Adaptive Control

Transatomic Power Corporation

October 5th, 2016

Reactivity Swing

“A well-designed core is one that has the absolute minimum reactivity swing,” [1] as surplus implies that neutrons that could have otherwise contributed to fission and conversion are lost through external absorption and leakage.

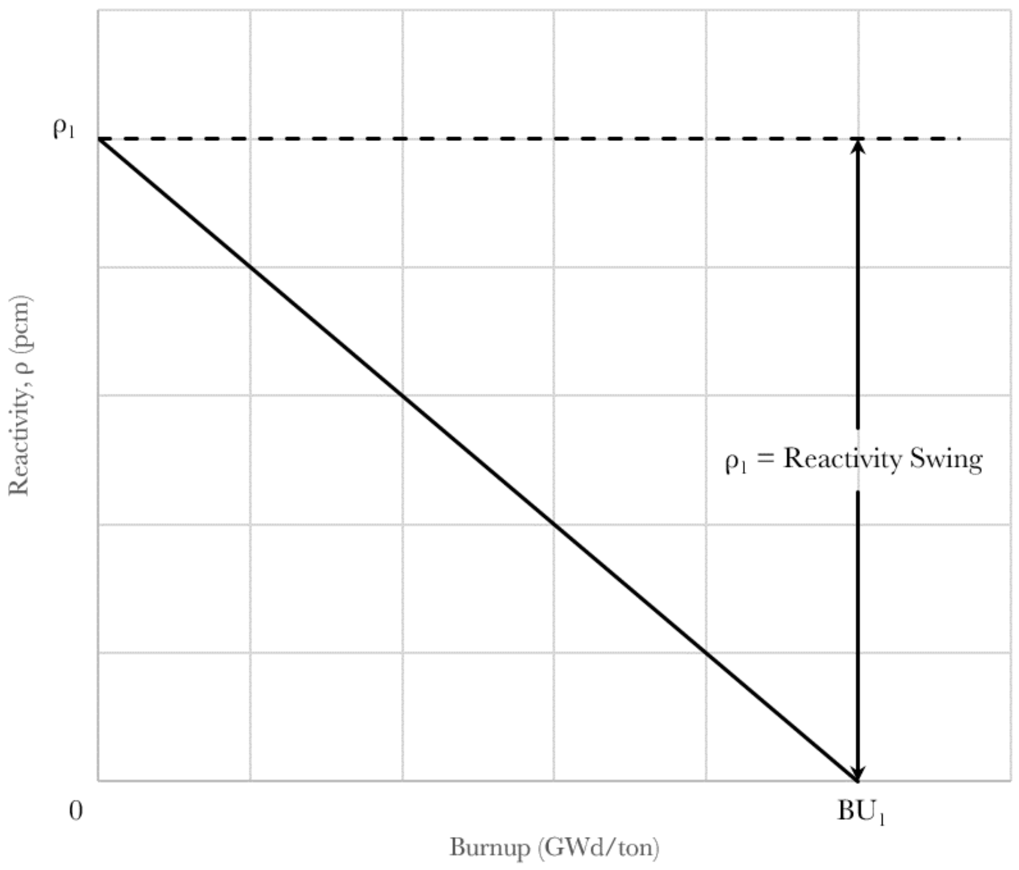

Figure 1. A graphical depiction of the concept of reactivity swing.

[1] R. G. Cochran and N. Tsoulfanidis, "In-Core Fuel Management," in *The Nuclear Fuel Cycle: Analysis and Management*, La Grange Park, Illinois, American Nuclear Society, 1999, pp. 165-205.

The Effect of Salt Volume Fraction

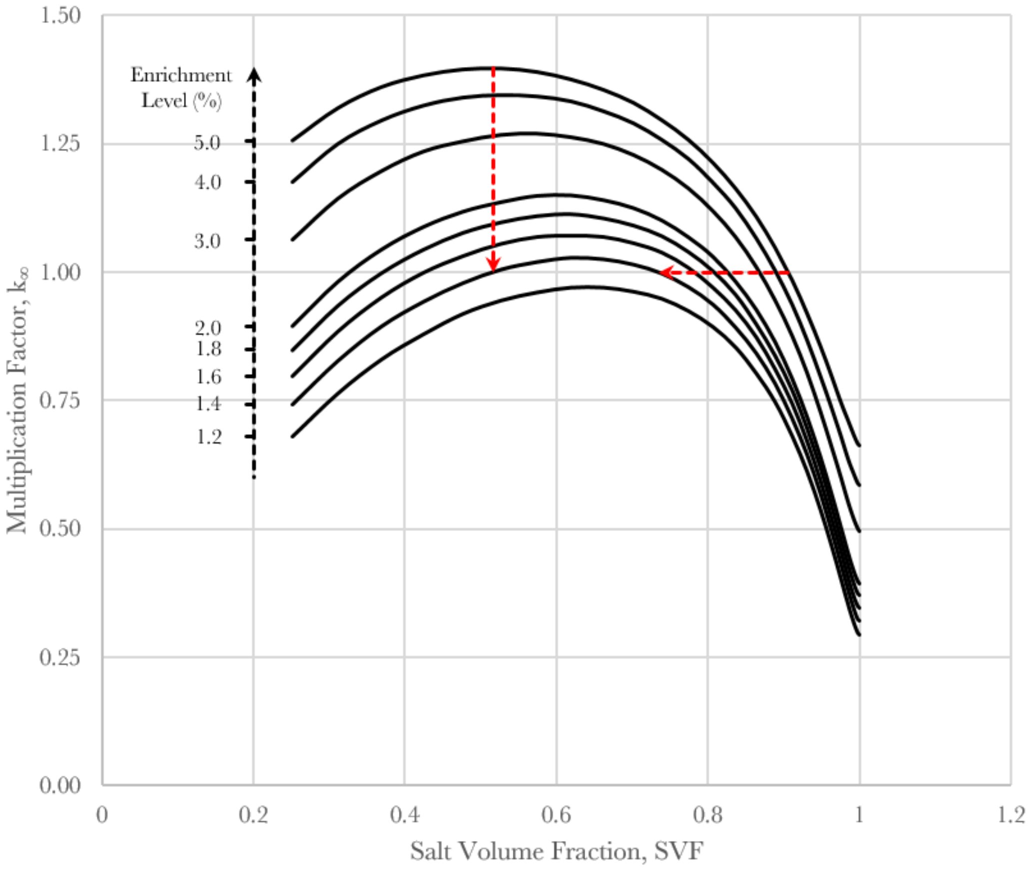

Figure 2. The Infinite multiplication factor, k_∞ as a function of salt volume fraction and ^{235}U enrichment for a representative TAP pin cell.

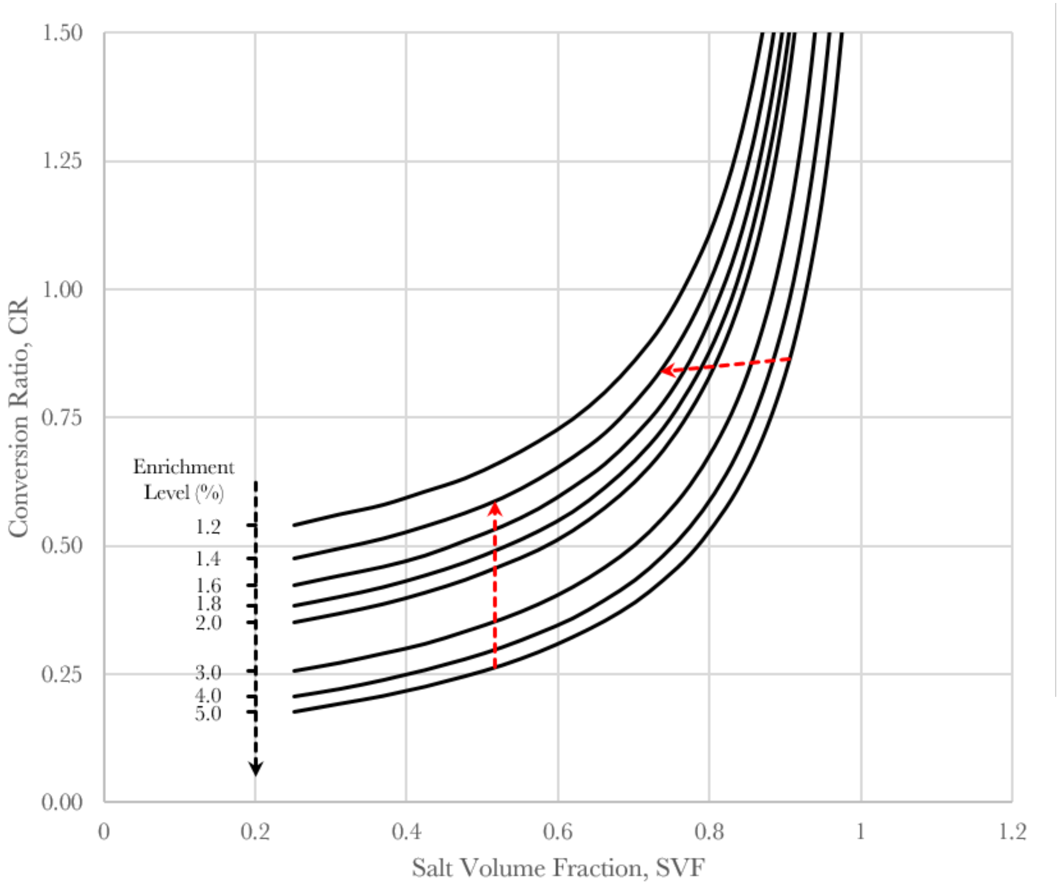

Figure 3. Conversion ratio as a function of salt volume fraction and ^{235}U enrichment for a representative TAP pin cell.

A New Take on Control Rods

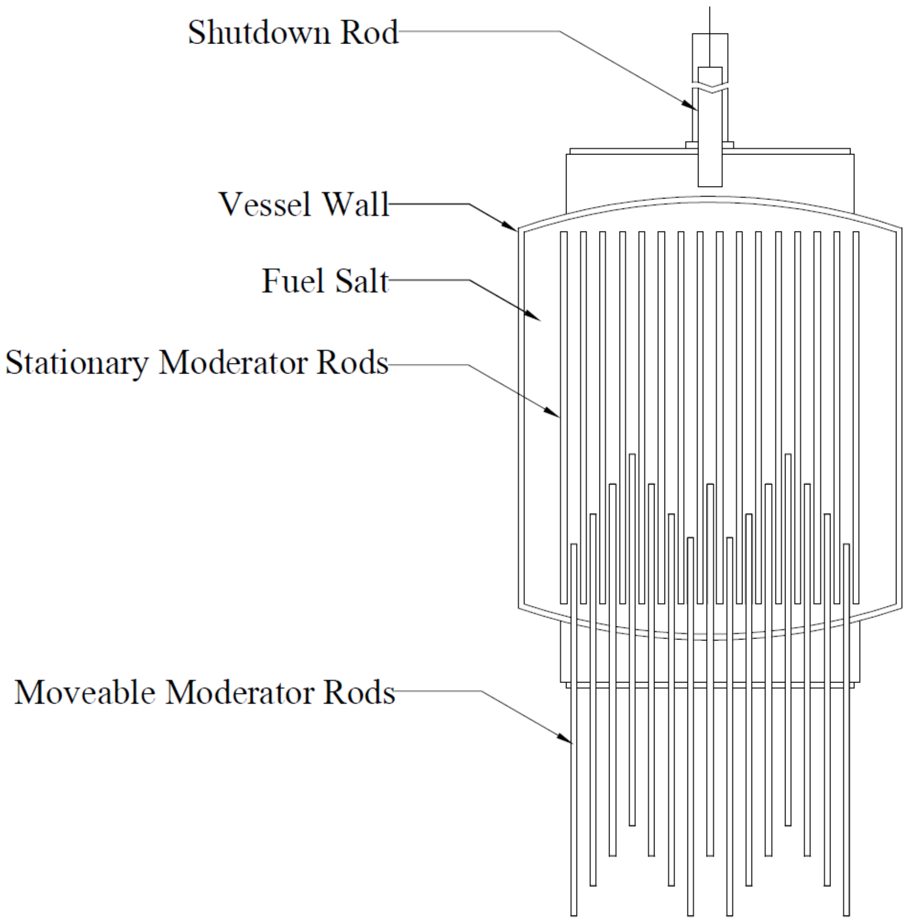

Figure 4. A conceptual depiction of a reactor vessel design that uses moveable or additional moderator rods for reactivity control.

Adaptive Reactivity Control

- Short Term
  - Moveable moderator rods
    - Similar to control rods in an LWR
- Long Term
  - Insert more moderated assemblies
    - Similar time interval to refueling and maintenance in an LWR

Simulating Operation

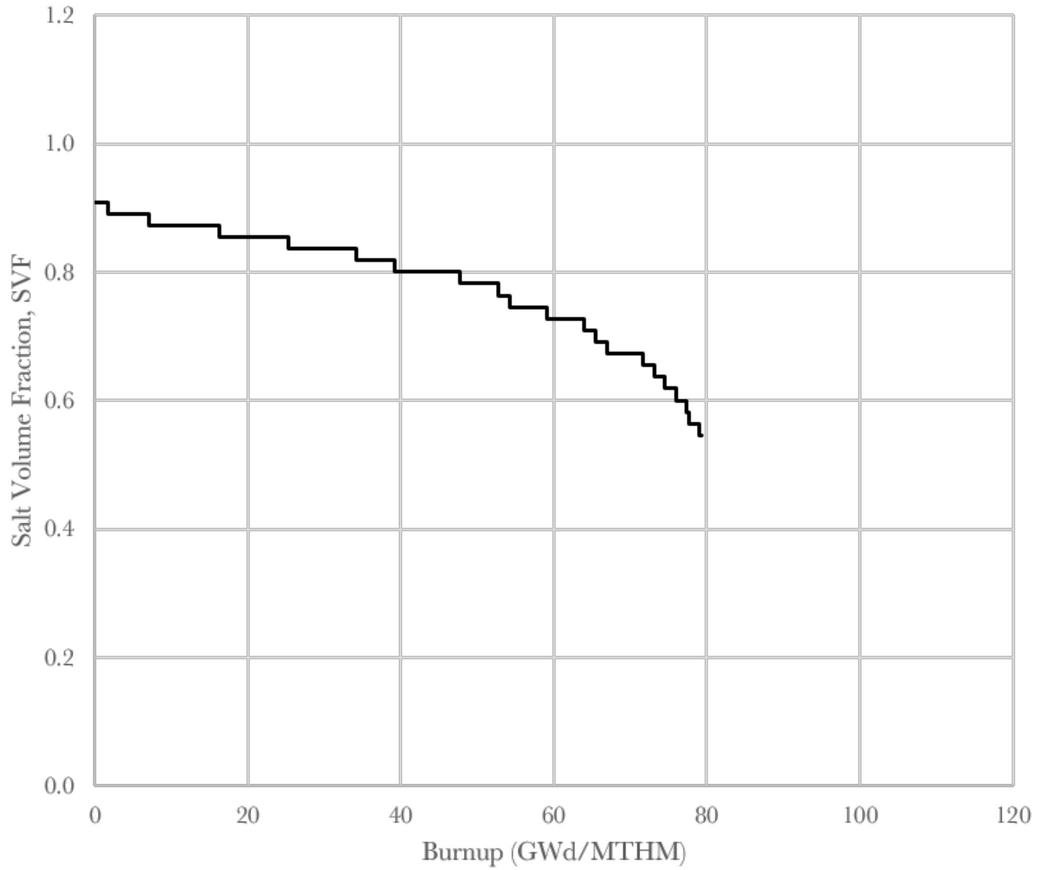

Figure 5. A visualization of the change in salt volume fraction as a function of burnup for the representative TAP operational scheme simulated in SERPENT 2.

Overview

- Software
  - Serpent 2
- Fuel Cycle
  - 5% Enriched ^{235}U Initial load & feed
- Modeling Considerations
  - Performed at the "assembly" level (5 by 5 rod array)
  - Fission product removal & fuel addition
  - Rod insertion

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Results: Spectrum

Beginning of life (BOL)

- A high initial fissile load (5%) can achieve criticality on the hardened spectrum

End of life (EOL)

- More thermal neutrons allow for low levels of enrichment to remain critical

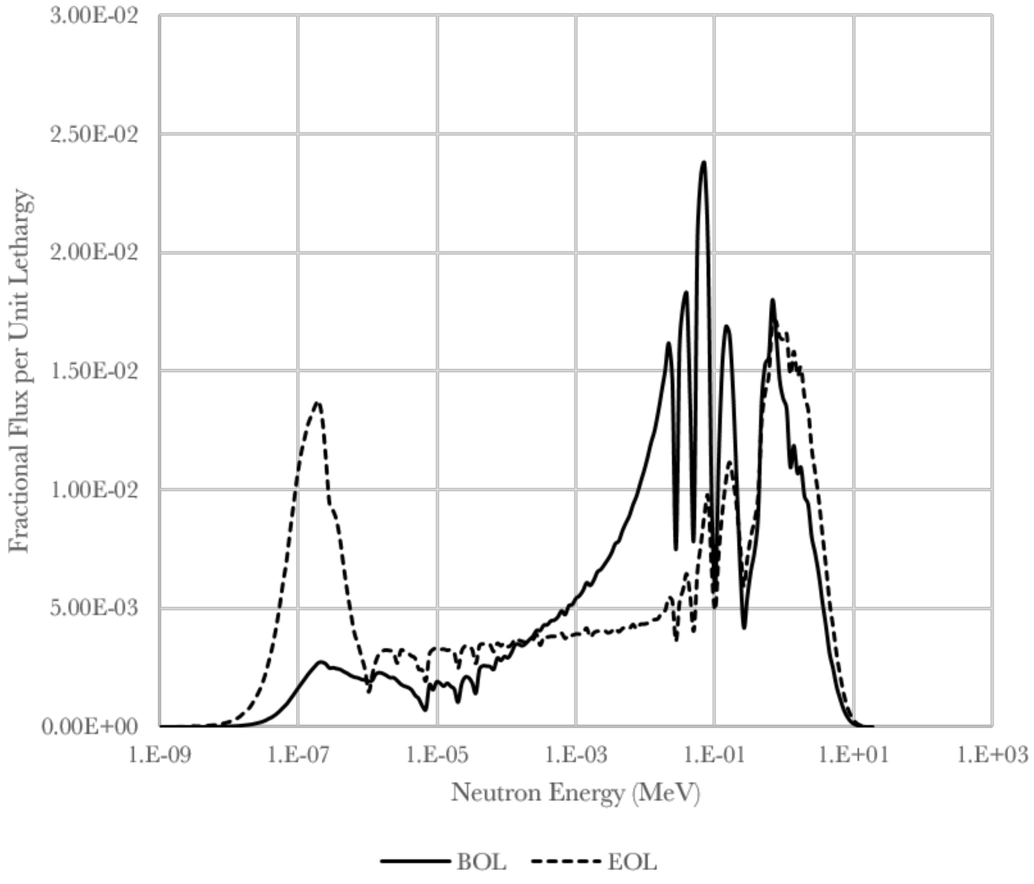

Figure 6. The development of the neutron spectrum with increasing burnup, with BOL and EOL signifying the spectrums at the beginning and end of life respectively

Results: Conversion Ratio

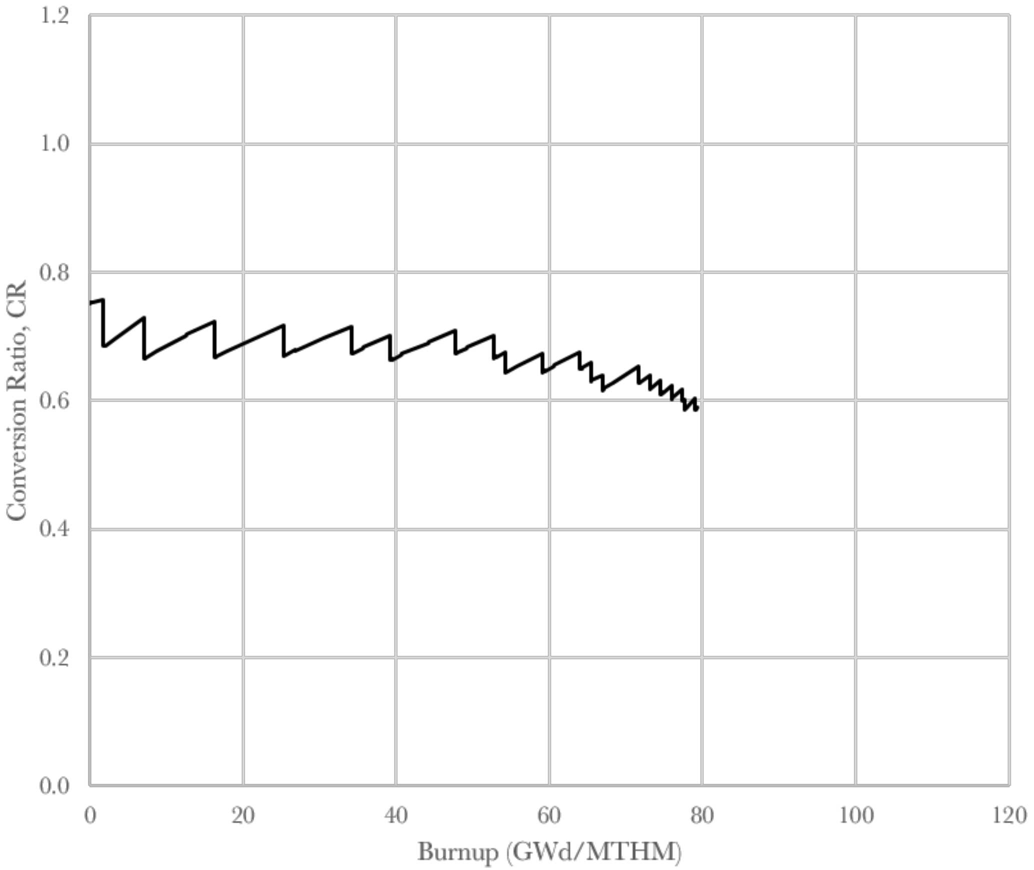

Figure 7. Conversion ratio as a function of a burnup for the representative TAP operational scheme simulated in SERPENT 2.

Beginning of Life (BOL)

- Hardened spectrum allows for significant conversion of fertile material

End of Life (EOL)

- Softening of the spectrum keeps the system critical, but plays a detrimental role in the progression of conversion ratio.

Results: Criticality

- Bulk rod insertion
  - A rod addition in the model represents rod insertion in every assembly in the core
  - Not modeling short term control
- Leakage limit reduction coincides with the spectrum evolution

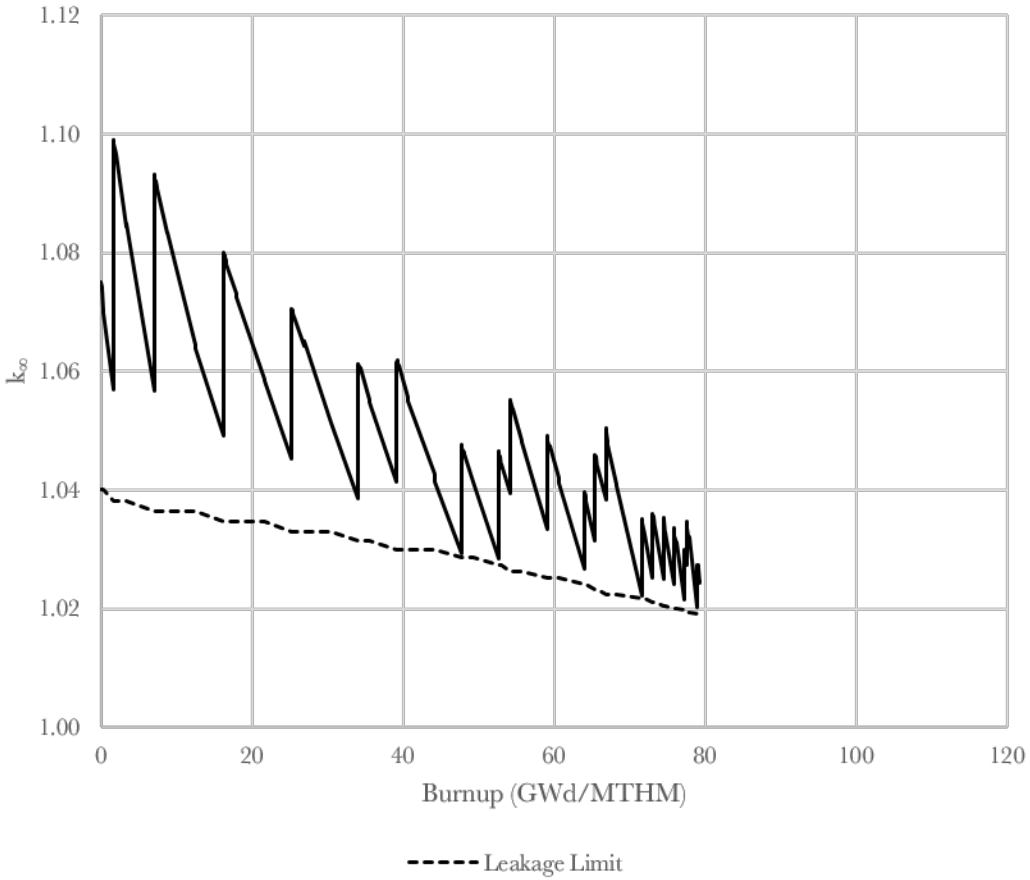

Figure 8. The infinite multiplication factor as a function of burnup for the representative TAP operational scheme simulated in SERPENT 2.

Waste vs Burnup

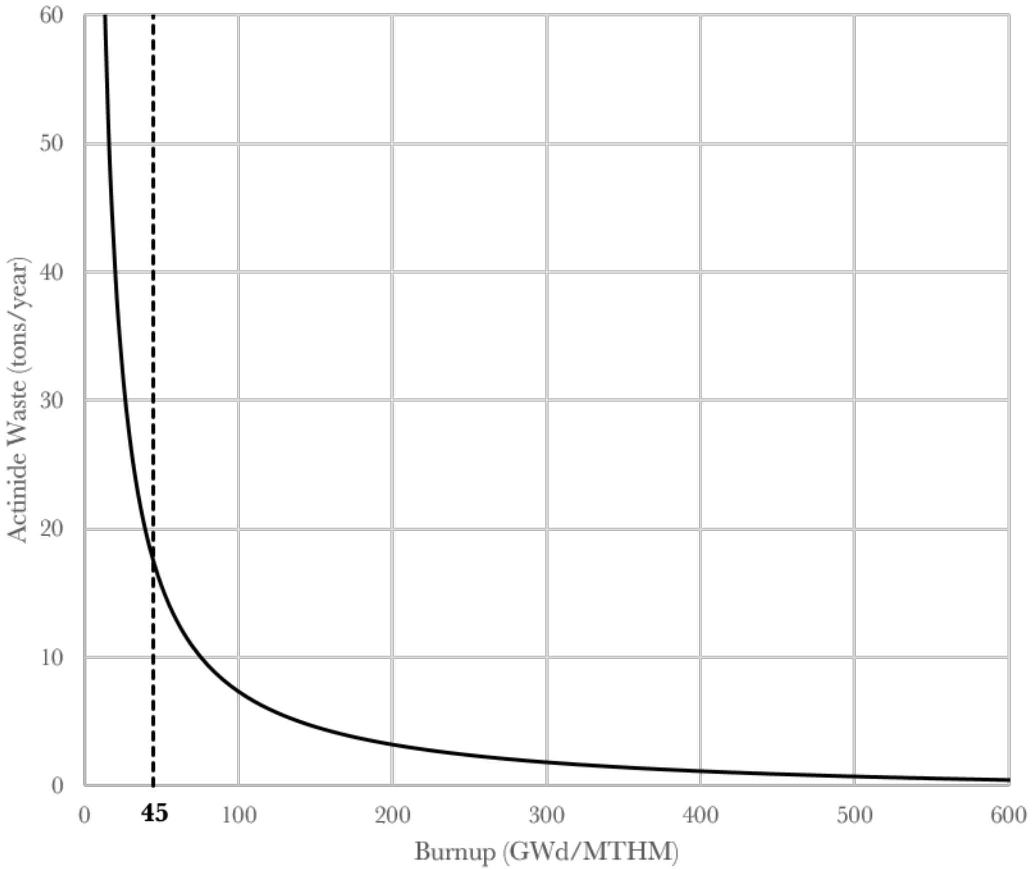

Figure 9. Actinide waste production rate as a function of burnup. Please note the rates are normalized to a thermal power level of 2.27 GWth.

- Fundamental Equations*
  - $\dot{W}_P = \frac{m_F}{CL}$
  - $Bu = \frac{P \cdot CL}{m_T}$
  - $m_C = \frac{P \cdot CL \cdot \bar{M}}{\bar{E}_F \cdot N_A}$
- Derived Formulation*
  - $\dot{W}_P = \frac{P}{Bu} - \frac{P \cdot \bar{M}}{\bar{E}_F \cdot N_A}$

* $\dot{W}_P$ = Waste production rate, m_F = Remaining mass at the end of cycle, m_T = Total actinide mass used over the course of cycle, m_C = mass consumed over the course of cycle, $\bar{M}$ = Average molar mass of fissioning nuclei, $\bar{E}_F$ = Average energy per fission, N_A = Avogadro's number

Summary

- Novel method of adaptive reactivity control
  - Moveable moderator rods
- Increased performance even without modeling short term control
  - Conventional 5% fuel cycle
  - Burnup > 80 GWd/MTHM
  - 50% Waste reduction compared to current LWR’s

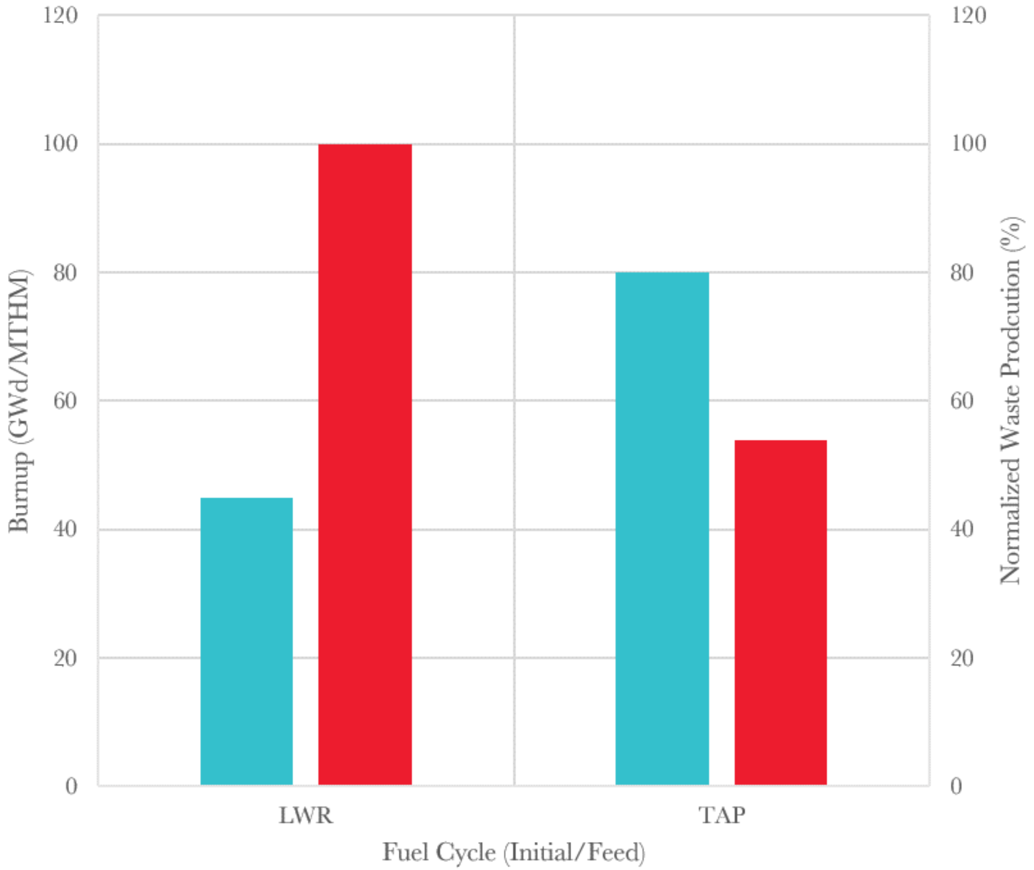

Figure 10. A comparison of the burnup (blue) and waste production (red) of an LWR and the TAP MSR, operating on the conventional 5% fuel cycle.

Questions?

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Additional: Parameters

- Neutron Population
  - 300 active cycles
  - 100 inactive cycles
  - 10000 neutrons per cycle
- Cross Section Data
  - ENDF-VII.1, 900 K
- Depletion Time Step*
  - $\Delta t = \Delta t_I \cdot 5^{n-1}$
  - $\Delta t_I = 0.1\ days$
  - $\Delta t_{Max} = 182.5\ days$
- Boundary Conditions
  - Reflective
- Conversion Ratio*
  - $CR = \frac{\left({}^{238}U + {}^{240}Pu\right)(n,\gamma)}{\left({}^{235}U + {}^{239}Pu + {}^{241}Pu\right)(n,\gamma + n,f)}$

* Δt = Time Step, Δt_I = Initial Time Step, Δt_{Max} = Maximum Time Step Size n = Step Number, (n, γ) = Radiative Capture, (n, f) = Fission

Additional: Isotopic Evolution

- Decreased conversion over the course of life cause the slope of the total fissile evolution to increase with time.
- Significant reduction in ^{238}U capture at the EOL compared to BOL leads to the sharp drop in ^{239}Pu.

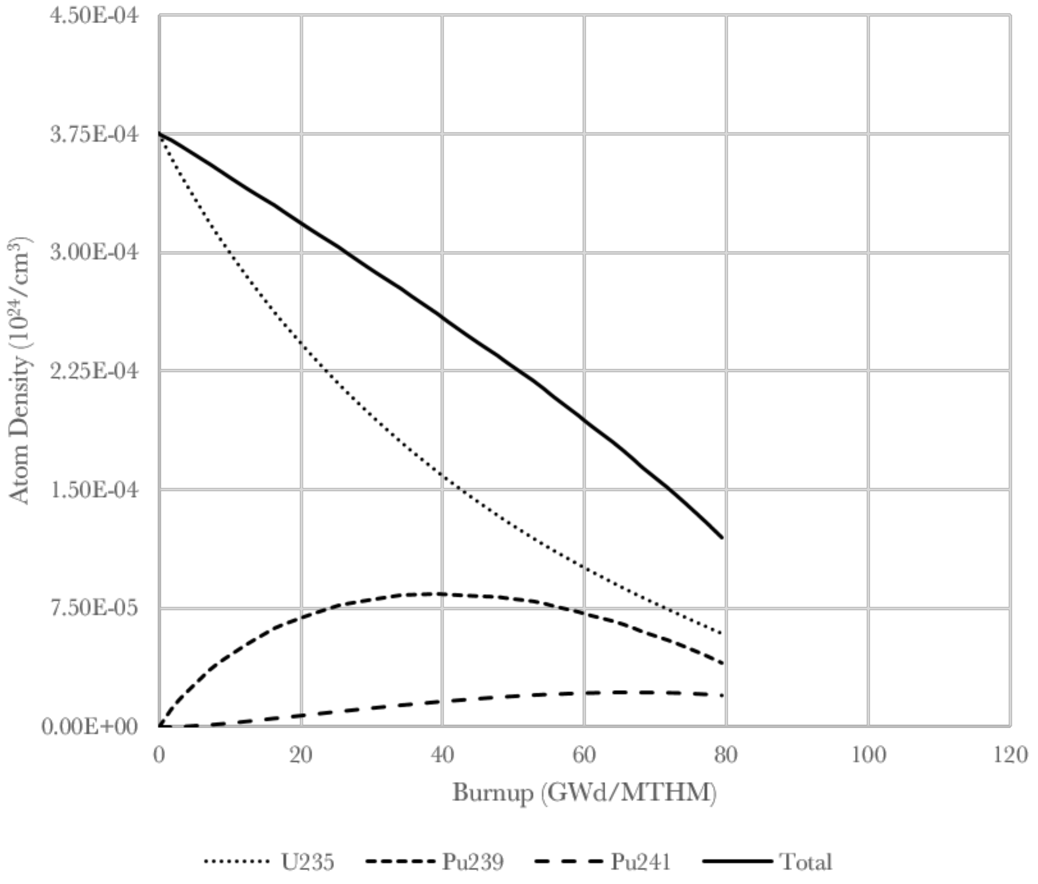

Figure 11. Illustrating the effect of conversion ratio, the above data shows the development of the primary fissile isotopes with increasing burnup.

Additional: Convergence

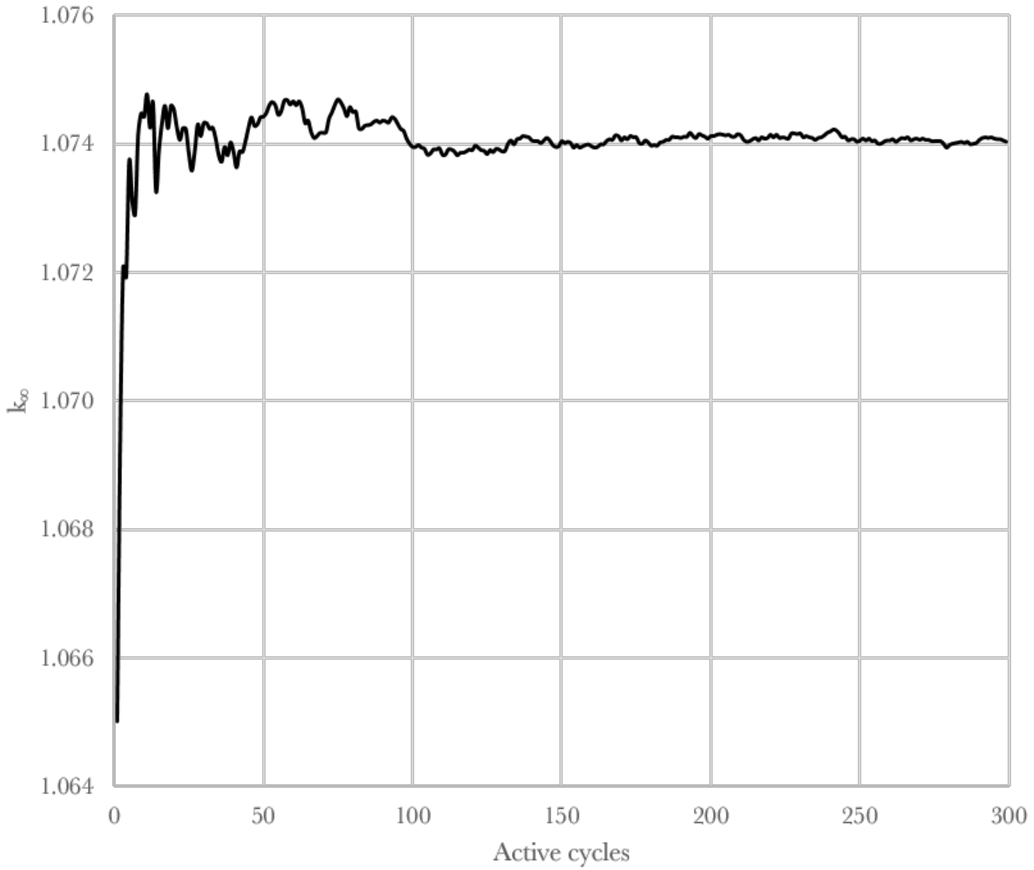

Figure 12. k_∞ as a function of the number of active cycles simulated

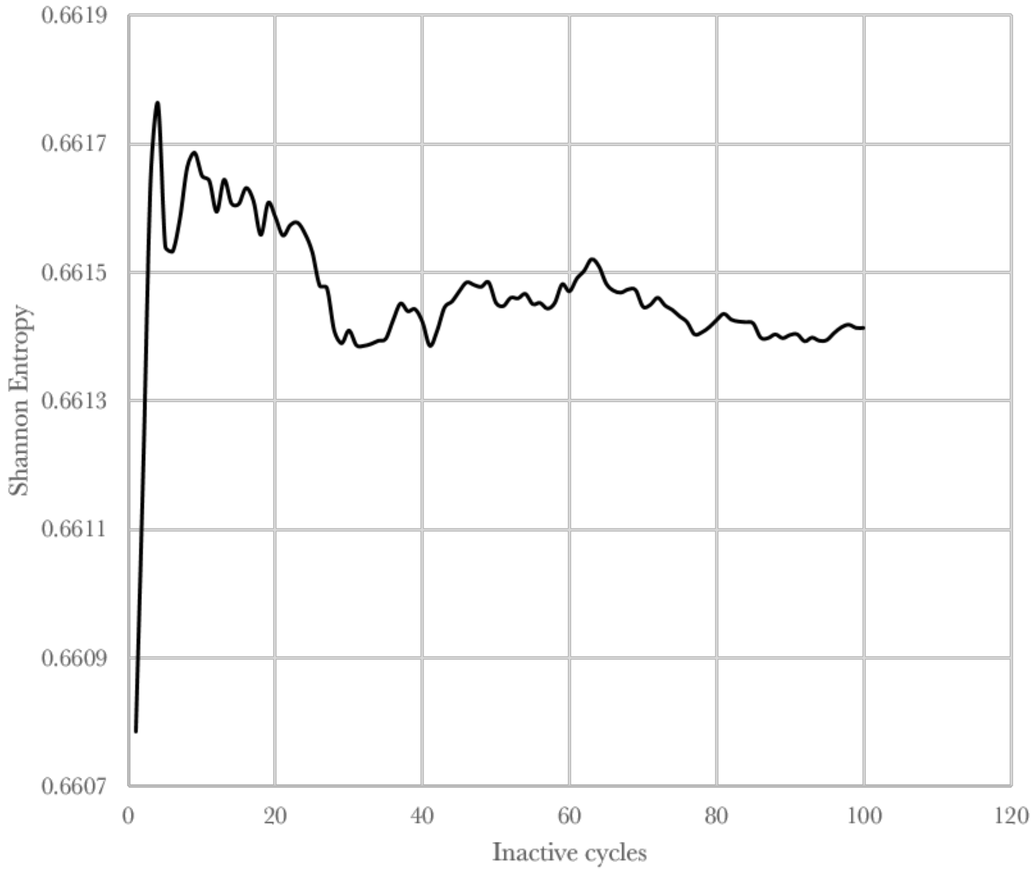

Figure 13. Shannon entropy as a function of the number of inactive cycles simulated

Additional: Literature

Table 1.5 Effective energy released per thermal fission for the three primary fissile isotopes

Isotope	Thermal Fission Energy Release (MeV)
^{235}U	192.9
^{239}Pu	198.5
^{241}Pu	200.3

Table 1.6 Average neutrons released per thermal fission for the three primary fissile isotopes

Isotope	Neutrons Released per Thermal Fission
^{235}U	2.42
^{239}Pu	2.87
^{241}Pu	2.93

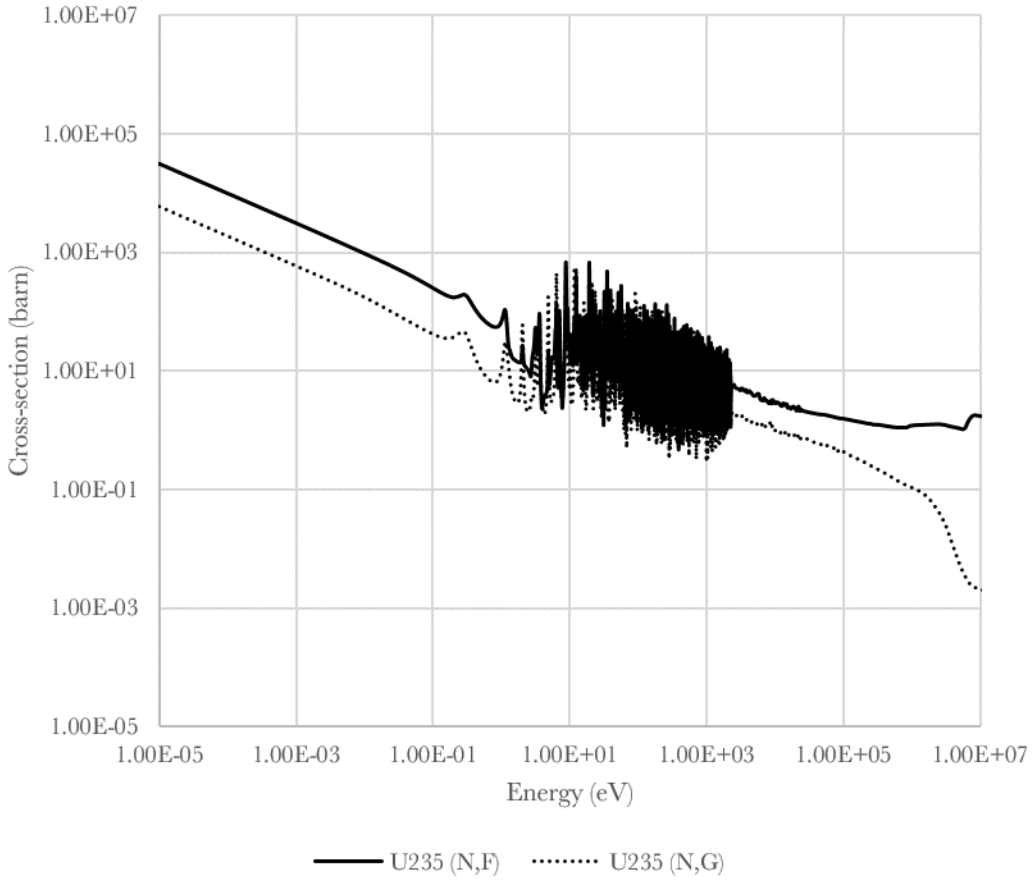

Figure 14. Fission and capture cross-sections for ^{235}U, data taken from ENDF-VII.1

Additional: Literature

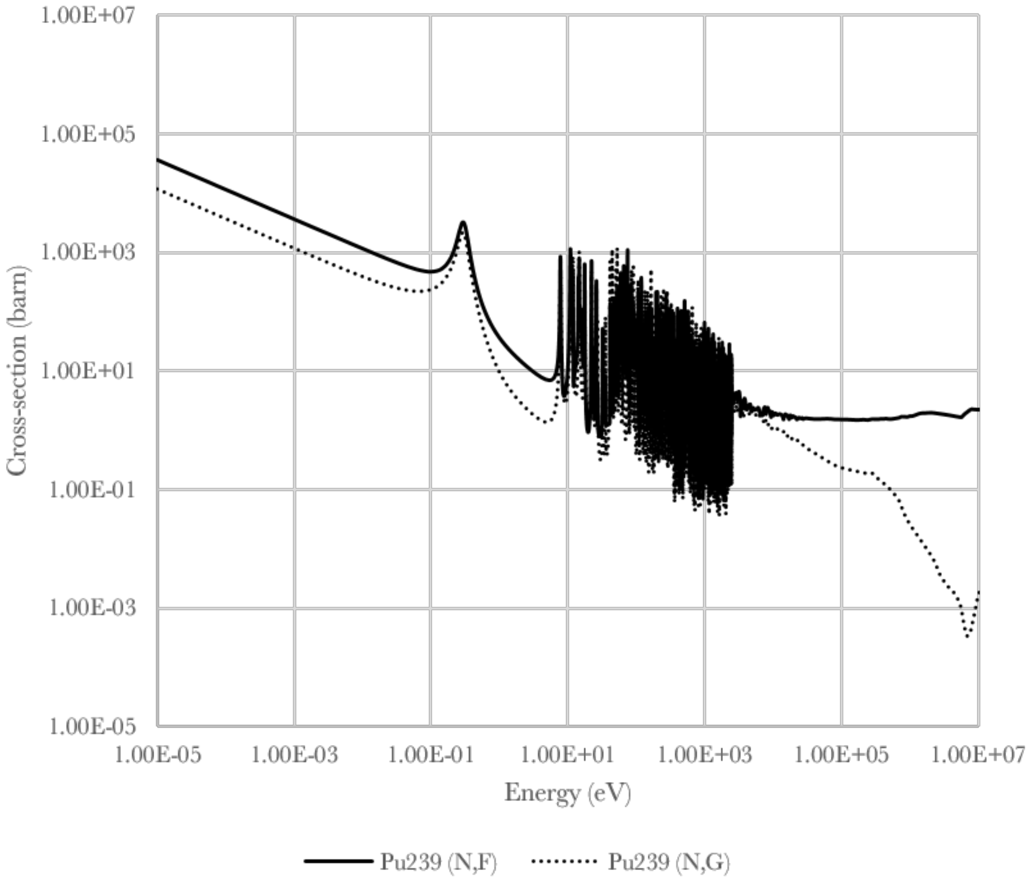

Figure 15. Fission and capture cross-sections for ^{239}Pu, data taken from ENDF-VII.1

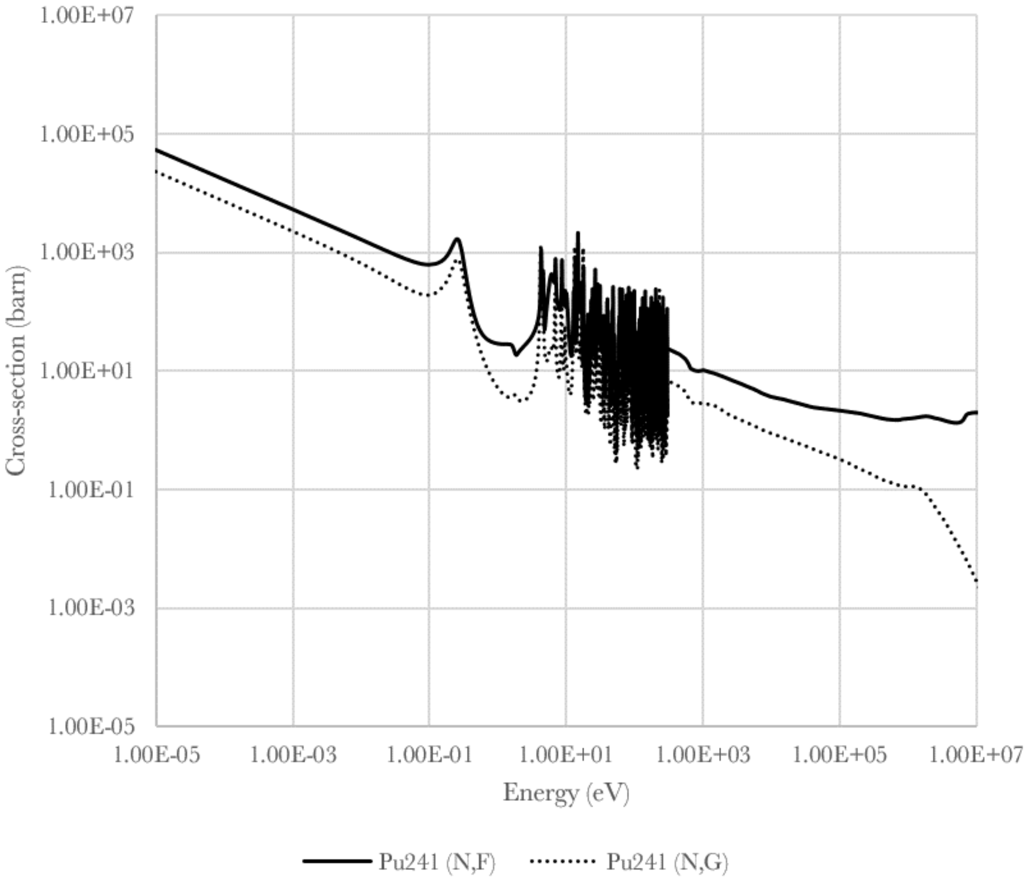

Figure 16. Fission and capture cross-sections for ^{241}Pu, data taken from ENDF-VII.1