Advanced Methods for Criticality Safety Assessment

Presented to:

Past, Present and Future Validation Methods in International Criticality Safety Assessment–Panel

American Nuclear Society Annual Meeting
Washington, D.C.
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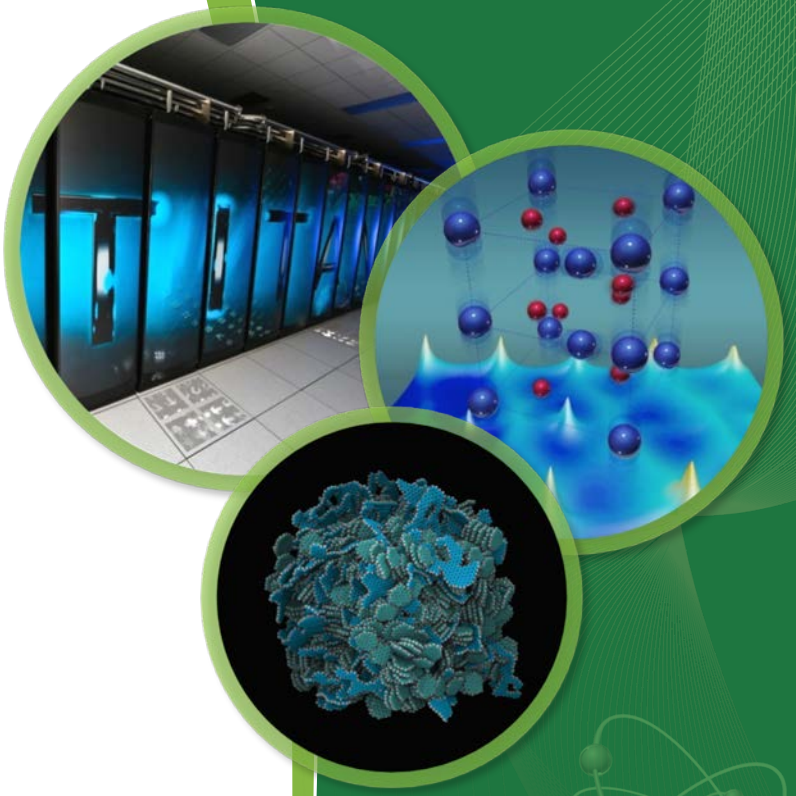

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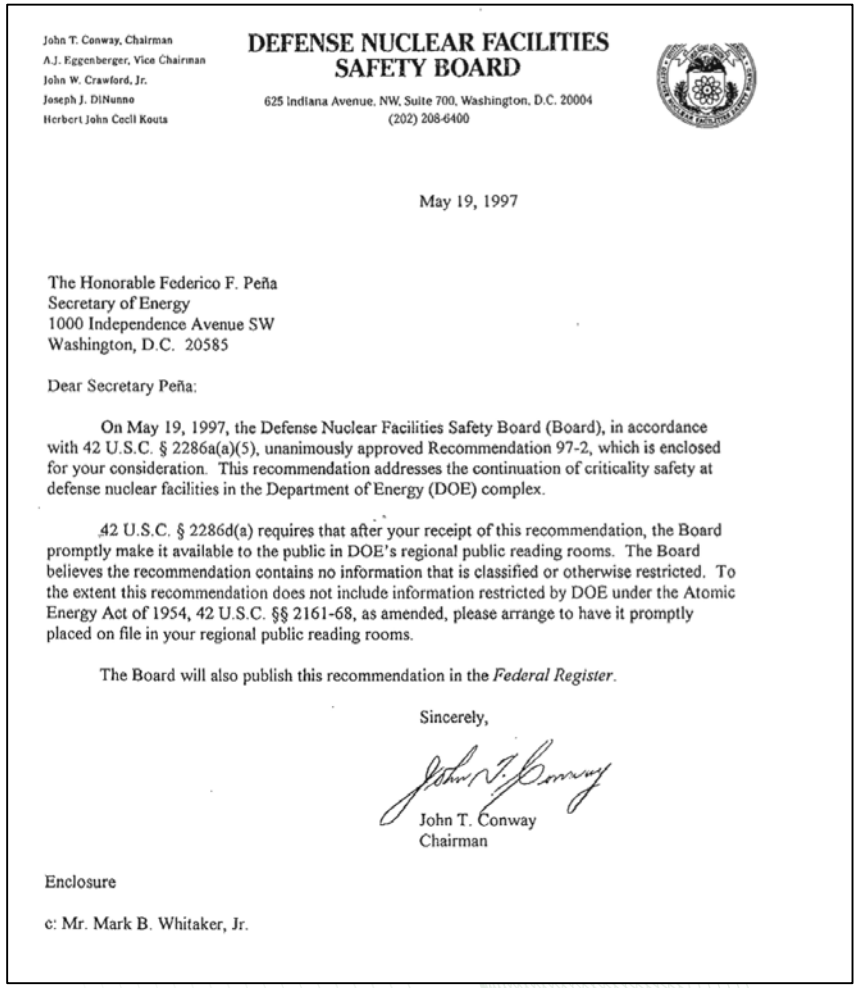
OAK RIDGE National Laboratory

Defense Nuclear Facilities Safety Board Recommendation 97-2 to U.S. Department of Energy

- Establish a program to **interpolate and extrapolate** such existing calculations and data as a function of physical circumstances that may be encountered in the future, so that useful guidance and bounding curves will result.
- The decreasing order of preference should be **experimental data, theory benchmarked against experimental data, and non-benchmarked criticality analysis with an adequate safety margin.**
- **Organize the records of calculations and experiments** conducted to ensure the criticality safety of DOE's past operations so as to provide guidance for criticality safety in similar situations in the future and avoid repetition of past problems.
- **Collect and issue** the experimental and theoretical data from the above in a publication as guidance for future activities.



John T. Conway, Chairman
A.J. Eggenberger, Vice Chairman
John W. Crawford, Jr.
Joseph J. DiNunno
Herbert John Cecil Kouts

DEFENSE NUCLEAR FACILITIES SAFETY BOARD

625 Indiana Avenue, NW, Suite 700, Washington, D.C. 20004
(202) 208-6400

May 19, 1997

The Honorable Federico F. Peña
Secretary of Energy
1000 Independence Avenue SW
Washington, D.C. 20585

Dear Secretary Peña:

On May 19, 1997, the Defense Nuclear Facilities Safety Board (Board), in accordance with 42 U.S.C. § 2286a(a)(5), unanimously approved Recommendation 97-2, which is enclosed for your consideration. This recommendation addresses the continuation of criticality safety at defense nuclear facilities in the Department of Energy (DOE) complex.

42 U.S.C. § 2286d(a) requires that after your receipt of this recommendation, the Board promptly make it available to the public in DOE's regional public reading rooms. The Board believes the recommendation contains no information that is classified or otherwise restricted. To the extent this recommendation does not include information restricted by DOE under the Atomic Energy Act of 1954, 42 U.S.C. §§ 2161-68, as amended, please arrange to have it promptly placed on file in your regional public reading rooms.

The Board will also publish this recommendation in the *Federal Register*.

Sincerely,

John T. Conway
Chairman

Enclosure

c: Mr. Mark B. Whitaker, Jr.

Knowledge management

"There are known knowns; there are things we know that we know. There are known unknowns; that is to say, there are things that we now know we don't know. But there are also unknown unknowns – there are things we do not know we don't know."

-United States Secretary of Defense, Donald Rumsfeld, 2002

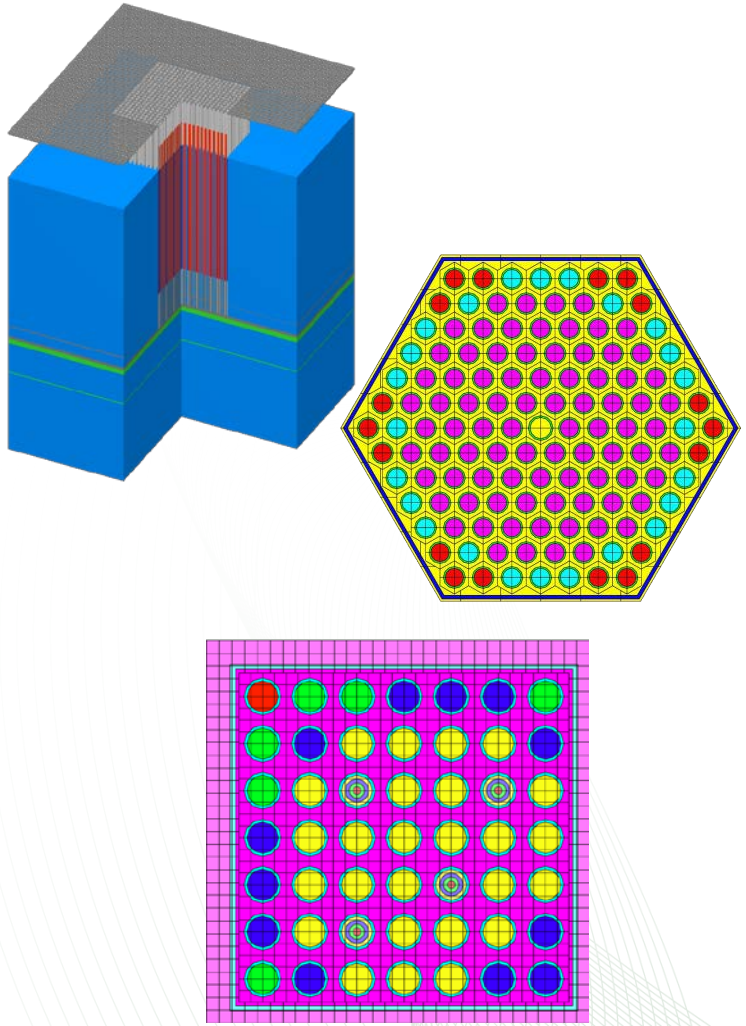
KNOWN KNOWNS **Measurements/Observations**	**KNOWN UNKNOWNS** **Uncertainty Quantification**
UNKNOWN KNOWNS **Communication**	**UNKNOWN UNKNOWNS** **Safety Margins**

"All models are wrong, some are useful."

-George E. P. Box – Statistician, Professor, Univ. of Wisconsin

Sensitivity/Uncertainty (S/U) Analysis for Validation

- Establish impact of data uncertainties on results (uncertainty quantification)
  - Bias margins for criticality safety
  - Design margins for advanced reactors
  - Safety margins for decay heat and criticality in UNF analysis
- Selection and design of benchmark experiments for validation (similarity analysis)
  - Maximizes information contained in existing integral experiments
  - Enables design of more relevant experiments
- Consolidation of measured and computed results for improved reliability (assimilation/adjustment)
  - Provides adjusted data that reduces bias and uncertainty in calculations
  - Recommend data improvements to nuclear data evaluators



Introduction to sensitivity coefficients

- Sensitivity coefficients provide insight on the sources and impact of uncertainty in nuclear engineering models.

Input Data → **Monte Carlo Code** → **Results**

Input Information:
Nuclear Data (Σ)
Number Densities (*N*)
Material Densities (ρ)

Input Uncertainty:
ΔΣ, Δ*N*, Δρ

→

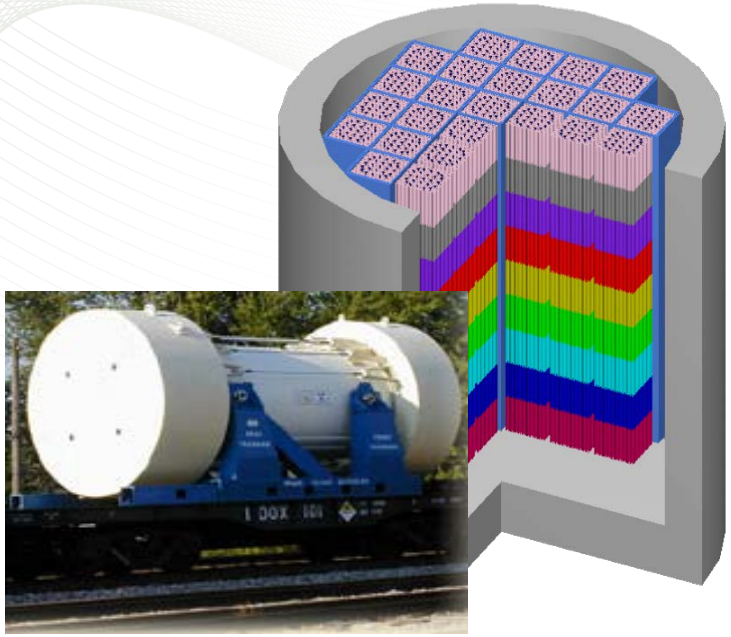
$$S_{R,\Sigma_x} = \frac{\delta R/R}{\delta \Sigma_x/\Sigma_x}$$

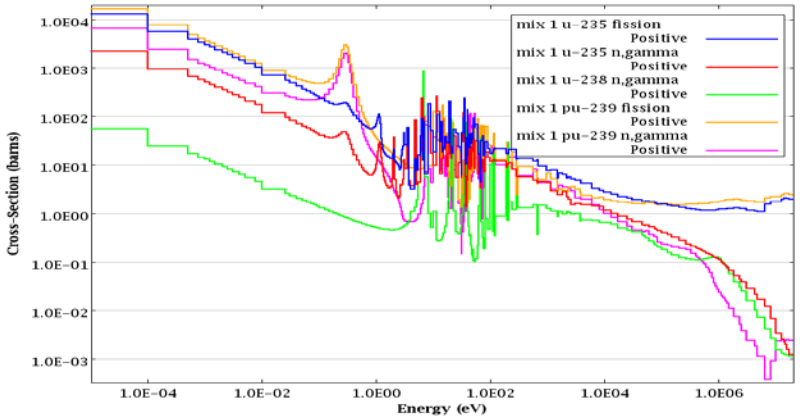
→

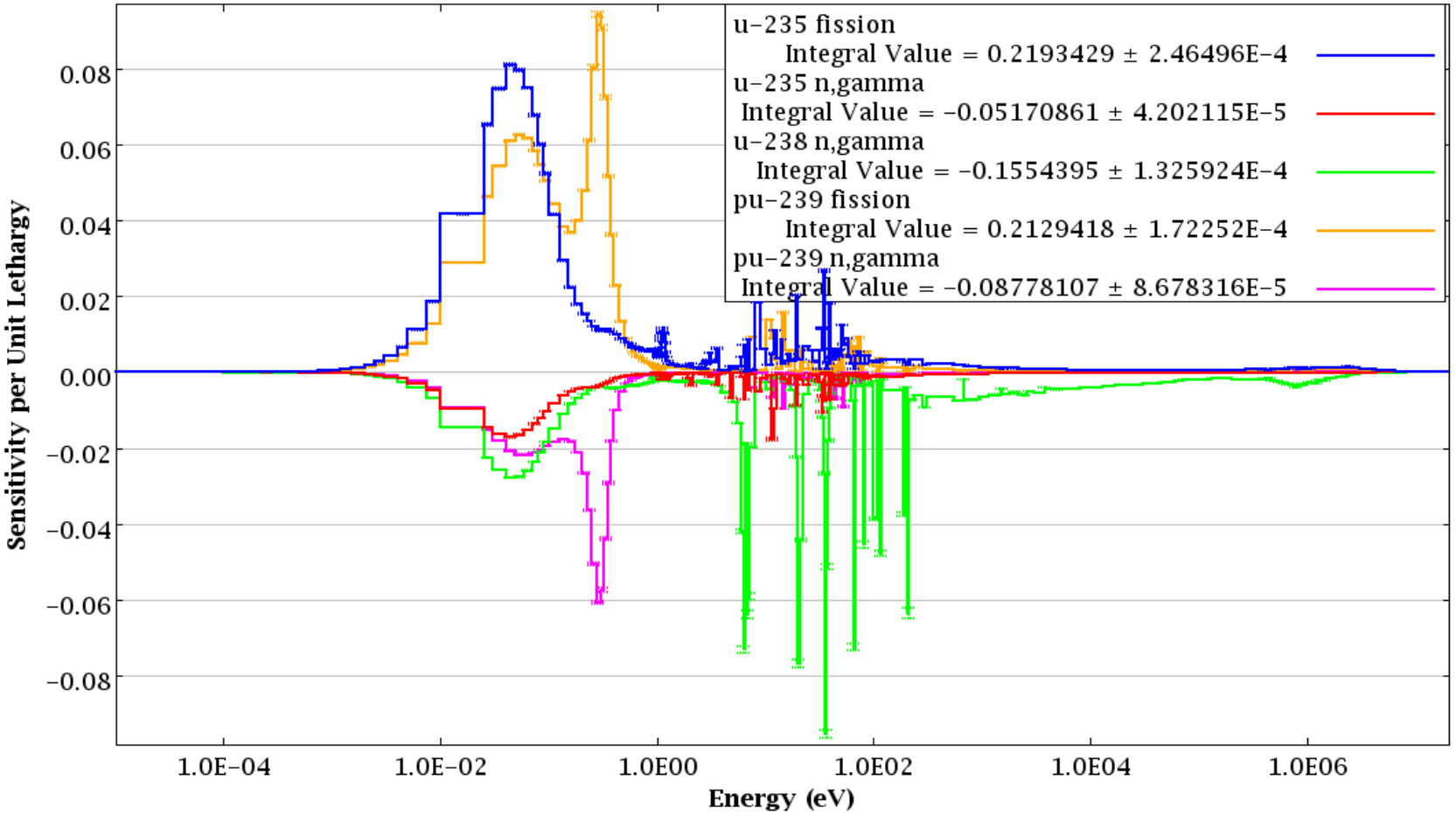
Output Information:
k_{eff}, Dose Rate,
Fission Rate, etc…

Output Uncertainty:
Δk_{eff}, Δ Dose Rate,
ΔFission Rate

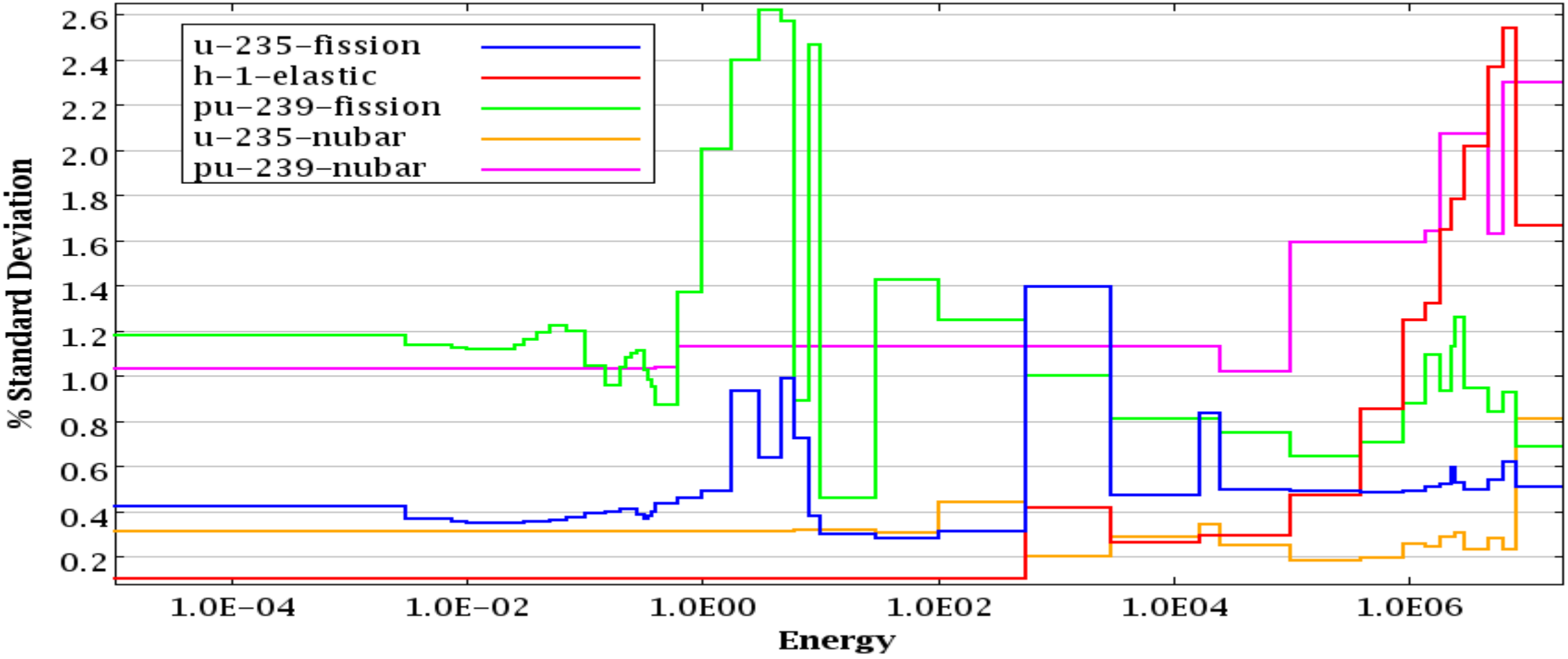
Sensitivities of the k_{eff} of a shipping cask to cross section data





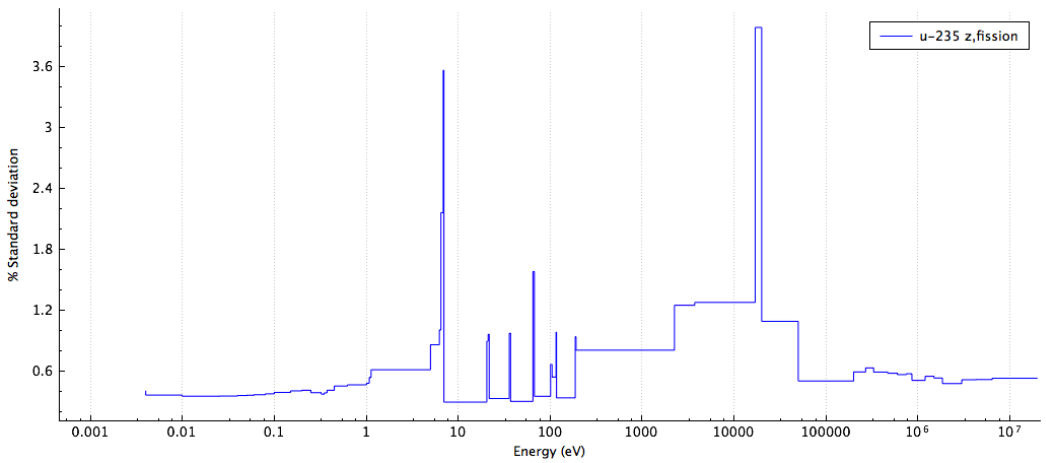


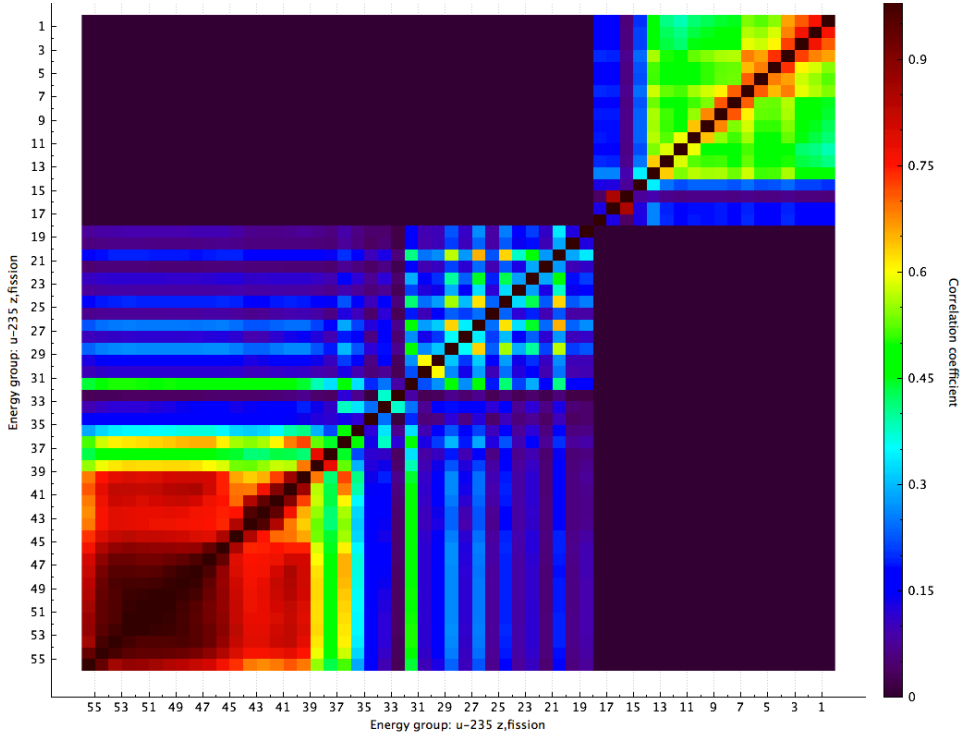
Uncertainties in cross-section data



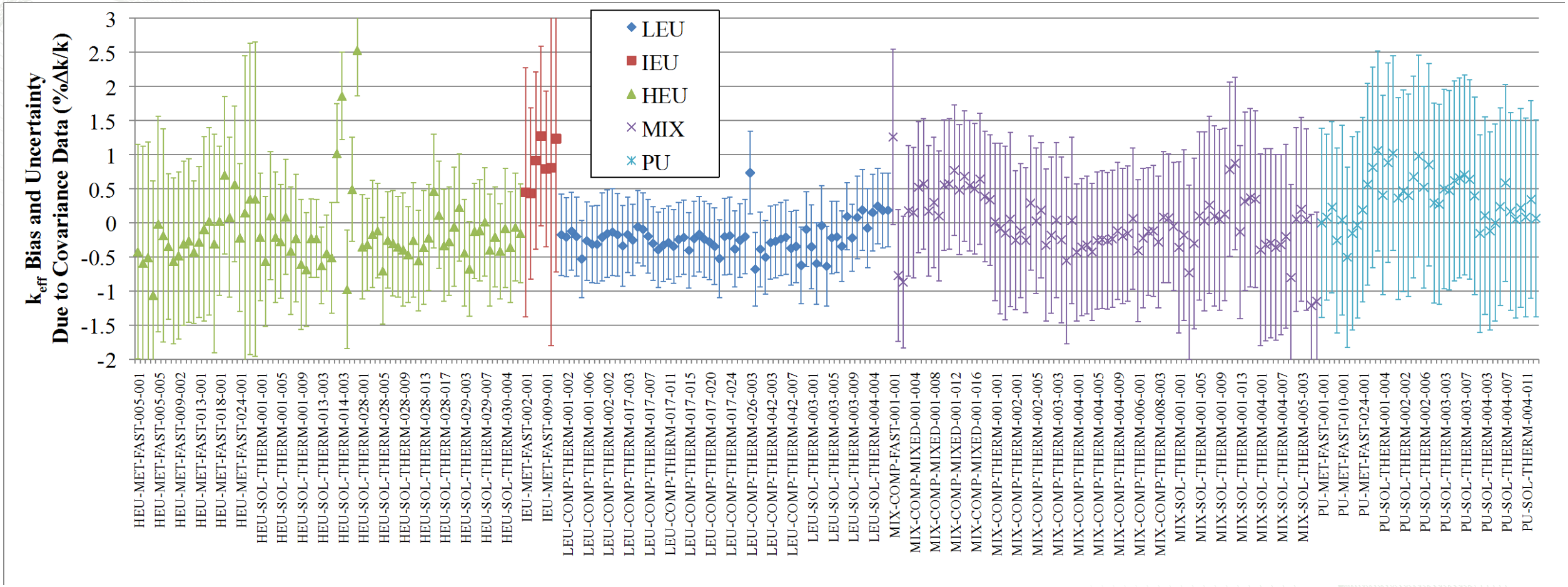
SCALE 6.2 covariance data library

- ENDF/B-VII.1 contains data for 187 isotopes.
- SCALE 6.1 data retained for ~215 missing nuclides.
- Modified ENDF/B-VII.1 ^{239}Pu nubar, ^{235}U nubar, H capture, and several fission product uncertainties, with data contributed back to ENDF/A repository.
- Fission spectrum (chi) uncertainties processed from ENDF/B-VII.1 and from JENDL 4.0 (minor actinides).
  - Previous SCALE chi uncertainties were generated from Watt spectrum data and data were missing for minor actinides.
- 56- and 252-group energy structures.

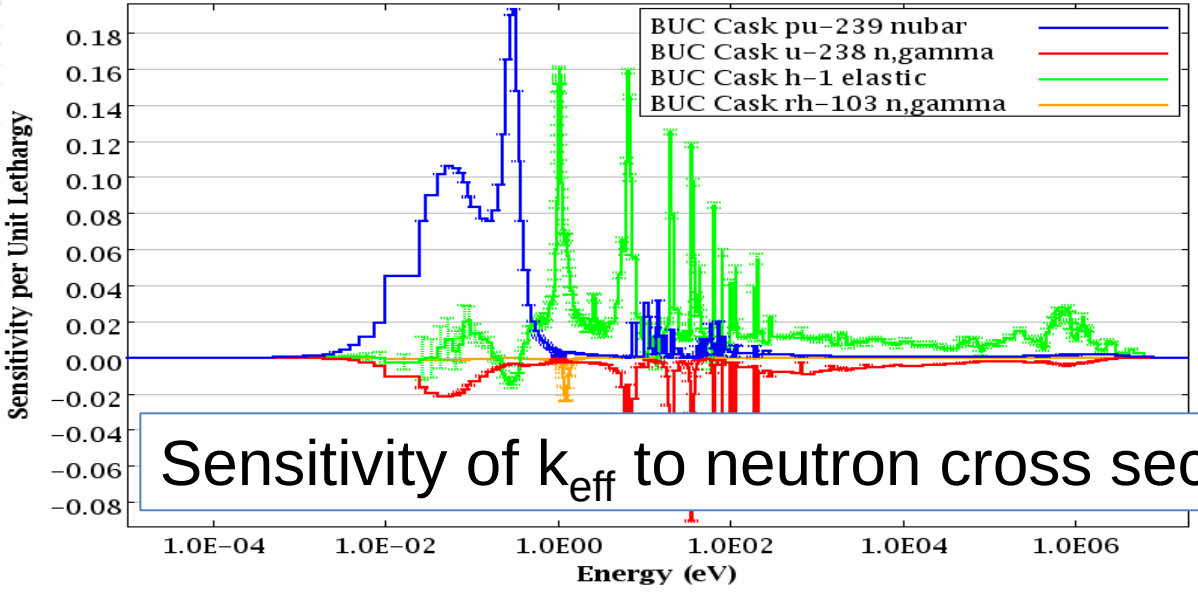




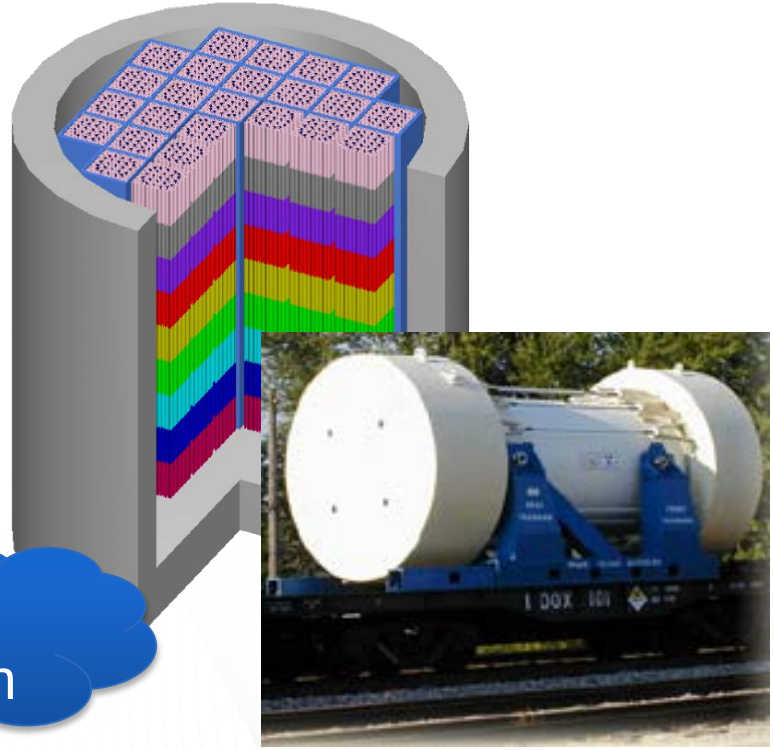
Uncertainties due to cross section covariance data for benchmark experiments



Identifying important processes and uncertainties

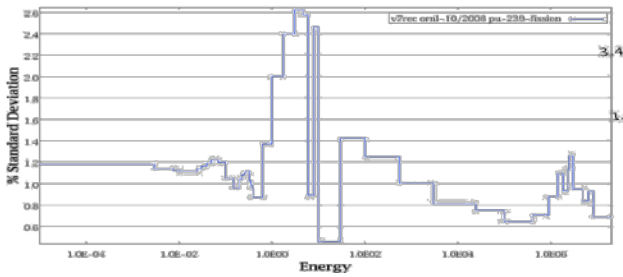


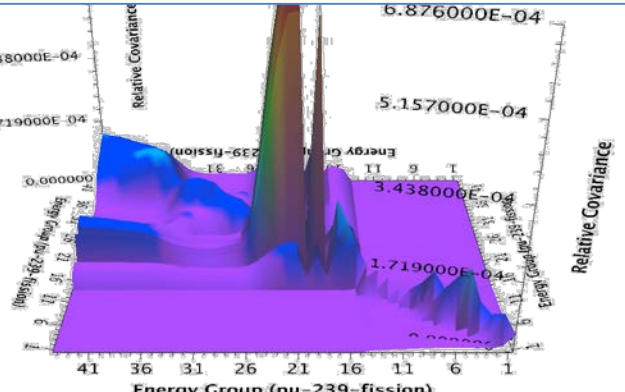
Sensitivity of k_{eff} to neutron cross sections



Known
Unknown

Covariance (uncertainty) for cross sections

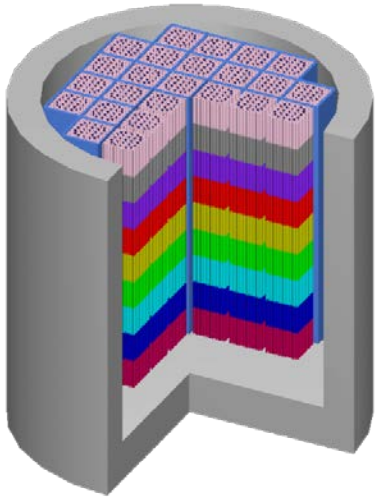


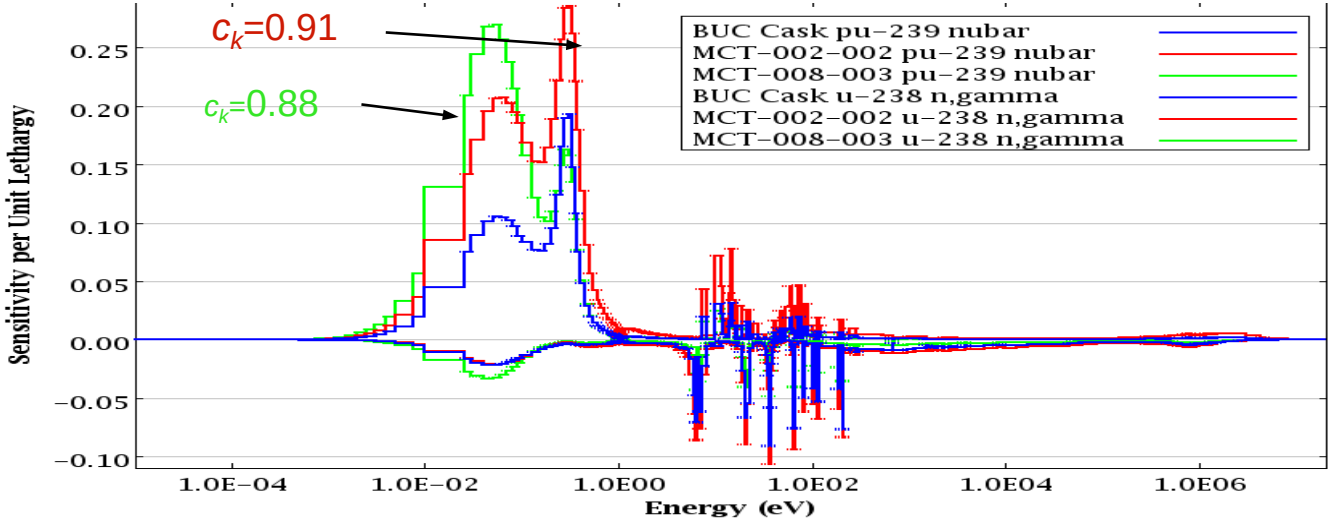


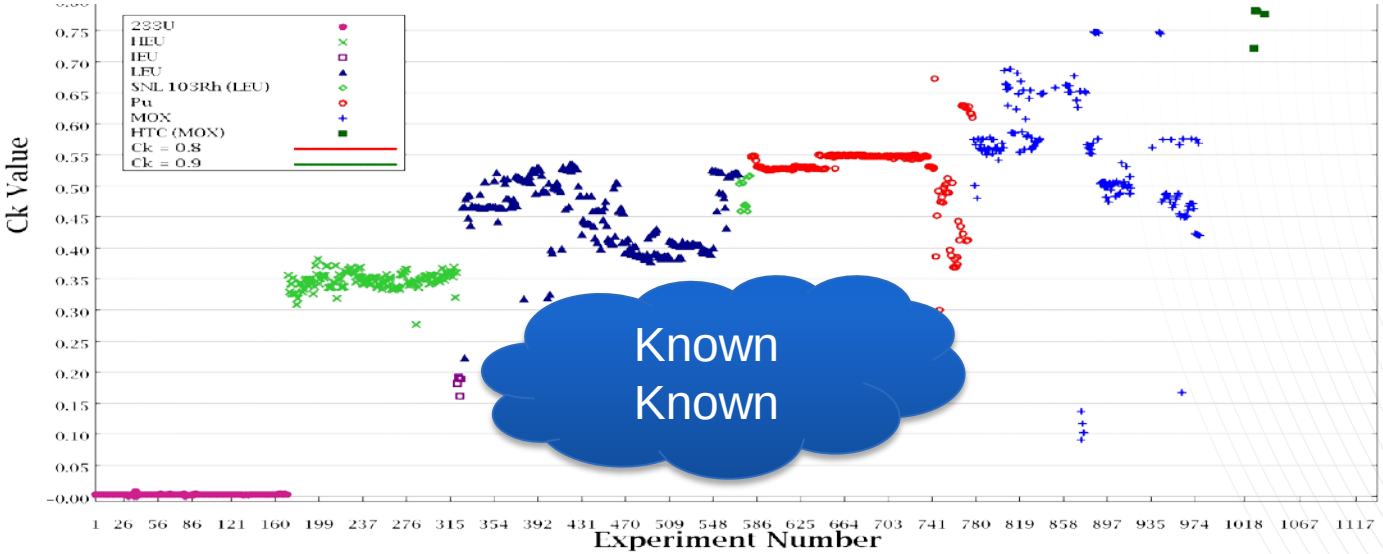
Covariance Matrix		Unc. in % dk/k
Nuclide-Reaction	Nuclide-Reaction	Due to this Matrix
^{239}Pu nubar	^{239}Pu nubar	4.0032E-01 ± 2.5625E-06
^{238}U n,gamma	^{238}U n,gamma	1.9457E-01 ± 1.2387E-05
^{239}Pu fission	^{239}Pu fission	1.5501E-01 ± 1.0838E-05
^{235}U nubar	^{235}U nubar	1.3981E-01 ± 5.0038E-07
^{239}Pu fission	^{239}Pu n,gamma	1.2261E-01 ± 4.3564E-06

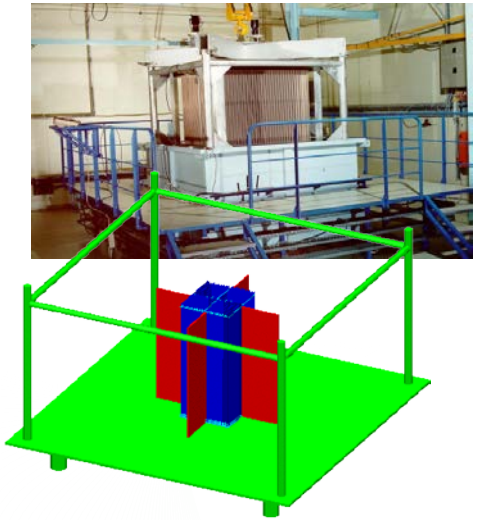
Similarity Analysis: Identifying experiments representative of targeted application

APPLICATION

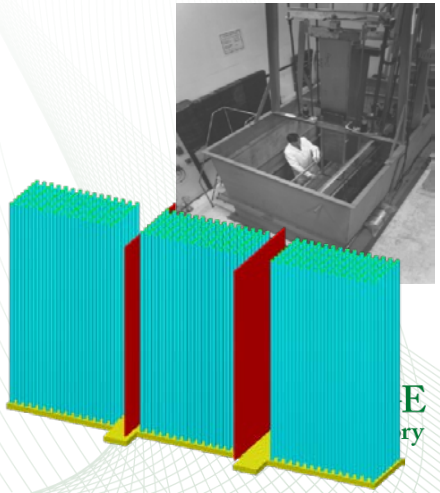




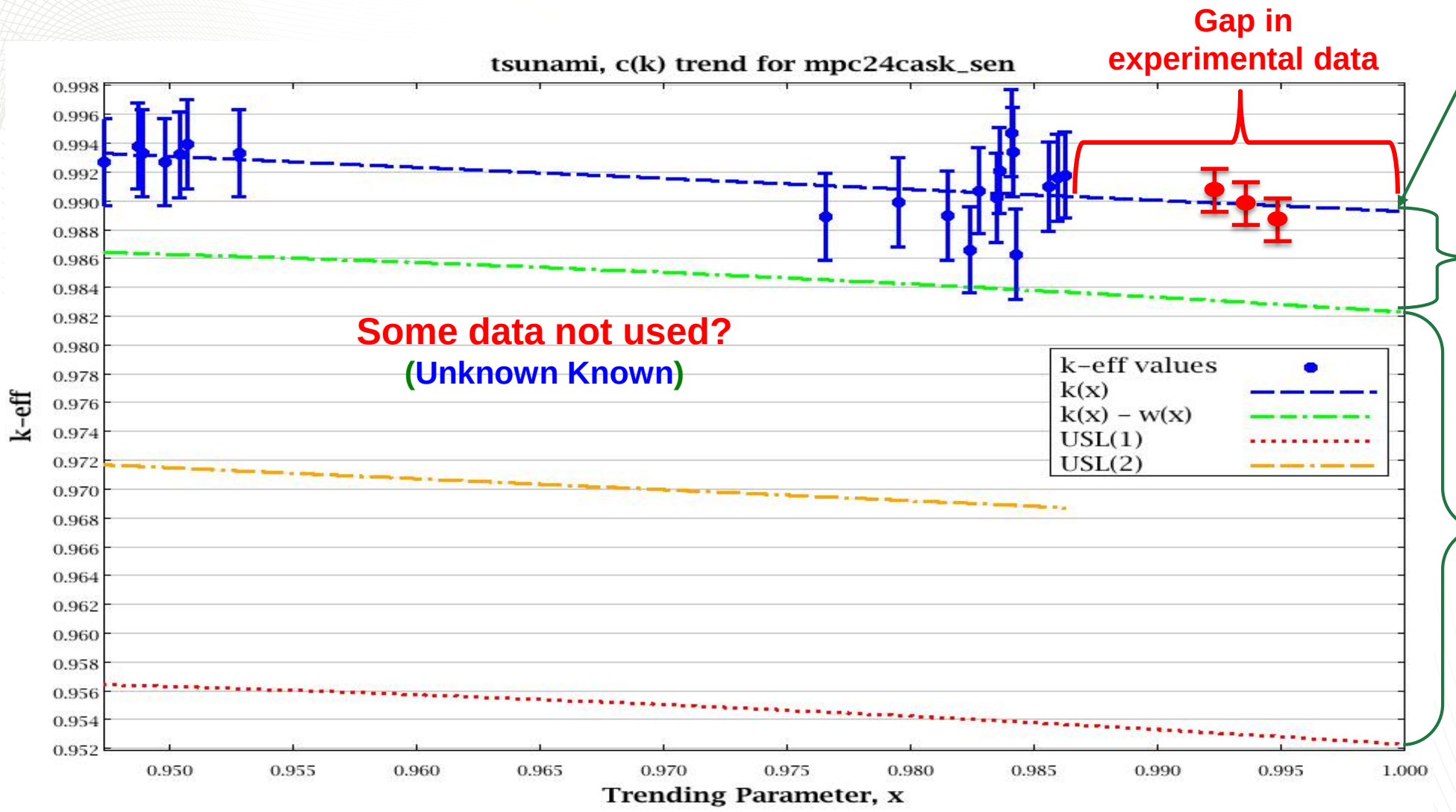




NUCLEAR CRITICALITY EXPERIMENTS



Setting safety limits – similarity as a trending parameter



Experiments projected to application
(Known Known)

Confidence band
(Known Unknown)

Safety margin
(Unknown Unknown)

Applications of S/U methods to DOE safety analysis

ORNL/TM-2001/262

OAK RIDGE NATIONAL LABORATORY
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Investigations and Recommendations on the Use of Existing Experiments in Criticality Safety Analysis of Nuclear Fuel Cycle Facilities for Weapons-Grade Plutonium

B. T. Rearden
K. R. Elam

UT-BATTELLE

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ORNL/TM-2008/196

Application of the SCALE TSUNAMI Tools for the Validation of Criticality Safety Calculations Involving ^{233}U

January 2009

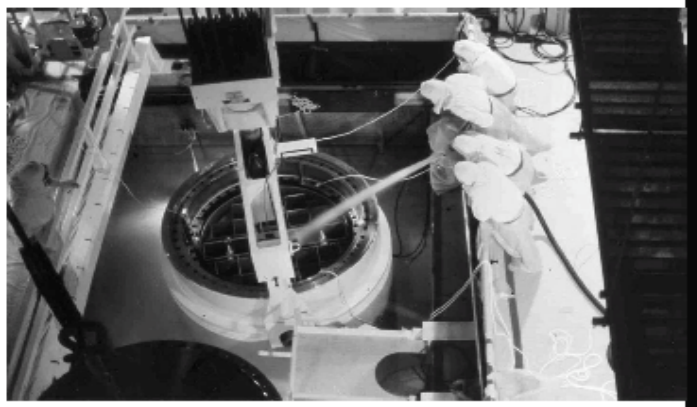
Prepared by
D. E. Mueller
B. T. Rearden
D. F. Hollenbach

UT-BATTELLE

OAK RIDGE National Laboratory

Regulatory basis for applicability

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment



Sensitivity and Uncertainty Analysis of Commercial Reactor Criticals for Burnup Credit

Office of Nuclear Regulatory Research

ORNL/TM-2005/48

OAK RIDGE NATIONAL LABORATORY
MANAGED BY UT-BATTELLE FOR THE DEPARTMENT OF ENERGY

Evaluation of Cross-Section Sensitivities in Computing Burnup Credit Fission Product Concentrations

FCSS ISG-10, Rev. 0 - 1 -

ISG-10 (2005)
$c_k \geq 0.95$ recommended

Justification for Minimum Margin of Subcriticality for Safety

Prepared by

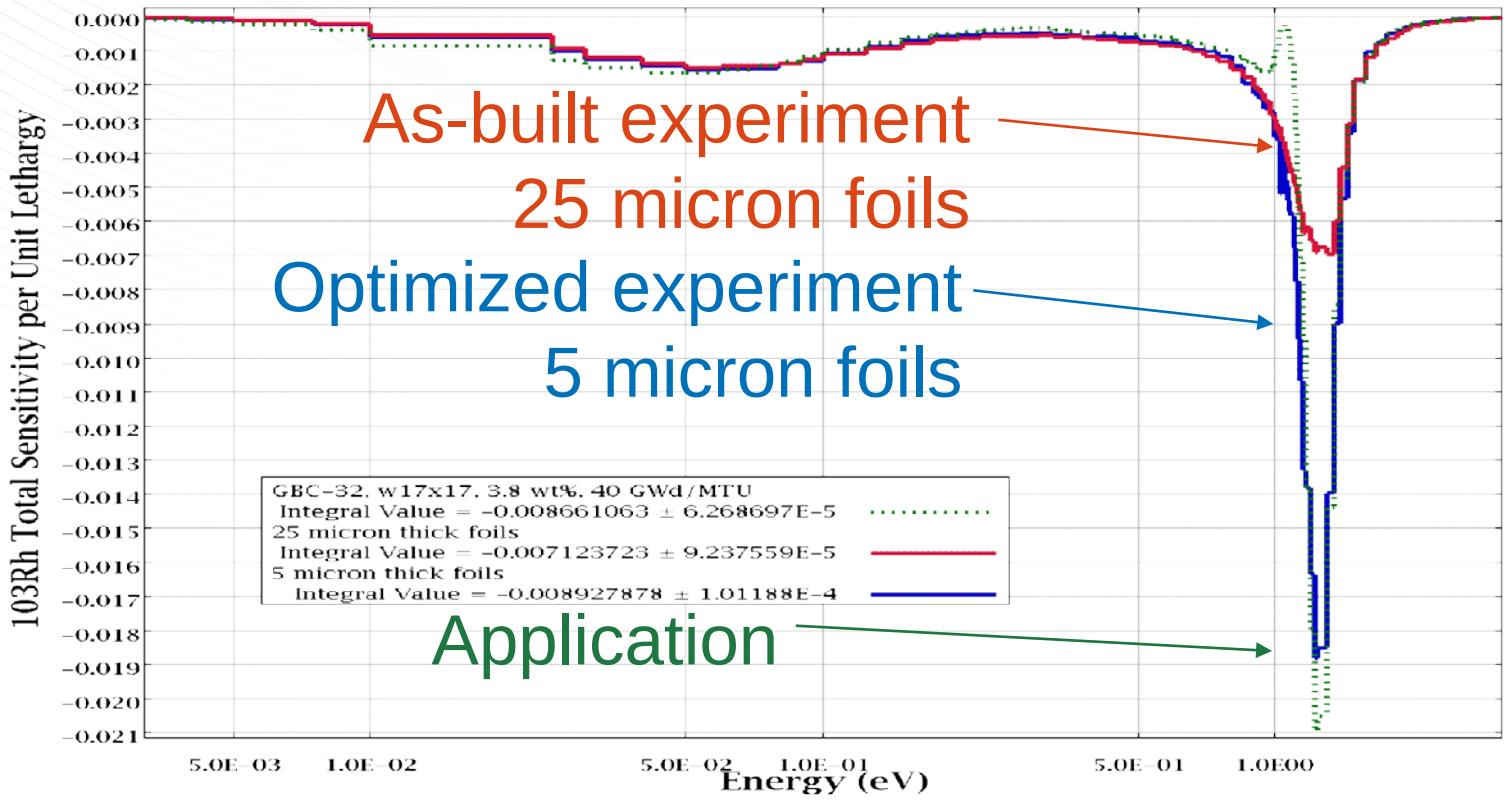
Division of Fuel Cycle Safety and Safeguards

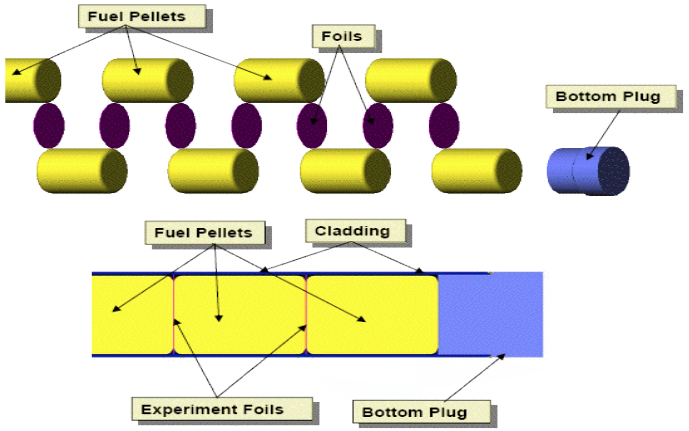
Office of Nuclear Material Safety and Safeguards

Issue

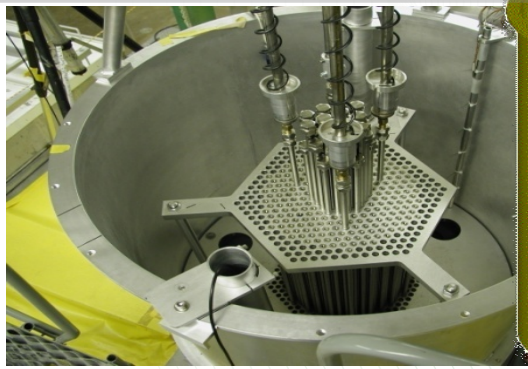
Technical justification for the selection of the minimum margin of subcriticality for safety for fuel cycle facilities, as required by 10 CFR 70.61(d)

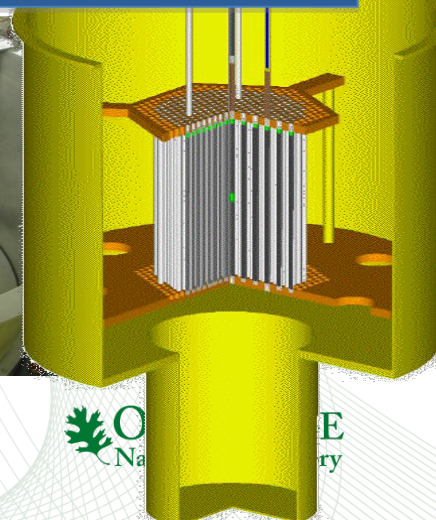
S/U methods used to help design new experiments in US and France





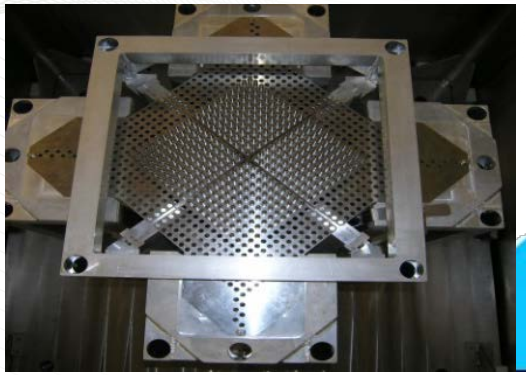
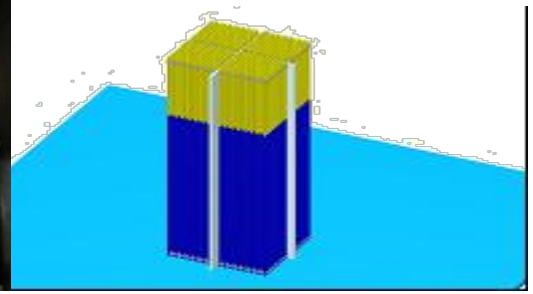
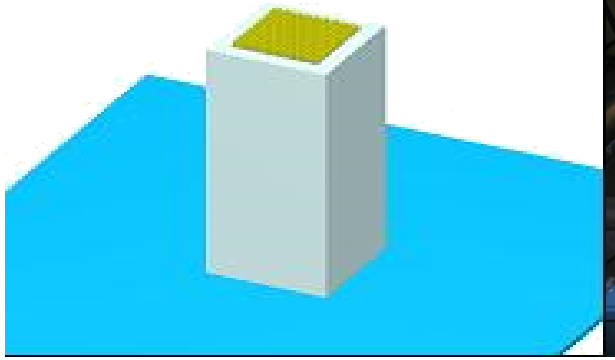
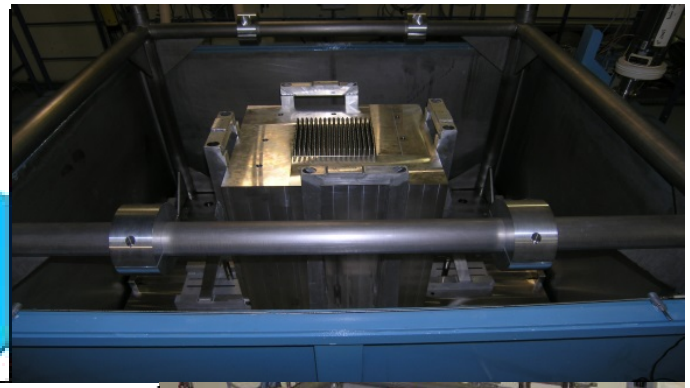
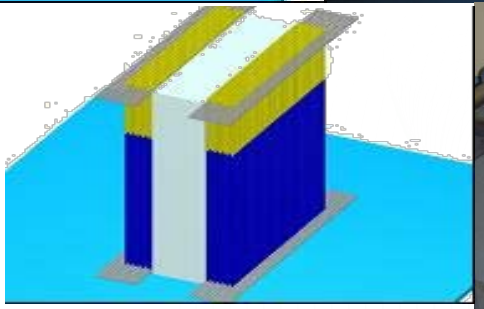
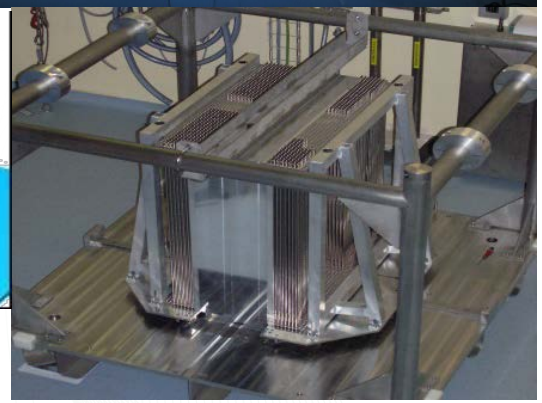
Rh-103 Critical Experiment Design for Burnup Credit





Known Unknown → Known Known

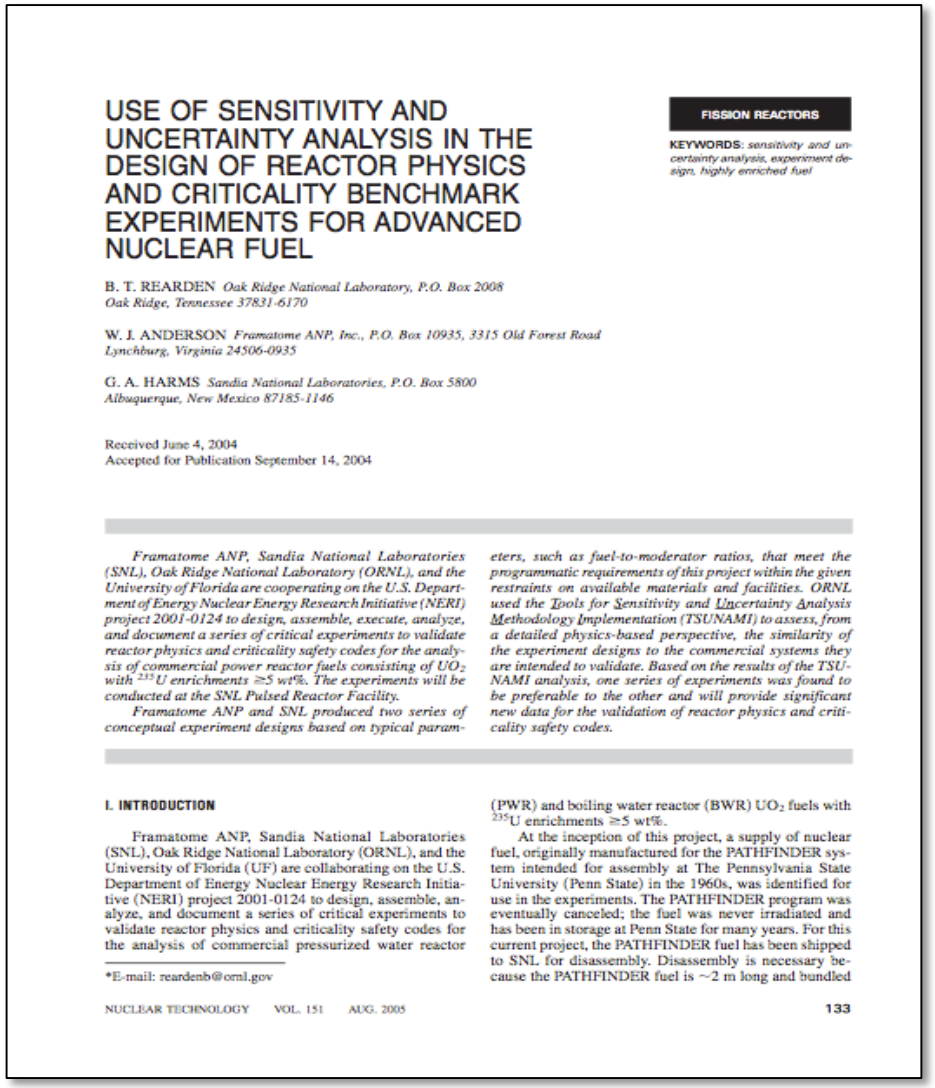
Application to design of MIRTE reference experiments

- Design of reference experiments (without material)
  - ↳ Need to optimize the number of reference experiments (to perform reproducibility exp. for uncertainty treatment)
  - Studies performed with SCALE 5.1
    - KENO V.A calculations for reference experiments design (criticality)
      - Keep lattices dimensions and reduce critical water height
      - Keep critical water height and reduce lattices dimensions
    - TSUNAMI calculations to obtain sensitivity coefficients
      - Comparison of sensitivity profiles for Uranium cross sections between experiments with and without material

Experiment design

- Required step(s) in NCSP $C_e dT$ process:



USE OF SENSITIVITY AND UNCERTAINTY ANALYSIS IN THE DESIGN OF REACTOR PHYSICS AND CRITICALITY BENCHMARK EXPERIMENTS FOR ADVANCED NUCLEAR FUEL

FISSION REACTORS

KEYWORDS: *sensitivity and uncertainty analysis, experiment design, highly enriched fuel*

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Received June 4, 2004
Accepted for Publication September 14, 2004

Framatome ANP, Sandia National Laboratories (SNL), Oak Ridge National Laboratory (ORNL), and the University of Florida are cooperating on the U.S. Department of Energy Nuclear Energy Research Initiative (NERI) project 2001-0124 to design, assemble, execute, analyze, and document a series of critical experiments to validate reactor physics and criticality safety codes for the analysis of commercial power reactor fuels consisting of UO_2 with ^{235}U enrichments ≥5 wt%. The experiments will be conducted at the SNL Pulsed Reactor Facility.

Framatome ANP and SNL produced two series of conceptual experiment designs based on typical parameters, such as fuel-to-moderator ratios, that meet the programmatic requirements of this project within the given restraints on available materials and facilities. ORNL used the Tools for Sensitivity and Uncertainty Analysis Methodology Implementation (TSUNAMI) to assess, from a detailed physics-based perspective, the similarity of the experiment designs to the commercial systems they are intended to validate. Based on the results of the TSUNAMI analysis, one series of experiments was found to be preferable to the other and will provide significant new data for the validation of reactor physics and criticality safety codes.

I. INTRODUCTION

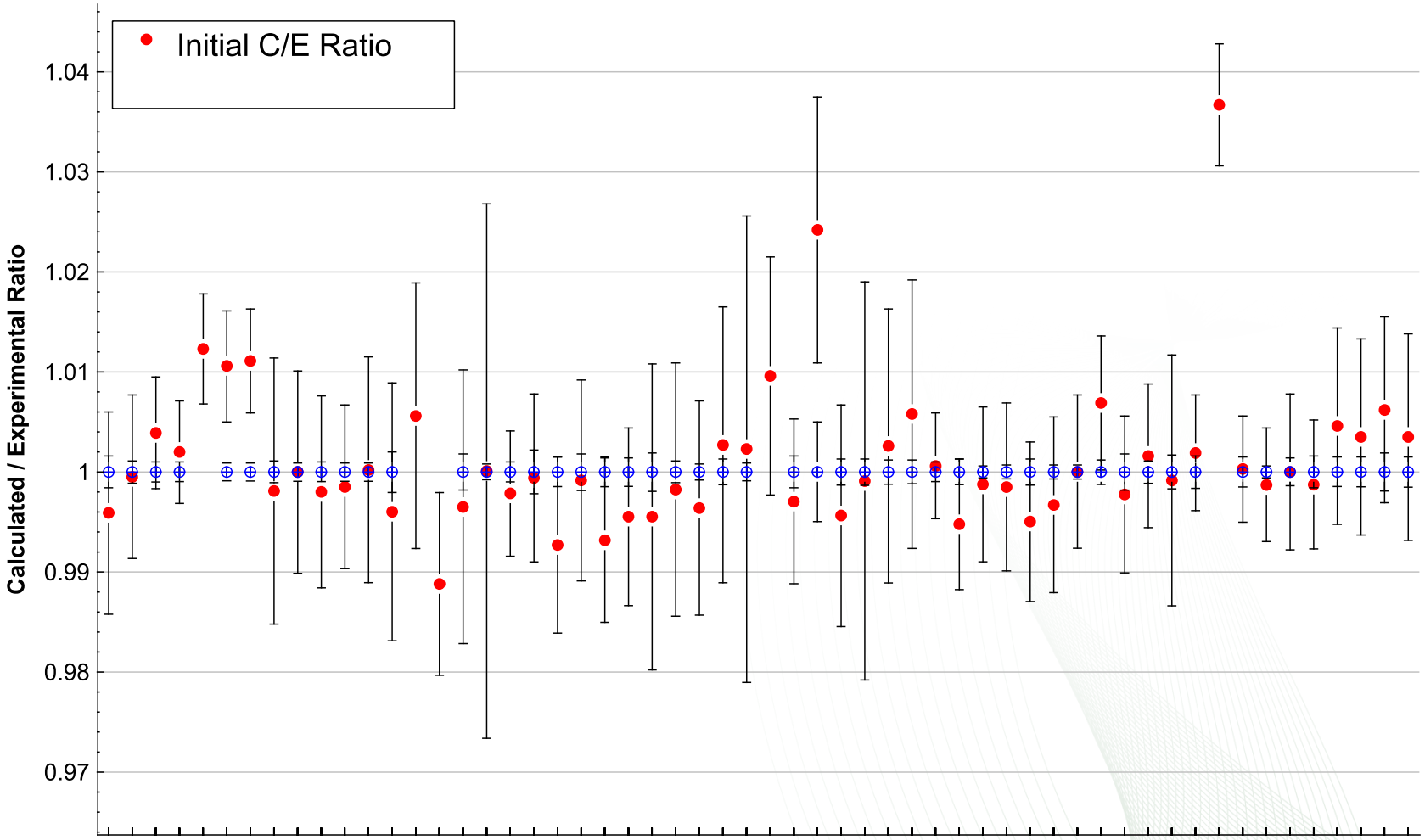
Framatome ANP, Sandia National Laboratories (SNL), Oak Ridge National Laboratory (ORNL), and the University of Florida (UF) are collaborating on the U.S. Department of Energy Nuclear Energy Research Initiative (NERI) project 2001-0124 to design, assemble, analyze, and document a series of critical experiments to validate reactor physics and criticality safety codes for the analysis of commercial pressurized water reactor (PWR) and boiling water reactor (BWR) UO_2 fuels with ^{235}U enrichments ≥5 wt%.

At the inception of this project, a supply of nuclear fuel, originally manufactured for the PATHFINDER system intended for assembly at The Pennsylvania State University (Penn State) in the 1960s, was identified for use in the experiments. The PATHFINDER program was eventually canceled; the fuel was never irradiated and has been in storage at Penn State for many years. For this current project, the PATHFINDER fuel has been shipped to SNL for disassembly. Disassembly is necessary because the PATHFINDER fuel is ~2 m long and bundled

*E-mail: reardenb@ornl.gov

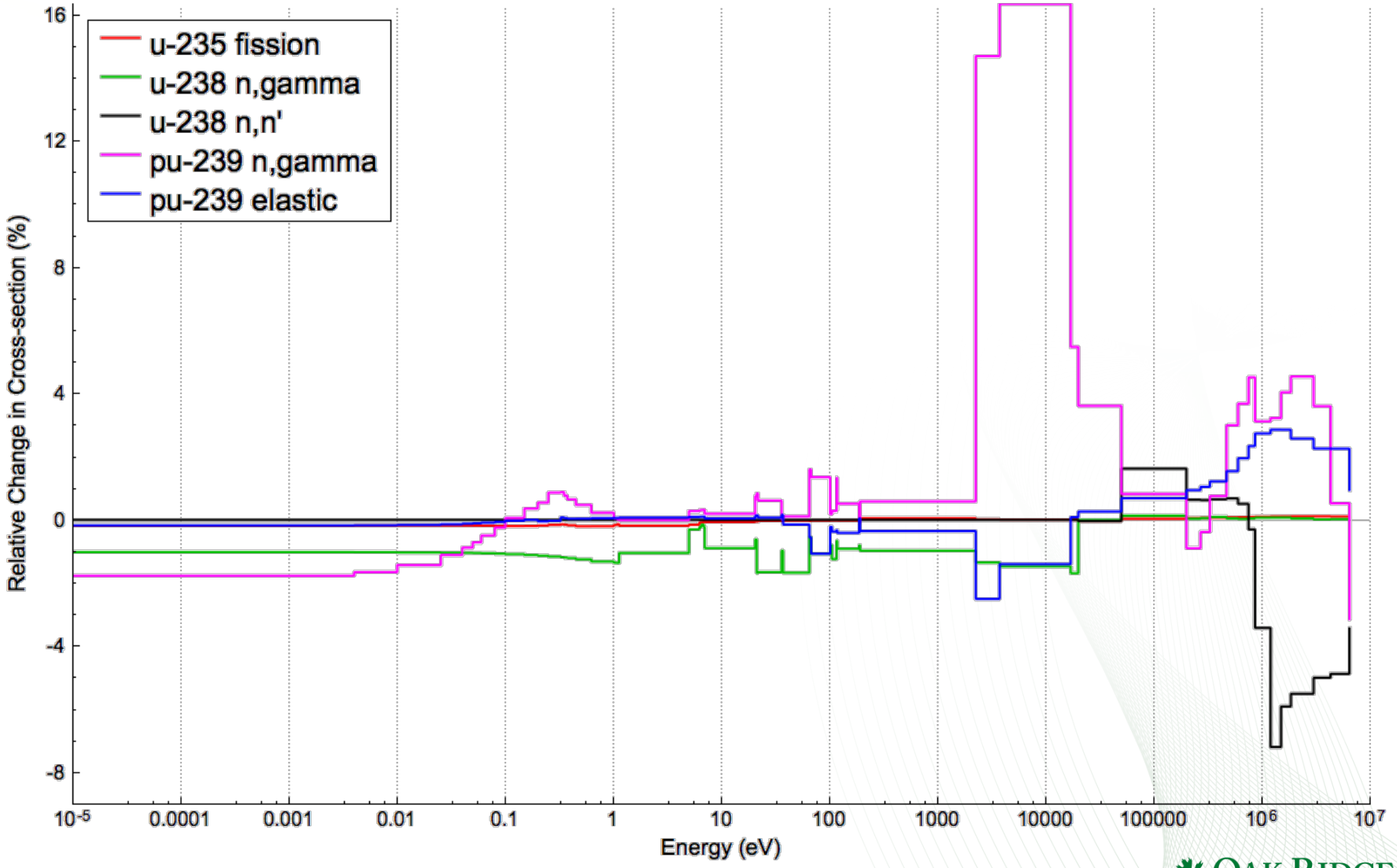
Data adjustment techniques – combining information from many different types of experiments

- Experimental benchmark data (E) is used to improve the accuracy of the initial computed responses (C).
- This assimilation consistently adjusts the underlying nuclear data.
- Inconsistent experiments are systematically removed from analysis.

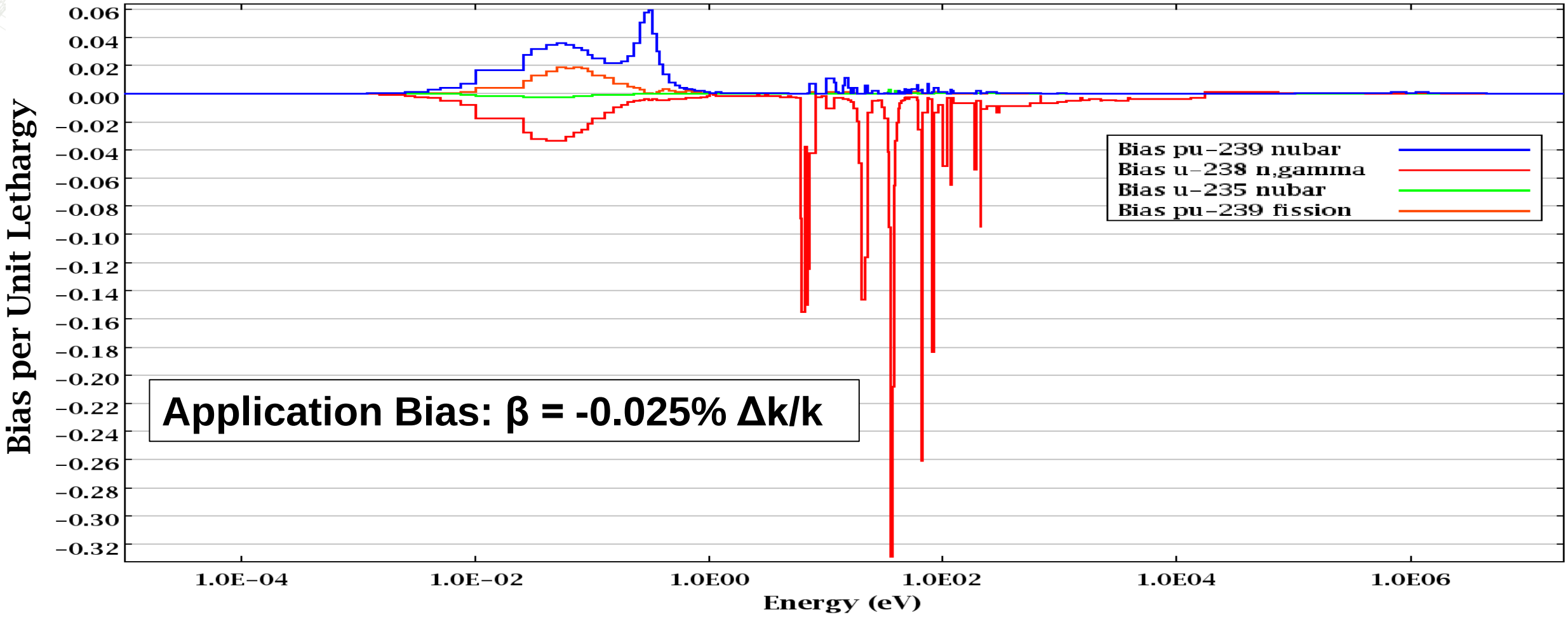


Cross section adjustments for consistent C/E for all benchmark experiments

- Experimental benchmark data (E) is used to improve the accuracy of the initial computed responses (C).
- This assimilation consistently adjusts the underlying nuclear data.



System bias is computed as well as energy-dependent bias is estimated for each nuclide and reaction



Inclusion of integral experiments reduces uncertainty in nuclear data

- Original Uncertainty is:
  0.520% Δk/k
- Adjusted Uncertainty is:
  0.119% Δk/k
- Interpretation: ~80% of uncertainty is quantified through validation with experiments.
- Remaining uncertainty highlights gaps in available validation data.

NUCLIDE	REACTION	CONTRIBUTION TO BIAS % dk/k
u-238	n,gamma	-2.1084E-01
pu-239	nubar	1.2761E-01
pu-239	fission	3.9872E-02
o-16	elastic	3.2243E-02
pu-239	n,gamma	-2.5810E-02
pu-239	chi	1.0248E-02
u-235	chi	2.9940E-04
fe-56	n,gamma	1.7158E-02
u-235	fission	-1.2351E-02
pu-240	n,gamma	-1.3162E-02
u-238	elastic	2.7715E-03
u-235	n,gamma	1.0599E-03
h-1	elastic	2.7348E-03
u-238	n,n'	-6.8963E-03
u-235	nubar	-4.1298E-03
fe-56	elastic	-6.0079E-03
h-1	n,gamma	4.1893E-03
u-238	nubar	3.1408E-03

Regulatory basis for fission product burnup credit

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

NUREG/CR-7108
ORNL/TM-2011/509

An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses—Isotopic Composition Predictions

Office of Nuclear Regulatory Research

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An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses—Criticality (k_{eff}) Predictions

Division of Spent Fuel Storage and Transportation
Interim Staff Guidance - 8
Revision 3

September 2012

Issue: Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transportation and Storage Casks

Introduction:

Title 10 of the Code of Federal Regulations (10 CFR) Part 71, *Packaging and Transportation of Radioactive Material*,[1] and 10 CFR Part 72, *Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste*,[2] require that spent nuclear fuel (SNF) remain subcritical in transportation and storage, respectively. Unirradiated reactor fuel has a well-specified nuclide composition that provides a straightforward and bounding approach to the criticality safety analysis of transportation and storage systems. As the fuel is irradiated in the reactor, the nuclide composition changes and, ignoring the presence of burnable poisons, this composition change will cause the reactivity of the fuel to decrease. Allowance in the criticality safety analysis for the decrease in fuel reactivity resulting from irradiation is termed burnup credit. Extensive investigations have been performed both within the United States and by other countries in an effort to understand and document the technical issues related to the use of burnup credit.

Expert Group on Uncertainty Analysis for Criticality Safety Assessment (UACSA)

- Expert group under OECD Nuclear Energy Agency Working Party on Nuclear Criticality Safety since 2007
- ~40 participants from ~15 nations
- Phase I – Comparison of validation methods
- Phase II – Assessment of technological uncertainties
- Phase III – Computation of sensitivity coefficients
- Phase IV – Integral experiment correlation data
- Phase V – Validation of MOX powder systems

Challenges for S/U-based validation techniques

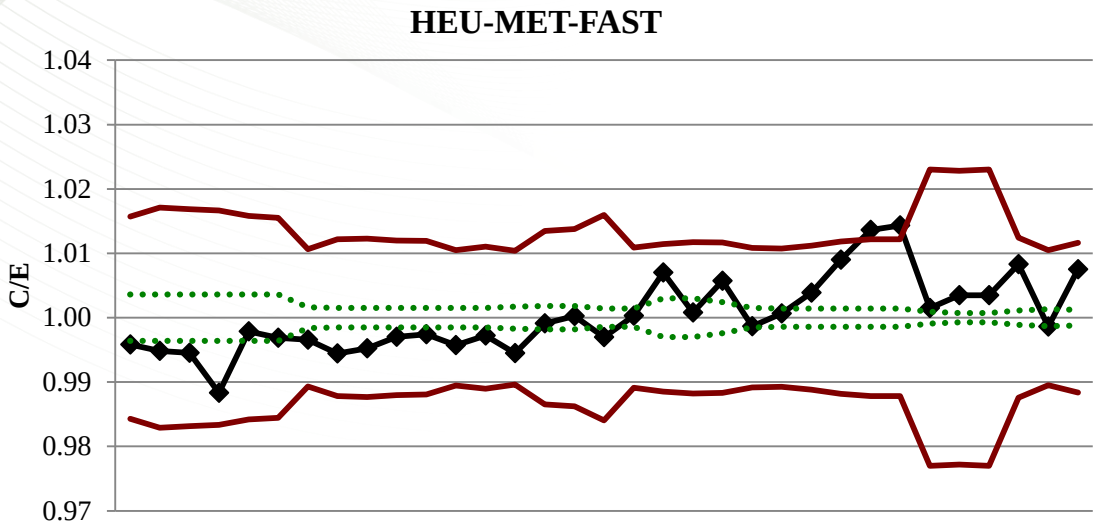
- Covariance data gaps and inconsistencies
- Inconsistent experimental uncertainty estimates and lack of correlations between experiments

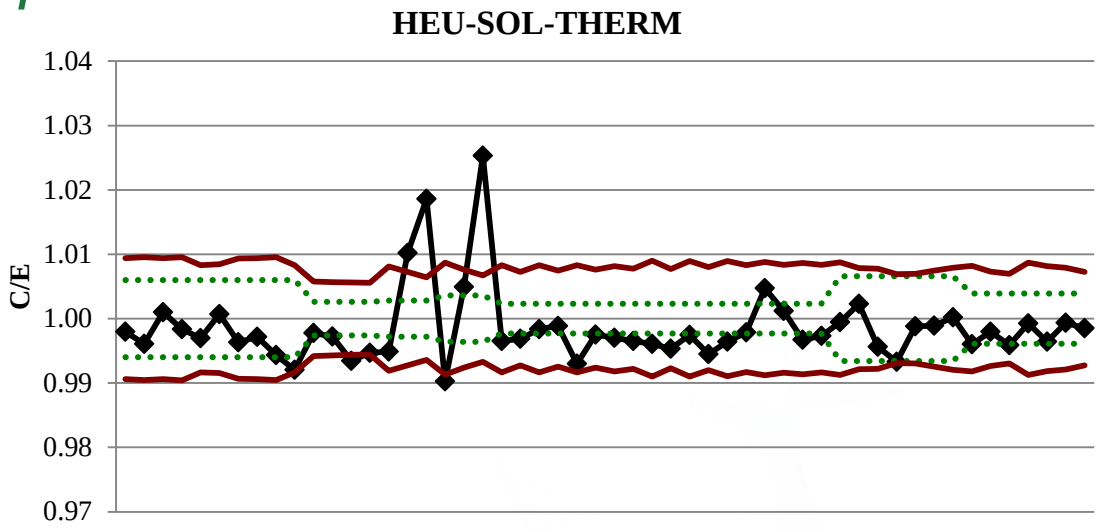
Covariance data testing

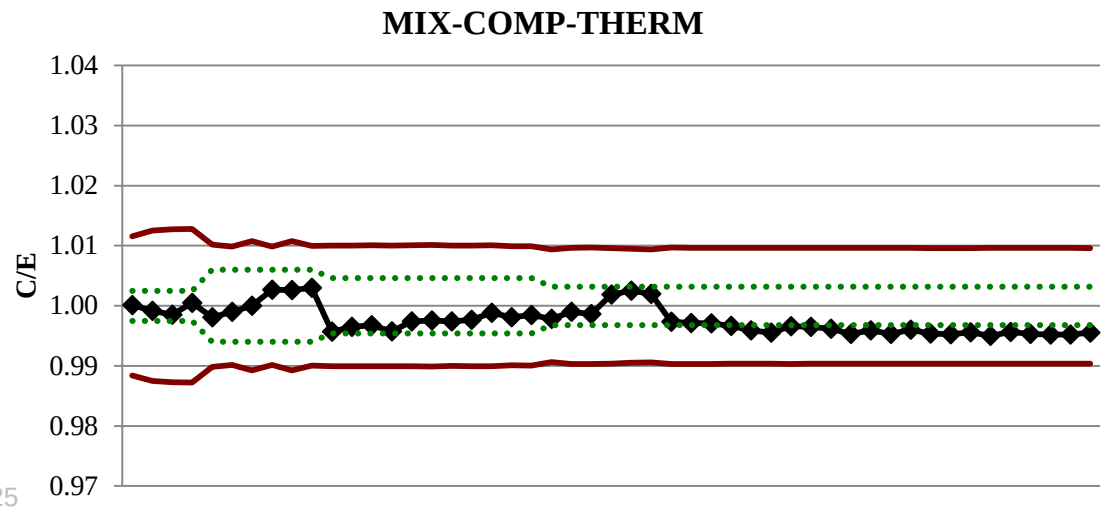
For CSEWG and new OECD/NEA WPEC SG-44

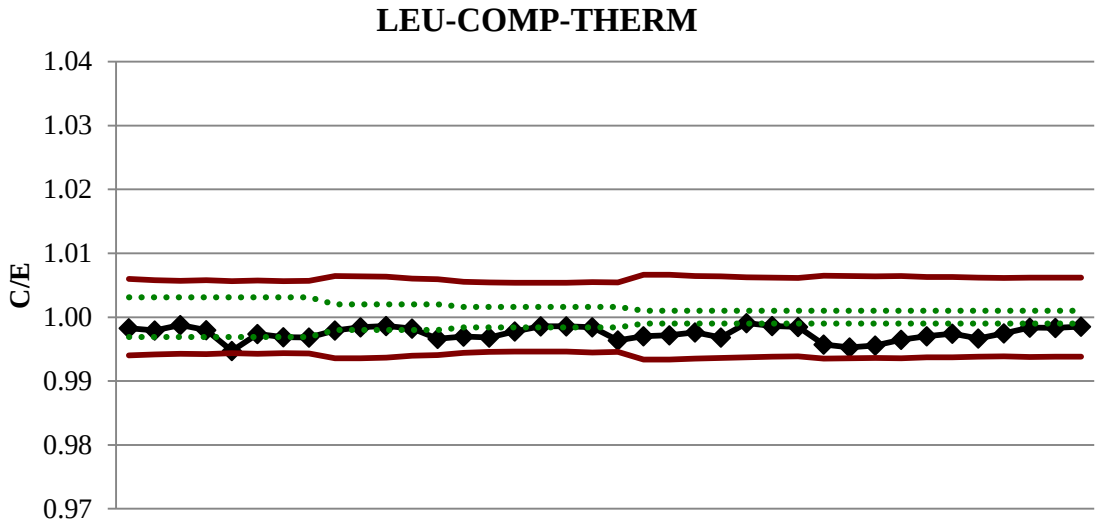
Investigation of Covariance Data in General Purpose Nuclear Data Libraries

Computational Bias
Experimental Uncertainty
Cross-section Uncertainty







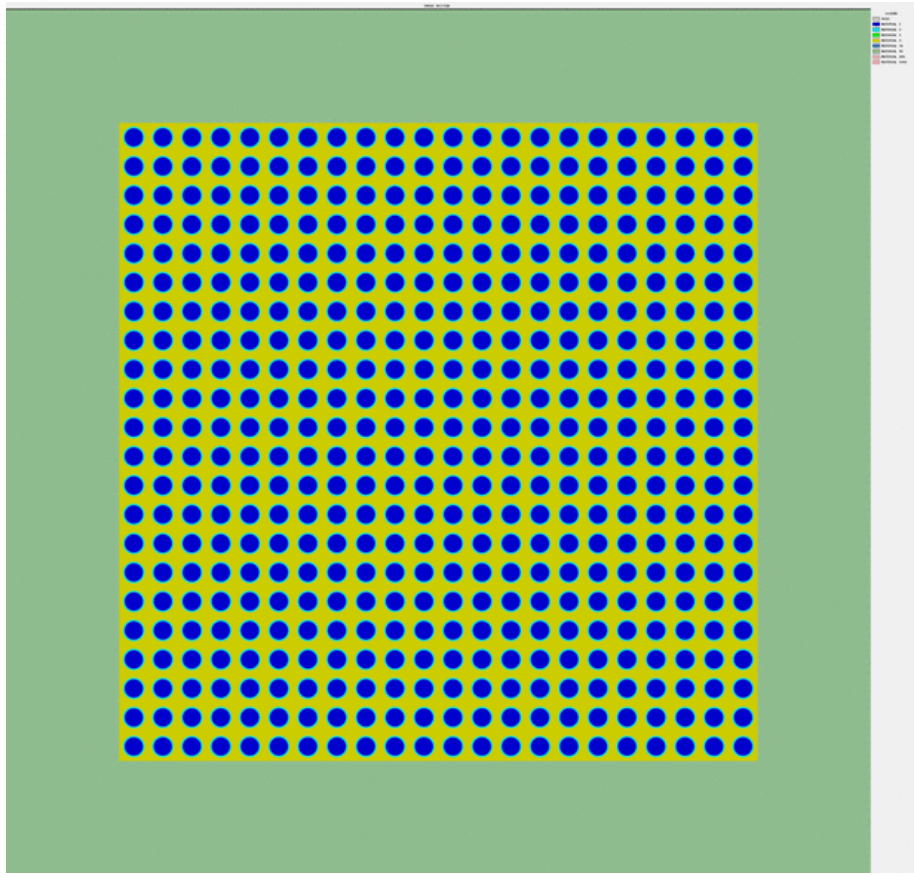


Different interpretations of experimental uncertainty

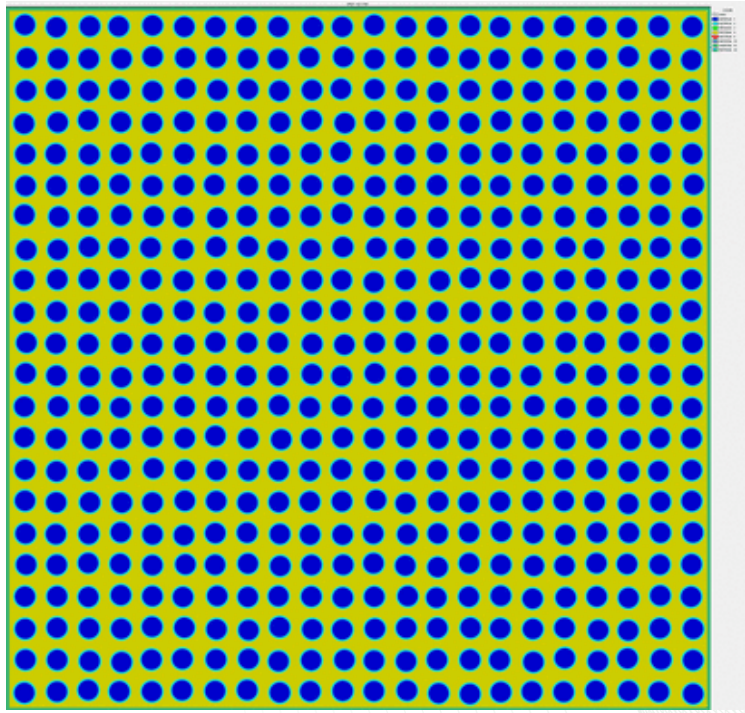
Rod spacing in LEU-COM-THERM-007 Case 1

Slideshow view to see animations!

Highly correlated

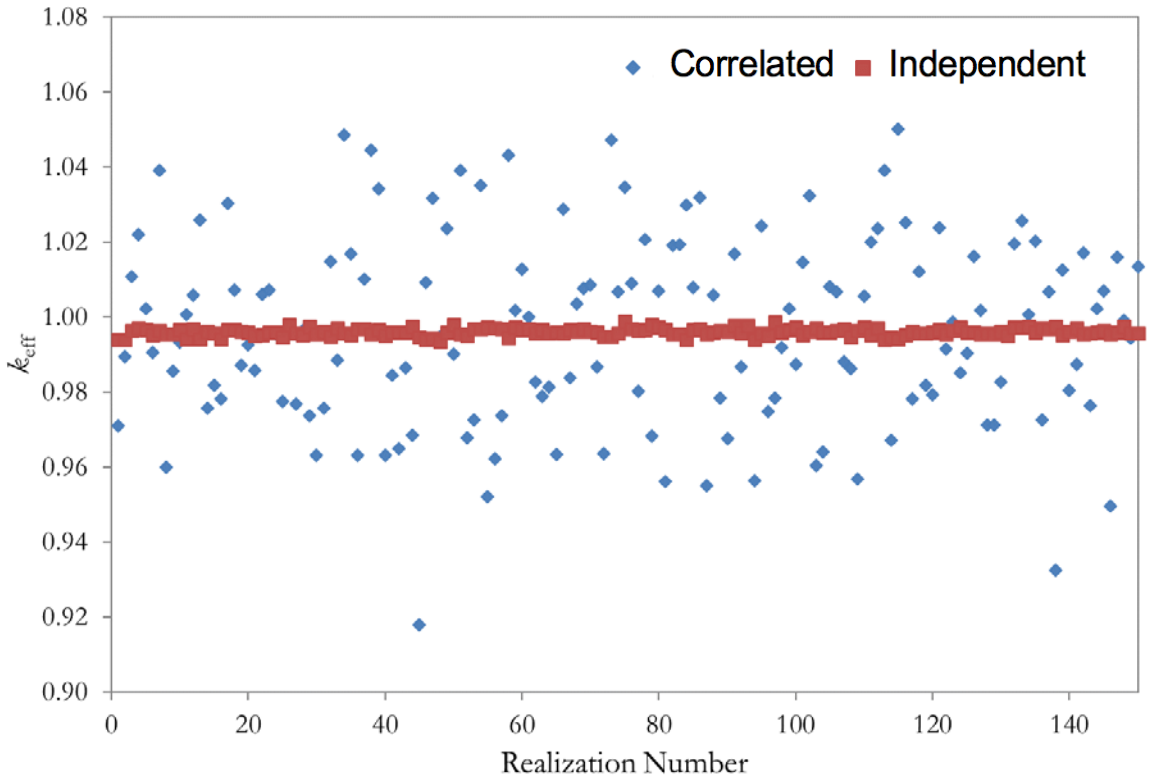


Mostly independent



Comparison of experimental uncertainty results

ICSBEP Evaluation			SCALE 6.2		Highly Correlated	Mostly Independent
Case	Expected k_{eff}	Experimental Uncertainty	Nominal k_{eff}	Nominal k_{eff} Uncert.	Sampled Standard Deviation	Sampled Standard Deviation
LCT-700-001	1.00000	0.00140	0.99392	0.00010	0.00461	0.00029



ICSBEP: 0.14%

Highly Correlated: 0.46%

Mostly Independent: 0.029%

W. J. Marshall, *Determination of Critical Experiment Correlations Via the Monte Carlo Sampling Technique,* The University of Tennessee (2017) (in review)

Correlations in critical experiments with different methods

Highly correlated

	7-1	7-2	7-3	39-1	39-2	39-3	39-4	39-5	39-6	39-7	39-8	39-9	39-10	39-11	39-12	39-13	39-14	39-15	39-16	39-17
7-1	1.000	0.933	0.391	0.978	0.975	0.974	0.974	0.956	0.957	0.974	0.971	0.978	0.969	0.977	0.972	0.980	0.979	0.973	0.977	0.978
7-2	0.933	1.000	0.557	0.923	0.920	0.925	0.930	0.925	0.929	0.933	0.920	0.936	0.940	0.925	0.924	0.928	0.933	0.928	0.937	0.931
7-3	0.391	0.557	1.000	0.405	0.390	0.409	0.417	0.459	0.463	0.415	0.389	0.434	0.451	0.384	0.406	0.405	0.382	0.399	0.418	0.420
39-1	0.978	0.923	0.405	1.000	0.978	0.970	0.973	0.957	0.958	0.976	0.972	0.970	0.973	0.979	0.976	0.976	0.981	0.977	0.972	0.977
39-2	0.975	0.920	0.390	0.978	1.000	0.972	0.970	0.953	0.954	0.975	0.967	0.968	0.963	0.974	0.974	0.976	0.977	0.970	0.972	0.977
39-3	0.974	0.925	0.409	0.970	0.972	1.000	0.967	0.945	0.954	0.971	0.963	0.971	0.966	0.974	0.969	0.974	0.971	0.967	0.972	0.970
39-4	0.974	0.930	0.417	0.973	0.970	0.967	1.000	0.956	0.954	0.971	0.965	0.968	0.965	0.972	0.971	0.973	0.974	0.968	0.973	0.978
39-5	0.956	0.925	0.459	0.957	0.953	0.945	0.956	1.000	0.946	0.958	0.944	0.955	0.955	0.953	0.949	0.952	0.952	0.954	0.951	0.958
39-6	0.957	0.929	0.463	0.958	0.954	0.954	0.954	0.946	1.000	0.956	0.953	0.960	0.961	0.955	0.954	0.963	0.955	0.957	0.958	0.960
39-7	0.974	0.933	0.415	0.976	0.975	0.971	0.971	0.958	0.956	1.000	0.973	0.974	0.970	0.979	0.978	0.974	0.979	0.974	0.979	0.978
39-8	0.971	0.920	0.389	0.972	0.967	0.963	0.965	0.944	0.953	0.973	1.000	0.964	0.970	0.973	0.966	0.970	0.973	0.964	0.964	0.966
39-9	0.978	0.936	0.434	0.970	0.968	0.971	0.968	0.955	0.960	0.974	0.964	1.000	0.967	0.976	0.968	0.976	0.976	0.969	0.975	0.974
39-10	0.969	0.940	0.451	0.973	0.963	0.966	0.965	0.955	0.961	0.970	0.970	0.967	1.000	0.964	0.968	0.969	0.966	0.964	0.968	0.970
39-11	0.977	0.925	0.384	0.979	0.974	0.974	0.972	0.953	0.955	0.979	0.973	0.976	0.964	1.000	0.973	0.980	0.979	0.977	0.977	0.977
39-12	0.972	0.924	0.406	0.976	0.974	0.969	0.971	0.949	0.954	0.978	0.966	0.968	0.968	0.973	1.000	0.978	0.976	0.968	0.972	0.976
39-13	0.980	0.928	0.405	0.976	0.976	0.974	0.973	0.952	0.963	0.974	0.970	0.976	0.969	0.980	0.978	1.000	0.977	0.979	0.976	0.976
39-14	0.979	0.933	0.382	0.981	0.977	0.971	0.974	0.952	0.955	0.979	0.973	0.976	0.966	0.979	0.976	0.977	1.000	0.976	0.977	0.979
39-15	0.973	0.928	0.399	0.977	0.970	0.967	0.968	0.954	0.957	0.974	0.964	0.969	0.964	0.977	0.968	0.979	0.976	1.000	0.970	0.973
39-16	0.977	0.937	0.418	0.972	0.972	0.972	0.973	0.951	0.958	0.979	0.964	0.975	0.968	0.977	0.972	0.976	0.977	0.970	1.000	0.976
39-17	0.978	0.931	0.420	0.977	0.977	0.970	0.978	0.958	0.960	0.978	0.966	0.974	0.970	0.977	0.976	0.976	0.979	0.973	0.976	1.000

Pitch sampled: all pitches are the same and are the same for all cases

Coefficients range from 0.96 to 0.98 (For cases with the same pitch)

Mostly independent

	7-1	7-2	7-3	39-1	39-2	39-3	39-4	39-5	39-6	39-7	39-8	39-9	39-10	39-11	39-12	39-13	39-14	39-15	39-16	39-17
7-1	1.000	0.034	0.023	0.012	0.005	-0.040	0.069	-0.009	0.071	0.067	0.082	0.088	0.049	0.044	0.042	0.063	0.088	0.139	-0.021	0.082
7-2	0.034	1.000	0.074	-0.045	0.020	0.040	0.181	0.086	0.065	0.041	-0.028	-0.034	0.009	-0.030	0.018	0.047	-0.041	0.023	0.061	-0.028
7-3	0.023	0.074	1.000	0.118	0.063	0.094	0.061	0.086	0.201	0.079	0.100	0.134	0.047	0.091	0.012	0.125	0.050	0.117	0.172	0.055
39-1	0.012	-0.045	0.118	1.000	0.121	0.138	0.076	0.071	0.124	0.034	0.100	0.085	0.135	0.023	0.037	0.037	0.087	0.083	0.115	0.149
39-2	0.005	0.020	0.063	0.121	1.000	0.034	0.075	0.037	0.130	0.041	0.055	0.049	0.009	0.025	0.095	0.100	-0.050	0.124	-0.003	0.115
39-3	-0.040	0.040	0.094	0.138	0.034	1.000	0.079	0.077	0.044	0.007	0.048	-0.064	0.145	0.076	0.061	0.090	0.067	0.059	0.088	0.116
39-4	0.069	0.181	0.061	0.076	0.075	0.079	1.000	-0.051	0.090	-0.012	-0.017	0.036	0.026	-0.021	0.034	0.088	0.042	-0.004	0.025	-0.018
39-5	-0.009	0.086	0.086	0.071	0.037	0.077	-0.051	1.000	0.138	0.081	0.043	0.140	0.112	0.059	0.085	0.131	0.184	0.001	0.161	0.093
39-6	0.071	0.065	0.201	0.124	0.130	0.044	0.090	0.138	1.000	0.103	-0.014	0.035	0.149	0.051	0.062	0.116	0.013	0.074	0.153	0.127
39-7	0.067	0.041	0.079	0.034	0.041	0.007	-0.012	0.081	0.103	1.000	0.131	0.007	0.004	0.024	-0.003	0.111	0.053	0.081	0.173	0.035
39-8	0.082	-0.028	0.100	0.100	0.055	0.048	-0.017	0.043	-0.014	0.131	1.000	-0.067	0.047	-0.016	0.063	0.004	0.030	0.013	0.050	0.070
39-9	0.088	-0.034	0.134	0.085	0.049	-0.064	0.036	0.140	0.035	0.007	-0.067	1.000	0.082	0.041	0.070	0.000	0.046	-0.081	-0.009	0.077
39-10	0.049	0.009	0.047	0.135	0.009	0.145	0.026	0.112	0.149	0.004	0.047	0.082	1.000	0.080	0.069	-0.004	0.041	0.115	0.119	0.047
39-11	0.044	-0.030	0.091	0.023	0.025	0.076	-0.021	0.059	0.051	0.024	-0.016	0.041	0.080	1.000	0.115	0.022	-0.087	-0.048	0.112	0.046
39-12	0.042	0.018	0.012	0.037	0.095	0.061	0.034	0.085	0.062	-0.003	0.063	0.070	0.069	0.115	1.000	0.132	0.112	0.006	0.065	0.069
39-13	0.063	0.047	0.125	0.037	0.100	0.090	0.088	0.131	0.116	0.111	0.004	0.000	-0.004	0.022	0.132	1.000	0.184	0.206	0.232	0.138
39-14	0.088	-0.041	0.050	0.087	-0.050	0.067	0.042	0.184	0.013	0.053	0.030	0.046	0.041	-0.087	0.112	0.184	1.000	0.148	0.051	0.204
39-15	0.139	0.023	0.117	0.083	0.124	0.059	-0.004	0.001	0.074	0.081	0.013	-0.081	0.115	-0.048	0.006	0.206	0.148	1.000	0.090	0.037
39-16	-0.021	0.061	0.172	0.115	-0.003	0.088	0.025	0.161	0.153	0.173	0.050	-0.009	0.119	0.112	0.065	0.232	0.051	0.090	1.000	-0.023
39-17	0.082	-0.028	0.055	0.149	0.115	0.116	-0.018	0.093	0.127	0.035	0.070	0.077	0.047	0.046	0.069	0.138	0.204	0.037	-0.023	1.000

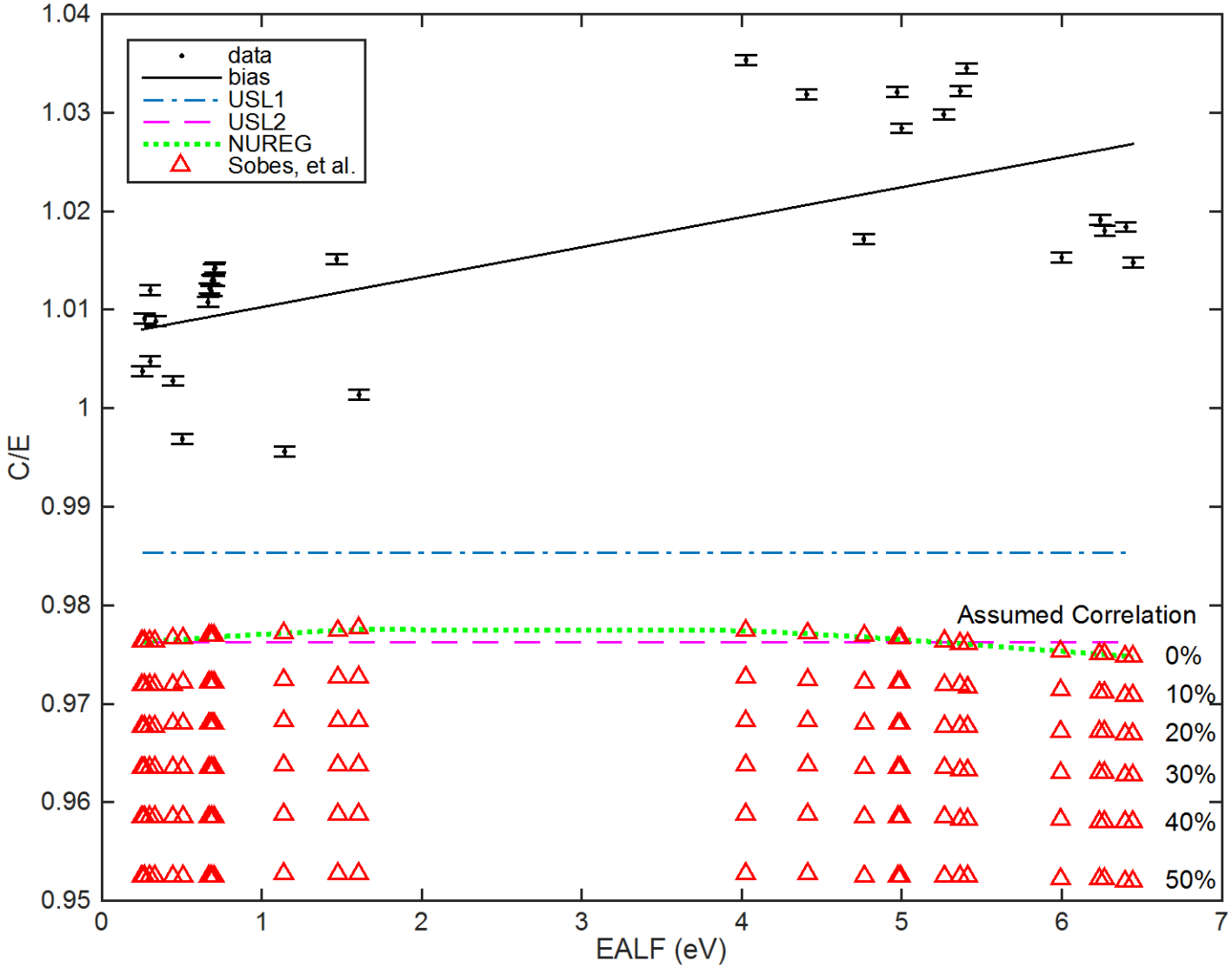
All fuel rod positions are sampled independently and differently in each case

Coefficients range from ~0 to ~0.23

Fuel rod position modeling makes a difference!

W. J. Marshall and B. T. Rearden, "Determination of Critical Experiment Correlations for Experiments Involving Arrays of Low-Enriched Fuel Rods," *Proc. of ANS NCSD 2017 - Criticality safety - pushing the boundaries by modernizing and integrating data, methods, and regulations*, Carlsbad, NM, September 10–15, 2017.

Effect of experimental correlations on upper subcritical limit



- Using data from a previous transportation package criticality safety assessment, the inclusion of experimental correlations impacts the USL by as much as **3% Δk/k**

V. Sobes, B. T. Rearden, D. E. Mueller, W. J. Marshall, J. M. Scaglione, and M. E. Dunn, "Upper Subcritical Calculations Based on Correlated Data," ICNC 2015 – International Conference on Nuclear Criticality Safety, Charlotte, NC, September 13–17, 2015.

Comments on advanced methods for criticality safety assessment

- Provides systematic approach for interpolating and extrapolating among existing experiments.
- Allows for combining information from many diverse experiments.
- Extracts and projects bias information from replacement experiments.
- Provides margin in gaps where experiments are not available.
- Assists in design of new experiments targeted to meet application needs.
- Tools are readily available for production use.
- Challenges still exist in uncertainties and correlations in benchmark experiments and nuclear data.

Backup Slides

SCALE Tools for Sensitivity and Uncertainty Analysis Methodology Implementation (TSUNAMI)

- Understanding the Sensitivity of a System to Different Processes:
  TSUNAMI-1D/2D/3D
- Nuclear Data Uncertainty Propagation:
  TSUNAMI-1D/2D/3D or **TSUNAMI-IP**
- Identifying Neutronically Similar Systems:
  TSUNAMI-IP
- Estimating Computational Biases:
  TSUNAMI-IP or **TSURFER**
- Adjusting Cross Section Data by Assimilating the Results of Irradiation Experiments:
  TSURFER
- Sensitivity Analysis for Replacement Experiments:
  TSAR

MCNP Whisper methodology

- Non-parametric, extreme-value theory method for determining USL's and margins of subcriticality (MOS).
  - Developed by Kiedrowski at LANL in 2014
- Whisper USL's are designed to be conservative.
  - Adding more benchmark cases can only increase the MOS.

Whisper methodology

- Whisper weights the importance of experiments based on their similarity to the target application.

$$weight_i = \frac{\left(c_k^i - c_k^{threshold}\right)}{\left(\max(c_k) - c_k^{threshold}\right)}$$

- Whisper includes additional benchmark experiments until a cumulative weight of 25 is obtained.
  - Treatments exist for applying the Whisper method to cases with few similar benchmarks.
- Covariance data derived from multi-lab “low-fidelity” project.

Whisper methodology

- Whisper uses data assimilation methods to identify inconsistent benchmark experiments and omit them from the USL calculation.
- Whisper uses the adjusted response uncertainty to provide additional subcritical margin.
  - Performing a convergence study on the adjusted response uncertainty is helpful for Whisper analyses.

$$MOS = MOS_{software} + MOS_{data} + MOS_{application}$$

- A detailed discussion of the Whisper methodology is available in:

B.C. Kiedrowski, et. al., “Whisper: Sensitivity/Uncertainty-Based Computational Methods and Software for Determining Baseline Upper Subcritical Limits,” *Nucl. Sci. Eng.* (2015).

Propagating cross section uncertainties

- Sensitivity coefficients can be combined with cross section uncertainties to quantify the uncertainty in a response.

$$S_{k,\Sigma_x} \cdot Cov_{\Sigma_x,\Sigma_y} \cdot S^T_{k,\Sigma_y} = \sigma_k^2$$

$$\left(\frac{\delta k/k}{\delta\Sigma/\Sigma}\right) \quad \left(\frac{\Delta\Sigma}{\Sigma}\right)^2 \quad \left(\frac{\delta k/k}{\delta\Sigma/\Sigma}\right) \quad \left(\frac{\Delta k}{k}\right)^2$$

The Sandwich Equation

Benchmark similarity assessment

- The similarity coefficient, *c(k)* or c_k, describes the amount of nuclear data-induced uncertainty that is shared by two systems.

$$S_{R_1,\Sigma_x} \cdot Cov_{\Sigma_x,\Sigma_y} \cdot S^T_{R_2,\Sigma_y} = \sigma^2_{R_1,R_2}$$

$$\left(\frac{\delta R/R}{\delta\Sigma/\Sigma}\right) \qquad (\Delta\Sigma/\Sigma)^2 \qquad \left(\frac{\delta R/R}{\delta\Sigma/\Sigma}\right) \qquad (\Delta R/R)^2$$

$$c_k = \frac{\sigma^2_{R_1,R_2}}{\sigma_{R_1}\sigma_{R_2}}$$