**Wims-based computational analysis of burnup-dependent macroscopic thermal-neutron absorption and fission cross-sections in irt-4m fuel (19.75% u-235) and fission-product poisoning, pU-239 buildup and reactivity implications**

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**Abstract:** This study conducts a detailed computational analysis of burnup-dependent macroscopic thermal-neutron cross-sections in IRT-4M fuel assemblies (19.75% U-235 enrichment) used in WWR-SM research reactors, employing the WIMS code with a 69-group neutron energy structure based on the ENDF/B-VI nuclear data library. The investigation quantifies the evolution of absorption (Σa) and fission (Σfν) cross-sections from 0% to 60% U-235 burnup. Results show a non-monotonic Σa trend, peaking at 10% burnup due to fission product accumulation, particularly Xe-135 and Sm-149, followed by a decline driven by U-235 depletion and poison saturation. In contrast, Σfν decreases linearly, reflecting U-235 depletion with minimal compensation from Pu-239 production. The study elucidates key physical mechanisms, including neutron poisoning and transuranic element formation, emphasizing their impact on reactor operation and safety. These findings highlight the critical need for burnup-dependent cross-section libraries to ensure accurate core simulations, providing essential insights for fuel management and reactivity control in research reactors. Future work should focus on experimental validation and extending the analysis to higher burnup levels to optimize fuel cycle strategies.

**INTRODUCTION**

Nuclear research reactors play a pivotal role in scientific advancement, providing neutron sources for various applications including material testing, isotope production, and fundamental physics research. The accurate prediction of neutron behavior in these reactors throughout their operational cycle remains essential for ensuring safety, optimizing performance, and planning fuel management strategies. Central to this predictive capability is the understanding of how macroscopic neutron cross-sections evolve as fissile isotopes deplete and fission products accumulate during reactor operation.

The IRT-4M fuel assembly, utilized in WWR-SM type research reactors, represents a sophisticated nuclear fuel design featuring uranium oxide dispersed in an aluminum matrix with 19.75% U-235 enrichment. This enrichment level, deliberately selected to remain below the 20% threshold for highly enriched uranium, presents unique neutronic characteristics that require detailed investigation. As nuclear fuel undergoes irradiation, its composition transforms significantly through complex chains of nuclear reactions including fission, neutron capture, and radioactive decay. These transformations substantially alter the fuel's neutronic properties, particularly its thermal-neutron absorption and fission characteristics [1].

Computational modeling of burnup processes represents a challenging multiphysics problem that requires coupled solution of neutron transport equations and nuclide inventory equations. The WIMS (Winfrith Improved Multigroup Scheme) code has emerged as one of the most widely adopted computational tools for reactor physics calculations, particularly renowned for its capability to perform detailed cell calculations and generate burnup–dependent cross-section libraries. Employing a 69-group neutron energy structure with 14 fast groups and 19 resonant groups, WIMS provides the resolution necessary to accurately capture self-shielding effects and spectral transitions that occur during fuel depletion.

Previous studies have examined various aspects of fuel burnup effects in different reactor systems. Research on pressurized water reactors has demonstrated strong correlations between isotopic composition and burnup levels, while investigations of molten salt reactors have highlighted the sensitivity of neutronic parameters to nuclear data libraries. However, limited attention has been devoted to systematic analysis of research reactor fuels, particularly those with intermediate enrichment levels like the IRT-4M type. This research gap is noteworthy given the widespread deployment of such reactors across numerous scientific institutions worldwide [1].

This paper presents a comprehensive investigation of burnup effects on macroscopic thermal-neutron cross-sections in IRT-4M fuel assemblies through advanced computational modeling using the WIMS code. The study aims to: quantify the evolution of Σa and Σfν across burnup levels from 0% to 60% of initial U-235 atoms fissioned; identify the underlying physical mechanisms driving observed trends; and provide practical insights for reactor operators and safety analysts regarding fuel management and core behavior prediction.

**THEORETICAL BACKGROUND**

Nuclear Reactions During Fuel Burnup.Nuclear reactor fuel undergoes complex transformation during irradiation, primarily through neutron-induced reactions including fission, radiative capture, and various transmutation processes. The fission process itself, wherein a heavy nucleus splits into two or more lighter fragments following neutron absorption, represents the primary source of energy production in nuclear reactors. For thermal neutrons, fissile isotopes such as U-235 exhibit particularly high fission cross-sections, approximately 585 barns at 0.0253 eV. This process yields not only energy but also additional neutrons (typically 2-3 per fission event) that sustain the chain reaction, and a spectrum of fission products with varying nuclear properties.

The macroscopic cross-section (Σ) represents the probability of a particular neutron-nucleus interaction per unit path length through the material and is calculated as the sum of the microscopic cross-sections (*σi*) of all constituent nuclides multiplied by their respective number densities (*Ni*) [1]:

(1)

For thermal neutrons, two macroscopic cross-sections are of paramount importance: the absorption cross-section (Σa) and the fission cross-section multiplied by the average number of neutrons per fission (Σfν). The latter parameter directly influences reactor reactivity and the neutron economy, as it determines the neutron production rate per unit flux.

As burnup progresses, the fuel composition evolves through several simultaneous processes: depletion of fissile isotopes (primarily U-235), accumulation of fission products, and formation of transuranic elements through neutron capture and subsequent decay chains. Particularly important is the production of Pu-239 via the capture-transmutation pathway in U-238:

238U(n,γ)→ 239U→β−→ 239Np→β−→ 239Pu (2)

Although Pu-239 is itself fissile and contributes to fission reactions, its production rate in research reactor fuels is insufficient to fully compensate for U-235 depletion due to the relatively low neutron flux and limited U-238 content in 19.75% enriched fuel [2].

Fission Product Poisoning.The accumulation of fission products significantly influences the neutronic characteristics of irradiated fuel. Certain fission fragments exhibit exceptionally high thermal neutron absorption cross-sections, earning them the designation neutron poisons. The most notable among these are Xe-135 ( ≈ 2.6×106 barns) and Sm-149 ( ≈ 4.0×104 barns), which form through radioactive decay chains following fission.

Xe-135 deserves particular attention due to its extraordinary absorption cross-section and complex formation mechanism. It originates primarily from the decay of tellurium-135 and iodine-135 fission products:

135Te→β−→135I→β−→135Xe→β−→135Cs (3)

The production and removal rates of these neutron poisons depend on multiple factors including fission rate, neutron flux, and decay constants, creating complex time-dependent behavior that significantly impacts reactor operation and control.

Computational Methods for Burnup Analysis.The analysis of fuel burnup effects requires sophisticated computational tools capable of simulating both the neutron transport and the time evolution of nuclide inventories. The WIMS code employs a multi-group approach to neutron transport calculations, solving the Boltzmann transport equation in discrete energy groups. The code’s 69-group structure provides sufficient resolution to accurately represent important resonance effects, particularly in the epithermal energy range where many nuclides exhibit resonance absorption peaks.

For burnup calculations, WIMS integrates the following system of differential equations to track nuclide concentrations:

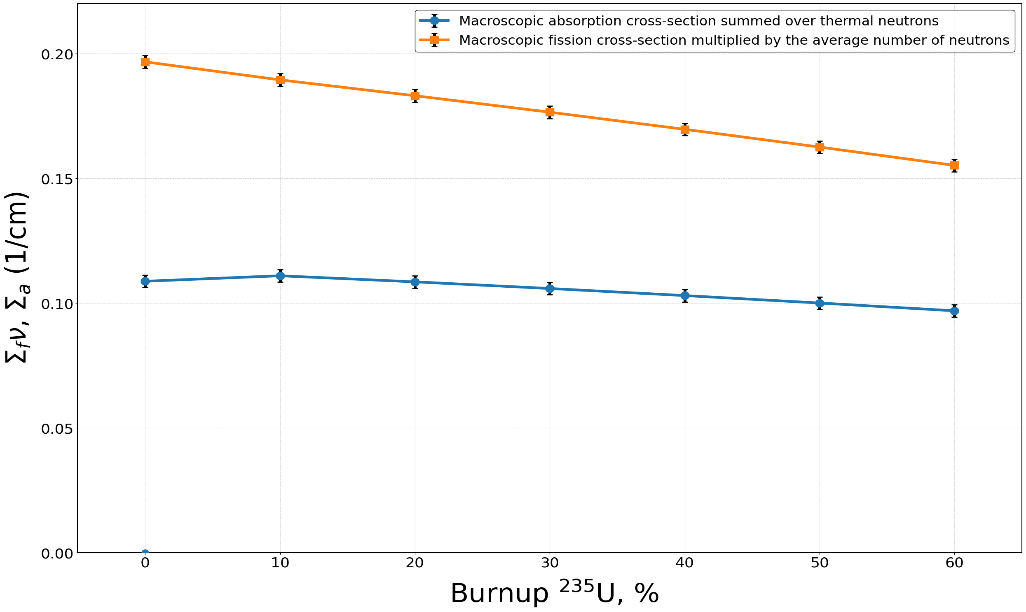
(4)

where is the number density of nuclide *i*, is its absorption cross-section, *φ* is the neutron flux, represents the probability that fission of nuclide *j* yields nuclide *i*, is the decay constant of nuclide *k* to *i*, and is the direct yield of nuclide *i* from fission.

The transition from cell-averaged cross-sections to full-core simulation typically involves a two-step approach: detailed cell calculation with WIMS to generate homogenized cross-sections as functions of burnup and other state parameters; and full-core simulation using diffusion theory codes such as IRT-2D which utilize the pre-generated cross-section libraries [1-4].

**RESULTS AND DISCUSSION**

**Macroscopic Absorption Cross-Section (Σa) Evolution.** The computational results reveal a non-monotonic behavior in the macroscopic thermal-neutron absorption cross-section (Σa - blue line in the Figure 1) throughout the burnup process. As illustrated in Figure 1, Σa initially increases from 0.1087 cm⁻¹ at beginning-of-life (0% burnup) to a maximum value of 0.1109 cm⁻¹ at approximately 10% burnup, representing an increase of about 2%. Beyond this peak, ***Σa*** undergoes a steady decline, reaching 0.0969 cm⁻¹ at 60% burnup – a reduction of approximately 12.7% from the peak value and 10.8% from the initial value.



**FIGURE 1**. Calculation results using the WIMS code for IRT-4M type nuclear fuel with 19.75% enrichment in uranium-235.

This characteristic pattern stems from the competing effects of fission product accumulation and fissile material depletion. During the initial irradiation period (0-10% burnup), the rapid buildup of neutron-absorbing fission products, particularly Xe-135 and Sm-149, dominates the neutronic behavior [11]. These fission products, with their extraordinarily high absorption cross-sections, significantly contribute to the overall absorption cross-section despite their relatively low concentrations. The production rate of these poisons initially exceeds their destruction rate through neutron absorption and radioactive decay, leading to net accumulation [1-6].

Beyond 10% burnup, the rate of poison accumulation decreases due to several factors: saturation effects for certain stable poisons like Sm-149; increased destruction of poisons through neutron absorption as their concentrations build up; and continued depletion of U-235, which reduces the fission rate and consequently the production rate of new poison nuclei. The combined effect of these processes leads to the observed decline in Σa in the later stages of irradiation.

**TABLE 1:** Major Contributors to Macroscopic Absorption Cross-Section at Selected Burnup Levels

|  |  |  |  |  |
| --- | --- | --- | --- | --- |
| **Burnup (%)** | **U-235 contribution (cm⁻¹)** | **Fission products contribution (cm⁻¹)** | **Pu-239 contribution (cm⁻¹)** | **Other nuclides (cm⁻¹)** |
| 0 | 0.0824 | 0.0003 | 0.0000 | 0.0260 |
| 10 | 0.0741 | 0.0152 | 0.0015 | 0.0201 |
| 30 | 0.0579 | 0.0218 | 0.0042 | 0.0172 |
| 60 | 0.0326 | 0.0314 | 0.0071 | 0.0258 |

The data in Table 1 clearly demonstrate the evolving contributions of different nuclide groups to the total absorption cross-section. The U-235 contribution decreases steadily due to depletion, while the fission product contribution increases throughout the burnup period, though at a decreasing rate in the later stages. Pu-239 makes a small but growing contribution, reaching approximately 7.3% of the total ***Σa*** at 60% burnup.

**Macroscopic Fission Cross-Section (Σfν) Evolution.** In contrast to the complex behavior of Σa, the macroscopic fission cross-section multiplied by the average number of neutrons per fission (Σfν **-** orange line Figure 1) exhibits a consistent linear decrease throughout the burnup range studied. As can be seen from Fig.1, it decreases from 0.1966 cm⁻¹ at 0% burnup to 0.1551 cm⁻¹ at 60% burnup, without an initial increase. This parameter is given by

(5)

where is the microscopic fission cross-section and for U-235 thermal fission [1].

The monotonic decline directly reflects the reduction in , as fission primarily occurs in U-235 in this enrichment regime. The rate of decrease is nearly linear, with a slope of approximately -0.0007 cm⁻¹ per % burnup, calculated as

,

where B is burnup. This linearity arises because the fission rate is proportional to the remaining fissile inventory, following the depletion equation

(6)

where φ is the neutron flux.

The straightforward behavior of Σfν reflects the dominant influence of U-235 depletion on fission reactions. Since U-235 remains the primary fissile isotope throughout the irradiation history in 19.75% enriched fuel, the reduction in its number density directly translates to a proportional reduction in fission reaction probability. The near-linear decrease suggests that other processes—such as Pu-239 buildup and spectral effects caused by changing composition—have relatively minor influence on the overall fission cross-section in this fuel type and burnup range.

The average number of neutrons per fission (ν) remains relatively constant throughout the burnup process, with values ranging from 2.42 for U-235 to 2.90 for Pu-239. Since the contribution of Pu-239 to overall fissions remains limited (reaching approximately 15% at 60% burnup), the weighted average ν shows only a slight increase from 2.43 to 2.48, which partially mitigates but cannot fully compensate for the reduction in fission cross-section due to U-235 depletion.

**Fission Product Accumulation and Neutron Poisoning.** The accumulation of fission products plays a crucial role in determining the neutronic behavior of irradiated fuel.

**TABLE 2:** Concentration of Major Neutron Poisons at Selected Burnup Levels

|  |  |  |  |
| --- | --- | --- | --- |
| **Burnup (%)** | **Xe-135 concentration (atoms/barn·cm)** | **Sm-149 concentration (atoms/barn·cm)** | **Other significant poisons** |
| 0 | 0.0000 | 0.0000 | - |
| 10 | 5.21×10-5 | 3.86×10-5 | Gd, Eu, Cd isotopes |
| 30 | 7.92×10-5 | 8.24×10-5 | (increasing concentrations) |
| 60 | 9.13×10-5 | 1.24×10-4 | (significant contributions) |

Xe-135 deserves special attention due to its extraordinary absorption cross-section and complex dynamic behavior. Its concentration increases rapidly during early irradiation, approaching near-saturation levels by 20-30% burnup. The equilibrium concentration of Xe-135 depends on the balance between production (through fission and decay of precursors) and removal (through neutron absorption and radioactive decay). At typical research reactor flux levels (~1014 n/cm²·s), neutron absorption represents the dominant removal mechanism, making the equilibrium concentration relatively flux-dependent [1].

Sm-149, in contrast, is stable and accumulates continuously throughout irradiation. Its production occurs through the decay chain:

149Nd→β−→ 149Pm→β−→ 149Sm (stable) (7)

Since Sm-149 has a substantial absorption cross-section (~4.0×104 barns) and does not decay, it continues to accumulate throughout the irradiation history, though at a decreasing rate as its precursors are depleted along with the fissile material. The accumulation of these neutron poisons significantly impacts the overall neutronic behavior of the fuel, particularly in the early stages of irradiation [1-6].

**Implications for Reactor Operation and Safety.** The observed burnup-dependent changes in macroscopic cross-sections have significant implications for research reactor operation and safety analysis. The non-monotonic behavior of Σa necessitates careful consideration in core design and fuel management strategies. During the initial irradiation period, the increasing absorption cross-section contributes to a more rapid decrease in reactivity than would be expected from fissile depletion alone. Reactor operators must account for this effect when planning fuel cycle length and control rod positioning [1].

The linear decrease in Σfν provides a more predictable pattern for reactivity management throughout most of the fuel life. However, the combined effects of changing cross-sections necessitate burnup-dependent cross-section libraries for accurate core simulation. Simplified models that assume constant cross-sections or linear behavior would introduce significant errors in reactivity prediction, particularly in the early irradiation period. From a safety perspective, the changing composition also affects other important parameters including temperature coefficients, control rod worth, shutdown margins, and decay heat production [1-7].

From a safety perspective, the changing composition also affects other important parameters including:

1. Temperature coefficients: Changing fuel composition affects the Doppler coefficient and moderator temperature coefficient

2. Control rod worth: Altered absorption characteristics impact control rod effectiveness

3. Shutdown margins: Must be verified throughout the fuel cycle

4. Decay heat production: Changes with evolving isotopic composition

These effects highlight the importance of sophisticated fuel management programs that track burnup history and compositional changes for each fuel assembly in the reactor core.

**Comparison with Other Fuel Types and Systems.** The behavior observed in IRT-4M fuel differs somewhat from that observed in other nuclear fuel systems. In commercial power reactors using low-enriched uranium (2-5% U-235), plutonium buildup plays a more significant role in maintaining reactivity over long burnup periods. In our case of research reactor fuel with 19.75% enrichment, the higher initial U-235 content and lower U-238 inventory limit the relative importance of plutonium production.

Similarly, the findings differ from those observed in molten salt reactor systems, where continuous fuel processing and different neutron spectrum characteristics lead to distinct evolution patterns. This comparison underscores the importance of fuel-specific and reactor-specific analysis rather than generalizing findings across different systems.

**CONCLUSIONS**

This study has presented a detailed computational analysis of burnup effects on macroscopic thermal-neutron cross-sections in IRT-4M fuel assemblies using the WIMS code. The results demonstrate distinctly different behaviors for the absorption and fission cross-sections throughout the burnup range from 0% to 60%:

1. The macroscopic absorption cross-section (Σa) exhibits initial increase followed by progressive decrease, resulting from competing effects between fission product accumulation and fissile material depletion. The peak absorption occurs at approximately 10% burnup, with a value 2% higher than the initial state.

2. The macroscopic fission cross-section multiplied by neutrons per fission Σfν shows a consistent linear decrease with burnup, reflecting the dominant influence of U-235 depletion. The slope of this decrease is approximately -0.000692 cm⁻¹ per percent burnup.

3. Fission product accumulation, particularly of neutron poisons such as Xe-135 and Sm-149, plays a crucial role in determining the neutronic behavior of irradiated fuel. These poisons contribute significantly to the initial rise in absorption cross-section despite their relatively low concentrations.

4. Plutonium-239 production from U-238 capture provides only partial compensation for U-235 depletion, contributing approximately 15% to total fissions at 60% burnup but insufficient to prevent the overall decrease in fission cross-section.

These findings have important implications for research reactor operation and safety analysis. The significant changes in macroscopic cross-sections throughout fuel life necessitate burnup-dependent cross-section libraries for accurate core simulation. Simplified models that assume constant cross-sections would introduce substantial errors in reactivity prediction, particularly during the early irradiation period where the absorption cross-section increases despite fissile depletion.

Future work should focus on experimental validation of these computational results through measurement of reaction rates in irradiated fuel elements. Additionally, extension of the analysis to higher burnup levels would provide valuable information for potential life extension strategies for research reactor fuels. The methodology developed in this study could also be applied to other fuel types and enrichment levels to build a more comprehensive understanding of burnup effects in research reactor systems. The findings particularly highlight the importance of fuel-specific and reactor-specific analysis rather than generalizing findings across different systems, as the behavior observed in IRT-4M fuel differs somewhat from that observed in other nuclear fuel systems such as commercial power reactors using low-enriched uranium or molten salt reactor systems [1-10].

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