2026
IJS-DP: 12881

VERITAS

Visual Explorer for Real vs. Idealized neutron Transport and Analytic Solutions

DEVELOPER MANUAL

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1. Introduction

The VERITAS (Visual Explorer for Real vs. Idealized neutron Transport and Analytic Solutions) simulator is a high-fidelity computational tool designed to model the process of neutron moderation across a variety of media and environmental conditions. By leveraging the OpenMC Monte Carlo particle transport code, VERITAS provides users with an intuitive yet rigorous platform for visualizing the stochastic nature of neutron interactions.

The software is primarily intended as an educational resource for students learning nuclear physics fundamentals, as well as a versatile tool for entry-level research and preliminary design studies.

The simulator is documented in three manuals:

- User Manual – Describing the operation of the simulator from the user's perspective.
- Physics Manual – Describing the underlying physical models and nuclear data processing.
- Developers Manual – Description of the software architecture and integration of the Dash/OpenMC framework.

This document is the User Manual for the VERITAS simulator.

2. Getting Started: The Interface

2.1. Landing Page Overview

Upon launching *VERITAS*, the user is presented with a comprehensive landing page designed to provide both technical context and operational guidance. The layout is structured into several interactive modules, accessible via the sidebar and central dashboard:

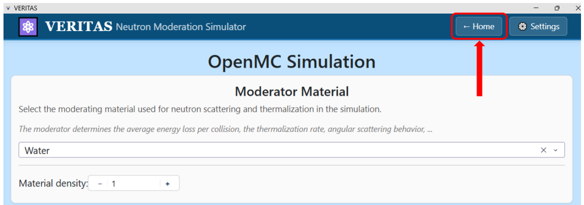
- **Quick Start Guide:** A step-by-step workflow for configuring and executing a simulation, ranging from material selection to results analysis.
- **Physics Introduction:** A concise overview of the fundamental principles of neutron moderation, thermalization, and the stochastic nature of Monte Carlo transport.
- **Feature Description:** A showcase of the simulator's capabilities, including 3D track visualization, energy loss heatmaps, and analytic solution comparisons.
- **System Background:** Details regarding the project's origins, its role in the Reactor Physics course at the University of Ljubljana, and the integration of the *IDECS/pydecs* custom cross-section pipeline.
- **Credits & Acknowledgments:** Recognition of the creators, mentors, and testers at the Jožef Stefan Institute, as well as the open-source community behind *OpenMC*.

The **Footer** serves as a navigational hub, providing direct links to the Reactor Physics Division at the Jožef Stefan Institute (IJS), the Faculty of Mathematics and Physics (UL), and the official documentation for the primary software libraries—*OpenMC*, *Python*, and *Plotly Dash*—that power the simulator. Details regarding the components used to create custom cross-sections, specifically *IDECS* and *pydecs*, are discussed in depth in the following sections.

2.2. Application Interface Navigation

Transitioning from the landing page to the core simulation environment is done by clicking the **Enter Simulator** button. Conversely, users can return to the initial landing page at any time by selecting the **Home** icon located within the global **Header** of the **VERITAS** interface.

(a) Accessing the simulation environment via the **Enter Simulator** button.



(b) Returning to the landing page using the **Home** navigation link.

Figure 1: Navigation workflow between the landing page and the simulation interface.

2.3. User settings

Users can access the application settings through the settings button located in the header. Clicking this button opens a modal that allows the user to configure the **VERITAS** storage capacity for simulation setups and custom materials, the logical port number, the cross-section library location, and the primary setup storage directory.

When entering directory or file locations, absolute Windows paths must be used (e.g., `C:\VERITAS\Crosssections\cross_sections.xml`). This ensures the `VERITAS.bat` script and the `setup.ps1` initialization file can correctly map the host directories to the internal Docker environment.

The application supports the use of any logical port within the range of 1024 to 65535. While the default is 8300, users may select any port in this range, provided it is not already occupied by another system service. Please note that changes to the port number, cross-section library path, and setup storage location will take effect only after the application is restarted.

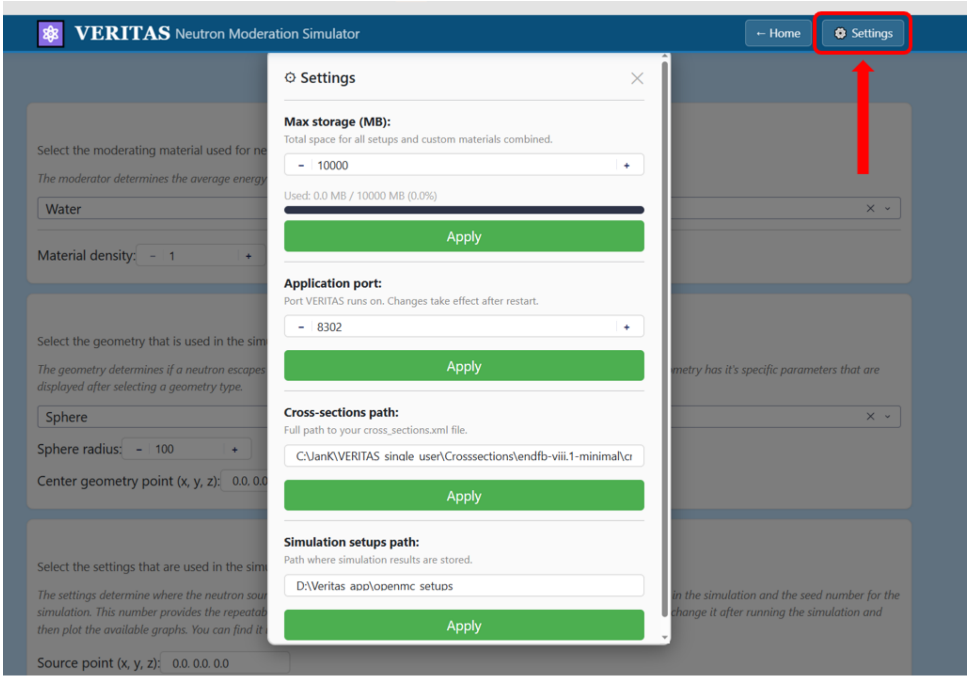


Figure 2: User settings modal.

2.4. Simulation Configuration and Settings

The **VERITAS** app page provides a comprehensive suite of parameters allowing the user to define the physical environment and source characteristics of the neutron transport simulation.

2.4.1. Material Selection and Properties

Users may define the moderator medium by selecting from a library of standard materials, including **Graphite**, **Hydrogen**, **Light Water**, **Heavy Water**, and **Beryllium**. Additionally, the interface supports custom-defined materials generated through the **pydecs** cross-section pipeline. For each selection, the user can either input a specific mass density (g/cm^3) or utilize the software's predefined default values.

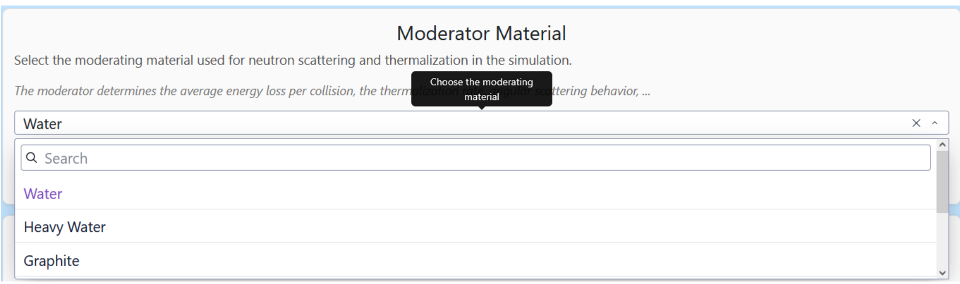


Figure 3: Materials card with material options being displayed.

2.4.2. Geometry Specification

The simulation volume is defined by selecting one of three fundamental geometric shapes: **Sphere**, **Box**, or **Cylinder**. The interface allows for precise manipulation of the dimensions (radius, height, or side lengths) and the spatial coordinates of the geometry's center point. This boundary defines the active region where neutron moderation and scattering processes are tracked. When neutrons hit this boundary they are terminated and counted as escaped.

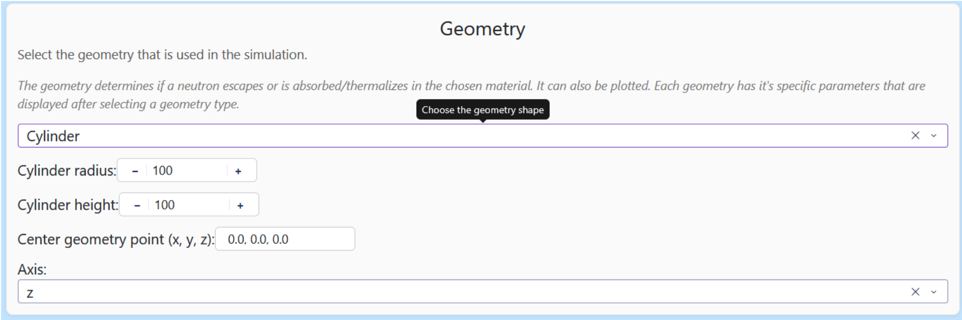


Figure 4: Geometry card with cylindrical geometry selected.

2.4.3. Settings and Source Parameters

To control the stochastic nature of the simulation, the user can configure the primary neutron source, including:

- **Source point:** The point of origin for the neutron tracks.
- **Source type:** The spatial distribution type of the source. The user can pick between a point or planar source. Selecting the planar source option exposes additional geometric configuration attributes:
  - **Plane normal axis:** Defines the primary orientation plane of the source along with its corresponding normal vector (e.g., `XY plane (normal = Z)`).
  - **Half-widths (u, v) [cm]:** Establishes the precise physical dimensions of the rectangular source area along its local orthogonal axes.
  - **Beam direction:** Controls the angular emission profile, allowing the user to choose between unidirectional (`Forward (one side)`) or bidirectional (`Both sides`) particle propagation.
- **Energy:** The initial kinetic energy of the generated neutrons.
- **Total neutrons:** The number of neutrons that will be simulated.
- **Termination energy:** The energy at which neutrons get terminated and are no longer part of the simulation. Helps to reduce file size and unnecessary simulation noise at lower energies.
- **Max. File Size:** Sets the upper storage threshold for the generated `tracks.h5` file. To prevent disk overflow and maintain system performance, the application employs a practical dynamic scaling that automatically adjusts the **Maximum Particle Track Length** and the **Total Particle Count** in accordance with the specified limit. This ensures the high-resolution trajectory data remains within the physical storage capacity of the session directory.
- **Reproducibility:** Options to use a randomized seed for unique results or a fixed user-defined seed to ensure reproducible simulation histories.

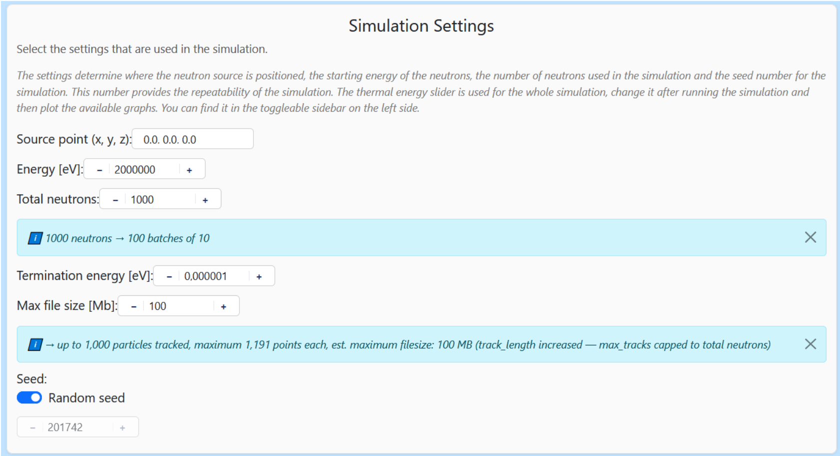


Figure 5: Settings card overview.

To assist with the user experience, every card has a short description above the input fields. Every UI component also includes a **floating tooltip**. These hover-activated descriptions provide immediate context and physical definitions for each input field.

2.4.4. Custom Material Synthesis via pydecs

If the user selects **Custom Material** from the material dropdown menu, an additional configuration interface appears. This module facilitates the synthesis of idealized nuclear data using the *pydecs* pipeline. A comprehensive technical description of the *pydecs* architecture is provided in the **Developer's Manual**. Users may choose from three fundamental material models:

- **Pure Scatterer:** Designed for studying idealized moderation mechanics. Users define the **elastic cross-section** (in barns) and the **nuclear spin**.

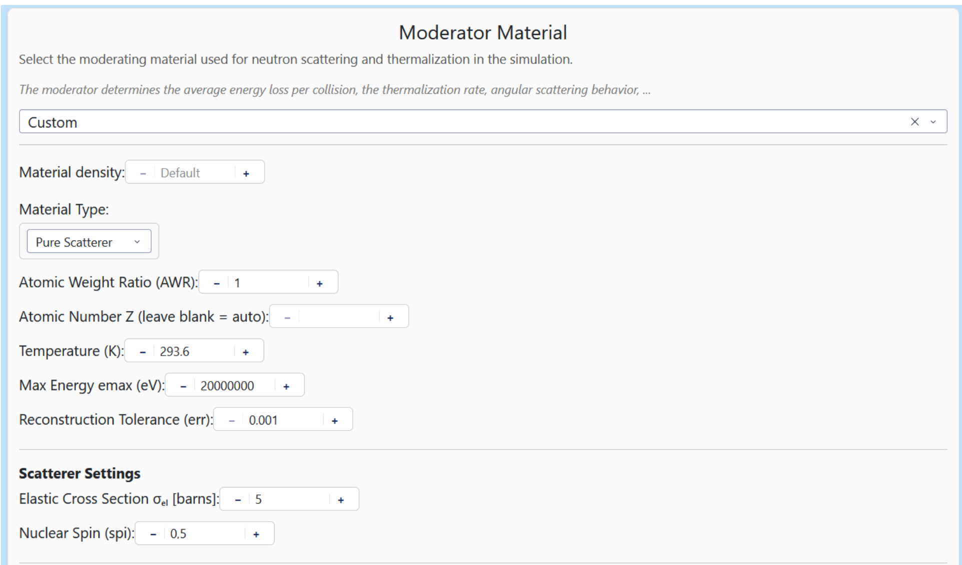


Figure 6: Configuration interface for a Pure Scatterer material model.

- **Pure Absorber:** Provides three distinct functional sub-modes for absorption modeling:
  - *1/v Absorber*: Requires a reference cross-section at 10^{-5} eV and an exponent α, such that $\sigma(E) \propto E^{-\alpha}$.

- *Constant Absorber*: A continuous, energy-independent absorption cross-section.
- *Step/Threshold Absorber*: A constant cross-section applied exclusively within a user-defined energy range $[E_{min}, E_{max}]$.

- **Scatterer with Absorption Windows:** Allows for the construction of complex energy-dependent profiles. Users can define multiple absorption windows by specifying E_{min}, E_{max}, and the corresponding cross-section (in barns) in a comma-separated format, alongside a constant background elastic cross-section.

For all custom materials, global parameters including the **nuclide atomic weight**, **temperature** (for Doppler broadening calculations), and the **maximum energy** for cross-section evaluation must be specified.

Generation and Deployment Following parameter definition, the user must assign a unique name to the material and click the **Generate and run NJOY** button. If the provided name is already in use, the application prompts the user to either overwrite the existing preset or generate an incremental version (e.g., *example_v2*). **VERITAS** then triggers the backend NJOY2016 processing scripts to produce a valid ACE (*Advanced Combined Energetic*) file.

Upon successful completion, a green confirmation message is displayed (e.g., *"✓ example.ace generated successfully"*). To utilize this material in a simulation, it must be selected from the **Created Nuclides** dropdown menu at the bottom of the material configuration card.

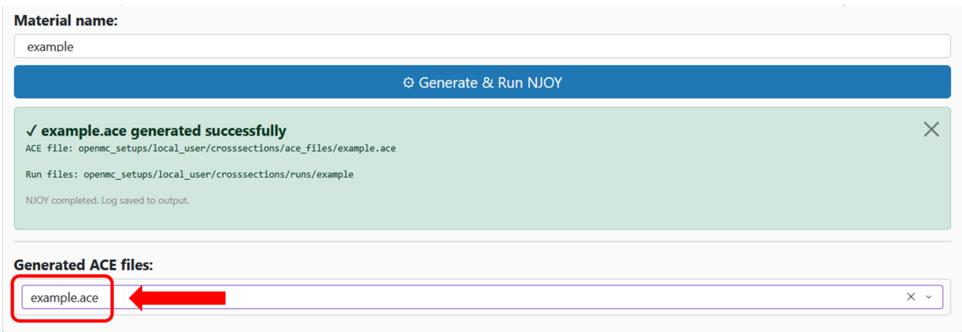
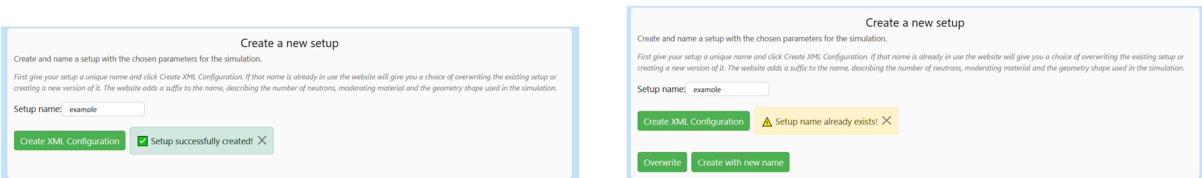


Figure 7: Successful generation and selection of a custom material within the interface.

2.5. Setup Configuration and OpenMC Execution

Once the material, geometry, and simulation parameters are defined, the user must assign a unique identifier to the configuration. Following the logic of the custom material pipeline, if a duplicate name is detected, the application provides a prompt to either overwrite the existing configuration or save a new version (e.g., *setup_v2*).



(a) Confirmation of successful setup creation.

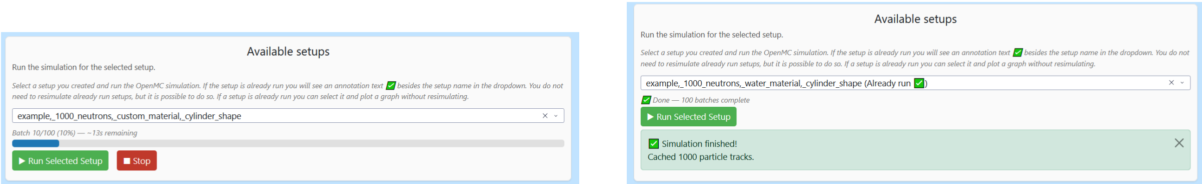
(b) Alert indicating a duplicate setup name.

Figure 8: Interface prompts for setup naming and conflict resolution.

The final step in the simulation workflow is to select the desired configuration from the **Setups** dropdown and execute the transport calculation using **OpenMC**. Configurations that have been completed in previous sessions are marked with an *"Already run ✓"* suffix for easy identification.

VERITAS provides real-time feedback during the execution and caching phases. To optimize time management, users have the option to terminate the simulation or caching process at any time via the **Stop** button if the task proves overly time-consuming.

To visualize results, the specific setup must be actively selected in the dropdown menu. If a user attempts to generate plots without an active selection, **VERITAS** will issue a notification alert.



(a) Active OpenMC execution with progress tracking and stop functionality.

(b) Completion status indicating the setup is ready for analysis.

Figure 9: Simulation execution states: from active processing to completion.

2.6. Thermal Energy Threshold Settings

Located on the left-hand side of the interface is the **Thermal Settings** sidebar, which is toggleable and remains open by default. This module allows the user to define a specific thermal energy cutoff used during the post-processing and visualization of simulation results.

While the standard default value for thermal energy is 0.025 eV, the interface provides a range from 0.001 eV to 10 eV. It is important to note that this threshold is purely a diagnostic parameter; it dictates the boundary at which neutrons are classified as ”thermalized” within the generated plots and statistics. Adjusting this slider does not alter the underlying physics or the stochastic transport processes within the **OpenMC** simulation itself.

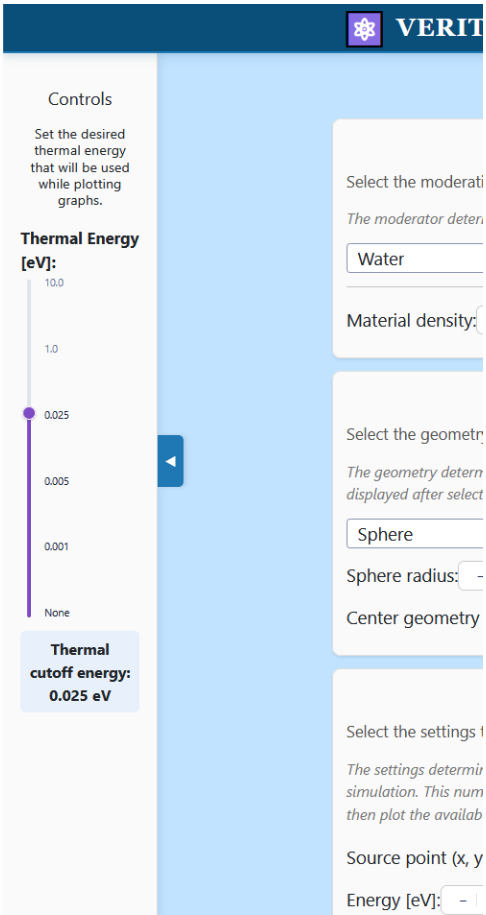


Figure 10: Toggleable sidebar featuring the thermal energy threshold slider.

3. Plotable graphs

3.0.1. Particle Selection Indices

To facilitate targeted data analysis, particle indices can be specified across the majority of the visualization modules. **VERITAS** supports three methods of index selection:

- **Individual Selection:** Inputting a single integer (e.g., n) plots a specific particle history.
- **Range Selection:** Using the format x-y (where $y > x$) plots all particle histories within that interval.
- **Stepped Range Selection:** Using the format x-y-z (where $y > x$ and $y - x \geq z$) plots every z-th particle within the specified range.

3.0.2. Plot Downloads

All plots can be downloaded in either `.png` or `.html` format.

- **HTML Format:** Saves a fully interactive graph that preserves zoom, pan, and tooltip capabilities, identical to the live interface in VERITAS.
- **PNG Format:** Exports a standard static image suitable for reports.

For sections containing multiple plots organized in tabs, the application automatically downloads the plot that is currently active in the UI .

3.1. Cross-Section Viewer

The first diagnostic tool within **VERITAS** is the **Cross-Section Viewer**. This module visualizes energy-dependent microscopic cross-sections for various neutron interactions, including **elastic scattering**, **absorption**, **heating**, and **fission** for every nuclide present in the selected moderator material.

The viewer is specifically designed for comparative analysis between the complex profiles of standard (real-world) isotopes and the idealized configurations synthesized via the *pydecs* pipeline. To accommodate the wide dynamic range of neutron energies and interaction probabilities, users can independently toggle the x and y axes between **logarithmic** and **linear** scales.

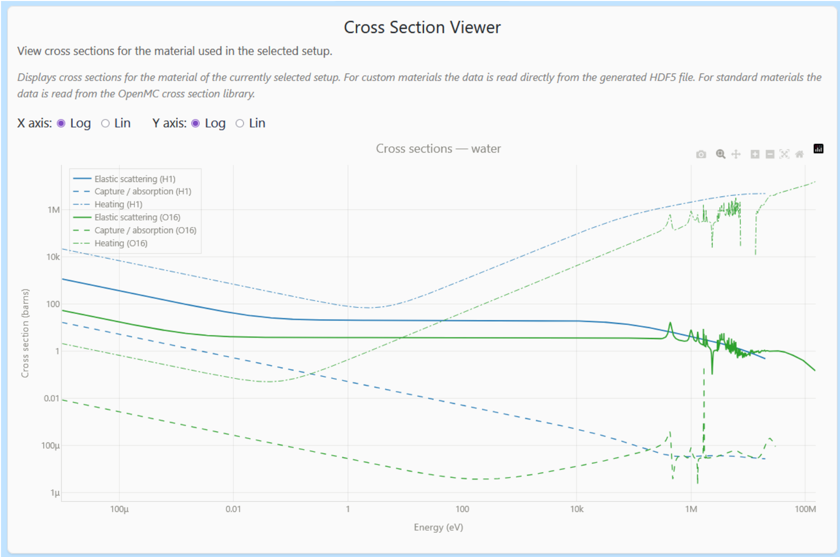


Figure 11: Cross-Section Viewer interface. For standard materials, nuclides are distinguished by color and interaction types by line style. For custom materials consisting of a single nuclide, interaction types are distinguished by area infill color.

The viewer becomes active once a configuration is chosen from the setup dropdown menu. For convenience, a tooltip displaying the currently active setup name appears centered directly beneath the section header.

3.2. 3D Particle Track and Geometry Visualization

The **Particle Track Visualization** tool allows users to render the spatial trajectories of neutrons within the moderator. After defining the desired particle indices, the user can plot the histories alongside an optional overlay of the moderator geometry.

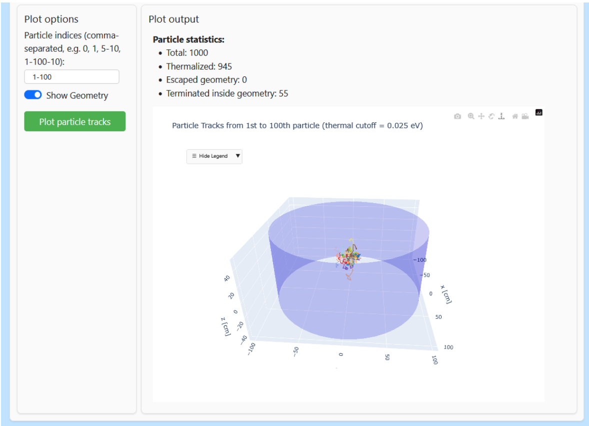
The plot legend provides metadata for each history, indicating the collision count at which a neutron reached thermal energy, or the energy and spatial coordinates at which a neutron was absorbed or escaped the system. Due to hardware rendering limits, it is recommended to visualize no more than 100 particles simultaneously, particularly when geometry is enabled or when modeling materials with high collision counts. Default number of particles shown without a user specified input is 10.

If a single particle is selected, the trajectory is color-coded using a continuous energy scale, allowing for the visualization of the neutron's energy degradation across successive collisions.

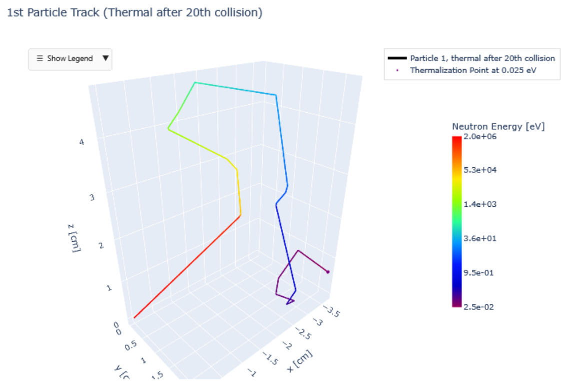
3.2.1. Particle Transport Statistics

Positioned immediately above the trajectory visualization is a concise **Particle Statistics** summary. This dashboard provides a high-level overview of the simulation's results, displaying:

- **Total Neutrons:** The total number of source particles simulated.
- **Thermalized:** The count of neutrons that successfully reached the defined thermal energy threshold.
- **Escaped:** The number of neutrons that exited the geometry boundaries.
- **Terminated:** The number of neutrons that were absorbed within the moderator volume.



(a) Multi-particle visualization with individual track identification.



(b) Single-particle visualization with energy-dependent color coding.

Figure 12: 3D trajectory visualization and global transport statistics.

3.3. Energy Transfer per Collision Analysis

The **Energy Loss per Collision** plot tracks the degradation of neutron kinetic energy as a function of scattering events. This visualization is essential for observing the transition from fast neutron energies to the thermal equilibrium state. Particle selection follows the indexing logic defined in Section 3.0.1; if no specific indices are provided, the module defaults to displaying the first 10 particle histories.

In the plot legend, each history is indexed and annotated with its final state: either the specific collision count at which thermalization was achieved, or the energy level at which the neutron escaped or was absorbed. To better visualize the rapid energy loss in the initial collisions versus the slower moderation at lower energies, the user can toggle the collision (x-axis) scale between **linear** and **logarithmic** modes via the plot options menu.

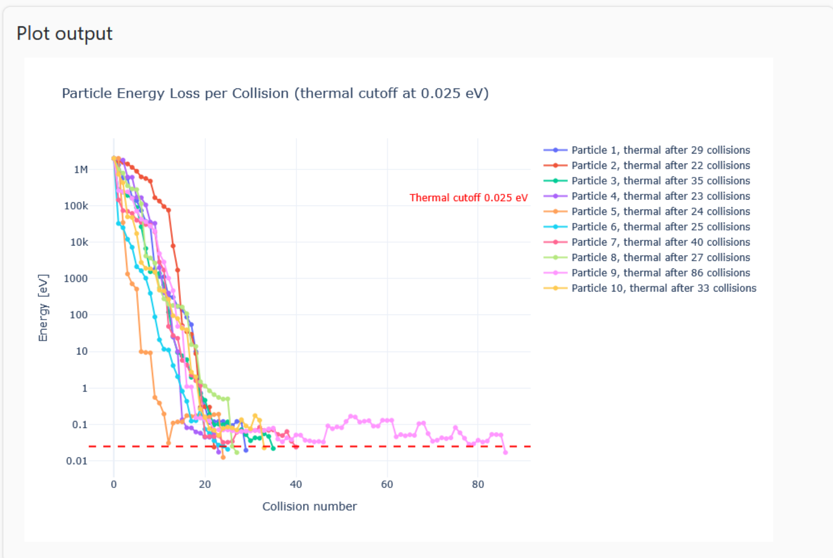


Figure 13: Energy loss distribution showing the moderation history for selected particles.

3.4. Thermal Up-scattering and Down-scattering Heatmap

The **Up/Down-scattering Heatmap** provides a statistical overview of energy transfer, visualizing the density of scattering events across the energy spectrum. By default, the heatmap incorporates data from all simulated particles, though specific subsets can be isolated using the indexing methods described in Section 3.0.1.

To allow for detailed spectral analysis, the interface offers several configurable visualization options:

- **Energy Change (ΔE):** The scale for the change in energy can be toggled between **linear** and **logarithmic** distributions.
- **Collision count:** The collision count within each energy bin can be displayed on a **linear** or **logarithmic** scale to highlight rare scattering events.
- **Normalization:** Energy values can be presented as absolute values (eV) or normalized relative to the neutron's initial kinetic energy (E/E_0).
- **Range Filtering:** Users may manually define the ΔE range (in eV) to focus on specific interaction regimes, such as the thermal peak or the epithermal slowing-down region.

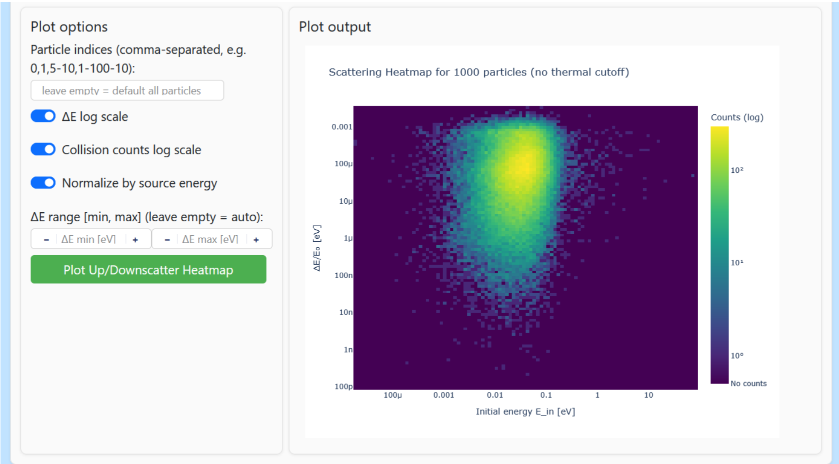


Figure 14: Heatmap distribution of energy transitions, in this example only upsactterings are being displayed.

3.5. Energy Transition Probability Density ($E \to E'$)

The **Energy Transition Probability** plot visualizes the stochastic distribution of neutron energies after a single scattering event. This module is designed to compare the simulated energy degradation against the theoretical limits of elastic scattering.

According to the laws of conservation of momentum and energy in the laboratory system, the energy E' of a neutron after a single elastic collision with a nucleus of mass A is bounded by the range $E \geq E' \geq \alpha E$. Here, the parameter α is defined as:

$$\alpha = \left(\frac{A-1}{A+1} \right)^2$$

where A represents the atomic mass of the target nucleus. On average, the energy after a single collision is expected to follow the distribution governed by the scattering kernel for the specific moderator material. Note that selecting an excessively wide initial energy bin may result in a smeared distribution, potentially obscuring the underlying physical characteristics of the scattering kernel.

To refine the visualization, the following configuration options are available:

- **Initial Energy Bin:** Users can specify the energy range (in eV) from which the scattering events are sampled. If no range is defined, the system defaults to the fast neutron regime of 1.9–2.0 MeV.
- **Bin Resolution:** The number of discrete bins used to display the probability distribution can be adjusted. To maintain statistical significance and visual clarity, this value must be an integer between 10 and 100.

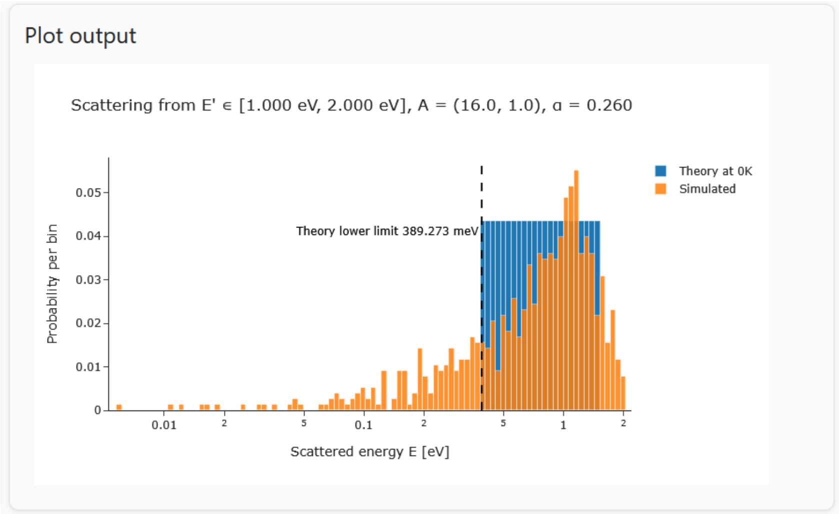


Figure 15: Probability distribution of neutron energy E' following a single scattering event from a defined incident energy E.

3.6. Distribution of Collisions Required for Thermalization

This module allows users to generate a histogram visualizing the number of collisions required for individual neutrons to reach the thermal energy threshold. To facilitate a comparative study of moderation mechanics, **VERITAS** overlays a theoretical poisson distribution curve.

3.6.1. Theoretical Consistency and Upscattering

For standard materials, the empirical data aligns most accurately with the theoretical model when the thermal energy cutoff is set above 1 eV. This is because the simplified analytical models utilized for comparison typically do not account for the effects of thermal up-scattering, which becomes physically significant in the regime below 1 eV. In this lower energy range, the stochastic nature of thermal motion (thermalization) deviates from the idealized slowing-down theory.

3.6.2. Statistical Analysis

To assist in evaluating the reliability of the simulation data, the user can toggle the display of statistical confidence intervals. The interface allows for the visualization of standard deviations (σ), with a slider from 0σ to 5σ.

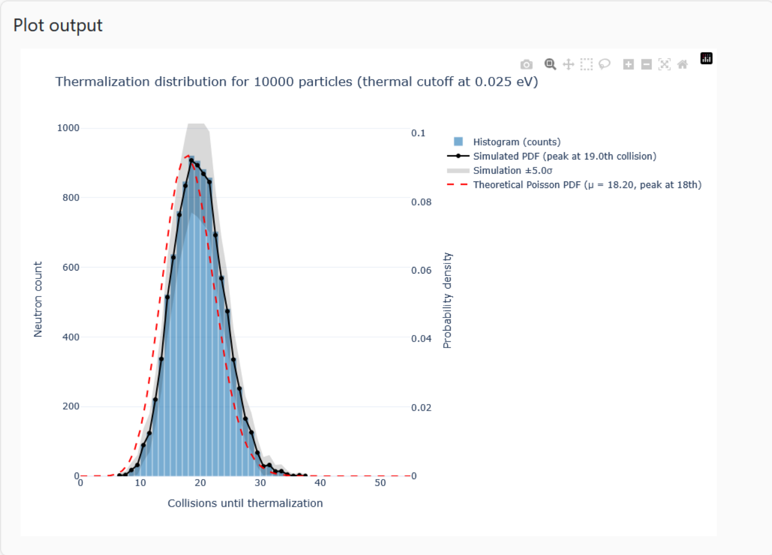


Figure 16: Histogram of collision counts required for thermalization, featuring theoretical overlay and confidence interval bars.

3.7. Mean Free Path vs. Initial Energy

This section provides a visualization of the Mean Free Path (MFP) relative to the incident energy of the neutrons. The visualization serves two purposes: it illustrates the stochastic nature of individual interactions while simultaneously providing the macroscopic trend of the medium.

Individual collision events for the selected particles are rendered as semi-transparent points in the background, while the ensemble average is computed and overlaid as a prominent red trendline. By default, the plot incorporates data from the first 1000 simulated particles, though specific subsets can be isolated using the indexing methods described in Section 3.0.1.

To effectively represent the several orders of magnitude spanning the fast and thermal energy regimes, the interface provides **Log x scale** and **Log y scale** toggles. The horizontal axis represents the **Initial Energy (eV)**, while the vertical axis measures the **Mean Free Path (cm)**.

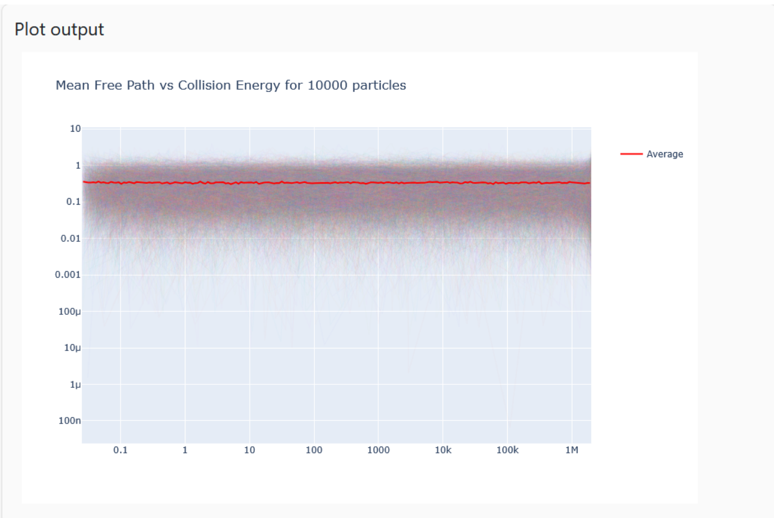


Figure 17: MFP distribution for 10,000 particles visualized in log-log mode, showing the relationship between incident energy and distance between collisions.

3.8. Scattering Cosine and Angular Distribution

This module analyzes the angular deflection of neutrons following a collision event. For elastic scattering on a nucleus of mass number A, assuming isotropy in the Center-of-Mass (CM) frame, the analytical average cosine of the scattering angle in the laboratory frame is given by:

$$\langle \cos\theta \rangle = \frac{2}{3A}$$

By default, the distribution incorporates data from all simulated particles, though specific subsets can be isolated using the indexing methods described in Section 3.0.1. The interface is organized into tabs, allowing users to toggle between the **Angle θ histogram** and the **Cosine $\cos\theta$ histogram.**

It should be noted that while the cosine distribution provides a direct physical mapping of the scattering kinematics, the angular histogram serves as a visual approximation. Users should be aware that due to the non-linear nature of the inverse cosine function, the average of the angles does not strictly equate to the arc-cosine of the average cosine:

$$\langle \theta \rangle \neq \arccos(\langle \cos\theta \rangle)$$

(a) Distribution of the scattering cosine ($\cos\theta$).

(b) Distribution of the scattering angle (θ) in degrees/radians.

Figure 18: Angular distribution analysis highlighting the preference for forward-scattering in the laboratory frame.

3.9. Scattering Heatmap: Angle vs. Energy

The scattering heatmap provides a 2D intensity distribution correlating the scattering cosine ($\cos\theta$) with the incident energy immediately prior to collision. This visualization is essential for observing how the angular distribution evolves as neutrons degrade from the fast to the thermal energy regime.

By default, the heatmap incorporates data from all simulated particles, though specific subsets can be isolated using the indexing methods described in Section 3.0.1. To improve visual clarity, the interface includes a **Log scale counts** option. This is particularly useful because the high collision density at higher energies can otherwise overshadow the details of the moderation process and thermal scattering events.

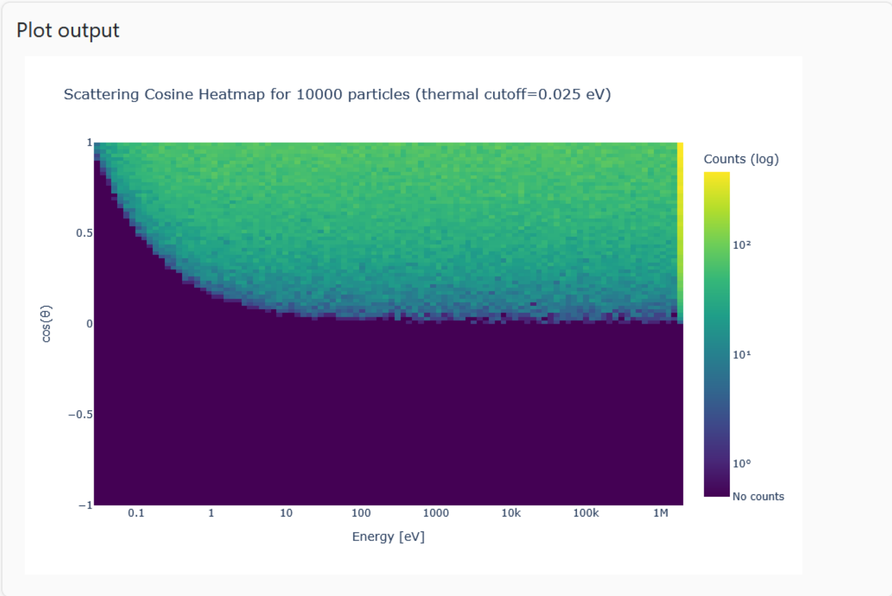


Figure 19: Scattering cosine heatmap for a custom material. The distribution clearly illustrates the prevalence of forward-scattering and no backscattering events.

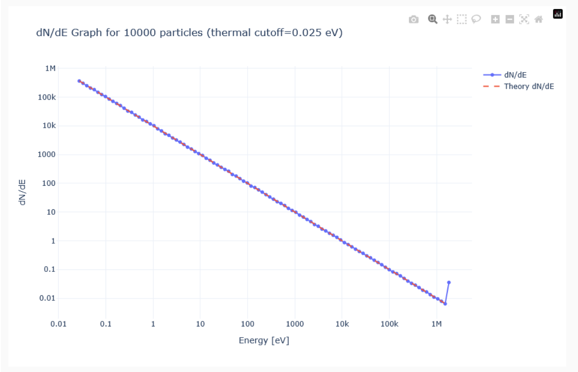
3.10. Neutron Spectra: dN/dE and $d\Phi/dE$

This section characterizes the energy distribution of the neutron population (dN/dE) and the energy-dependent fluence ($d\Phi/dE$). These metrics are fundamental for evaluating the performance of a moderator and identifying the transition between the fast, epithermal, and thermal flux components.

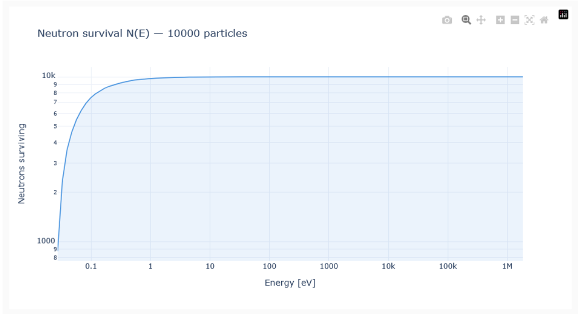
By default, the spectra incorporate data from all simulated particles, though specific subsets can be isolated using the indexing methods described in Section 3.0.1. The interface provides several specialized plotting options to assist in the analysis:

- **Log y scale:** Facilitates the examination of spectra that span multiple orders of magnitude.
- **Show Theory:** Overlays the classical analytical $1/E$ slowing-down model for comparison with the simulated results.
- **Use lethargy x scale:** Transforms the horizontal axis to the lethargy variable ($u = \ln(E_0/E)$), providing a more uniform visualization of the slowing-down density across the energy spectrum.

The data is organized into three distinct tabs: the dN/dE **distribution**, the $d\Phi/dE$ **fluence**, and the $N(E)$ **survival graph**. The survival graph is particularly valuable when simulating materials with significant absorption resonances or ”absorption windows”; it allows users to quantify the fraction of the neutron population lost to capture as they moderate through specific energy ranges.



(a) dN/dE distribution with the theoretical $1/E$ overlay in log-y mode.



(b) $N(E)$ survival graph showing population depletion due to absorption or escape from geometry.

Figure 20: Spectral analysis plots comparing simulated energy distributions against analytical models and tracking particle survival.

3.11. Analytic vs. Simulated ξ Comparison

This section evaluates the average logarithmic energy decrement per collision, ξ. The simulation results are compared against the asymptotic theoretical value for a single-species moderator:

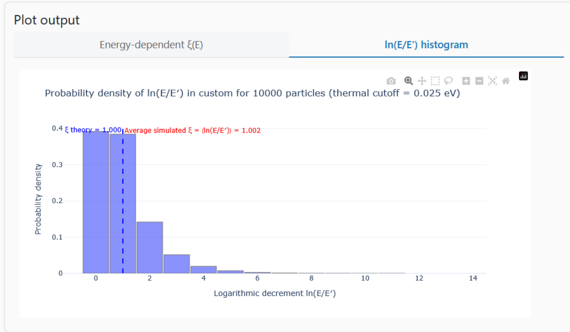
$$\xi = 1 + \frac{(A-1)^2}{2A} \ln\left(\frac{A-1}{A+1}\right) \approx \frac{2}{A+2/3}$$

By default, the comparison incorporates data from all simulated particles, though specific subsets can be isolated using the indexing methods described in Section 3.0.1. The interface provides several controls to refine the statistical analysis:

- **Number of bins:** Allows the user to adjust the discretization of the energy range to balance statistical noise against resolution.
- **Upscattering allowed:** Toggles the inclusion of thermal upscattering. Disabling this usually results in a closer match to the simplified analytical model, as standard theory often assumes a stationary target nucleus (no thermal motion).
- **Bin Weighting:** The theory is presented both with and without bin weighting. Weighting is essential for accurately capturing the characteristic ”dip” in ξ as neutrons approach thermal equilibrium with the moderator atoms in real materials.

The analysis is presented via two distinct plots. The first is a **Probability Density Histogram** of the energy decrement across all recorded collisions. The second is a **Log Decrement vs. Initial Energy** graph, featuring error bars and theoretical overlays to visualize how moderation efficiency changes across energy regimes.

(a) Average energy decrement ξ vs. incident energy, showing the thermal transition.



(b) Probability density histogram of the logarithmic energy decrement.

Figure 21: Comparison of simulated energy moderation parameters against analytical predictions.

3.12. Relative Energy Loss ($\Delta E/E$)

This section quantifies the moderation efficiency of the medium by analyzing the fractional energy transferred from the neutron to the target nucleus during each collision. The relative energy loss is defined as:

$$\frac{\Delta E}{E} = \frac{E_{\text{in}} - E_{\text{out}}}{E_{\text{in}}}$$

By default, the plot incorporates data from all simulated particles, though specific subsets can be isolated using the indexing methods described in Section 3.0.1. To facilitate a deep dive into the moderation kinematics, the following interface options are available:

- **Log x and Log y scales:** For visualizing the transition from the fast regime (where energy loss is high and deterministic) to the thermal regime (where energy loss fluctuates near zero).
- **Show Theory:** Overlays the theoretical limits for elastic scattering. This includes the maximum possible fractional energy loss, $\Delta E_{max}/E = (1-\alpha)$, and the average fractional energy loss.
- **E normalized:** A toggle that switches the vertical axis between the relative energy loss ($\Delta E/E$) and the absolute energy difference (ΔE). This allows users to distinguish between the efficiency of a single collision and the total magnitude of energy removed.
- **Upscattering allowed:** When enabled, this allows for negative ΔE values (energy gain), representing thermal upscattering where the neutron gains kinetic energy from the thermal motion of the moderator atoms.

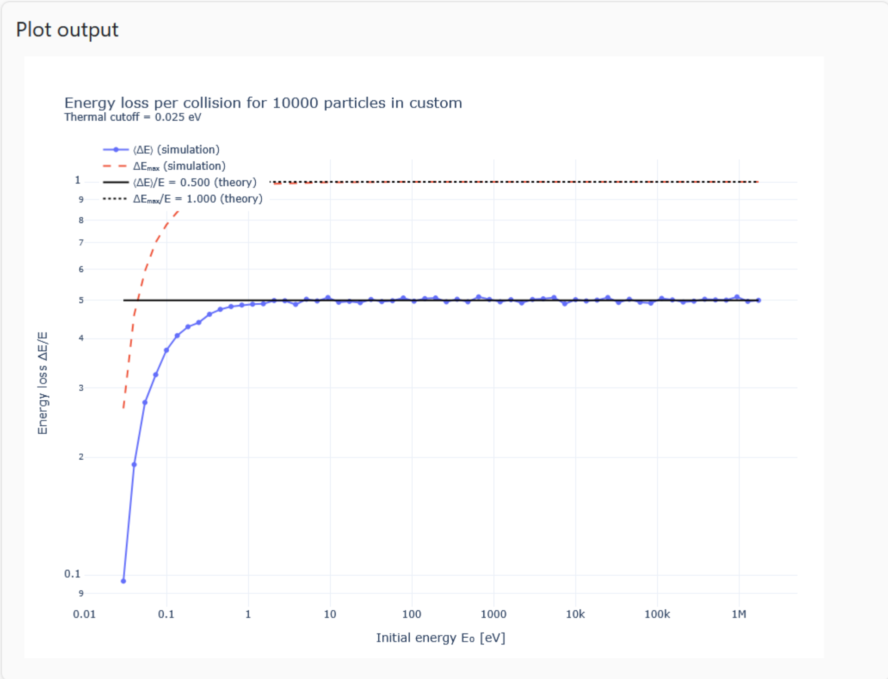


Figure 22: Relative energy loss per collision as a function of incident energy, illustrating the transition from constant average slowing-down to thermal equilibrium.

3.13. Time-Dependent Distribution: $N(t)$ and $\Delta N/\Delta t$

This section monitors the temporal evolution of the neutron population within the medium. Analyzing the population decay and interaction rates allows for the determination of the characteristic slowing-down time and the mean lifetime of neutrons in the system.

By default, the temporal analysis incorporates data from all simulated particles, though specific subsets can be isolated using the indexing methods described in Section 3.0.1. The interface provides two primary analytical views:

- $N(t)$ **Survival Plot:** Displays the remaining number of active neutrons as a function of time. Users can enable a **Theory Fit**, which applies an exponential decay model:

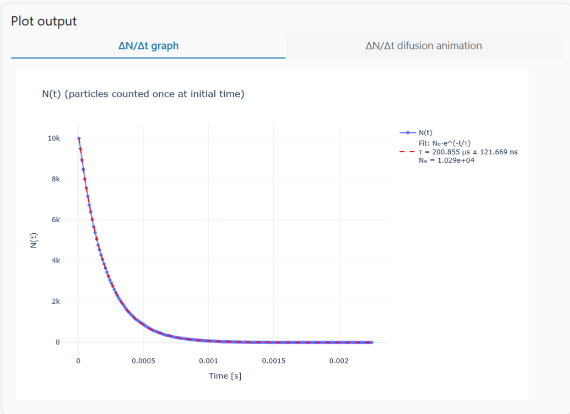
$$N(t) = N_0 e^{-t/\tau}$$

  where τ represents the decay constant or diffusion parameter. This fit is essential for quantifying the macroscopic absorption and leakage characteristics of the geometry.

- $\Delta N/\Delta t$ **Interaction Rate:** Visualizes the frequency of collisions over time, providing insight into the rate of energy loss and the approach to thermal equilibrium.

To accommodate both prompt and delayed neutron behaviors, the interface supports **Log/Lin** toggles for both the X and Y axes.

In the second tab, the **Spatial Diffusion Simulation** tracks the radial distribution of neutrons over time. This visualization illustrates the migration of the neutron cloud from the source center, displaying the probability density of neutrons at varying distances from the origin as a function of elapsed time.



(a) $N(t)$ survival curve with the exponential theory fit enabled.

(b) Snapshot of the spatial diffusion distribution over time.

Figure 23: Temporal and spatial analysis of the neutron population decay and diffusion.