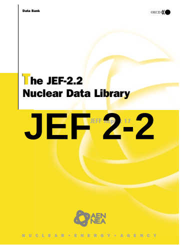

JEF/DOC-309

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Würenlingen/Villigen, 4 January 1990

Subject: Thermal Neutron Region Slowing Down Sources

Dear Mrs. Mattes

I am sending you some plots comparing slowing down sources into the thermal neutron energy range for hydrogen and oxygen in water as well as graphite computed at General Atomics (GA) in San Diego and at the Paul Scherrer Institute (PSI) in Würenlingen. Both sets of sources are based upon the ENDF/B scattering law data. The "GA" data sets shown here were generated by me with some algorithms based upon the "short collision time" approximation for long range energy transfers and a 1/E flux shape above the thermal neutron energy range (it should be noted that the 1/E flux shape assumption is not necessary, the GGC-5 code [1] uses the same algorithms with problem specific fine group fluxes above the thermal neutron energy range). The "JEF" data sets were computed at PSI with the NJOY code and WIMS-D weighting spectra ("wet" for hydrogen and oxygen, "dry" for graphite) [2]. One sees that the "JEF" (PSI/NJOY) sources appear to be computed at a much lower temperature (probably zero K) than the "GA" sources.

The infinite medium thermal neutron fluxes computed with the MICROX-2 code [3] with the two different data sets are not very different (see attached plots) but will have a slightly different temperature dependence. This is because:

1. the GA slowing down sources are temperature dependent and the PSI/NJOY sources are not, and
2. fewer neutrons are scattered into the energy range below 0.025 eV when real temperature scattering is considered.

Note that the "short collision time" approximation used in the GA work is really a free gas scattering kernel with an effective temperature which is larger than the physical temperature and that the difference between the effective temperature and the physical temperature decreases with increasing physical temperature [5]. It was found necessary to use series expansions for the error functions and their complements that occur in the free gas scattering kernel equations in order to reliably compute the temperature dependent slowing down sources from energies well above the usual thermal neutron energy range. It should also be noted that some of the differences in the computed fluxes are probably caused by differences in nuclear data processing systems. The GA fast and thermal data sets were obtained with GA developed codes [4], not the NJOY system. The overall agreement in computed fluxes is good except at low energies in the hydrogen moderated systems in which the differences in slowing down sources are most important. The small difference in the computed very low energy fluxes in an infinite graphite medium is probably due to the use of somewhat more rigorous (and very time consuming) integrations of the thermal neutron scattering law data in the GA developed codes as compared to the methods used in the NJOY code system. The GA scattering kernel for graphite was generated with a weighting spectrum appropriate to large graphite moderated systems.

I don't know just what approximations are used to compute the slowing down sources into the thermal neutron energy range in other European code systems but I wonder if some effect of this type is responsible for at least part of the temperature coefficient calculational problems which have caused the discussions about the possibilities of an unusual shape of the U-235 cross sections at very low energies, unusual Doppler broadening temperatures for U-238, etc. I do know that the GA calculations of temperature coefficients for graphite moderated systems were improved a little bit when we took such effects into account and that the effect should be larger in water moderated systems.

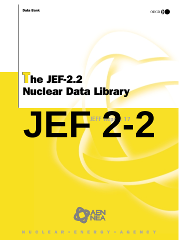
Yours truly

D. Mathews

D. Mathews

Cc: A. Baxter(GA)
R. Chawla
S. Pelloni
J. Stepanek
P. Vontobel

Encl: Slowing down sources (P_0) and infinite medium neutron fluxes for H_2O, H in H_2O, Oxygen and Graphite at room temperature (8 plots)



References

1. D. R. Mathews, et. al., "GGC-5, A computer Program for Calculating Neutron Spectra and Group Constants," Gulf General Atomic Co. Report GA-8871 (1971).

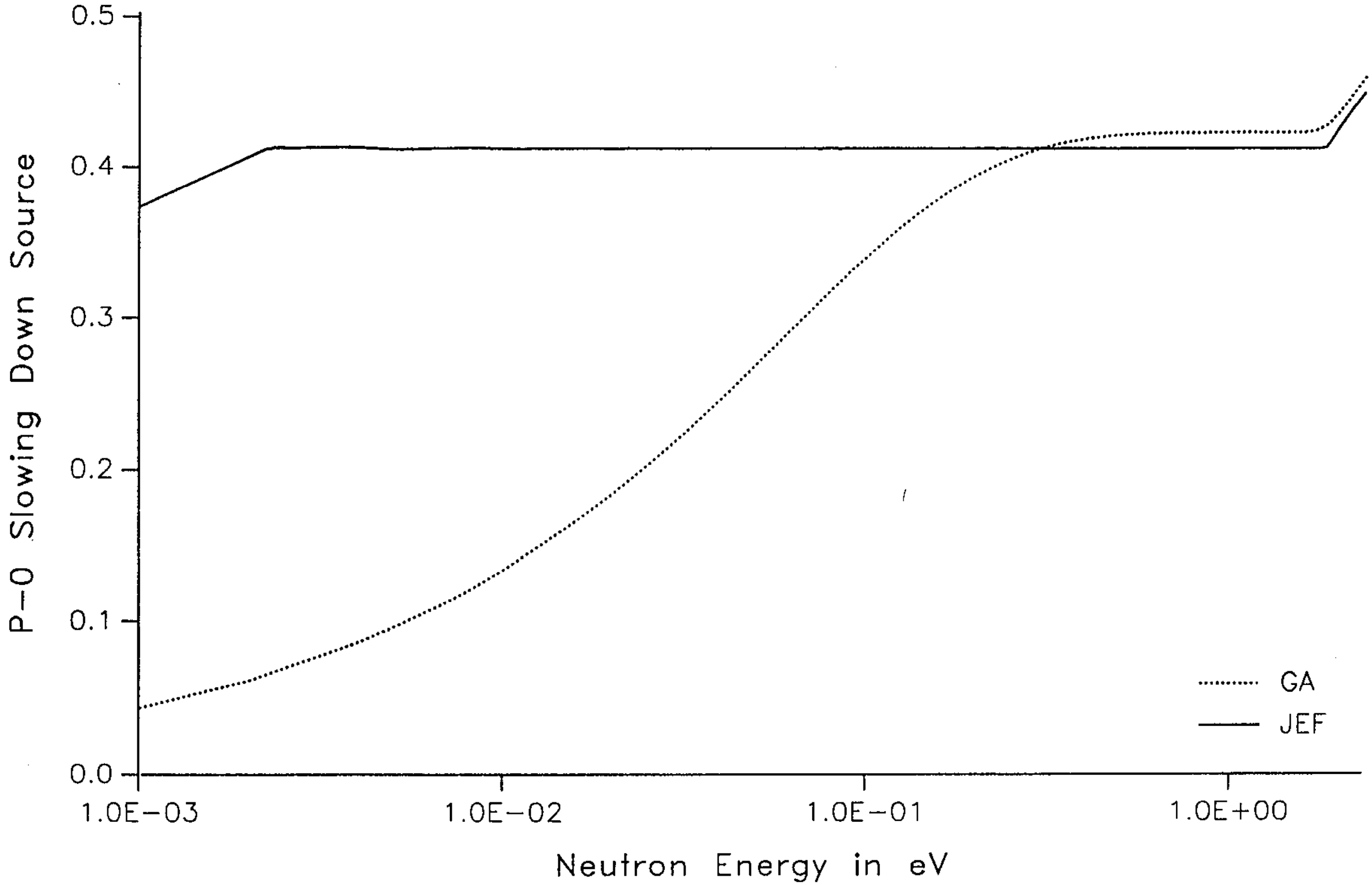
2. P. Vontobel and S. Pelloni, "JEF/EFF Based Nuclear Data Libraries," EIR-Bericht Nr. 636 (1987).

3. D. Mathews and P. Koch, "MICROX-2, An Improved Two-Region Flux Spectrum Code for the Efficient Calculation of Group Cross Sections," General Atomic Co. Report GA-A15009 (1979).

4. R. J. Archibald and D. R. Mathews, "The GAF/GAR/GAND Fast Reactor Cross Section Preparation System, Volume II, GAND2 and GFE2, Computer Programs for Preparing Input Data for the GAFGAR, GGC and MICROX Codes from an ENDF/B Format Nuclear Data File," Gulf General Atomic Co. Report GA7542 (Vol. II) (1973) plus internal GA memoranda by Archibald, Koch and Mathews in various combinations documenting the GAND3/GFE3, GAND4/GFE4 and GAND5/GFE5 codes used for the processing of the ENDF/B-III, ENDF/B-IV and ENDF/B-V data libraries, respectively.

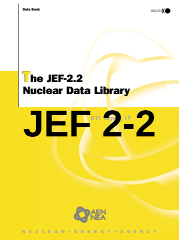
5. J. U. Koppel and D. H. Houston, "Reference Manual for ENDF Thermal Neutron Scattering Data," Gulf General Atomics Co. Report GA-8774 (Revised). Also available from the National Nuclear Data Center at Brookhaven National Laboratory as Report ENDF-269 (1978).

Thermal Section P-0 Slowing Down Sources
Room Temperature Water

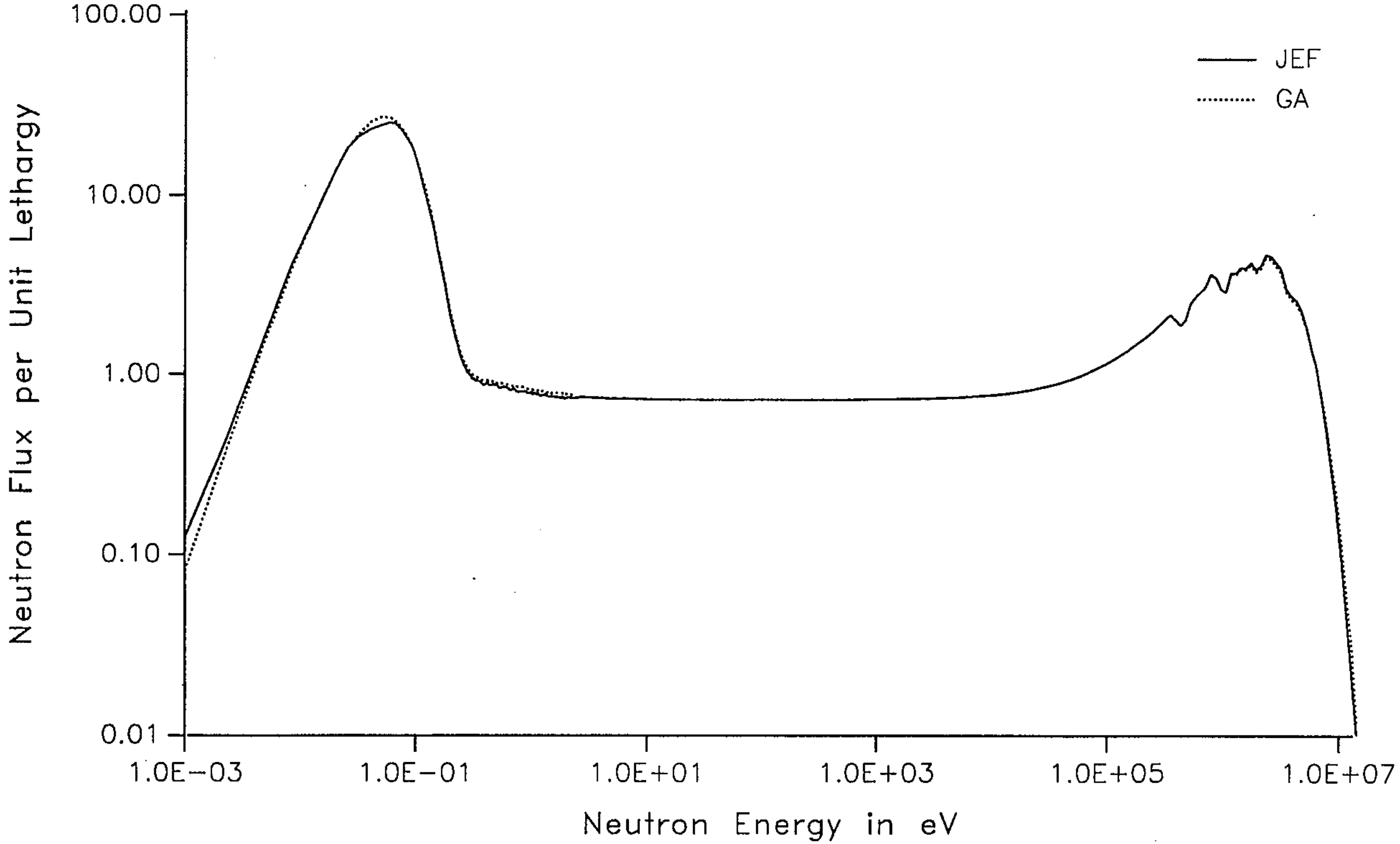


19AUG88 DRM

141105.19



Neutron Flux Versus Energy For a Unit Fission Source in Water
(room temperature) (MICROX-2 Ed. 13H)



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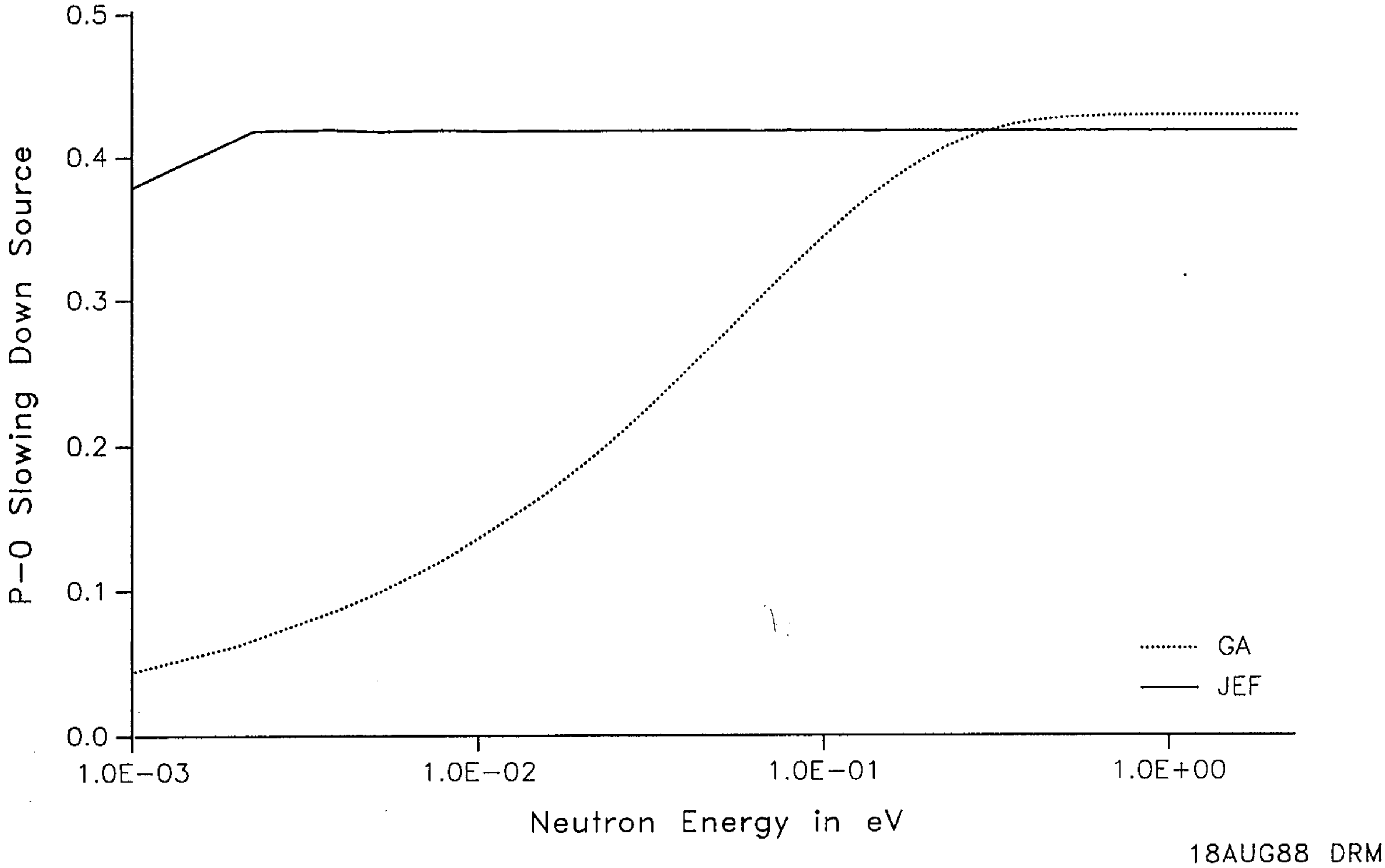
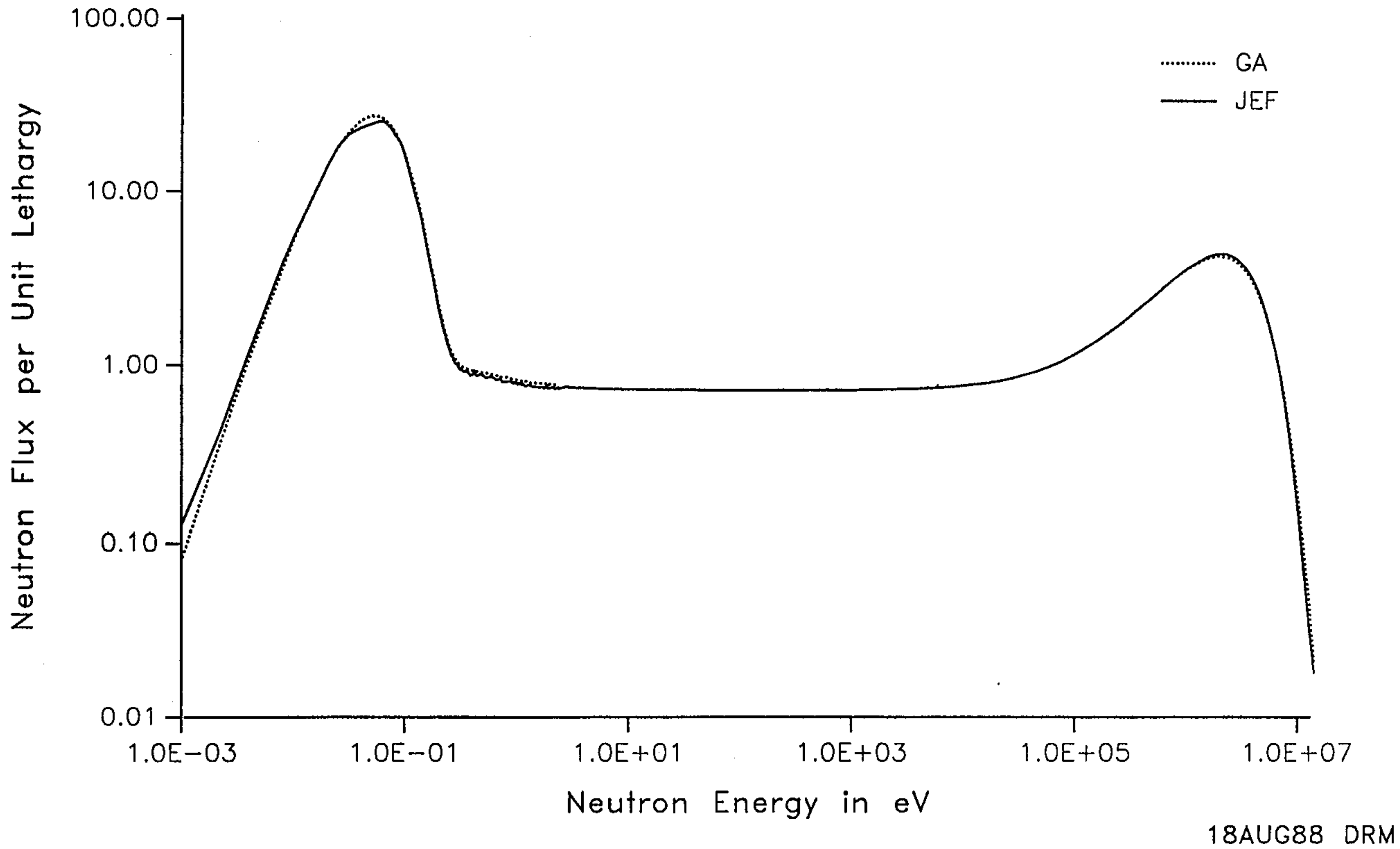
Thermal Section P-0 Slowing Down Sources
Room Temperature Hydrogen (as bound in water)




Neutron Flux Versus Energy For a Unit Fission Source in Hydrogen
(as bound in room temperature water) (MICROX-2 Ed. 13H)
100.00
10.00
1.00
0.10
0.01
Neutron Flux per Unit Lethargy
GA
JEF
1.0E-03
1.0E-01
1.0E+01
1.0E+03
1.0E+05
1.0E+07
Neutron Energy in eV
18AUG88 DRM

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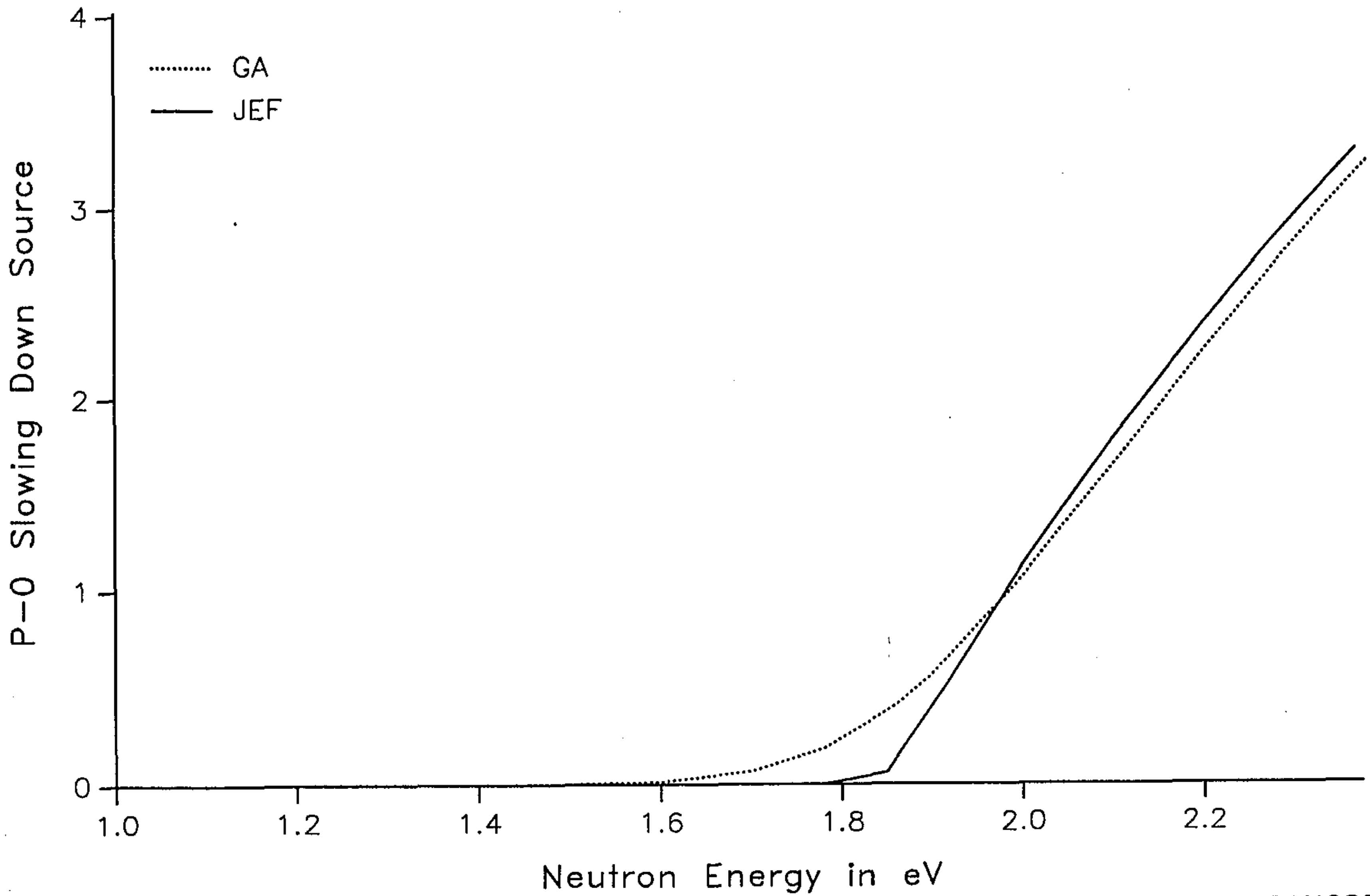
Thermal Section P–0 Slowing Down Sources
Room Temperature Oxygen



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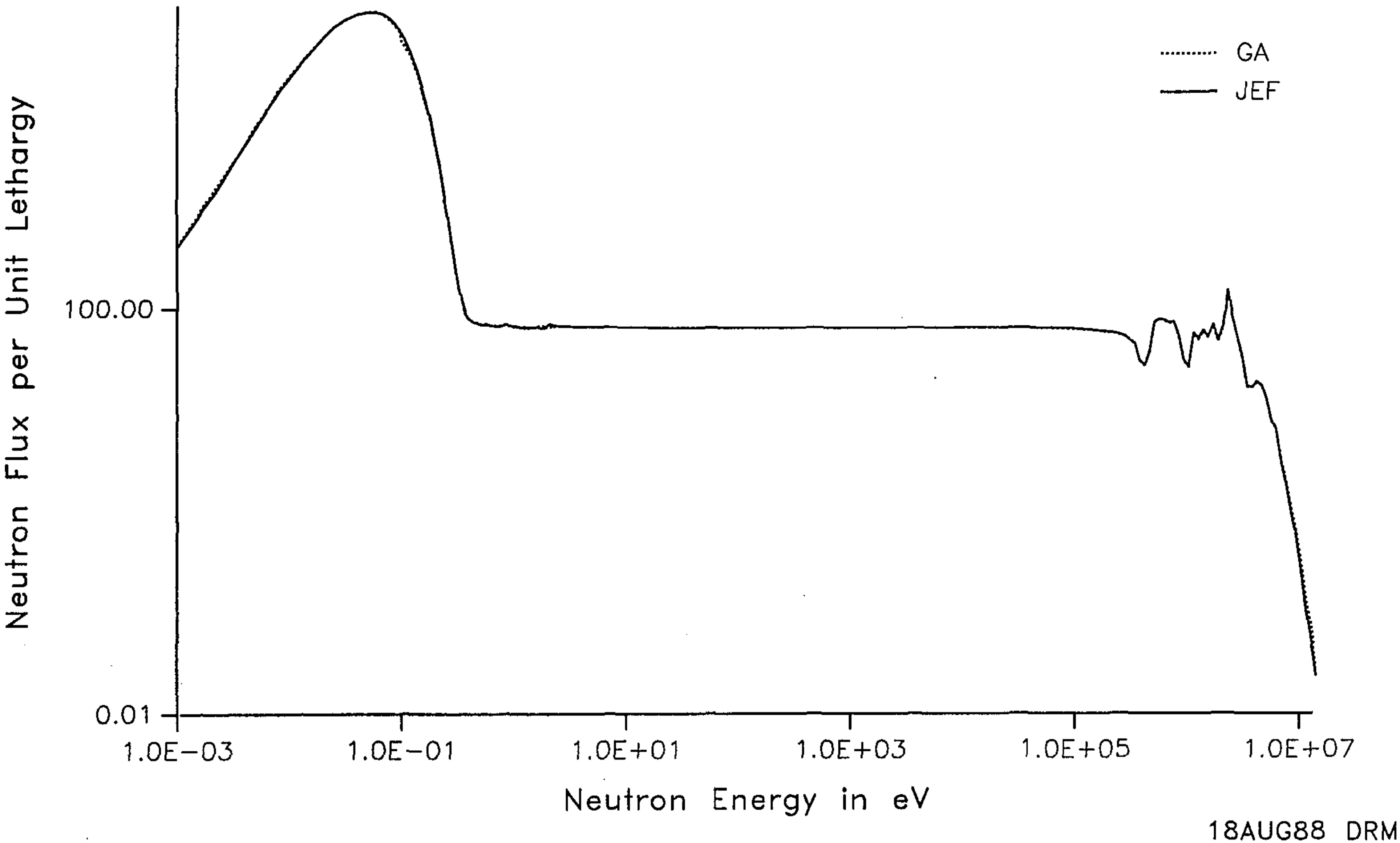
Neutron Flux Versus Energy For a Unit Fission Source in Oxygen (room temperature) (MICROX-2 Ed. 13H)



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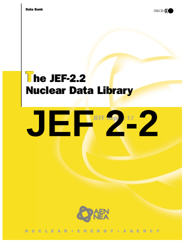
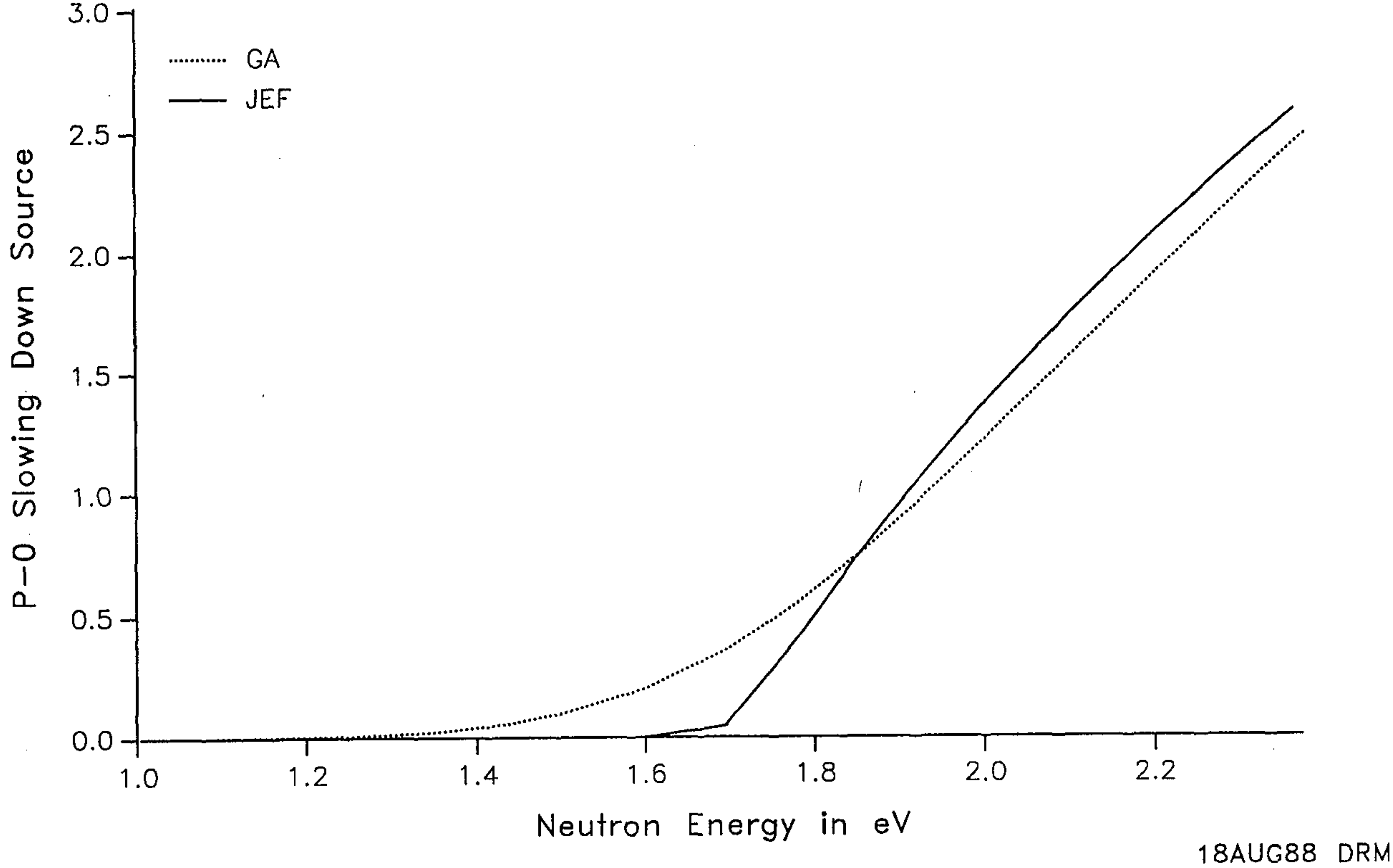


Thermal Section P-0 Slowing Down Sources
Room Temperature Graphite



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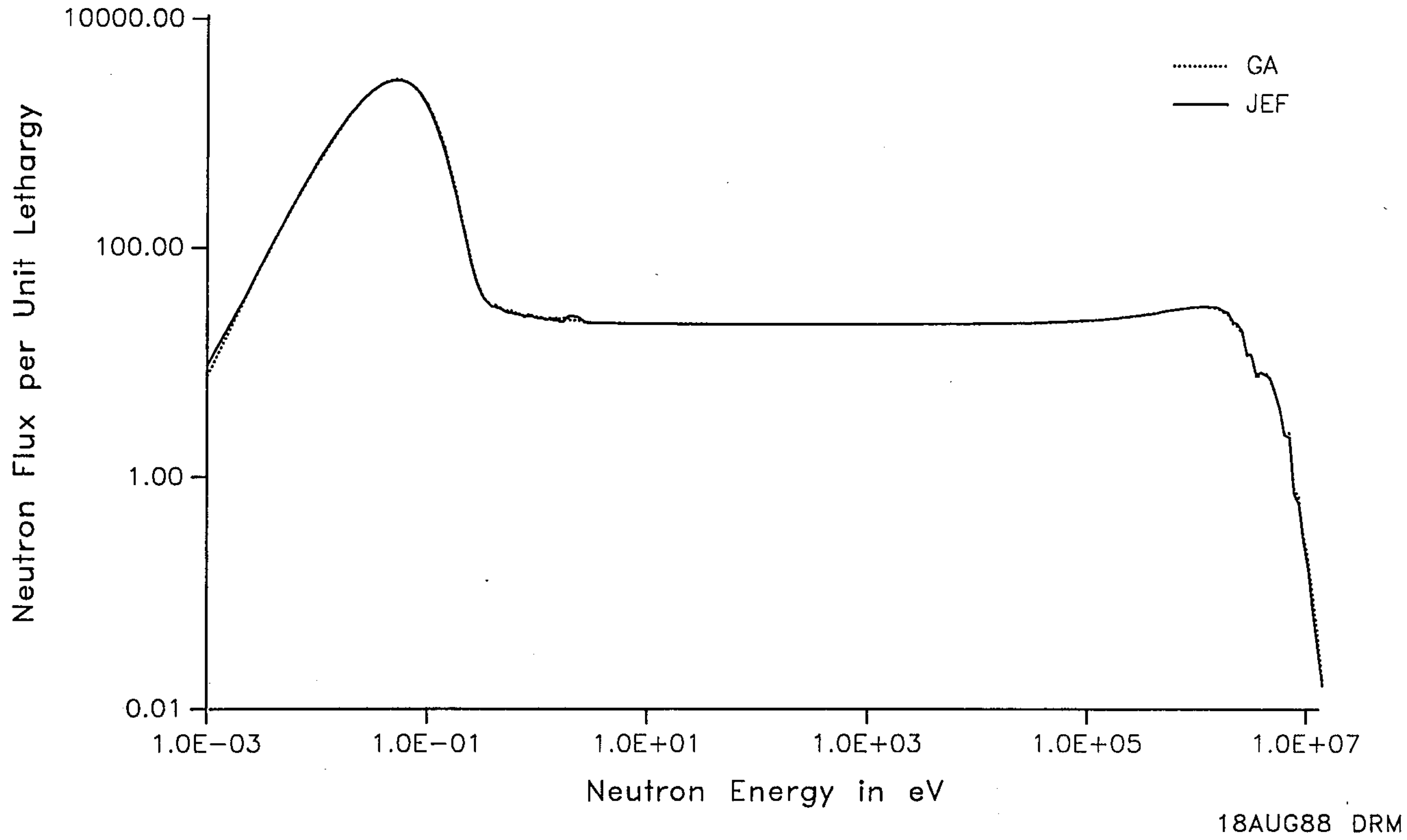
Neutron Flux Versus Energy For a Unit Fission Source in Graphite
(room temperature) (MICROX-2 Ed. 13H)



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