# Thermal Neutron Cross-Section Averaged over a Maxwellian

Maxwellian-averaged absorption cross-section of a 1/v absorber (with non-1/v factor) at a neutron temperature, from its 0.0253 eV value, with kT.

**Method:** Maxwellian averaging of 1/v cross-sections

## Input parameters

Unit system: SI

| Parameter | Value | Unit |
|---|---|---|
| Cross-section at 0.0253 eV (2,200 m/s) | 680.90 | b |
| Neutron temperature | 300.00 | °C |
| Non-1/v factor at T (1 for a 1/v absorber) | 0.97800 | |

## Calculated results

| Parameter | Value | Unit |
|---|---|---|
| Maxwellian-averaged cross-section | 422.39 | b |
| Ratio to the 2,200 m/s value | 0.62035 | |
| Neutron temperature as energy | 0.049390 | eV |
| Most probable neutron speed | 3,073.9 | m/s |

### Notes for this case

• For a 1/v absorber in a Maxwellian flux, σ̄ = (√π/2) g_a σ₀ √(T₀/T) with T₀ = 293.61 K (kT₀ = 0.0253 eV). Use tabulated g_a for non-1/v nuclides (U-235 about 0.98 at 300 °C for absorption); real reactor spectra are hardened above the moderator temperature.

## Formula and method

$$\bar{\sigma} = \frac{\sqrt{\pi}}{2} g_a(T) \sigma_0 \sqrt{\frac{T_0}{T}}, \quad v_p = \sqrt{\frac{2kT}{m_n}}$$

### Variables

| | | |
|---|---|---|
| $\sigma_0$ | Cross-section at 0.0253 eV (2,200 m/s) | m² |
| $T$ | Neutron temperature | K |
| $g_a$ | Non-1/v factor at T (1 for a 1/v absorber) | - |
| $\bar{\sigma}$ | Maxwellian-averaged cross-section | m² |
| $\bar{\sigma}/\sigma_0$ | Ratio to the 2,200 m/s value | - |
| $kT$ | Neutron temperature as energy | J |
| $v_p$ | Most probable neutron speed | m/s |

### Method and standards

Thermal cross-section averaging.

### Assumptions and limits

- Maxwellian spectrum at the neutron temperature.

### References

Maxwellian averaging of 1/v cross-sections

## Report details

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|---|---|
| Report reference | FCC-Q0A4-YVPV (the same inputs always give the same reference) |
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| Release checks | 3 reference cases of this calculator, computed independently, are checked on every release |

Results are for engineering estimation. Verify critical designs against the referenced standards and your own data.

### This calculator online

https://calc.firecad.com/boilers-power-plant/nuclear-power/thermal-neutron-cross-section-average

Open this case on calc.firecad.com →

Reopens the calculator with the inputs of this report.