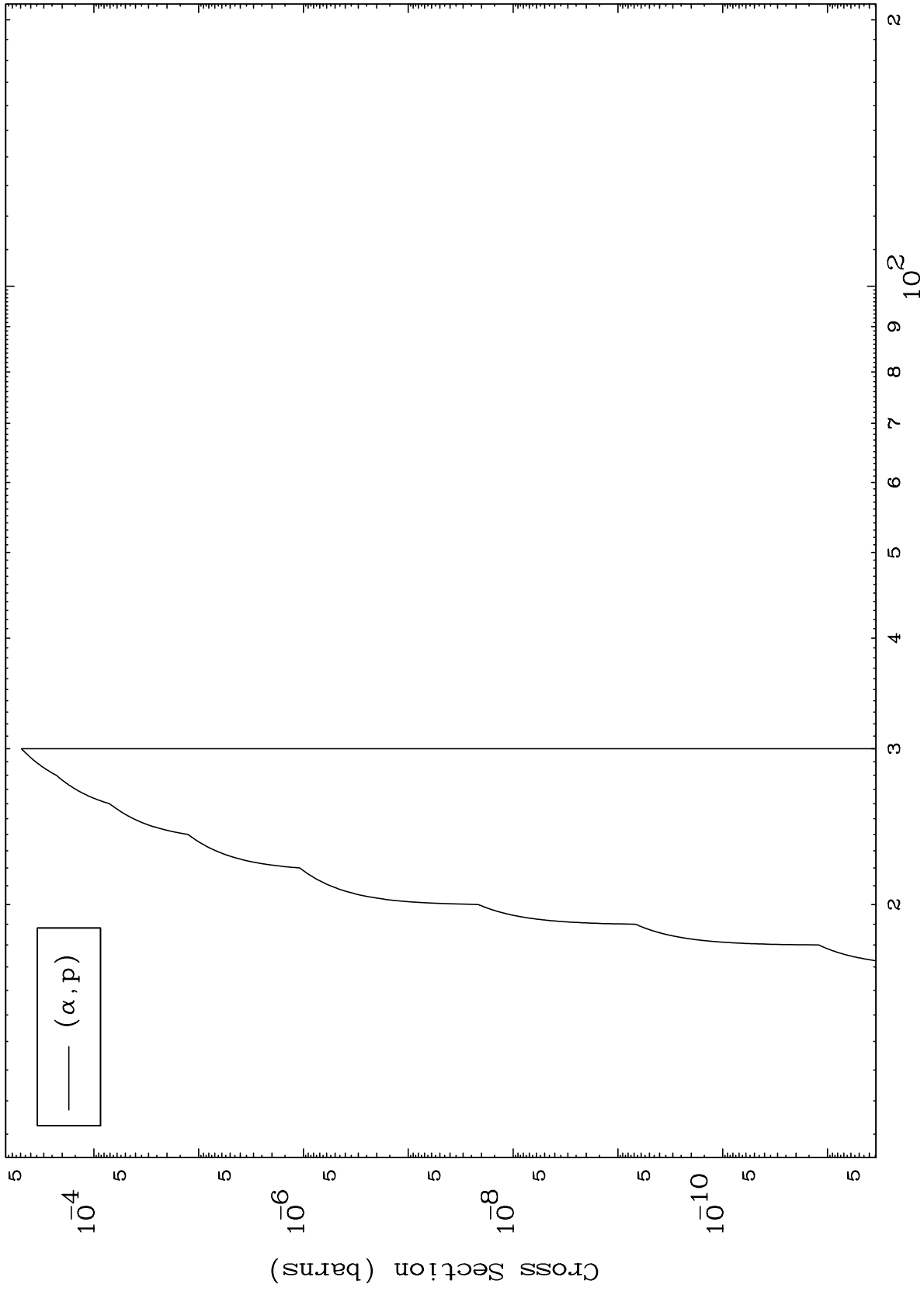
Incident Energy (MeV)

97-Bk-246

MAT 9743

( $\alpha, p$ ) Levels  
0 Kelvin Cross Sections

97-Bk-246



5

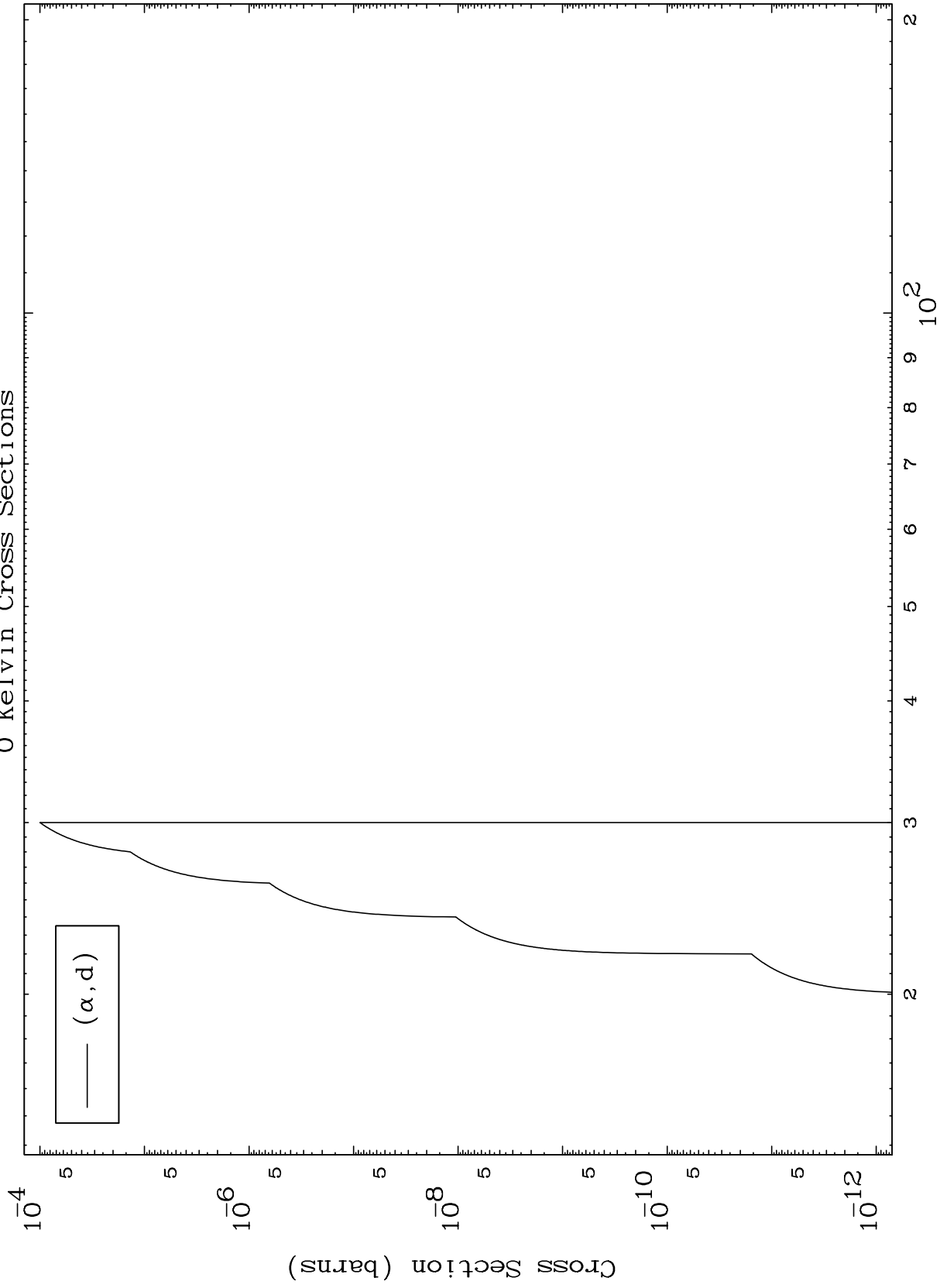
Incident Energy (MeV)

97-Bk-246

MAT 9743

( $\alpha, d$ ) Levels  
0 Kelvin Cross Sections

97-Bk-246



6

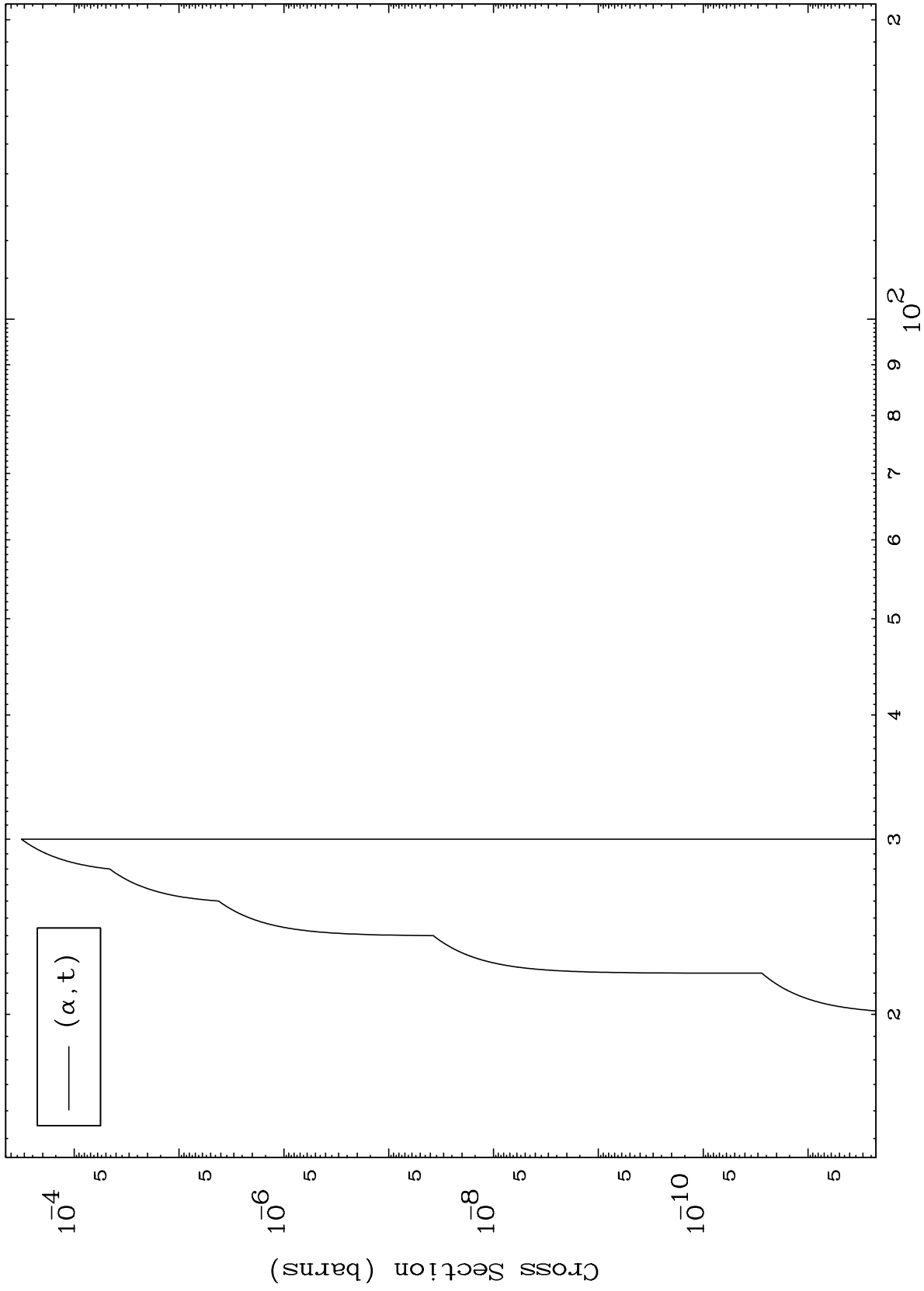
Incident Energy (MeV)

97-Bk-246

MAT 9743

( $\alpha, t$ ) Levels  
0 Kelvin Cross Sections

97-Bk-246



7

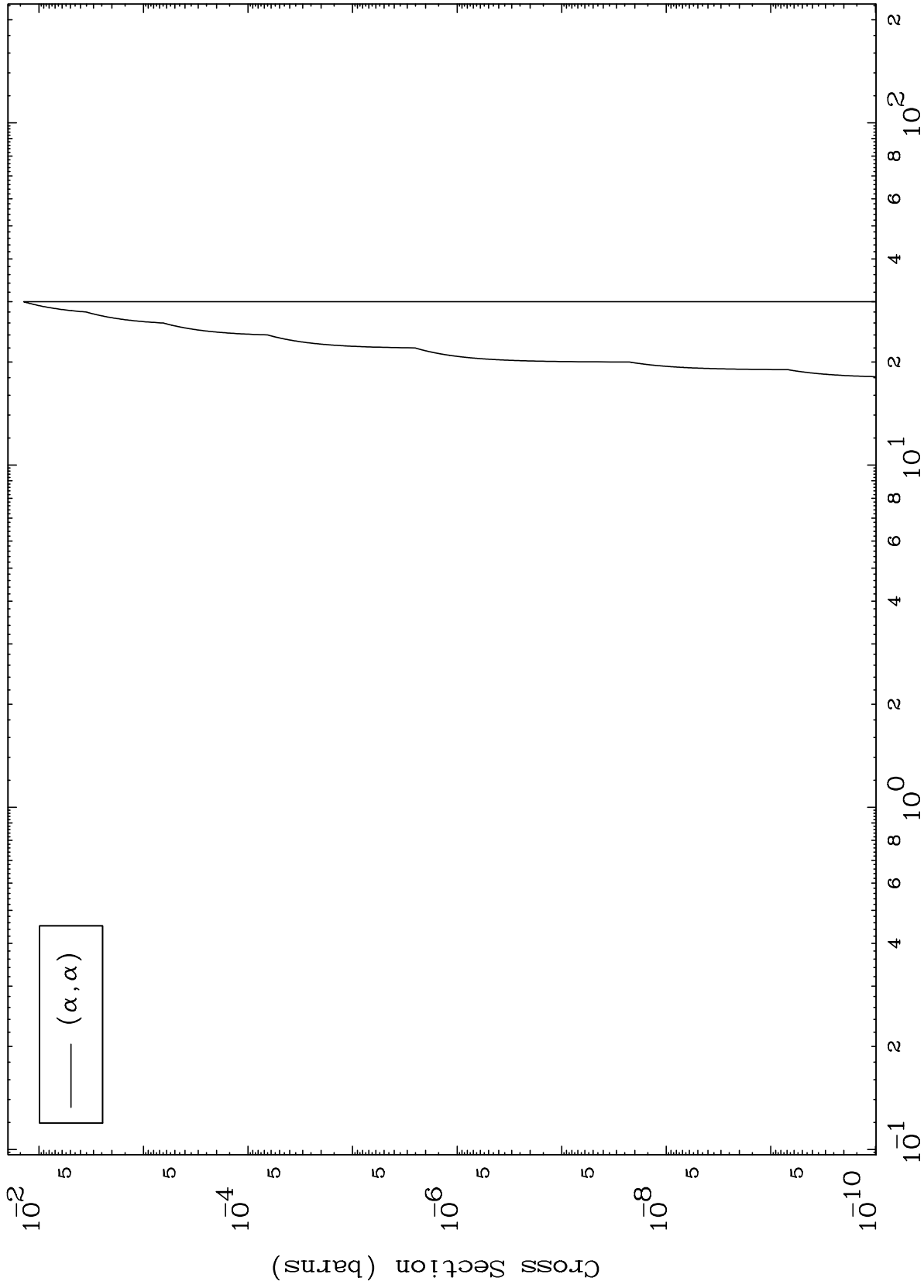
Incident Energy (MeV)

97-Bk-246

MAT 9743

( $\alpha, \alpha$ ) Levels  
0 Kelvin Cross Sections

97-Bk-246



8

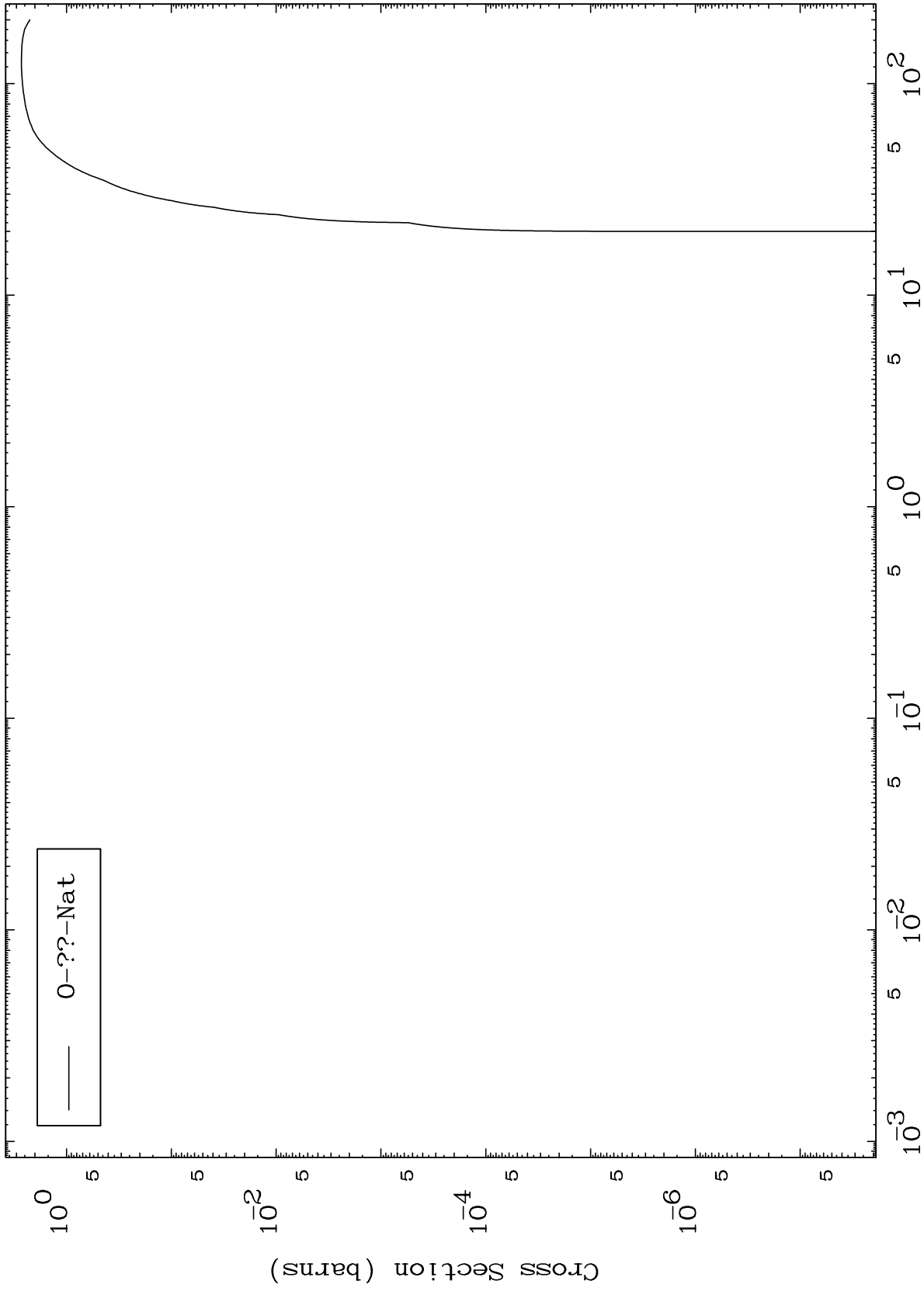
Incident Energy (MeV)

97-Bk-246

MAT 9743

$\alpha$  Fission  
Radionuclide Production Cross Section

97-Bk-246



Incident Energy (MeV)

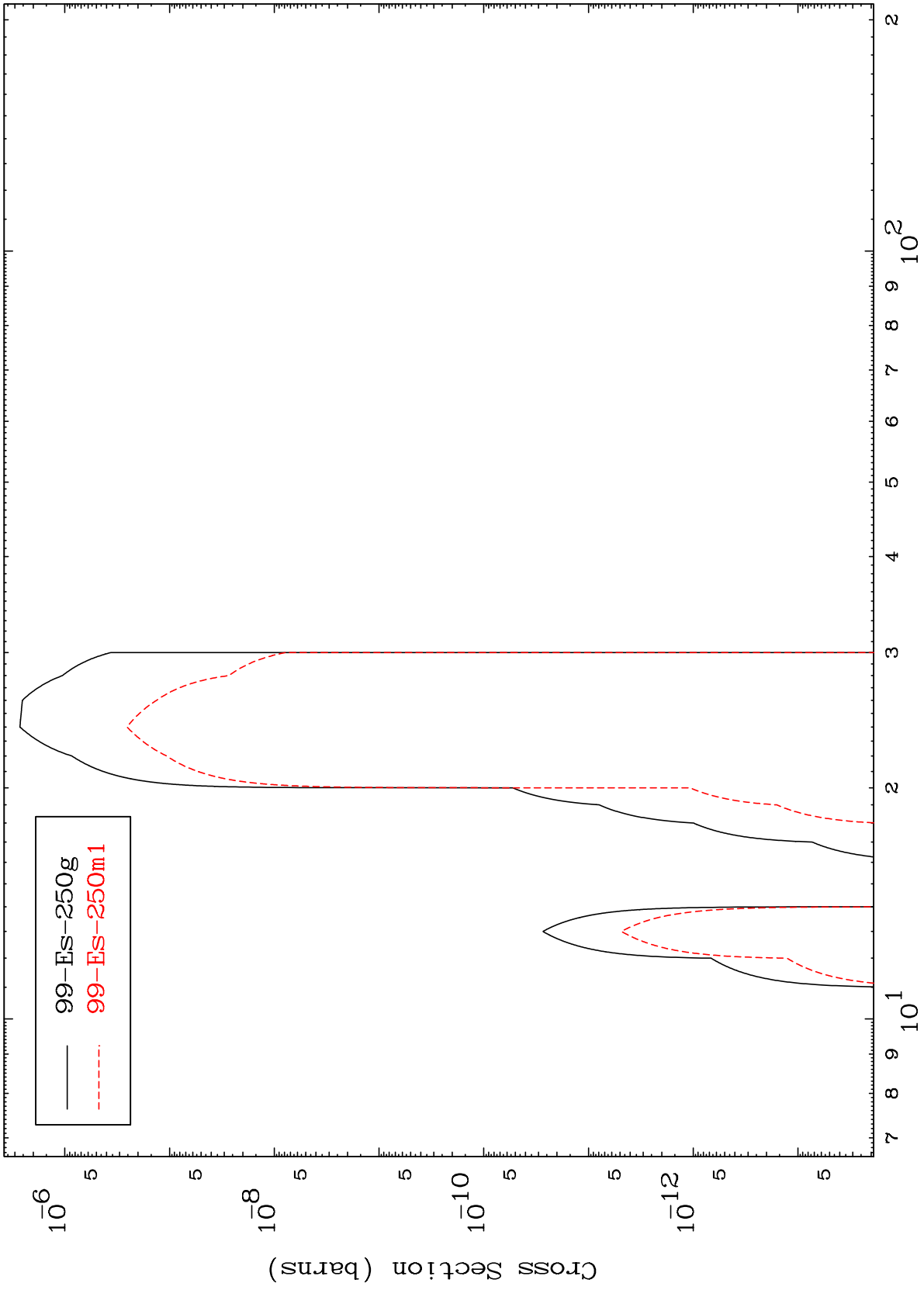
97-Bk-246

9

MAT 9743

97-Bk-246

( $\alpha, \gamma$ )  
Radionuclide Production Cross Section



10

Incident Energy (MeV)

97-Bk-246