Italian National Agency for New Technologies, Energy and Sustainable Economic Development

Alma Mater Studiorum Università di Bologna

INFN Istituto Nazionale di Fisica Nucleare
Sezione di Bologna

Impact of ENDF Evaluation on Nuclear Technologies Using n_TOF Data

n_TOF National Meeting, 13-14/11/2025

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Overview

- **Scope:**

  To quantify the impact of newly measured neutron cross sections (Xs) on the prediction of neutronic parameters using validated computational models.

- **Methodology Framework:**

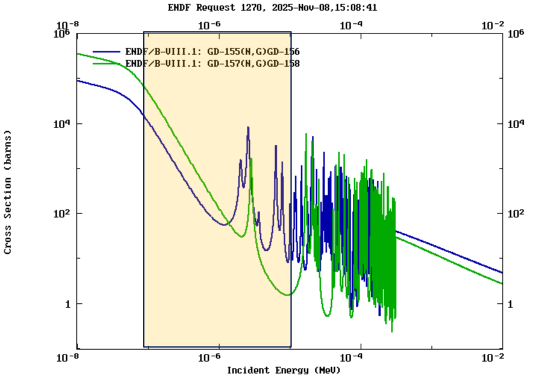
  1. **Benchmark experiments**: use integral data from the ICSBEP and IRPhE databases for high-fidelity simulations of critical facilities and reactor configurations;
  2. **Evaluated Xs**: process from experimental measurements to produce new reliable ENDF-formatted cross-section files;
  3. **Uncertainties:** include modelling, geometry, composition, and nuclear data uncertainties;
  4. **Computational tools:** use of transport codes to calculate neutronic parameters for the experimental facilities, using the evaluated cross sections and accounting for the total propagated uncertainty.

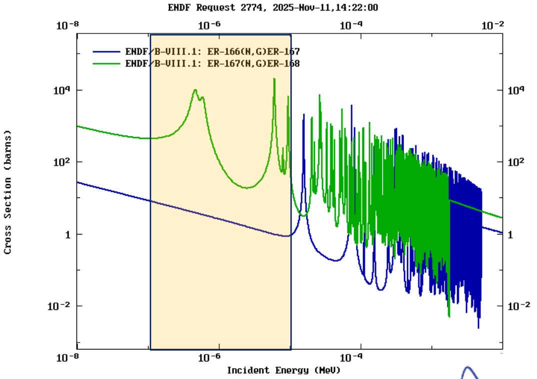
- **Outcome:**

  Compare calculated and experimental integral parameters to assess whether n_TOF data improve the agreement with benchmark results compared to existing evaluated libraries.

Motivation

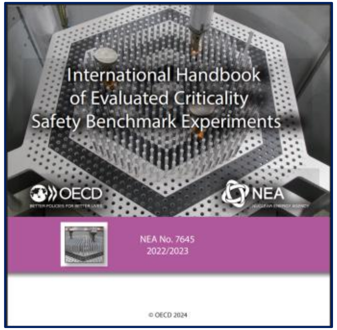
- **The neutron capture Xs of burnable absorber** (e.g., Gd, Er in LWR) **and reflector materials** (Cu in TAPIRO) are **being investigated at n_TOF, as they represent key parameters influencing** integral observables in many reactor types.
- The **strong resonance absorption** of these isotopes makes integral results, such as k_{eff} and spectral indices, highly sensitive to nuclear data.
- **Reliable validation of the cross sections** is essential for:
  1. Reactivity predictions in core design;
  2. Reactivity prediction in safety analysis;
  3. Fuel cycle and burnup analyses;
  4. Reflector characterization.
- **The comparison between experimental and calculated integral parameters ensures the consistency of evaluated Xs data with measurements, supporting their use in reactor safety and reactor core design applications**

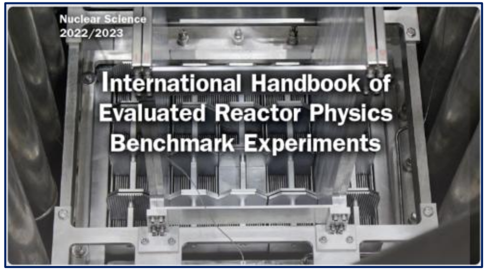




Experimental facilities: ICSBEP & IRPhE

- The **ICSBEP and IRPhE projects**, coordinated by the OECD-NEA, **are the primary international sources of integral experimental data for reactor physics and criticality safety applications**.
- They have established **rigorous and standardized procedures for the benchmark experiments documentation for validation of calculation methods and nuclear data libraries.**
- Each evaluated experiment includes a **comprehensive facility description and a detailed uncertainty assessment, ensuring high reliability for code-to-experiment comparisons**.
- These database are primary used for:
  - **Nuclear data validation** to identify and resolve discrepancies between calculated and experimental results;
  - **Criticality safety and core design applications** to compare simulation and experimental data, verifying safety margins and ensuring high-fidelity core modelling.





The ICSBEP and IRPhe Database

Content and objective of ICSBEP and IRPhe project:

- **The ICSBEP and IRPhe, developed under the aegis of the OECD-NEA compiles benchmark specifications derided from experimental critical facilities and experimental reactor worldwide,** providing rigorously evaluated **criticality safety benchmark results;**
- The benchmark specifications are designed for **use by criticality and safety analysts and nuclear data evaluators to validate calculation methods and nuclear data libraries;**
- The last 2022/2023 edition contains **170 experimental series** that were performed at **57 nuclear facilities** (IRPHe) and **589 experimental series** for **5159 critical, near-critical or subcritical configurations** (ICSBEP).
- ICSBEP also provides **49 criticality alarm placement/shielding configurations**, and **238 fundamental physics measurements configurations;**

Ref: https://www.oecd-nea.org/jcms/pl_20291/international-criticality-safety-benchmark-evaluation-project-icsbep-handbook

Ref: https://www.oecd-nea.org/jcms/pl_20279/international-handbook-of-evaluated-reactor-physics-benchmark-experiments-irphe

Example: LEU-COMP-THERM-052

NEA/NSC/DOC(95)03/IV
Volume IV
LEU-COMP-THERM-052

Uranium Dioxide (4.738-WT% enriched) Fuel Rod Arrays moderated and reflected by Gadolinium Nitrate Solution

Evaluator

Daniel Doutriaux
IPSN

Internal Reviewers
Pascal Grivot / Gilles Poullot

Independent Reviewer
Virginia Daen
Consultant to INEEL

URANIUM DIOXIDE (4.738-WT.%-ENRICHED) FUEL ROD ARRAYS
MODERATED AND REFLECTED
BY GADOLINIUM NITRATE SOLUTION

IDENTIFICATION NUMBER: LEU-COMP-THERM-052 SPECTRA

KEY WORDS: acceptable, array, critical experiment, fuel rods, gadolinium, gadolinium nitrate, hexagonal, low enriched, poison, poisoned-water moderated, poisoned-water reflected, triangular pitch, uranium, uranium dioxide

1.0 DETAILED DESCRIPTION

1.1 Overview of Experiment

Critical experiments with 4.738%-enriched uranium dioxide rod arrays in a large tank of poisoned water were carried out in testing equipment called Apparatus B in the experimental criticality facility at the "Service de Recherches et d'Etudes en Criticité" in Valduc (C.E.A. France) in 1978. Experiments were subcritical approaches extrapolated to critical, with the multiplication factor reached being very close to 1.000 (within 0.1%).

The six configurations of the experimental program were

- either one hexagonal assembly of 1261 fuel rods (21 rods per side of the hexagon)
- or one pseudo-cylindrical assembly of 1285 fuel rods (10 rods added on each side of the hexagon and 6 rods removed at each corner)

at 1.35, 1.72 and 2.26-cm triangular pitch, moderated and reflected by a gadolinium nitrate solution. The gadolinium concentration was adjusted in such a way that the solution critical height (between 87.5 and 89.6 cm) obtained by an extrapolation method covered most of the fissile column (90 cm).

Six experiments are evaluated of which all are considered acceptable for use as benchmark experiments. It is expected that these experiments can be used to improve the gadolinium cross-section assessment, the gadolinium negative reactivity worth Δk_{eff} being in the range 19% to 29%.

The experimental program is described in the basic report (Reference 1). Some complementary results were reported in Reference 2 just after experiments were performed. Further data are provided by an additional report (Reference 3) written during the period 1998 - 2000 in the framework of a re-evaluation of all experimental programs performed in the Valduc facility with these rods. This last report includes data found in archives and results of most recent extra measurements (mass and dimensions) carried out on the fuel rods.

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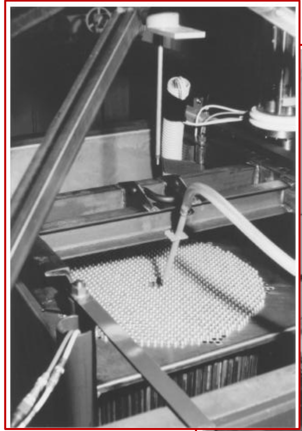
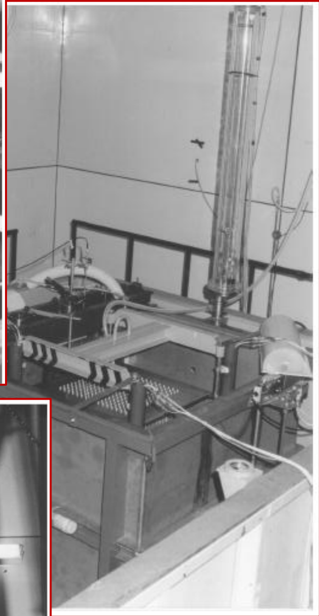
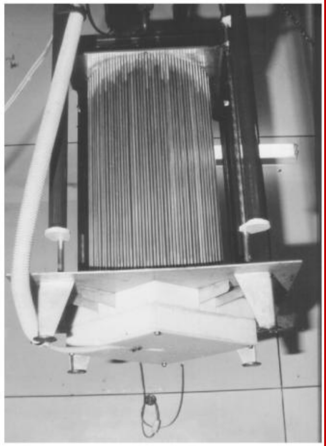
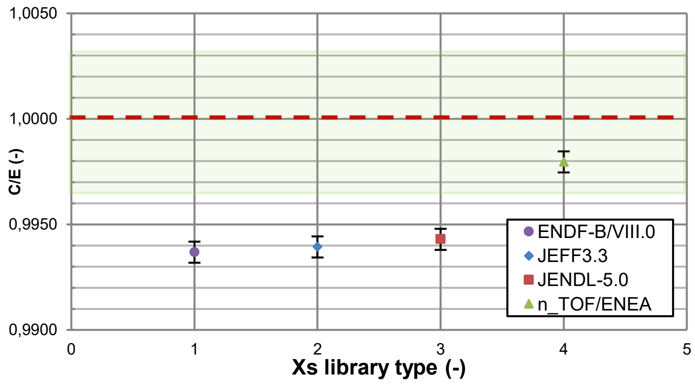


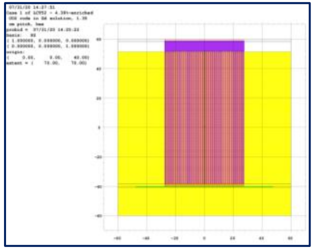


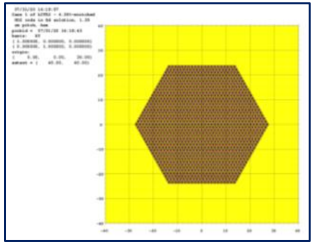


Methodology

- Produce the ENDF library with the n_TOF Xs of the measured isotopes;
- Select ICSBEP/IRPhE facilities that include these isotopes;
- Perform a series of simulation varying the cross sections (ENDF-B/VIII.1, JEFF4, JENDL 5, n_TOF) using the ICSBEP/IRPhe facilities;
- Analyse the impact of the different cross sections on the observables.







ENDF Generation

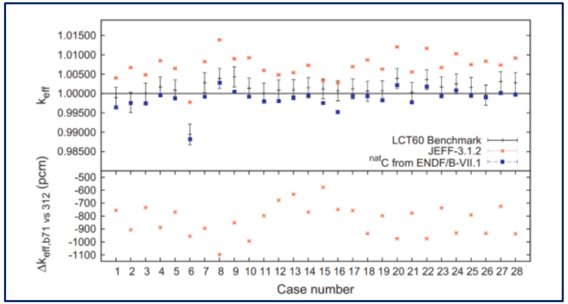
Database Selection

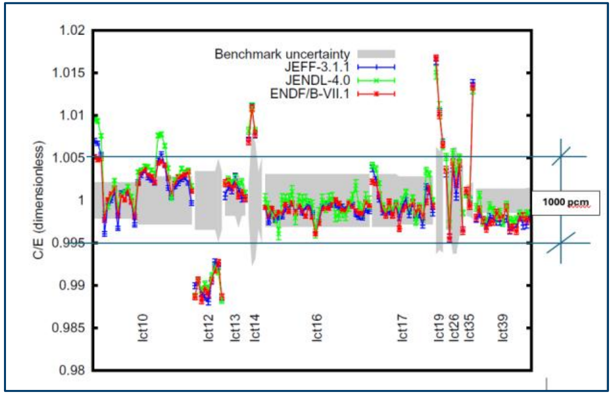
Code Simulation

Data Analysis

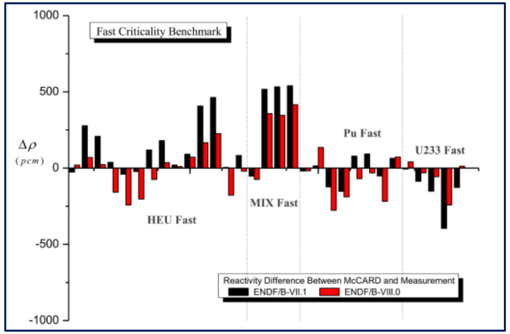
Analysis of the metric: literature review

- Example of possible observables:
  - Calc./Exp. ratio (C/E): k_{exp}/k_{calc}
  - Reactivity difference: $\rho_{exp} - \rho_{calc}$
  - Direct k_{eff} comparison: k_{exp}, k_{calc}





- Analysis of the associated experimental and simulation uncertainties;
- Analysis of complementary observables: reaction rate, sensitivity value and spectral indices to find physical motivation of possible bias;

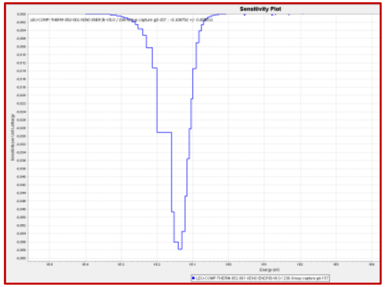


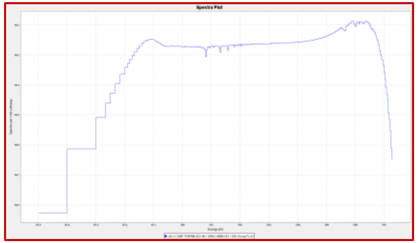
- Library ranking based on mean bias and standard deviation across benchmark cases. Given the j-th library and the i-th experimental facility:

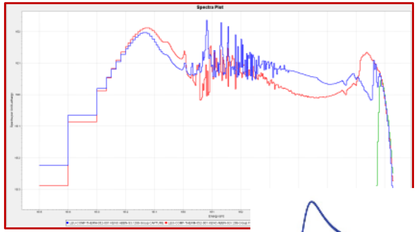
$$b_i^j = k_{calc,i}^j - k_{exp,i} \rightarrow \bar{B}_i^j = \frac{1}{N}\sum_{i=1}^{N} b_i^j \text{ and } \sigma_i^j = \sqrt{\frac{1}{N}\left(\sum_{i=1}^{N} b_i^j - \bar{B}_i^j\right)}$$

Overall Objectives

- **Verify the reliability of the newly measured cross sections for predicting criticality and other neutronic observables**;
- **Identify systematic deviations** and **potential areas for data improvement**;
- **Perform a sensitivity and uncertainty analysis** to quantify the impact of nuclear data variations on integral parameters;
- **Establish a verification methodology,** including statistical analysis (standard deviation, Chi-Square test, trend analysis);
- **Apply the methodology** to **key isotopes** measured or under measurement at n_TOF (Gd, Er, Cu).







Thank you for attention!

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