DEPARTMENT OF ENERGY (DOE)
NUCLEAR CRITICALITY SAFETY PROGRAM (NCSP)

Nuclear Data Advisory Group (NDAG)

SUMMARY RECORD OF THE FOURTH MEETING

Brookhaven National Laboratory
03 November 2003

Prepared by:

R. M. Westfall (ORNL)
Task Manager for the
NCSP Nuclear Data Element

R. D. McKnight (ANL)
NDAG Chair

Nuclear Data Advisory Group (NDAG)

Summary Record of the Fourth Meeting

Brookhaven National Laboratory (BNL), 03 November 2003

Introduction

R. McKnight (ANL) initiated the meeting by welcoming the participants and initiated self-introductions by attendees. The meeting included 15 NDAG members and 7 observer/presenters. The list of attendees is given in Appendix A.

Adoption of the Agenda

The group discussed inclusion of several additional presentations and modified the agenda accordingly. The final agenda is included as Appendix B.

NDAG Attendance / Participation

R. McKnight (ANL) requested the group review and discuss the subject of attendance to NDAG meetings. Although attendance was to have been limited to NDAG members, the meetings have previously included other presenters/observers. Although it is important to include relevant technical presentations, it is also important not to (a) devote too great a portion of the meeting to these presentations and (b) increase the number of participants to an unworkable number. This could easily occur when NDAG meets in conjunction with CSEWG (or similar) meetings. It was agreed that the NDAG agendas will be structured to include technical presentations in the first portion. This initial portion may include presentations by invited participants. At the conclusion of this portion of the agenda, the meeting will proceed to other NDAG matters and this portion of the meeting will be restricted to NDAG members only.

It was expressed that NDAG members unable to participate should be allowed to be represented by an alternate. It was agreed that members would notify the NDAG chair if they would be represented by an alternate.

Review of Current Activities

Presentations were made on the following subjects:

a. New LANL Actinide Evaluations: M. Chadwick reviewed nuclear data activities for criticality safety performed at ORNL and LANL. (More detailed discussions of these activities followed in presentations by R. MacFarlane, T. Kawano, and L. Leal. He also reviewed the suite of new Actinide evaluations performed at LANL, pointing out the many new features now included in the modeling (see Appendix C).

b. Data Testing Results for the New Actinide Evaluations: R. MacFarlane discussed performance of the new evaluations. He not only provided results with ENDF/B-VI.8 compared with the new preliminary ENDF/B-VII, but also presented results derived both with the new ICSBEP benchmark models and with the old CSEWG (ENDF-202) models (see Appendix D). There appear to be model biases for Flattop-25. R. McKnight took an action to look the potential source of this model bias.
c. LANL Gd Isotope Evaluations: T. Kawano discussed recent work to produce covariance data for the Gd isotopes above the resonance region based on experimental data and nuclear modeling fitting with the Kalman code (see Appendix E).
d. ORNL Gd Isotope Evaluations: L. Leal presented the results of new evaluations of the covariance data for the gadolinium isotopes in the resonance energy region. These evaluations are based on experimental data and fitting with the SAMMY code (see Appendix F). It was noted that action items from the previous NDAG meeting were: for ORNL to produce Gd covariance data in the resolved resonance region; for LANL to produce Gd covariance data above the resolved resonance region; and for LANL to merge the two evaluations to cover the entire range.
e. Covariance Data: D. Naberejnev presented results of a study to judge how important is covariance data in the resolved resonance range for the applications. This study revealed significant effects when full correlations were assumed. Such an exercise will be of value when actual covariance data are available for the analyses (see Appendix G). Discussion of this work not only resurrected the "oft-repeated refrain" of the lack of real evaluated covariance data in the resonance region but also highlighted the need for tools to processing data when they are available. This led to an extremely valuable discussion/tutorial by R. MacFarlane of the scope of the difficulties in processing these data for applications. M. Greene (ORNL) indicated to the group that official ENDF formats for representation of covariance information for the resolved resonance data would be approved by the CSEWG Methods and Formats Committee during the CSEWG meeting which would immediately follow the NDAG meeting. (Note: These formats were approved by CSEWG on 04Nov 2003.)
f. NEA WPEC High Priority Request List: R. McKnight reported steps taken by the NEA Working Party on International Evaluation Cooperation to produce a more useful High Priority Request List (for evaluated nuclear data). It was noted that the new procedure for generating the HPRL will follow the NDAG model – namely, data requests promoted as High Priority will have to be supported by sensitivity analyses which quantify the benefit to be derived through reduction of the nuclear data uncertainties.
g. Monte Carlo Benchmark Calculations with Revised ^{235,238}U Evaluations: A. C. Kahler (Bettis) extracted from a lager data testing study his results based on the new LANL ^{235,238}U evaluations (see Appendix H). These results were consistent with the results presented by R. MacFarlane. It was noted that inconsistencies appear for the Pb-reflected experiments in the ORNL L5 and L6 benchmark series. B. Briggs took an action to investigate these discrepancies.

(After the final technical presentation, only NDAG members participated in the remainder of the meeting.)

Minutes of the Third NDAG Meeting April 22, 2003

Minutes of the previous NDAG meeting held at Oak Ridge National Laboratory on April 22, 2003 were distributed. The actions of that meeting were reviewed. The summary record of that meeting was discussed and accepted without revision.

Review of NCSP Budget Execution Meeting September 25-26

J. Felty (SAIC/DOE) provided a brief summary of the NCSP Budget Execution meeting in Washington, DC on September 25-26, 2003. He also discussed briefly the budget as shown below in Table 1 (taken from the October 2003 DRAFT NCSP Five-Year Plan). He pointed out that the budget growth in out-years is expected to reflect only inflationary increases.

R. McKnight asked what policy had been set by RSICC regarding single-user licenses and fees of codes which are distributed by RSICC and also whether NCSP participants were subject to these fees. [At the budget meeting in DC, H. Hunter (RSICC) discussed the security concerns requiring implementation of single-user licenses and the necessity for the user fees. Also at that meeting, several people provided Hunter with feedback from the field.] J. Felty offered to obtain details of the new policy.

R. McKnight noted that one of the subjects brought up for discussion during the CSSG portion of the September 25-26 meeting had been nominations for annual recognition of outstanding program contribution. He suggested that work of R. E. MacFarlane, P. G. Young, and M. B. Chadwick of LANL in (i) identifying the possible solution to the long-standing "^{238}U reflector bias"; (ii) producing a new ^{238}U evaluation based on that knowledge; and (iii) tightly coupling testing of the new evaluation with the evaluation process represented exceptional technical work which has resulted in elimination of that long-standing bias (of direct importance to criticality safety) and furthermore has yielded an improved evaluation methodology in coupling improved modern theoretical modeling and focused testing of the new evaluated data. McKnight suggested, and the NDAG members concurred, that this recommendation to honor this recent LANL evaluation activity be forwarded to the CSSG.

M. Westfall (ORNL) discussed briefly the NCSP review meeting presented to D. Crandall (DP) at ORNL on September 2, 2003. He indicated this meeting had been very beneficial and served to re-invigorate ORNL management support of ORELA. The NDAG discussion then questioned the new role of Kerman, Dietrich, and Lynn, acknowledge by all as extremely valuable theoretical assets (it was noted by M. Chadwick that these individuals are valuable nuclear theory consultants to LANL in other programs) within the Nuclear Data element of NCSP. The NDAG discussion on this matter concluded the contribution by these consultants would strengthen the theoretical basis for our data evaluations and that benefits to NCSO could be maximized if utilized by all of the laboratories.

Table 1: Nuclear Criticality Safety Program Base Funding, Fiscal Years 2004 – 2008

	FY 2004 ($k)	FY 2005 ($k)	FY 2006 ($k)	FY 2007 ($k)	FY 2008 ($k)
Applicable Ranges of Bounding Curves and Data	784	800	700	400	400
Analytical Methods Development and Code Maintenance	2,036	2,500	2,600	2,650	2,650
International Criticality Safety Benchmark Evaluation Project	1,760	1,900	2,000	2,100	2100
Nuclear Data	3,155	3,300	3,400	3,450	3,450
Integral Experiments	1,372	1,400	1,450	1,700	1,800
Information Preservation and Dissemination	263	270	270	270	270
Training and Qualification	225	230	230	230	230
Criticality Safety Support Group	205	226	200	200	300
TOTAL	9,800	10,626	10,850	11,000	11,200

Review of NCSP Flowchart

J. Felty also reviewed the "flowchart" showing the organization of the NCSP which had been discussed and modified slightly. These charts are shown below in Figures 1 and 2 (also taken from the October 2003 DRAFT NCSP Five-Year Plan). It was noted how much of the "clutter" had been removed from these charts – greatly improving their readability. It was suggested there might be value in the re-introduction of words to indicate input from the criticality safety applications of data needs to the chart in Figure 1 at the "Start" point (to reinforce our attempt to address needs in a Top-Down manner). Felty indicated he would consider that suggestion.

Review of NCSP Elements

J. McKamy (who was not present) had made revisions to his memo "Analysis of Nuclear Criticality Safety Technology supporting the Environmental Management Program." Copies were distributed to the NDAG members and C. Hopper (ORNL) led discussion of the contents. It was noted that priority needs identified one year ago (see List of Priority Data Needs, J. McKamy, Revision 0, August 30, 2002), including

Figure 1: How the Nuclear Criticality Safety Program Works

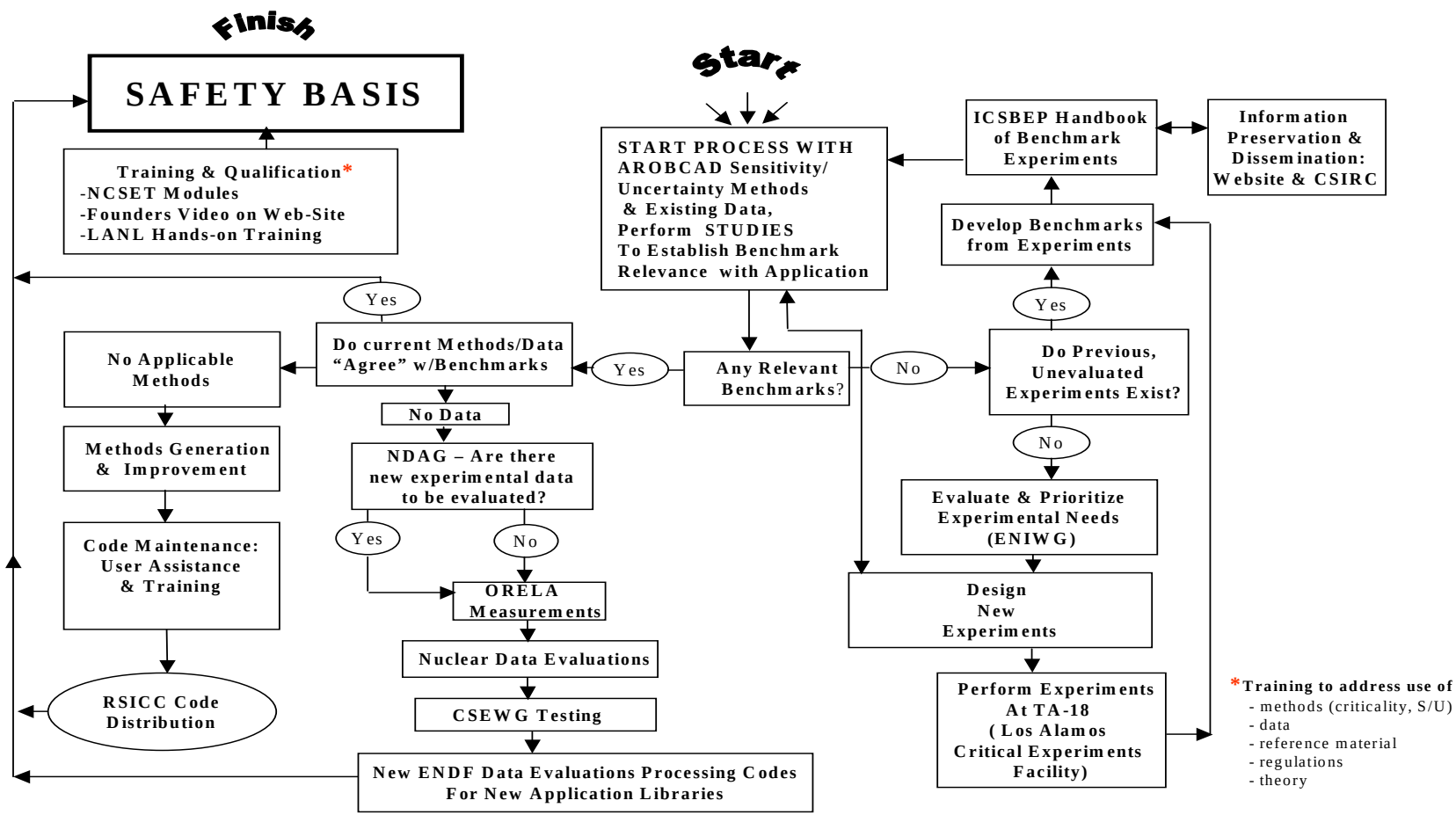
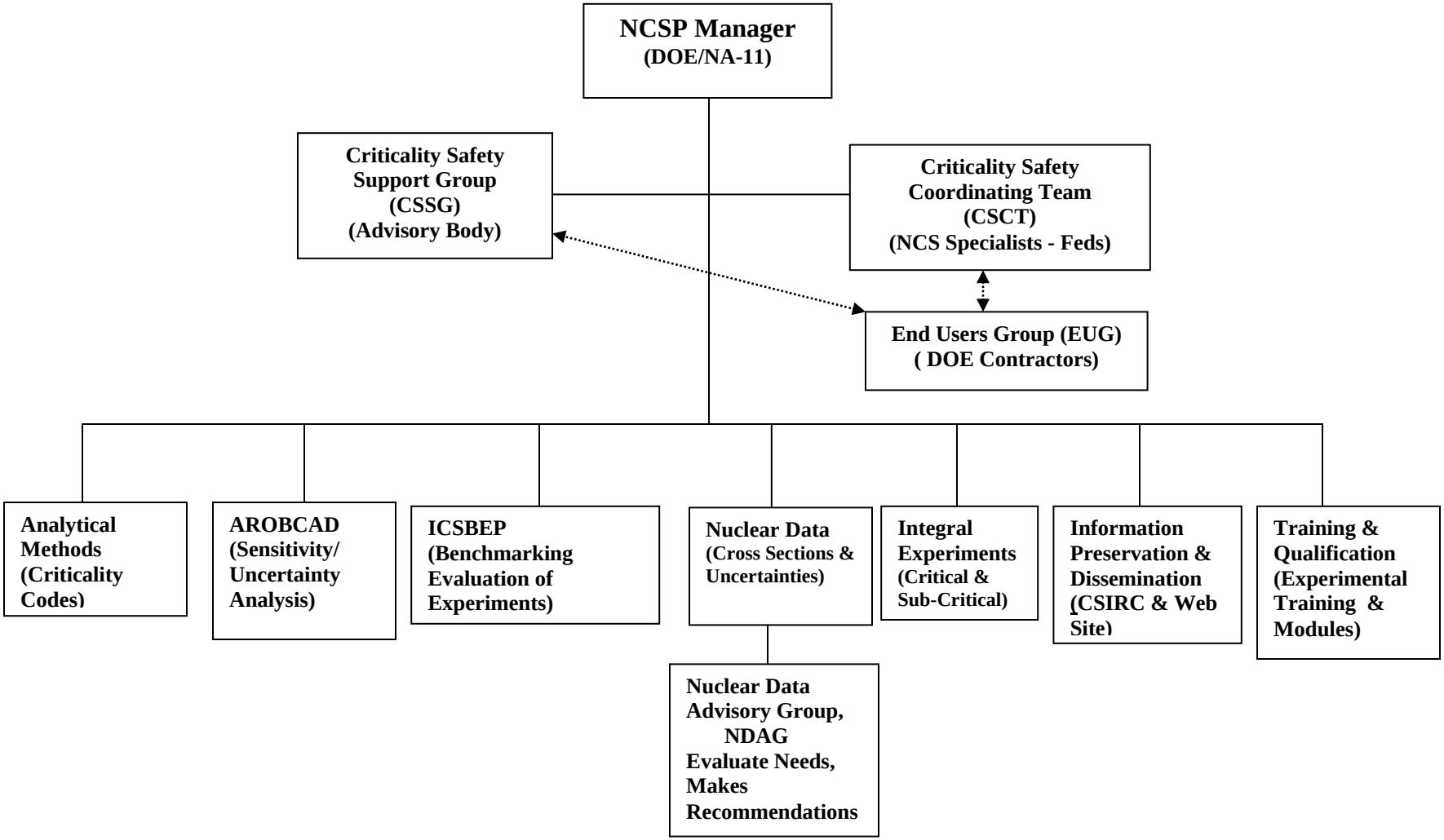


Figure 2: Nuclear Criticality Safety Program Organization



gadolinium, constituents of stainless steel, beryllium, are still of primary importance. It was noted that NASA space nuclear applications are supporting some effort in nuclear data evaluation and sensitivity analyses. The NASA effort has indicated a need for rhenium and niobium data. E. Fujita (ANL) noted that A. B. Smith (ANL) had just provided a new evaluation of rhenium to the NNDC.

The group noted that the NCSP (as evidence by the ORNL and LANL work on gadolinium presented earlier in the day) has addressed the priority need for Gd data. However, it was also stressed that the Priority Needs List still lacks any quantitative information with regard to requirements of the applications. C. Hopper took an action item to produce accuracy requirements for gadolinium and provide this information to the group prior to the next NDAG meeting.

L. Leal (ORNL) presented material [which had been collected by K. Guber (ORNL), who was unable to be present at the NDAG meeting] on the status of measurement data for the nuclides identified as priority needs. The presentation materials of Guber and Leal are provided in Appendix I. Guber had collected and reviewed measurement data from CINDA and CSISRS and also reviewed which measured data had been used in evaluations by the major data projects. The NDAG members again requested that this effort be extended to produce quantitative estimates (i.e., "guesstimates") of the uncertainties in the priority data – which would then allow assessment of data accuracies versus accuracy requirements.

(It may be noted that, at this point in the meeting, the time was now ~5 PM. Discussion of this and remaining agenda topics continued but with a shortage of time to allow thorough discussion.).

M. Chadwick reviewed (i) status of the actinide cross sections; (ii) status of covariance data and processing; and (iii) the recent ORNL/LANL evaluation of covariance data for gadolinium. C. Dunford (BNL) had prepared a short review of the BNL evaluation activities primarily related to the fission products (which in included in Appendix J). It was noted that this effort is providing improved data important to the criticality safety community in assessment of burnup credit.

D. Hayes (LANL) gave a presentation (see Appendix K) of current and proposed measurements activities at LACEF. R. McKnight recommended inclusion of a neutron balance (reaction rates by isotope and region) in the LANL Preliminary Experiment Design Documents. NDAG discussion suggested inclusion of sensitivity analysis as part of the standard pre-analysis performed by LANL in planning new experiments. Hayes questioned the necessity for such pre-analysis since the proposed experiment has been "requested." There appeared to remain some difference in the expectations of the pre-analysis for critical experiments.

Copies of "Existing Benchmark Data that Support Priority Data Needs," which had been updated by B. Briggs (INEEL) prior to the NDAG meeting, were distributed to all (and included herein as Appendix L). Although time did not allow more than cursory

discussion of this summary, it may be noted that CSEWG participants found several occasions to refer to this list (to assess availability of benchmarks for data validation), demonstrating the value of list. (Furthermore, in a brief meeting/discussion related to nuclear data needs and deliverables for the space nuclear program held right after the conclusion of the CSEWG meeting, the participants found the above summary document extremely valuable.)

The final program element discussed centered on a presentation by C. Hopper on "AROBCAD Activities having Potential Interest to NDAG" (include in Appendix M). Hopper discussed sensitivity/uncertainty analyses performed for (i) NRC funded work for ^{10}Boron use in shipping casks; (ii) dilution experiments (C & Fe) for waste exemptions for DOE & NRC issues; (iii) NASA/DOE Space Fission Power Systems; and (iv) storage of ^{237}Np at Oak Ridge Y-12 Plant. Discussion within NDAG noted that the sensitivities presented in these studies need to be combined with the data uncertainties to allow assessment of the data needs.

Other Business

Time did not allow a separate discussion of ENDF/B-VII plans and deliverables (this topic was, however, a major part of the following CSEWG meeting).

B. Briggs reported on discussions which occurred at the ICNC2003 meeting in Tokai, Japan with Cogema regarding possible use (read "purchase") of their experimental data in support of Burnup Credit.

Conferences / Meetings of Interest to the Critcality Safety Community

It was noted that ND2004 will be held in Santa Fe, NM on September 26 – October 1, 2004. It was also suggested that a paper presenting the work of NDAG should be submitted to the conference.

Time and Place of the Next Meeting

It is planned to convene the next NDAG meeting in May, 2004 at Los Alamos National Laboratory. Although the next full meeting of CSEWG will be next November at Brookhaven National Laboratory, it is necessary to convene a meeting of the CSEWG Evaluation and Data Validation Committees about 6 months prior to the November CSEWG meeting. Because there will be considerable overlap between the participants of these two groups, it is planned to coordinate scheduling of the NDAG and CSEWG committee meetings. R. Little and R. McKnight will discuss scheduling of these meetings while together for the November 21 DOE NCSP meeting in New Orleans.

C. Dunford suggested that too much time had been devoted to the technical presentations in the morning and that future NDAG meetings should shorten that portion of the agenda. Although it was noted that several very valuable items were discussed in the early portion of the meeting, there was consensus among NDAG members that this

suggestion to reduce time for technical presentations at the NDAG meeting in order to allow more time for review of the NCSP program elements was important and should be implemented.

Recommendations to the CSSG

It was noted that the NCSP web site, while extremely useful for the criticality safety community, has need of updating some of the content – e.g., the NCSP Five Year Plan is dated August 1999, chair of End User group is shown to be Hans Toffer (again, circa 1999), and so forth. This resource is too valuable to allow to become so dated.

There was a question with regard to the Analytical Methods Program Element inquiring if the good work being done in the Expert Group on Source Convergence is being transferred into improvements in the NCSP Monte Carlo codes. It was noted that, in fact, these improvements are being implemented and are available to the users (customers).

Recall also (see summary of the morning NDAG discussions) the suggestion to CSSG to recommend special recognition be given to the LANL contributors to the new ^{238}U evaluation.

Summary of Action Items

Although the NDAG did not formally review the actions from the meeting, the following items are excerpted from the above meeting summary:

- McKnight will investigate the potential source of the model bias (ICSBEP vs CSEWG) for Flattop-25.
- Briggs will investigate inconsistencies that appear for the Pb-reflected experiments in the ORNL L5 and L6 benchmark series.
- Felty will obtain clarification of the RSICC policy for code distribution within the NCSP community.
- Hopper will produce accuracy requirements for gadolinium and provide this information to the group prior to the next NDAG meeting.
- Kawano will combine the ORNL (resonance) and LANL (above resonance) covariance data for Gd (This action was generated at the previous NDAG meeting).
- Guber will extend his evaluation of the measurement data for Gd to produce “guesstimates” for the uncertainties.

The meeting was adjourned at ~5:30 pm.

APPENDIX A: List of Attendees

Department of Energy Nuclear Criticality Safety Program: Nuclear Data Advisory Group (NDAG): Minutes of Meeting at Brookhaven National Laboratory, November 3, 2003

Members:

	Telephone	e-mail
R. D. McKnight (ANL)	(630) 252-6088	rdmcknight@anl.gov
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Maurice Greene (ORNL)	(865) 574-8712	greenenm@ornl.gov
Calvin Hopper (ORNL)	(865) 576-8617	hoppercm@ornl.gov
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Mike Westfall (ORNL)	(865) 574-5269	westfallrm@ornl.gov
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Jim Felty (SAIC/DOE NA-11)	(301) 903-5494	james.felty@dp.doe.gov

Observers / Presenters:

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A. C. (Skip) Kahler (Bettis)	(412) 476-5399	kahlerac@bettis.gov
Michael Dunn (ORNL)	(865) 574-8712	dunnme@ornl.gov
Nancy Larson (ORNL)	(865) 574-4659	larsonnm@ornl.gov
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Morgan C. White (LANL)	(505) 665-3684	morgan@lanl.gov

APPENDIX B:
Agenda Fourth Meeting of the NCSP Nuclear Data Advisory Group
Brookhaven National Laboratory (Berkner Hall, Room B)
November 3, 2003, 8:30 AM

II. Adoption of Agenda

III. NDAG Attendance / Participation (R. McKnight)

IV. Reports of Current Activities
   a. New LANL Actinide Evaluations and Data Testing Results (M. Chadwick, R. MacFarlane)
   b. LANL / ORNL Gd Isotope Evaluations (T. Kawano)
   c. Covariance Data (M. Greene, L. Leal, T. Kawano, D. Naberejnev, R. MacFarlane)
   d. NEA WPEC High Priority Request List (R. McKnight)
   e. Monte Carlo Benchmark Calculations with Revised ^{235,238}U Evaluations (A. C. Kahler)

Break

V. Approval of Summary Record of the Third NDAG Meeting, April 22, 2003

VI. NCSP Budget Execution Meeting, September 25-26, 2003 (J. Felty *et al*)

VII. Review of NCSP Flowchart (M. Westfall)

VIII. Review of NCSP Elements
(Note: For these topics, NDAG would like to review and discuss 2003 Activities, Plans for 2004 and out years, Issues, and Priorities.)
   a. Data Needs (J. McKamy)
   b. Measurements (L. Leal for K. Guber)
   c. Evaluation (M. Chadwick)
   d. Integral Experiments (D. Hayes)
   e. Benchmarks (B. Briggs)
   f. Validation (R. McKnight)
   g. Methods, Codes, Processing (M. Westfall)
   h. AROBCAD (C. Hopper)
   i. Other Elements (All)

IX. Other Business
   a. ENDF/B-VII (M. Chadwick *et al*)
   b. Cogema Data (B. Briggs)
   c. Other

X. Conferences / Meeting of interest to the CS Community
   a. ND2004 Santa Fe, NM
   b. NDAG Paper

XI. Time and Place of next NDAG Meeting

XII. Recommendations to CSSG

XIII. Summary of Action Items

APPENDIX C

Nuclear Data for Criticality Safety, NDAG

Evaluation Progress

Mark Chadwick

Los Alamos National Laboratory

Tasks Agreed at Last NDAG, ORNL, & Progress

- **Reestablish capability to evaluate, process, & use covariance data for priority materials**
  - use n+Gd data as test case, for LANL-ORNL collaboration
  - T-16 (Kawano) has evaluated n+152Gd reactions above resonance range, put into ENDF format, & combined with Leal's resonance parameter information. Currently being tested.
  - total, capture, inelastic, n,2n & n,3n uncertainty information
  - Required development & use of nuclear modeling, & Bayesian, tools.

  - MacFarlane has interacted with Larson on covariance data processing methods for NJOY; implemented formalism and validated against Larson test case
  - MacFarlane's work has led to an appreciation of some serious methods complications in processing and using covariance resonance data (needs discussion!)

  - 235,238U covariance data. Kawano has obtained JENDL multigroup data, and shared it with ORNL collaborators

ORNL Progress

Luiz Leal will present work on:

F-19, Cl-35,Cl-37, Pu-241, U-238, and the Gd isotopes (resolved and unresolved resonance

analysis and covariance generation for these energy regions)

Actinide data improvements & testing, for 235,238,233U

- **Continue nuclear data advances, with continued critical assembly data testing**
  - more studies of thermal & fast assemblies
  - test new IAEA/CSEWG 235U fission standard
  - test new ORNL unresolved resonance data (to 20keV & to 100 keV)

  - Much testing – see MacFarlane's reports. Overall very good results, as presented at the last NDAG.
  - Issues re. 235 prompt neutron spectrum at thermal (using rel.8 o16 data, better performance using existing endf-6 spectrum as opposed to Madland's latest)
  - New IAEA 235U fission cross section is similar to LANL's interim evaluation. Tested in Godiva. Should be easy to use.
  - MacFarlane has results for ORNL 235 unresolved resonance data

Neptunium nuclear data and criticality

- **Contribute to analysis of LACEF Np-U criticality experiment, to determine critical mass & nuclear data**

  - Baseline analysis using latest U data and Np data, gives k-eff=0.993.
  - Investigated sensitivity to neptunium fission, nubar, and inelastic scattering cross sections. New focus needed on the secondary neutron spectra (inelastic, and chi)
  - will study subcritical neutron multiplication data
  - Bias in 235U data needs to be fixed – spectral indices in Godiva for threshold fissioners tend to be ~3-5% low, indicating the Godiva neutron spectrum is calculated too soft.

  - First estimates suggest a critical mass ~ 57 kg. (Current ENDF file is about 67 kg). Uncertainties also need to be determined.

Improved Cross Sections For Criticality Work

Robert E. MacFarlane
Nuclear Physics Group T-16
Theoretical Division, Los Alamos National Laboratory

November 2003

Recent Work On Cross Sections

- **New LANL evaluations were prepared for**
  - 232,233,234,235,236, 237,238,239,241-U
  - 239-Pu
  - 237-Np
  - Released as pre ENDF/B-VII

- **Recent testing has raised various issues**
  - Improved results for fast crits for elastic and inelastic changes
  - Effects of improved inelastics on thermal cases
  - Effects of new U-235 thermal fission spectrum
  - Effects of Release-8 oxygen changes
  - Possible new U-238 resonance parameters

Improvements for U-235 Assemblies

CSWEG Godiva	CSEWG Godiva	CSEWG Flattop-25	CSEWG Flattop-25
k-eff=.9961	k-eff=.9993	k-eff=1.0018	k-eff=1.0024
28/25=.965	28/25=.961	28/25=.981	28/25=.970
37/25=.959	37/25=.954	37/25=.981	37/25=.971
23/25=1.001	23/25=.987	23/25=.990	23/25=.975
39/25=.977	39/25=.974	39/25=.985	39/25=.980
HEU-MET-FAST-001	**HEU-MET-FAST-001**	**HEU-MET-FAST-028**	**HEU-MET-FAST-028**
k-eff=.9965	k-eff=.9989	k-eff=.9970	k-eff=.9979
28/25=.970	28/25=.957	28/25=.974	28/25=.971
37/25=.961	37/25=.956	37/25=.980	37/25=.973
23/25=1.002	23/25=.988	23/25=.990	23/25=.975
39/25=.978	39/25=.975	39/25=.984	39/25=.980

Improvements for Pu-239 Assemblies

CSWEG Jezebel	CSEWG Jezebel	CSEWG Flattop-Pu	CSEWG Flattop-Pu
k-eff=.9978	k-eff=1.0004	k-eff=1.0033	k-eff=1.0019
28/25=.968	28/25=.977	28/25=.986	28/25=.979
37/25=.973	37/25=.972	37/25=.989	37/25=.976
23/25=1.001	23/25=.989		
39/24=.976	39/25=.973		
PU-MET-FAST-001	**PU-MET-FAST-001**	**PU-MET-FAST-006**	**PU-MET-FAST-006**
k-eff=.9976	k-eff=.9998	k-eff=1.0024	k-eff=1.0004
28/25=.977	28/25=.977	28/25=.990	28/25=.983
37/25=.973	37/25=.974	37/25=.992	37/25=.981
23/25=1.000	23/25=.975		
39/25=.976	39/25=.974		

Improvements for U-233 Assemblies

CSWEG Jezebel-23	CSEWG Jezebel-23	CSEWG Flattop-23	CSEWG Flattop-23
k-eff=.9925	k-eff=.9987	k-eff=1.0024	k-eff=.9999
28/25=1.011	28/25=.981	28/25=.999	28/25=.978
37/25=.988	37/25=.981	37/25=.994	37/25=.982
U233-MET-FAST-001	**U233-MET-FAST-001**	**U233-MET-FAST-006**	**U233-MET-FAST-006**
	k-eff=.9985		k-eff=.9984
	28/25=.986		28/25=.970
	37/25=.982		37/25=.978

Results for Bigten

CSWEG 1-D Bigten	CSEWG 1-D Bigten	IEU-MET-FAST-007	IEU-MET-FAST-007
k-eff=1.0086	k-eff=.9978	k-eff=1.0066	k-eff=.9949
28/25=1.041	28/25=.981	28/25=1.033	28/25=.962
37/25=1.034	37/25=.949	37/25=1.036	37/25=.942
23/25=.998	23/25=.971	23/25=.996	23/25=.971
39/25=.987	39/25=.973	39/25=.988	39/25=.972
8c/25=.959	8c/25=.970	8c/25=.958	8c/25=.974

Exp k-eff=.9960(30)

Exp k-eff=.9948(13)

Notation for Tables

- **MCNP accuracy for k-eff about +/- .0005 (two sigmas)**
- **Black for ENDF/B-VI.5**
- **Red for new evaluations**
- **Spectral indices C/E**
  - 28/25 means U-238 fissions / U-235 fissions
  - 37/25 means Np-237 fissions / U-235 fissions
  - 23/25 means U-233 fissions / U-235 fissions
  - 39/25 means Pu-239 fissions/ U-235 fissions
  - 8c/25 means U-238 captures / U-235 fissions

Results for Thermal UO2 Lattices

Case	ENDF/B-VI	LANL	+ORNL
LCT6-06	.99374	.99656	.99992
LCT6-11	.99501	.99692	.99949
LCT6-16	.99503	.99701	.99978

Observations for LEU Cases

- **Improvements due to improved inelastics**
  -Increases k quite a bit, but not enough
  -Other recent evaluations (Maslov, BRC) have similar effects

- **Effects of changes to U-235 thermal fission spectrum**
  - Causes k to decrease, canceling part of the improvement

- **Resonance parameter changes**
  - People have often adjusted U-238 capture to improve results
  - New ORNL fit to high-resolution ORELA transmission data with adjustments to give proper thermal cross sections appears to improve results nicely.

Oxygen and U-235 fission spectrum

- **Release-8 changes in oxygen elastic scattering**
  -Were made to decrease k for high-leakage HEU solutions
  -Result was very little bias with leakage for these cases
  -Also leads to decreases for other high-leakage cases as observed by Mosteller.

- **Effect of new U-235 thermal fission spectrum**
  •It has a higher average energy than before
  •This leads to a reduction in k values for some thermal cases

Conclusions

- **K-eff values are good**
  - -For both fast and thermal systems
  - -Considering the uncertainties in the experiments
- **Flux is a little soft for fast U-235 systems**
  - - Reevaluation of U-235 with latest methods may help in future

Covariance Generation for Gd Isotopes

T. Kawano
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Los Alamos, New Mexico 87545, USA

Estimation of Uncertainties

- cross sections above resonance region
- based on experimental data, or
- nuclear model fitting technique with KALMAN

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Total Cross Section

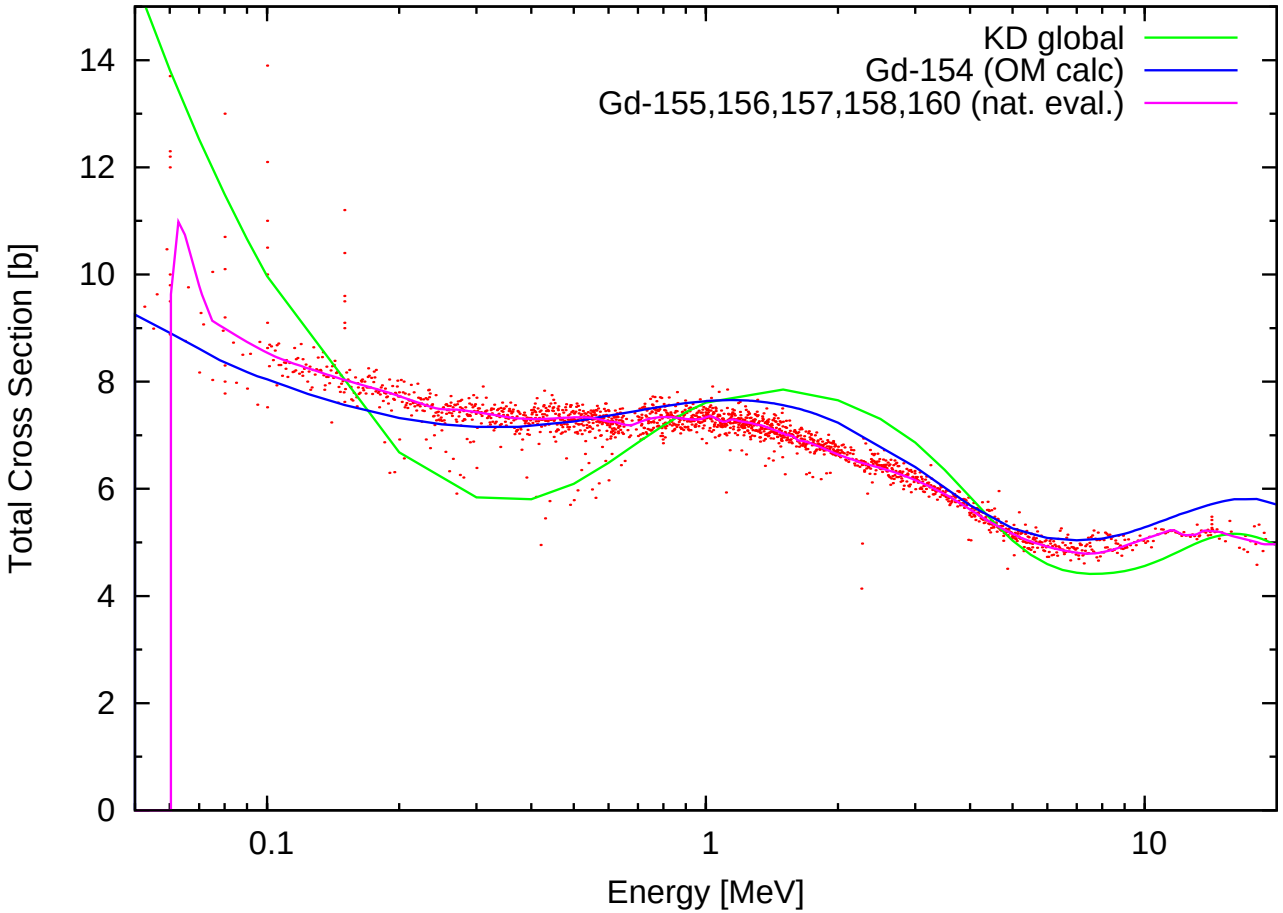
Status of ENDF/B-VI Evaluation

Isotope	Abn.[%]	Data Source
^{152}Gd	0.2	JENDL/FP, OM calc.
^{154}Gd	2.18	JENDL/FP, OM calc.
^{155}Gd	12.80	taken from exp. data of natural element
^{156}Gd	20.47	taken from exp. data of natural element
^{157}Gd	15.65	taken from exp. data of natural element
^{158}Gd	24.84	taken from exp. data of natural element
^{160}Gd	21.96	taken from exp. data of natural element

- ^{152}Gd and ^{154}Gd
  - Optical Model Fitting with KALMAN, and scale the error-bars
- ^{155}Gd, ^{156}Gd, ^{157}Gd, ^{158}Gd and ^{160}Gd
  - Least-Squares Fitting with SOK to the same data
  - Use the same covariance for all isotopes

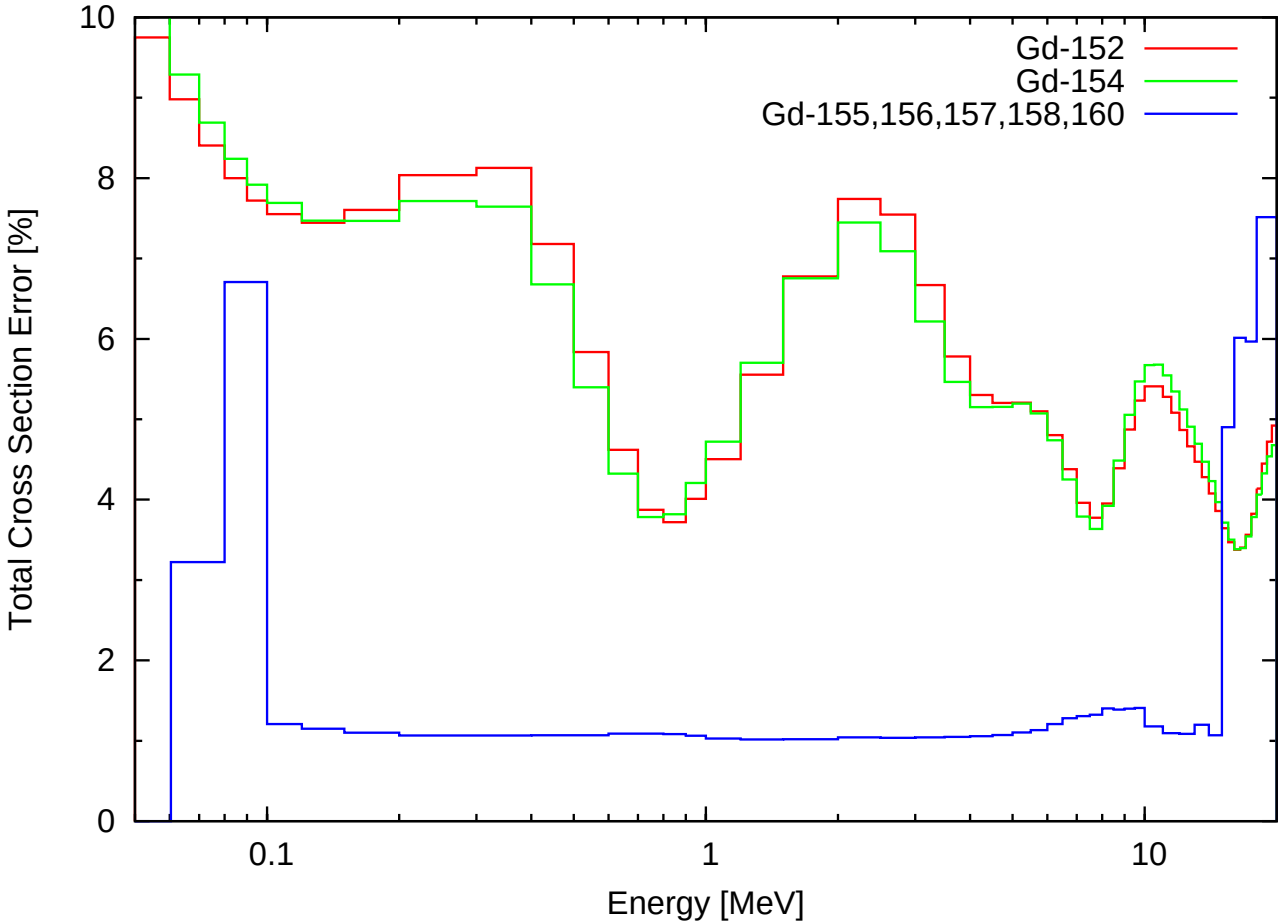
Los Alamos

Comparison of Total Cross Sections



(dots are the experimental data for natural elements)

Evaluated Errors



Capture Cross Section

Status of ENDF/B-VI Evaluation

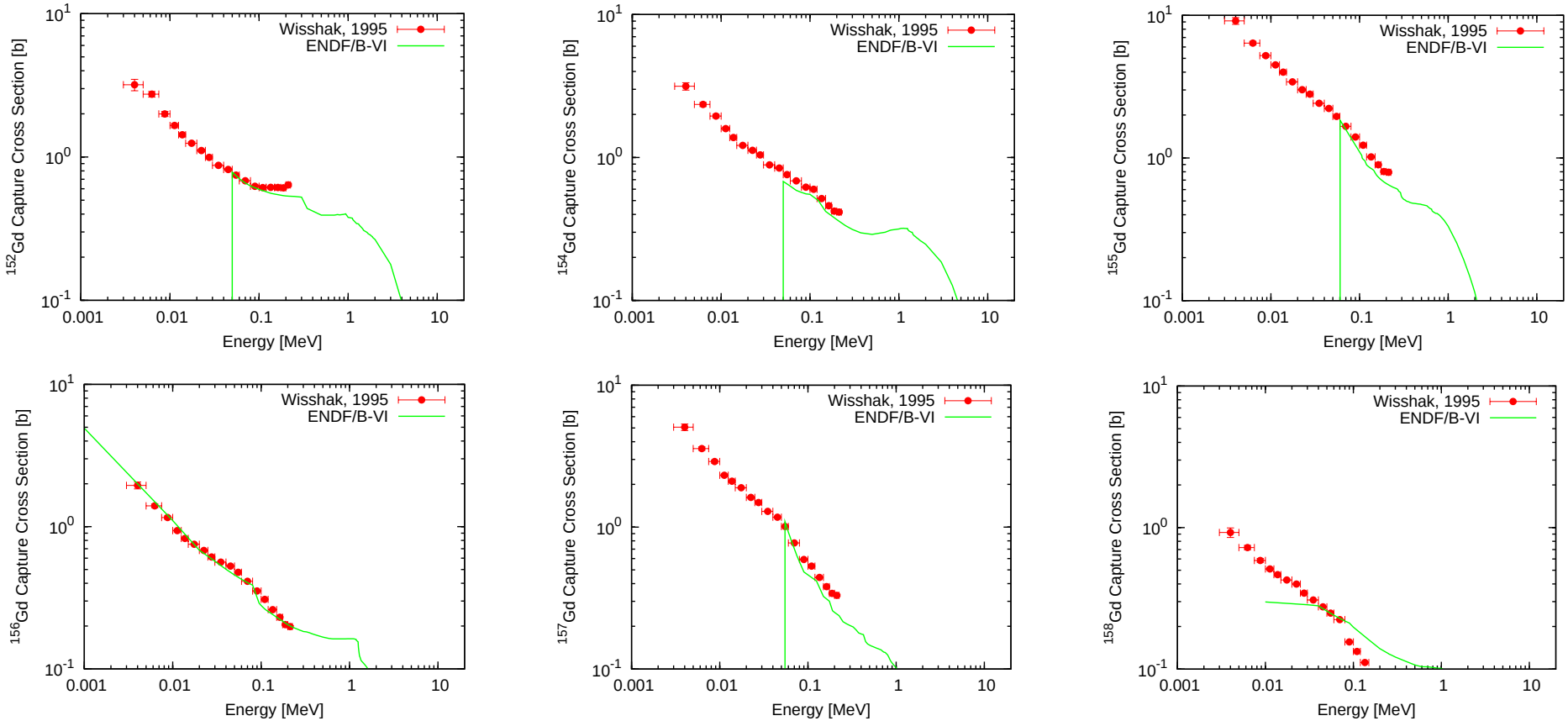
- ^{152}Gd and ^{154}Gd
  - Beer & Macklin below about 1 MeV, JENDL/FP (CASTHY) above
- ^{155}Gd, ^{156}Gd, ^{157}Gd, ^{158}Gd and ^{160}Gd
  - COMNUC calc. up to 5 MeV, and Shorin data above 5 MeV

Covariance Evaluation Method

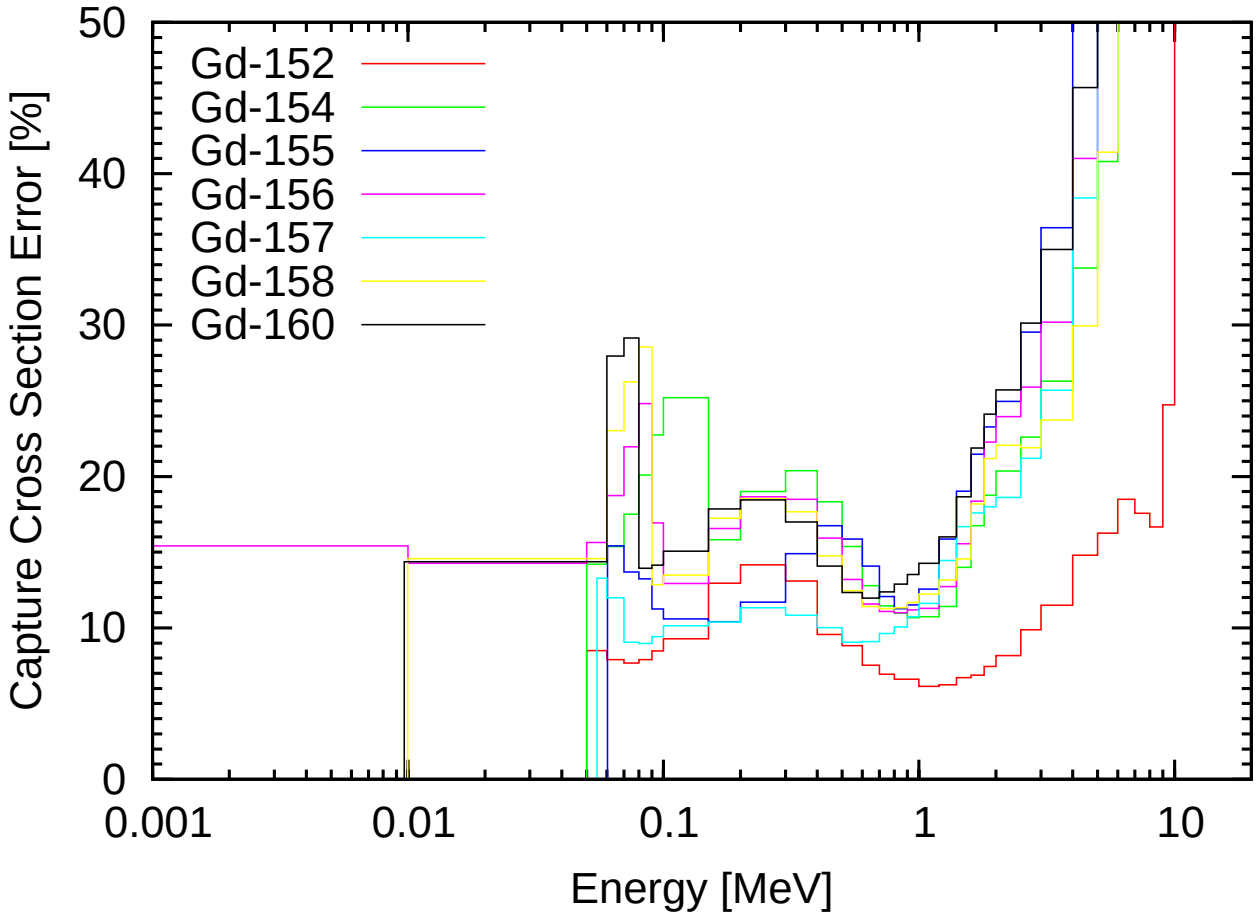
- Hauser-Feshbach-Moldauer model used
- Since we do not have parameters used in the ENDF/B-V and VI evaluations, those are taken from RIPL-2, and adjust them to (approximately) reproduce ENDF values.
- KALMAN calculation, and re-normalize the calculated uncertainties.

Los Alamos

Comparison with FZK data



Evaluated Errors



Inelastic Scattering Cross Section

Status of ENDF/B-VI Evaluation

- Inelastic scattering cross sections for ^{152}Gd and ^{154}Gd were taken from JENDL/FP file — CASTHY calculation.
- Evaluation of ^{155}Gd, ^{156}Gd, ^{157}Gd, ^{158}Gd and ^{160}Gd cross sections were done with COMNUC.

Covariance Evaluation Method

- Hauser-Feshbach-Moldauer model with RIPL-2 parameters, and compare with the cross sections in ENDF/B-VI.
- KALMAN calculation, and re-normalize the calculated uncertainties

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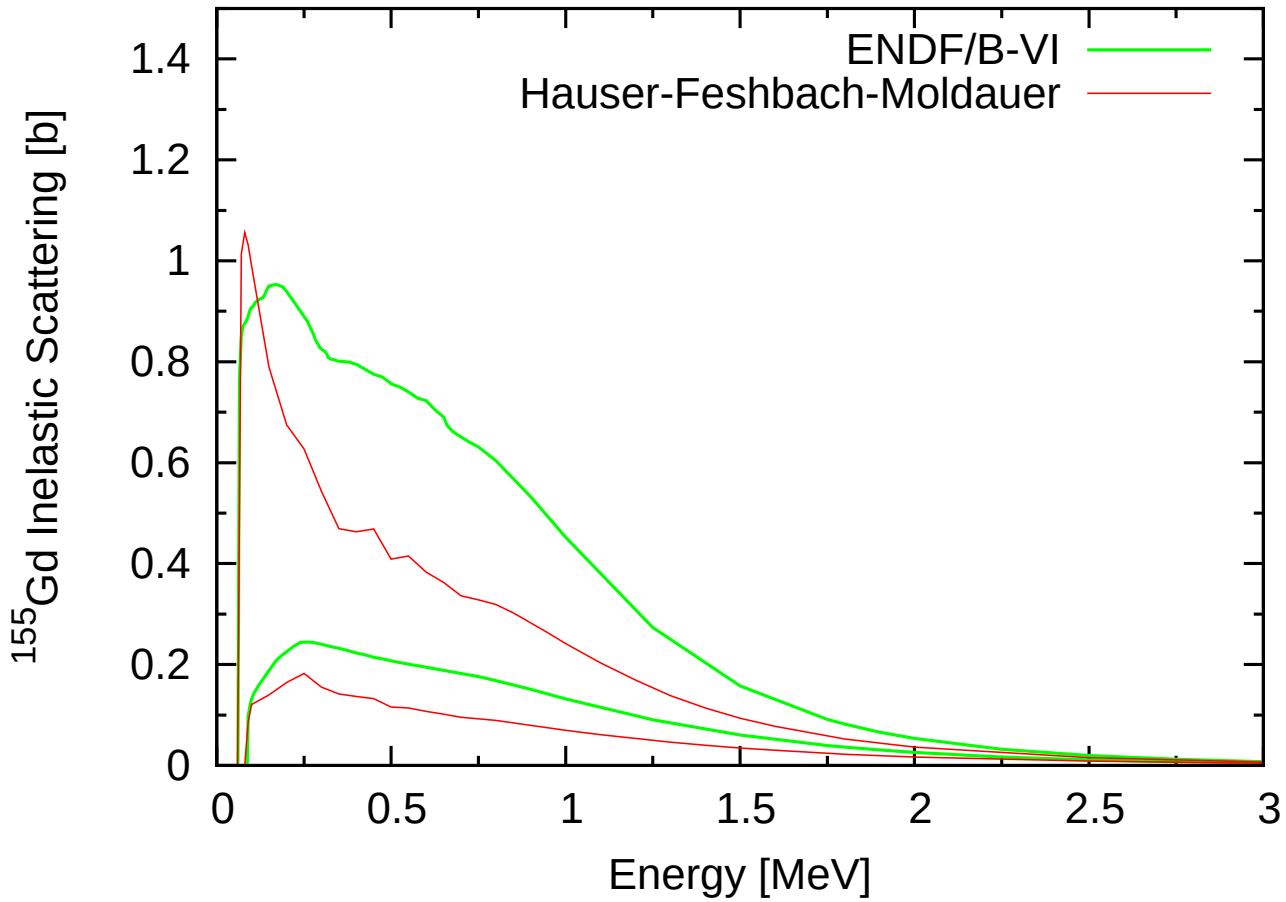
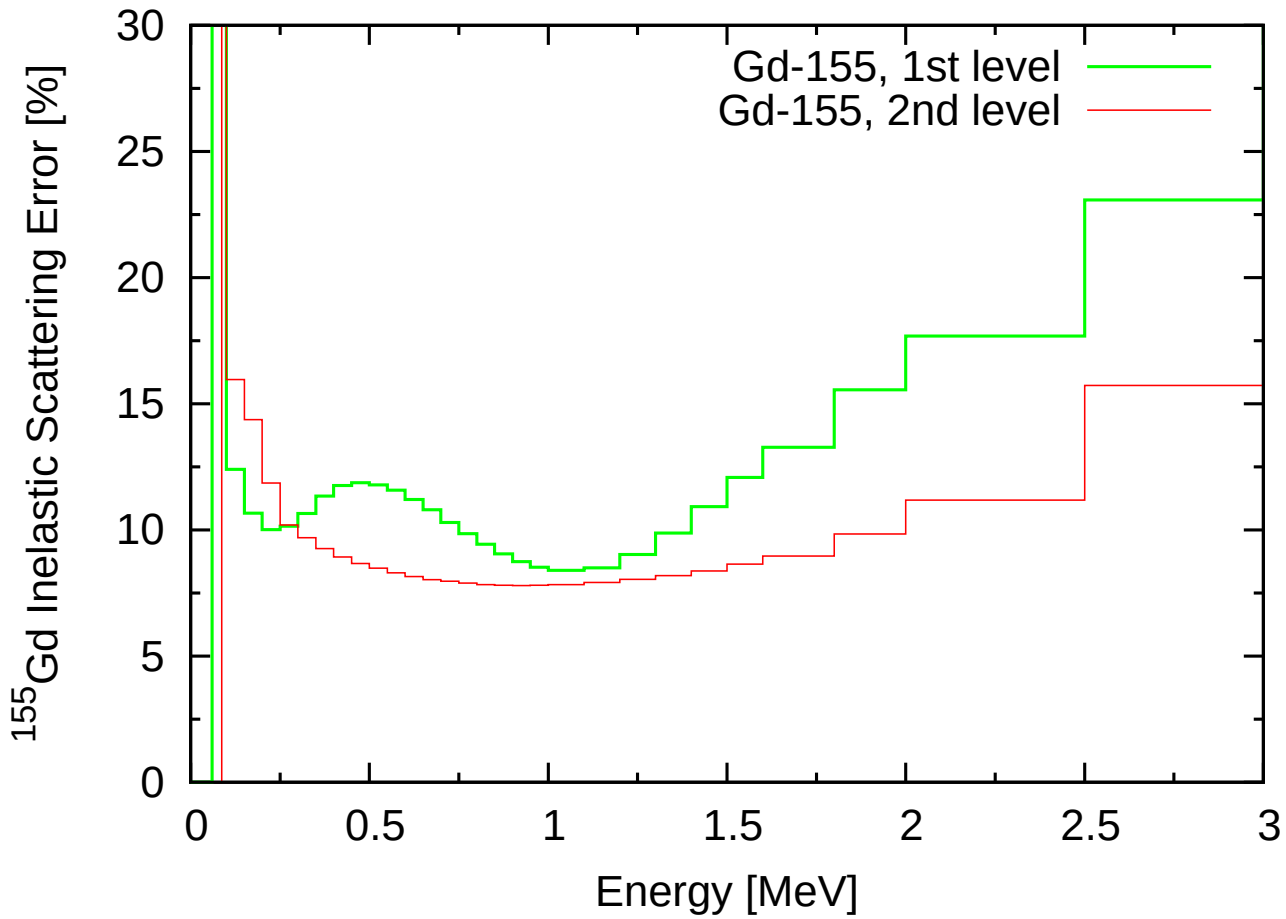
Comparison of Cross Sections



Evaluated Errors



Current Status of Covariance Evaluation

Covariance data were given for:

- Total, capture, inelastic scattering, $(n, 2n)$, and $(n, 3n)$ cross sections.
- Least-squares fitting to the total cross sections of natural element.
- KALMAN calculation, and re-normalize the calculated uncertainties.

Isotope	Total	Capture	Inelastic	$(n, 2n)$	$(n, 3n)$
^{152}Gd	○	○	○	○	○
^{154}Gd	○	○	○	×	×
^{155}Gd	○	○	○	×	×
^{156}Gd	○	○	○	×	×
^{157}Gd	○	○	○	×	×
^{158}Gd	○	○	○	×	×
^{160}Gd	○	○	○	×	×

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APPENDIX F

Data Evaluation for
^{19}F, ^{35}Cl, ^{37}Cl, ^{241}Pu, ^{238}U, and Gd Isotopes

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Nuclear Data Group
Nuclear Science and Technology Division
Oak Ridge National Laboratory

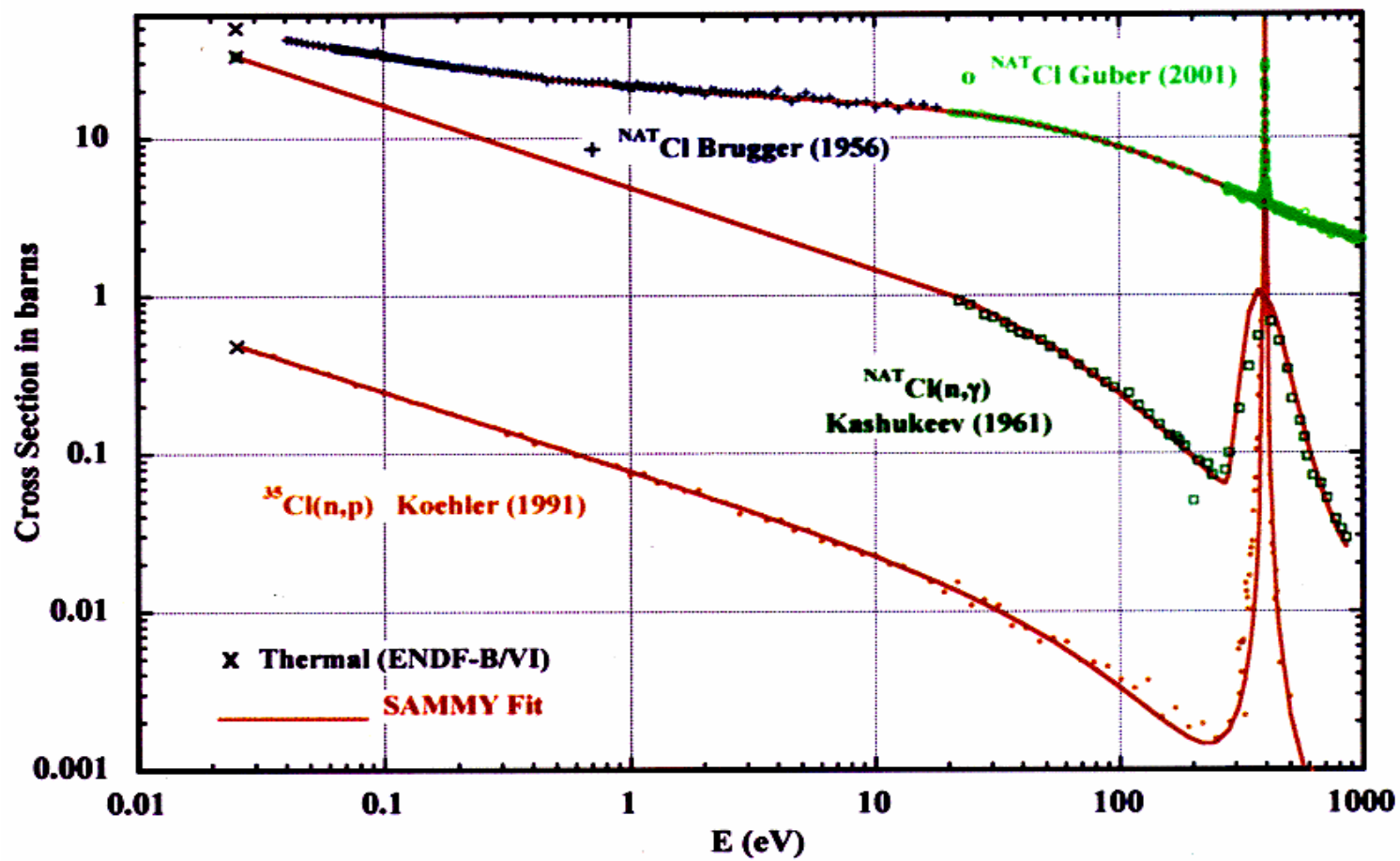
CSEWG
Brookhaven National Laboratory
November 4-6, 2003

^{35}Cl and ^{37}Cl Resonance Evaluation

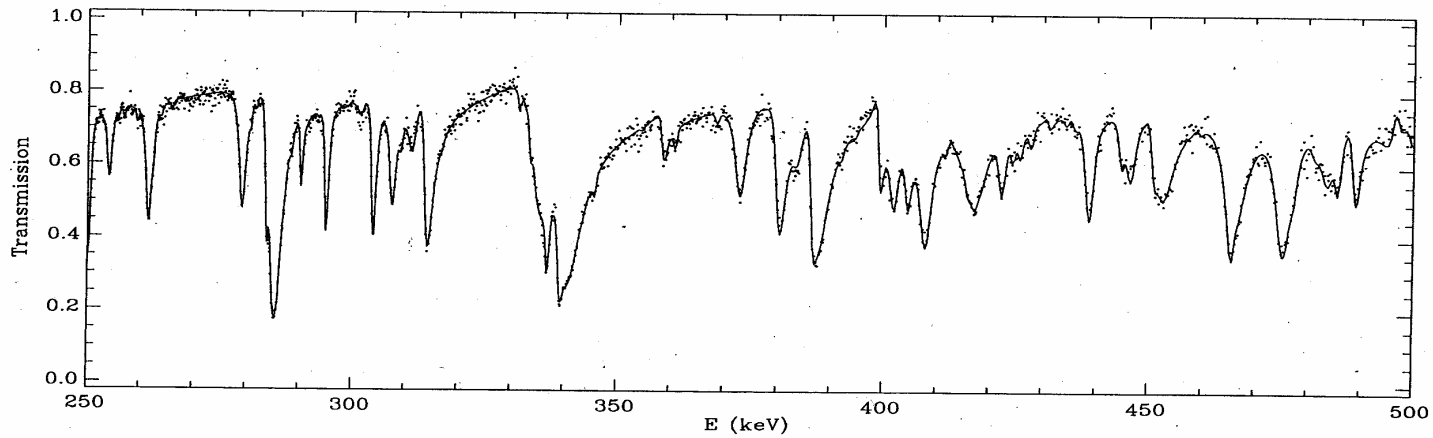
Features:

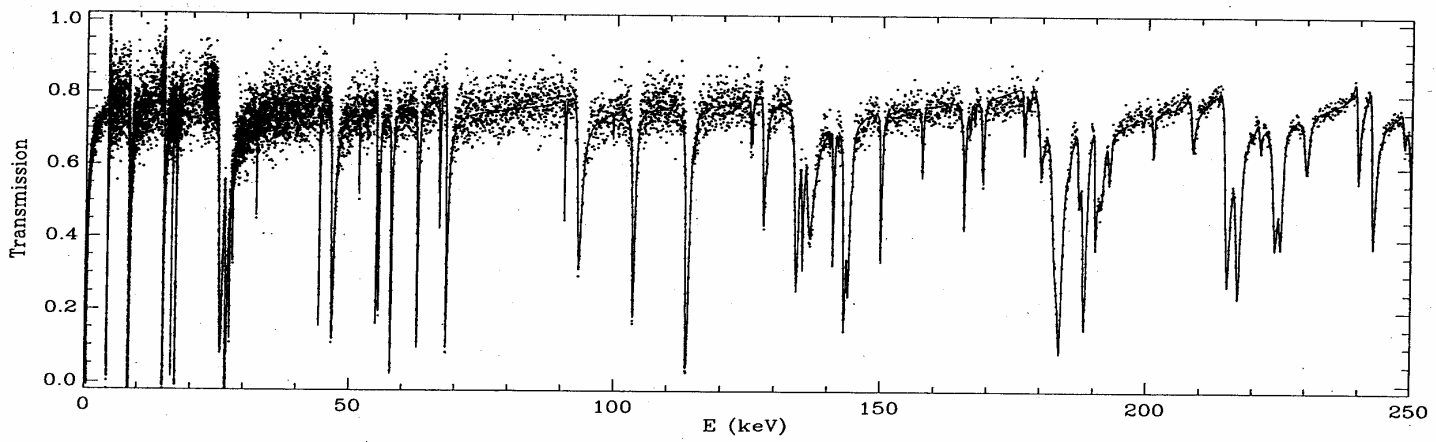
- Total cross section, capture cross section, and (n,p) cross section data measurements were used in the evaluation.
- Evaluation performed up to 1.2 MeV.
- 380 s- and p-wave. 244 for ^{35}Cl and 136 for ^{37}Cl.
- SAMMY: Reich-Moore formalism was used.
- Charge particle (n,p) exit channel not accommodated in existing ENDF format: Format modification proposed !!

Chlorine Cross Section

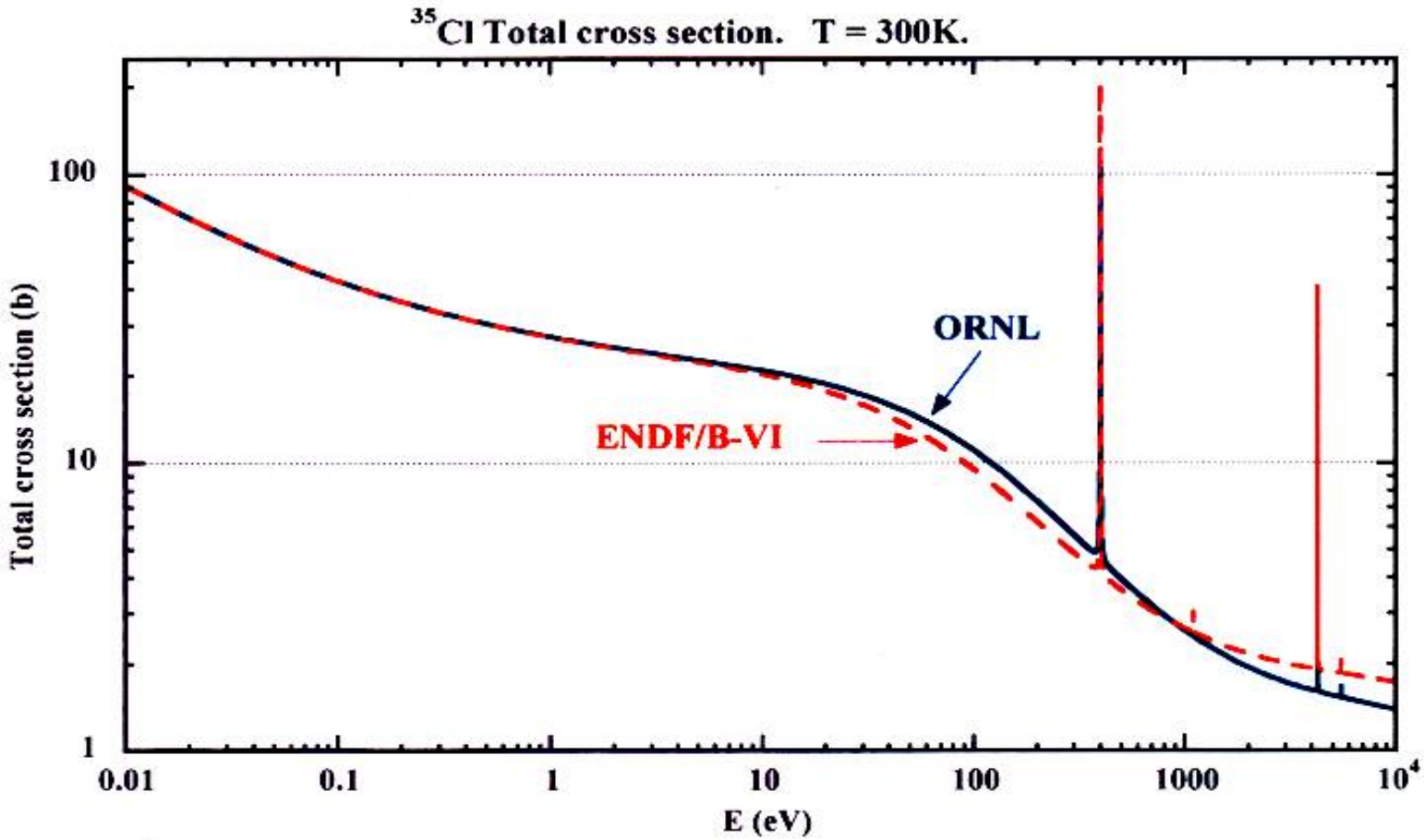


Chlorine Cross Section

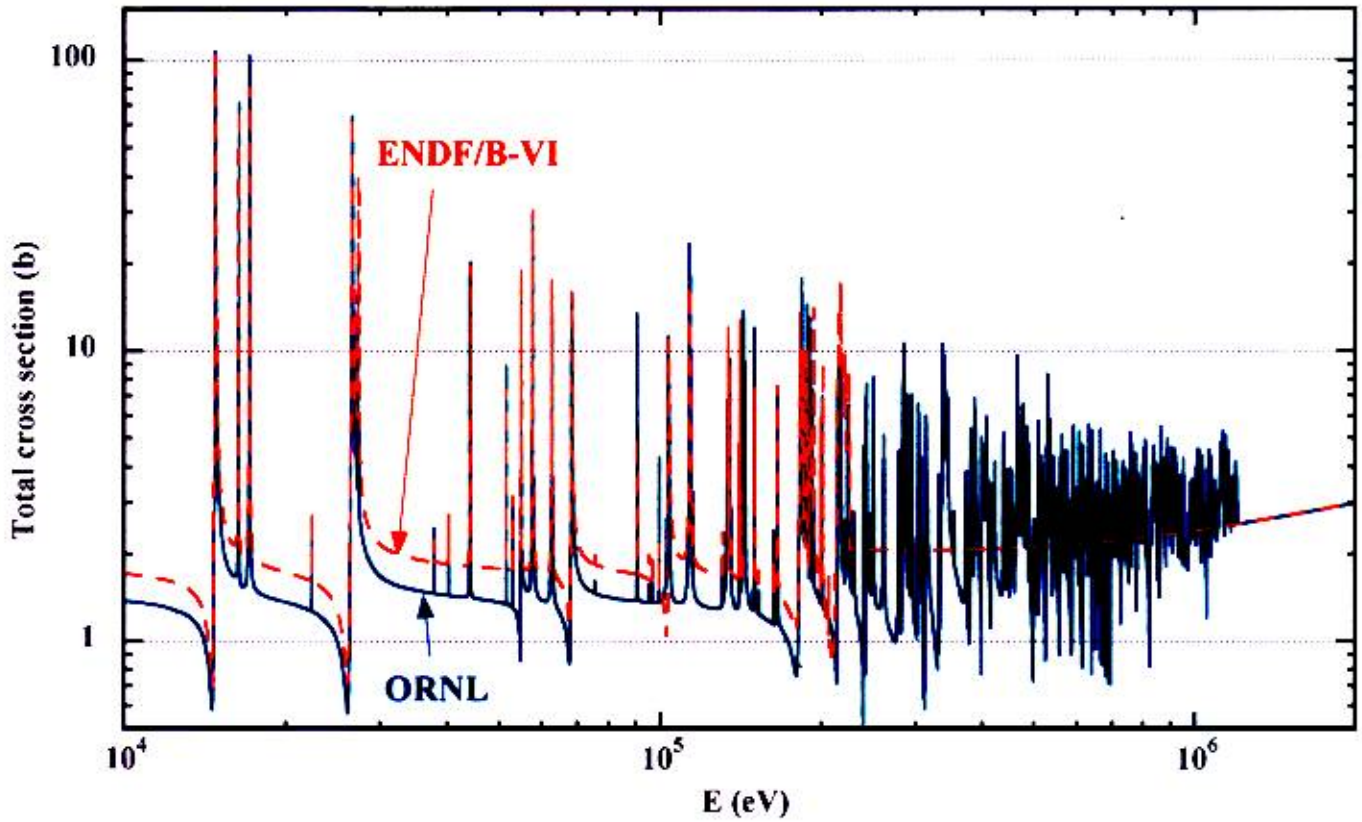




Comparison with ENDF



Comparison with ENDF



^{19}F Evaluation

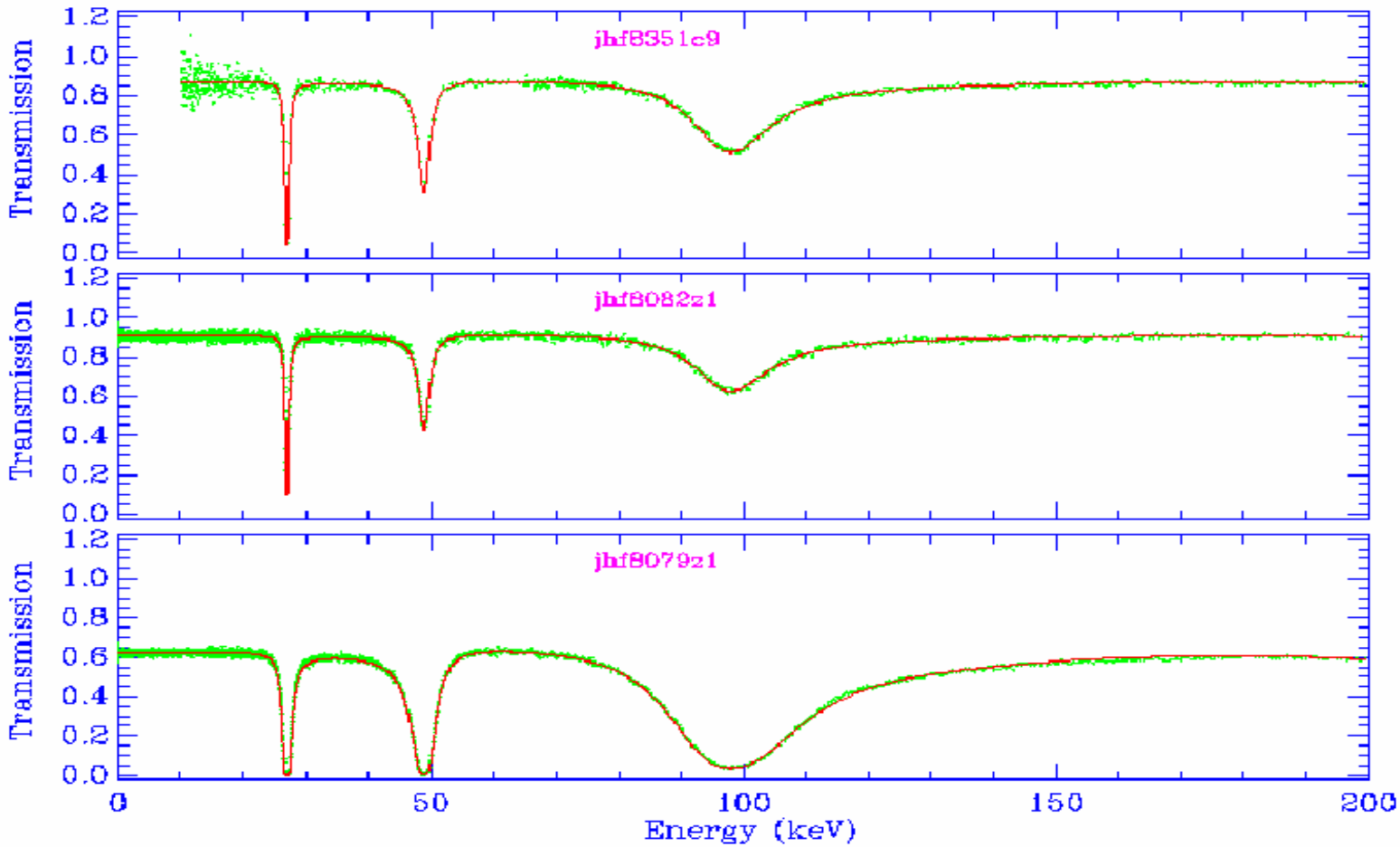
Experimental Data

- **Three Transmission Data Measurements of Larson *et al.* made at ORELA 80 meters flight path with sample thicknesses 0.13093 at/b, 0.016886 at/b, and 0.024184 at/b, respectively in the energy range 5 ev to 20 MeV.**

- **One Capture measurement done at ORELA 40 meters flight path performed by Guber *et al.* up to 700 KeV;**

- **Inelastic Cross Section Measurements Performed by Broder *et al.* at Obninsk up to 1 MeV. (**Format modification proposed**)**

Transmission



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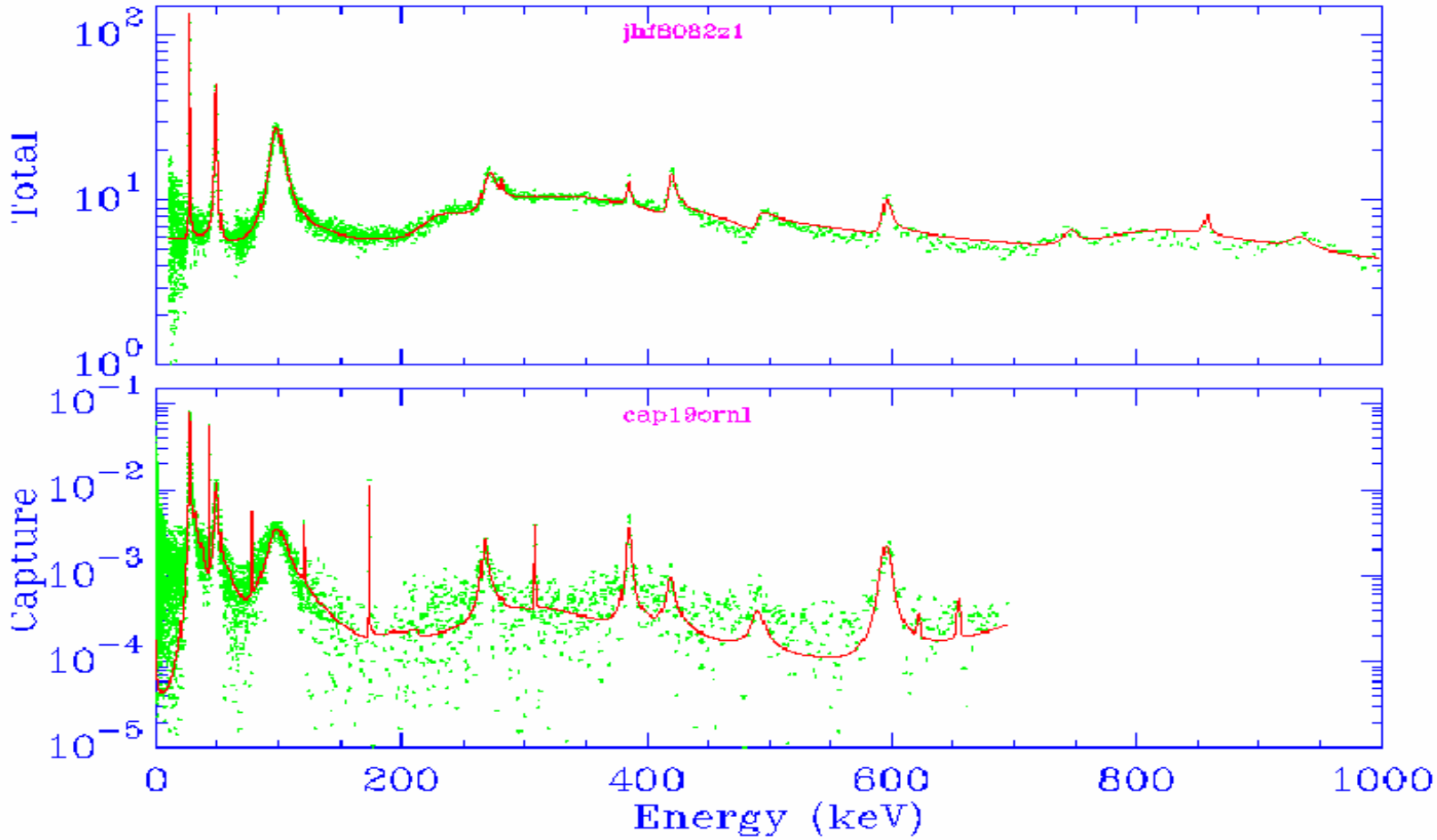
Total and Capture Cross Sections



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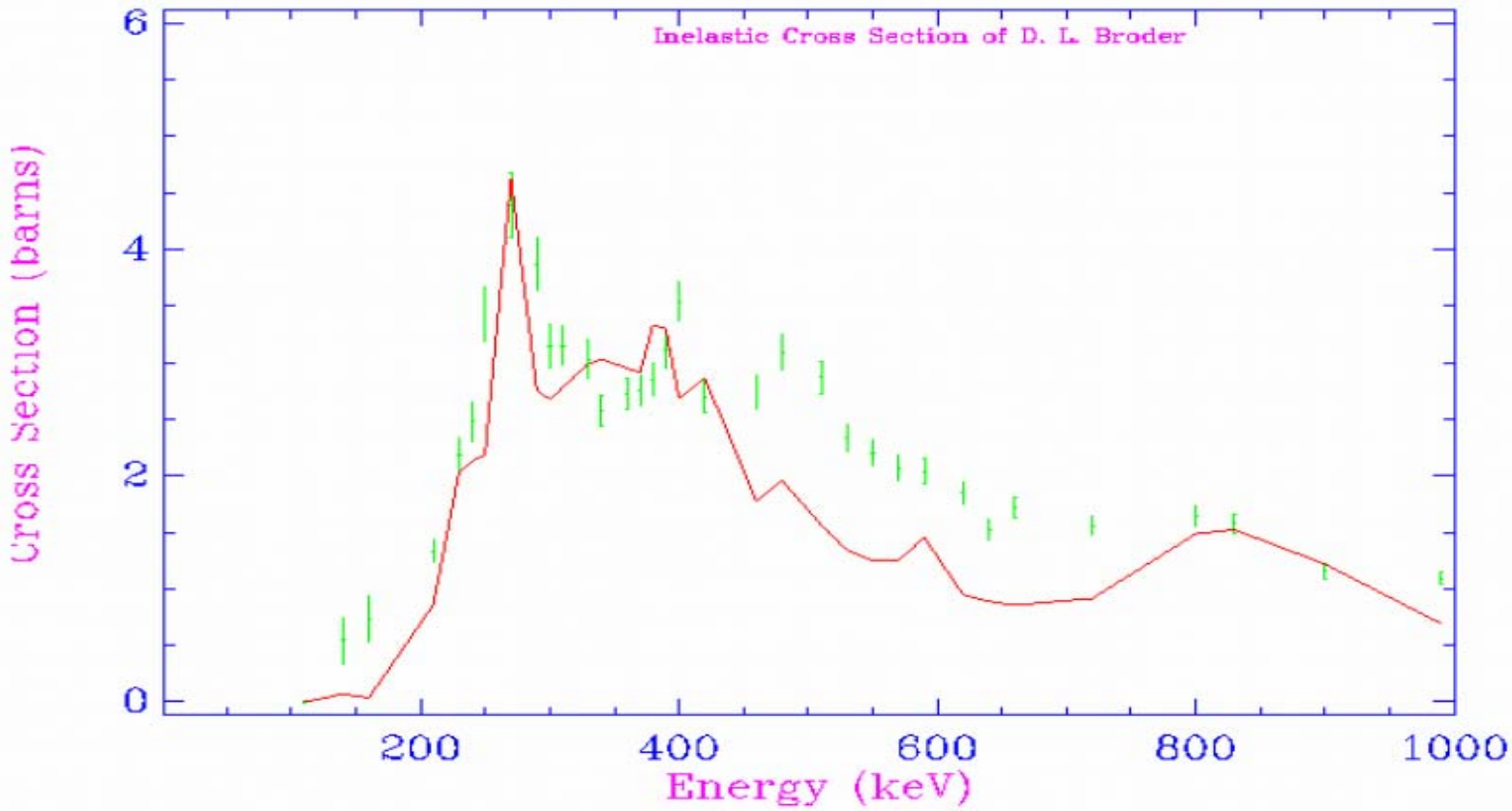
Inelastic Cross Sections



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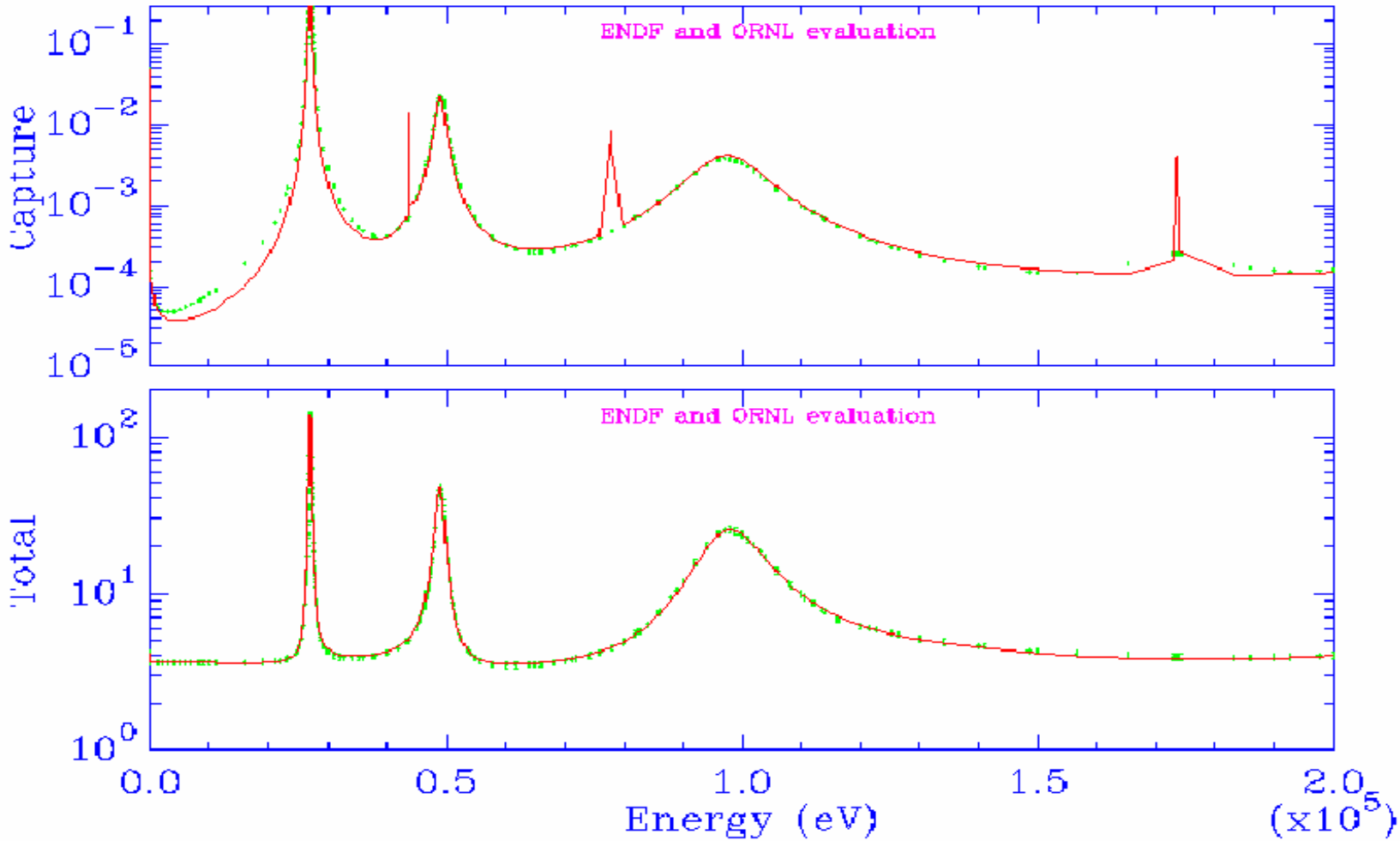
Comparison with ENDF evaluation



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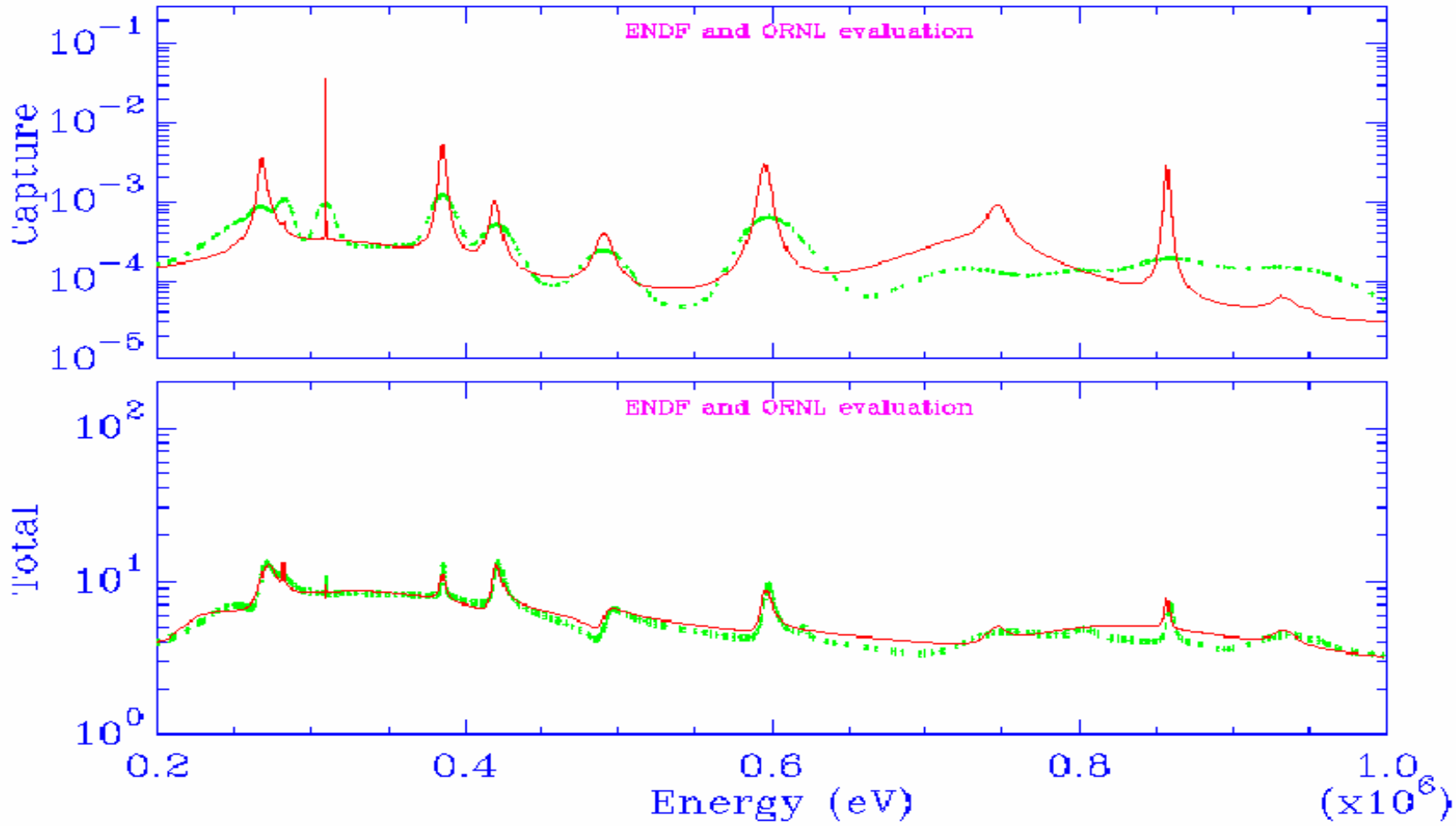
Comparison with ENDF evaluation



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Direct Capture Cross Section for ^{19}F

SAMMY allows the fit of the direct capture component of the cross section as suggested by Kerman and Dietrich

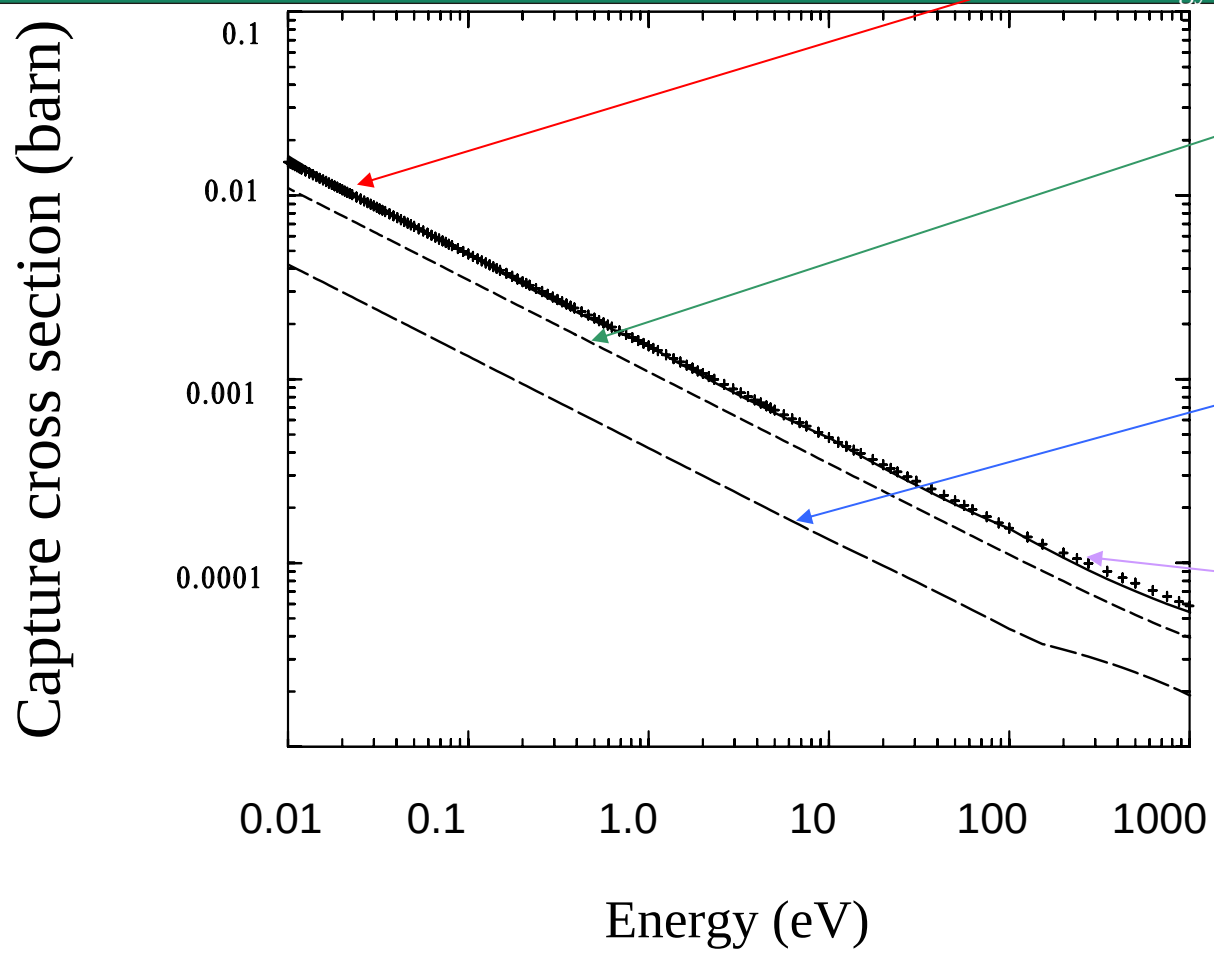
Total Capture = resonance component + direct capture component

$$\sigma_{\gamma}^{tot} = \sigma_{\gamma}^{res}(E_n, \Gamma_n, \Gamma_{\gamma}) + C \times \sigma_{\gamma}^{direct}$$

Example: ^{19}F

- **Model calculations by Goran Arbanas, ORNL**
- **Analysis by Luiz Leal, ORNL**
  - **Multiplier initially at 1.0, fitted to 0.547**

^{19}F (continued)



+ = data

dash = R-matrix contribution

long dash = direct component

solid line = sum of the two components

^{241}Pu Evaluation

Motivation

- **Investigation of the French Post-Irradiation Experiments (PIE) on PWR Indicates Significant Underestimation of the ^{241}Pu Capture Cross Section;**

- **Underestimation in the build-up prediction of ^{243}Am, ^{244}Cm, and ^{245}Cm;**

The Evaluation was performed in the energy range thermal to 20 eV;

Paper submitted to Nuclear Science and Engineering.

^{241}Pu Evaluation

Experimental Data Bank

- **Weston fission and capture data (1976);**
- **Wagemans fission data (1991);**
- **Young and Smith total cross section (1968);**

All data normalized in the energy range 0.01 eV to 1 eV according to Wagemans fission and Young total cross section.

Results

E_r 0.26 eV	Γ_γ meV	Γ_n meV	Γ_{f1} meV	Γ_{f2} meV
1988	32.555	0.04371	-78.04	-0.481
1993	33.350	0.04252	-50.42	25.36
This Work	34.630	0.04367	-42.71	32.06

Results

E_r 0.0253 eV	Standard (barns)	Present work (barns)
Fission	1012.68 ± 6.58	1012.20
Capture	361.29 ± 4.95	363.0
Scattering	12.17 ± 2.62	11.3
Total	1386.14 ± 8.64	1386.5

Results

(C/E-1 in % using JEF2.2, JENDL3.2, and ORNL evaluation)

BU (Gwd/t)	26.9	38.4	50.3	59.8
Pu241/U238(JEF2.2)	-6.3 ± 2.3	-5.0 ± 1.8	-3.8 ± 1.6	-3.1 ± 1.6
Pu242/U238(JEF2.2)	-10.5 ± 4.0	-9.7 ± 3.4	-8.8 ± 3.1	-8.6 ± 2.8
Am243/U238(JEF2.2)	-18.0 ± 6.2	-11.6 ± 5.2		-8.8 ± 4.4
Pu241/U238(JENDL3.2)	-5.4 ± 2.3	-3.8 ± 1.8	-2.5 ± 1.6	-1.6 ± 1.6
Pu242/U238(JENDL3.2)	-9.0 ± 4.0	-7.9 ± 3.4	-6.8 ± 3.1	-6.4 ± 2.8
Am243/U238(JENDL3.2)	-16.6 ± 6.2	-9.9 ± 5.2		-6.8 ± 4.4
Pu241/U238(ORNL)	-5.9 ± 2.3	-4.4 ± 1.8	-3.1 ± 1.6	-2.4 ± 1.6
Pu242/U238(ORNL)	-5.2 ± 4.0	-4.3 ± 3.4	-3.2 ± 3.1	-2.9 ± 2.8
Am243/U238(ORNL)	-12.8 ± 6.2	-6.5 ± 5.2		-3.7 ± 4.4

^{238}U Evaluation

- Resonance Analysis done in the Energy Region From 10^{-5} eV to 20 keV;
- Two Sets of Resonance Parameters: 0 to 10 keV and 10 keV to 20 keV;
- ENDF/B-VI Resonance Parameters below 10 keV were used for Starting Values;
- Extension of the ENDF/B-VI Resolved Resonance Region from 10 keV up to 20 keV;
- Evaluation done with the Reich-Moore formalism in the code SAMMY;
- High Resolution Data of Harvey et al. and Capture Cross Section Data of DeSaussure were used.
- Accurate Self-Shielding and Multiple Scattering Correction for Capture Data were done with an Improved Version of SAMMY;

Energy Range 1 keV to 10 keV

- Starting with the ENDF/B-VI Resonance Parameters (Moxon-Sowerby Evaluation);

- Experimental Data Used:
  - Olsen transmission data (1977) (four thicknesses)
  - Harvey Transmission Data: Three thicknesses: 0.1748, 0.0396, and 0.0125 atoms/b; Data measured at ORELA on the 200 m flight path
  - Capture Data of DeSaussure measured at ORELA on the 40 m flight path

Results for the evaluation energy region 1 keV to 10 keV

- Modification of the external resonance parameters for accurate description of the transmission data between resonances. The average scattering cross section increases by 2.8 %;

- Scattering Radius of 9.450 fm in agreement with the Olsen ORNL Evaluation;

- Better description of the Capture data. The average capture cross section increases by 3.8 % in the energy range 1 keV to 10 keV;

Average ^{238}U Cross Sections (1 keV to 10 keV)

Energy Range (keV)	Capture (b)		Elastic (b)	
	ENDF/B-VI	ORNL	ENDF/B-VI	ORNL
1 – 2	1.87	1.94	21.68	22.25
2 – 3	1.36	1.41	21.57	22.12
3 – 4	1.15	1.20	19.82	20.36
4 – 5	0.88	0.89	14.79	15.03
5 – 6	0.90	0.90	14.20	14.38
6 – 7	0.87	0.88	16.66	16.37
7 – 8	0.68	0.74	13.79	14.07
8 – 9	0.63	0.65	15.03	15.71
9 – 10	0.65	0.71	13.07	13.89
1 - 10	1.00	1.04	16.73	17.12

Energy Range 10 keV to 20 keV

- **No prior values of the resonance parameters existed;**

- **Experimental Data Used:**

  – **Harvey Transmission Data: Three thicknesses: 0.1748, 0.0396, and 0.0125 atoms/b; Data measured at ORELA on the 200 m flight path**

  – **Capture Data of DeSaussure measured at ORELA on the 40 m flight path**

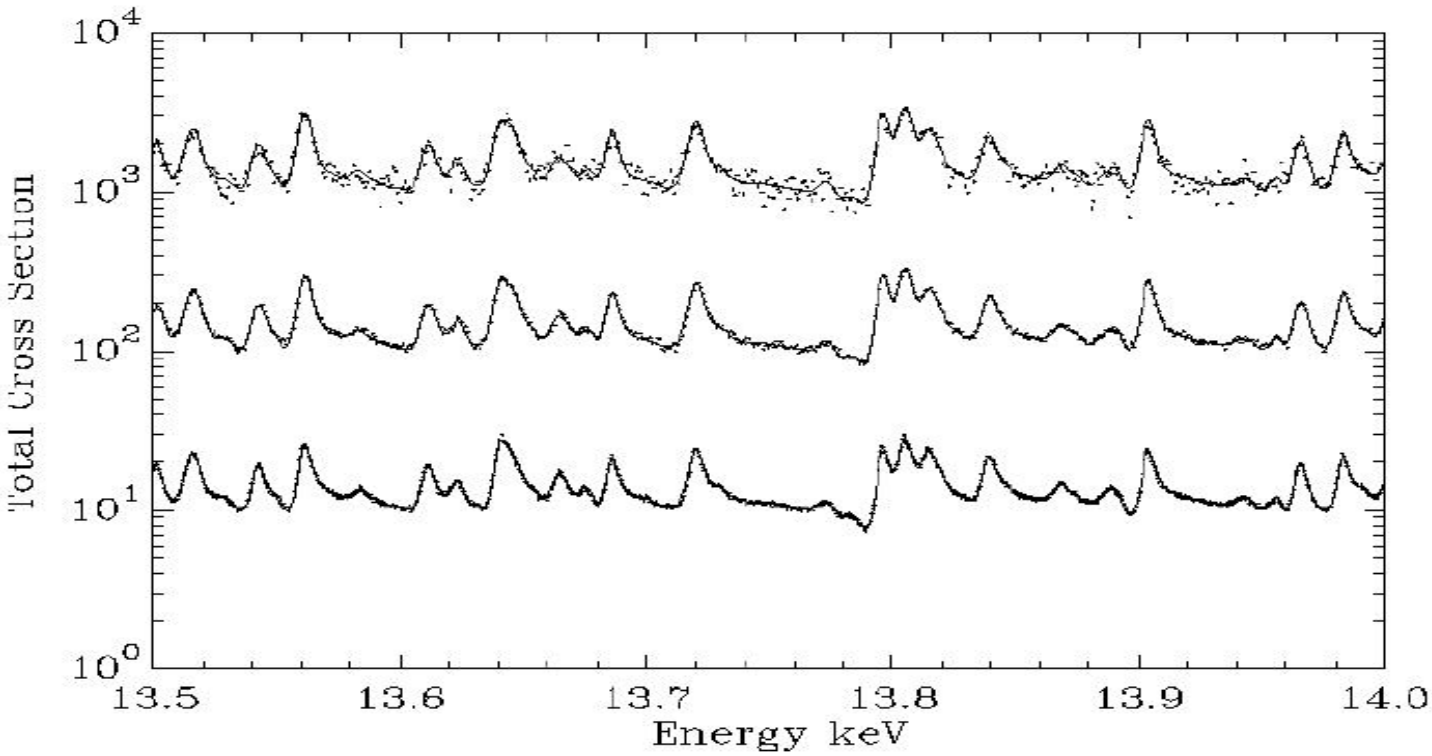
Results for the evaluation energy region 10 keV to 20 keV

- 457 s-wave resonances and 1228 p-wave resonances obtained with SAMMY analysis of Harvey transmission data and DeSaussure Capture cross section data

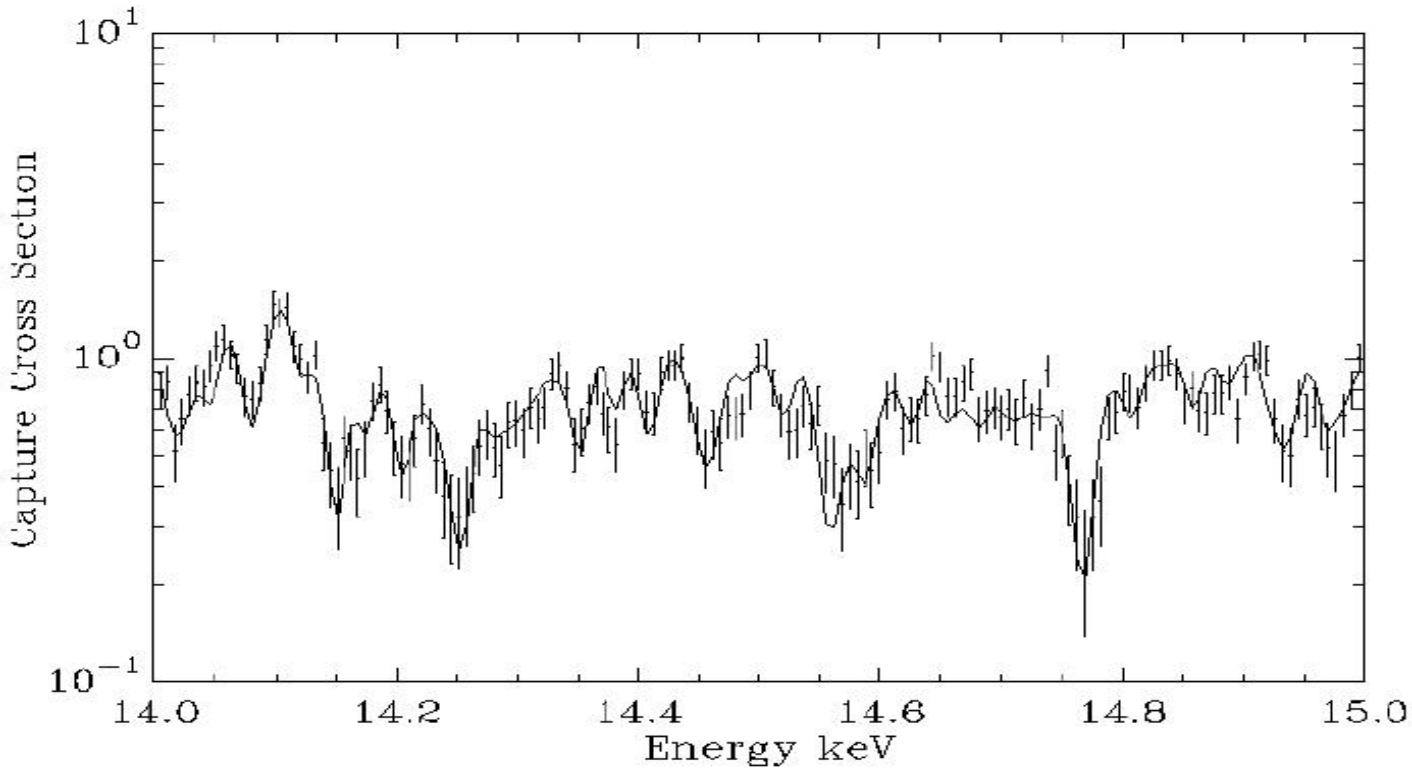
- The average capture cross section agrees with the ENDF/B-VI evaluation;

- The average elastic cross section is larger than the ENDF/B-VI evaluation by 3.1 %;

Effective Total Cross Section from the Transmission Data (upper curve x 100, middle curve x 10)



Effective Capture Cross Sections



Average ^{238}U Cross Sections (10 keV to 20 keV)

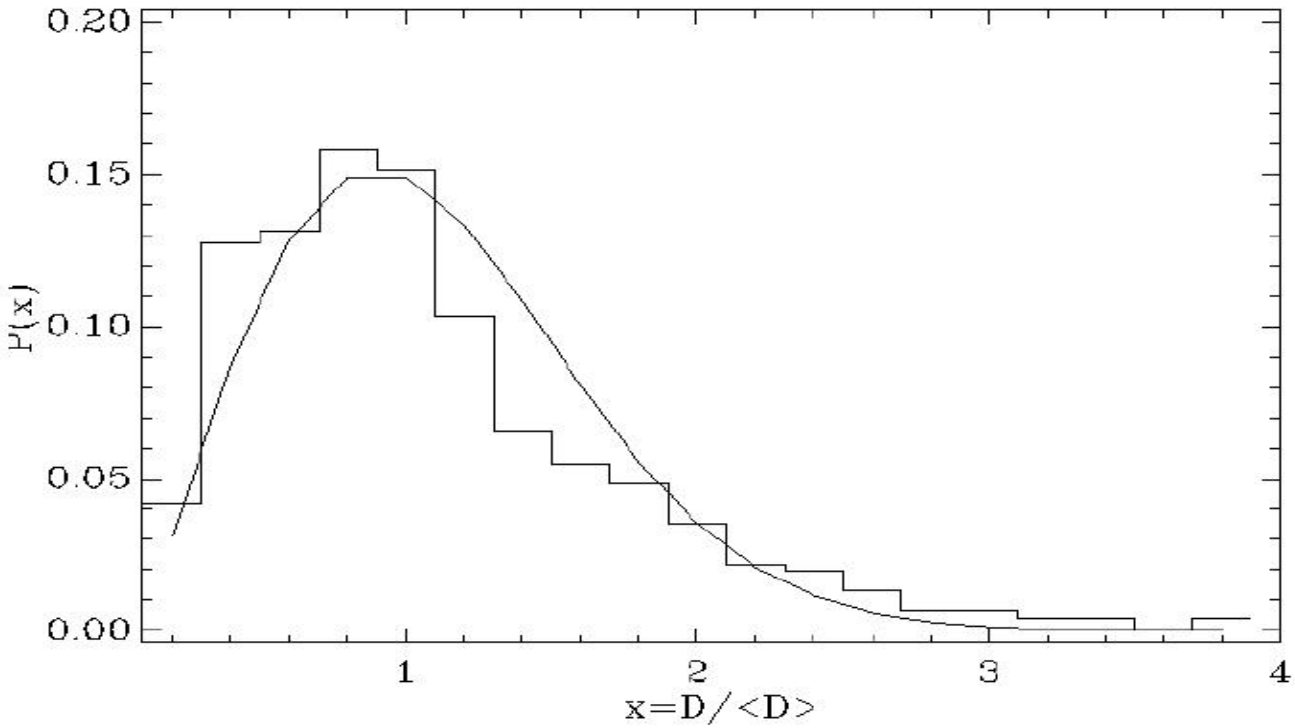
Energy Range (keV)	Capture (b)		Elastic (b)	
	ENDF/B-VI	ORNL	ENDF/B-VI	ORNL
10 – 11	0.71	0.68	14.67	14.72
11 – 12	0.68	0.65	14.53	13.91
12 – 13	0.66	0.65	14.40	15.32
13 – 14	0.64	0.72	14.29	14.56
14 – 15	0.62	0.62	14.18	15.54
15 – 16	0.60	0.58	14.09	14.60
16 – 17	0.58	0.58	14.01	13.79
17 – 18	0.57	0.58	13.93	16.25
18 – 19	0.55	0.53	13.86	14.57
19 - 20	0.54	0.52	13.79	12.91
10 - 20	0.615	0.611	14.17	14.62

Statistical Properties of Resonance Parameters in the Energy Range 10 keV to 20 keV

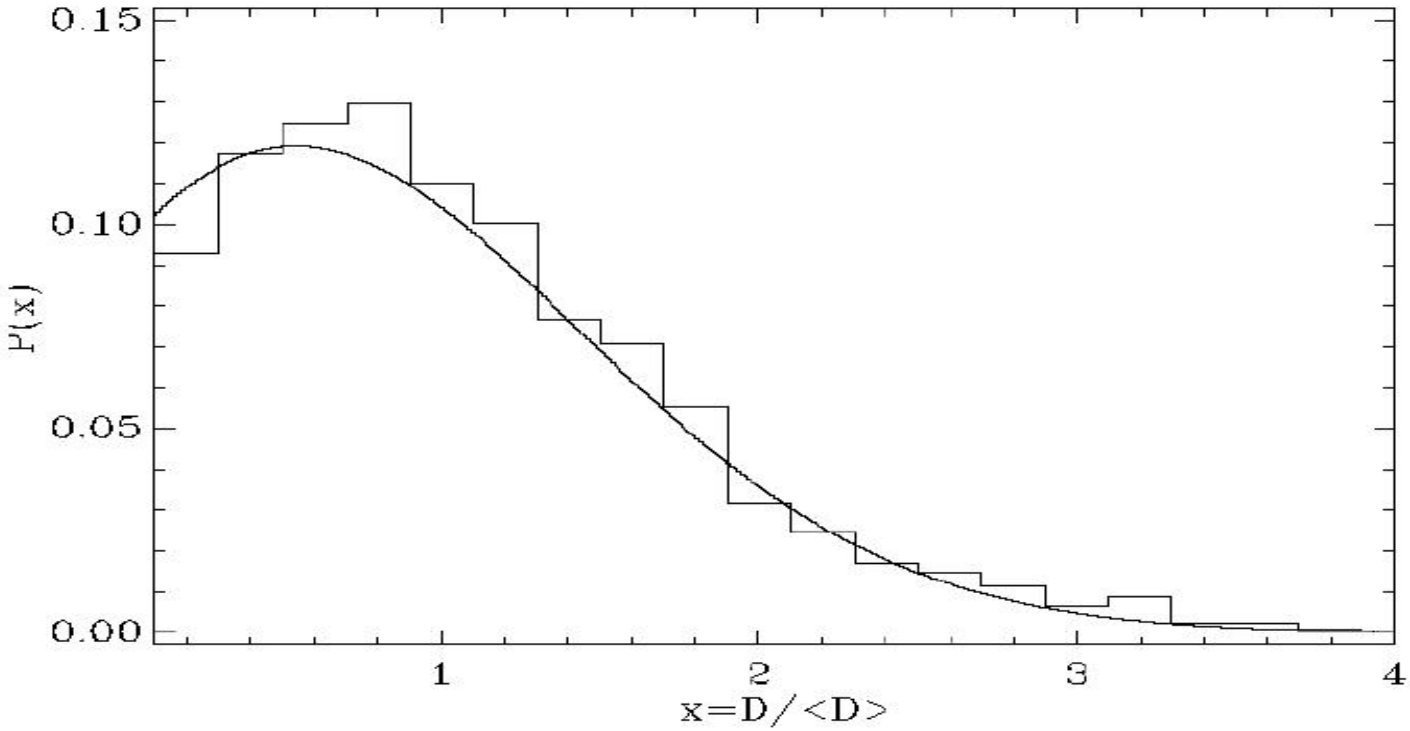
- Number of Resonances used to fit the data are 457 s-wave and 1228 p-wave;
- Below 10 keV ENDF/B evaluation uses 473 s-wave and 1228 p-wave;

- Spin Assignment:
  - Large resonances are assigned s-wave from their shape;
  - Small resonances are distributed among 3 families: s-wave, p-wave p1/2 and p3/2 following the 2J+1 spin dependence of the population;

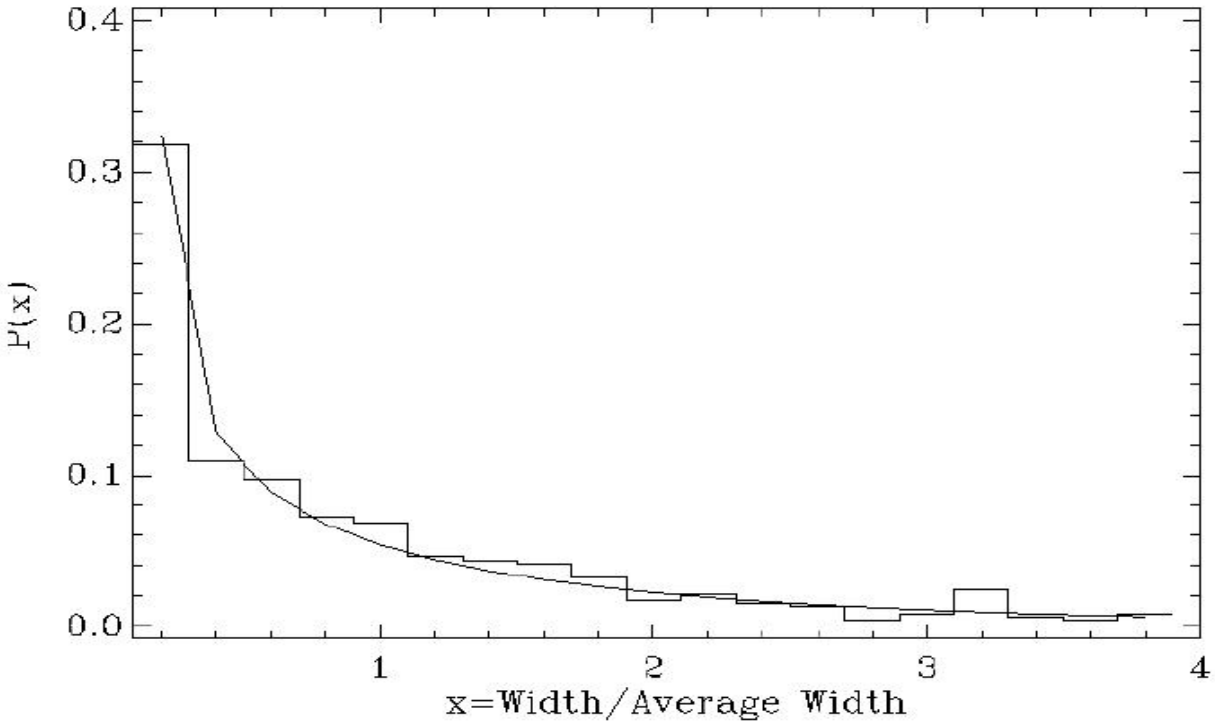
Distribution of s-wave Level Spacings



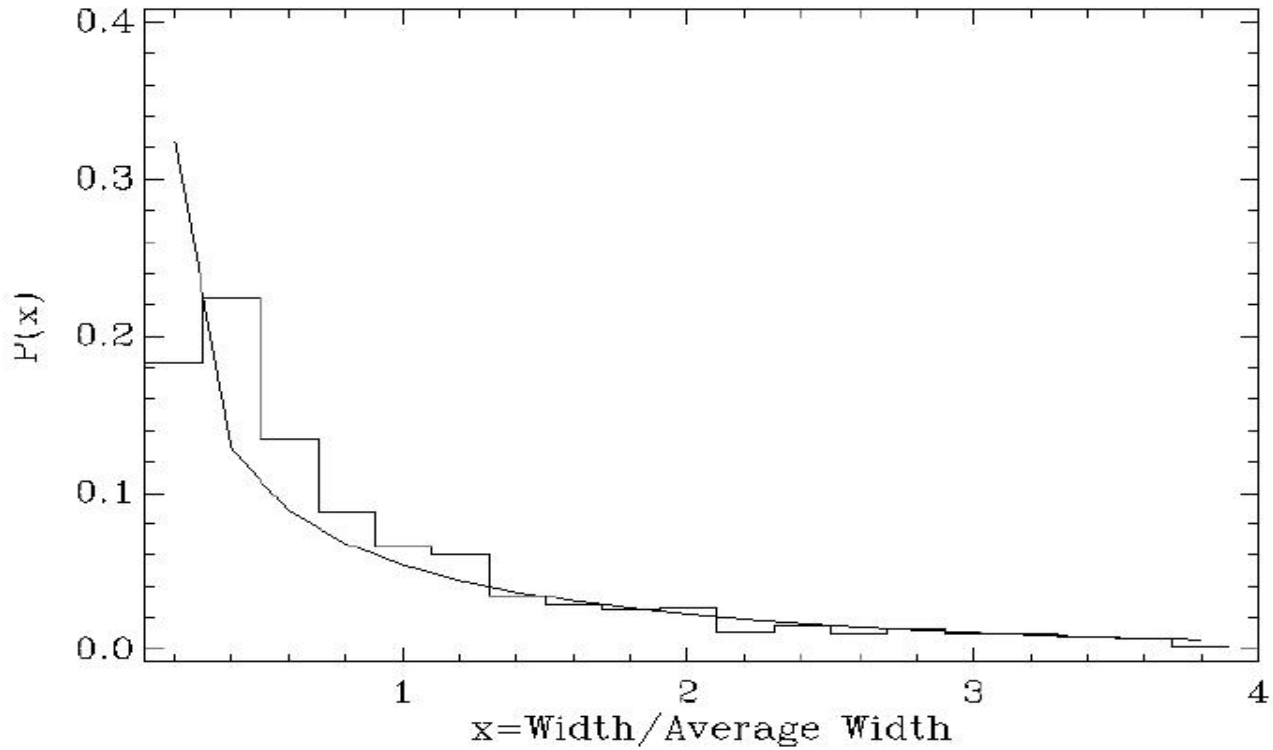
Distribution of p-wave Level Spacings



Porter-Thomas Distribution of s-wave neutron widths



Porter-Thomas Distribution of p-wave neutron widths



Neutron Strength Functions

ORNL Results 10 keV to 20 keV	ENDF/B-VI From Thermal to 10 keV
S_0 $(1.120 \pm 0.074)\ 10^{-4}$	S_0 $(0.947 \pm 0.07)\ 10^{-4}$
S_1 $(1.693 \pm 0.068)\ 10^{-4}$	S_1 $(1.577 \pm 0.06)\ 10^{-4}$

Benchmark results using ^{238}U ORNL Evaluation

Thermal benchmark (Leu-Comp-Therm Series)

Thermal Benchmark Experiments Performed at the Tank-type Critical Assembly TCA facility (JAERI, Japan) from 1963 to 1975

TCA

Tank-type Critical Assembly, light water moderated and fueled with low enriched uranium (2.6 wt % of UO_2) with a size of 1.8m in diameter and 2m long, controlled by reactor core water level. The water to fuel volume ratio ranges from 1.5 to 3.0

Benchmark Results

- Calculations done with MCNP4C code
- Base library used was the ENDF/B-VI release Five
- Calculations done with 1000 histories per cycle with 3000 cycles
- Two sets of results:

  a) Calculations with MCNP library based on ENDF/B-VI.5

  b) Calculations with MCNP library based on ENDF/B-VI.5 replacing ^{238}U evaluation with ORNL evaluation

Benchmark Results

Case	ENDF/B-VI.5	ENDF/B-VI.5 with ^{238}U ORNL Evaluation
LCT-01	0.99276 ± 0.00043	0.99506 ± 0.00043
LCT-02	0.99359 ± 0.00042	0.99429 ± 0.00043
LCT-03	0.99245 ± 0.00044	0.99568 ± 0.00042
LCT-04	0.99376 ± 0.00042	0.99534 ± 0.00041
LCT-05	0.99383 ± 0.00043	0.99471 ± 0.00042
LCT-06	0.99345 ± 0.00043	0.99654 ± 0.00042
LCT-07	0.99400 ± 0.00042	0.99604 ± 0.00042
LCT-08	0.99358 ± 0.00043	0.99610 ± 0.00043
LCT-09	0.99429 ± 0.00042	0.99582 ± 0.00041
LCT-10	0.99502 ± 0.00042	0.99767 ± 0.00041
LCT-11	0.99535 ± 0.00041	0.99704 ± 0.00041
LCT-12	0.99472 ± 0.00041	0.99619 ± 0.00042
LCT-13	0.99420 ± 0.00042	0.99557 ± 0.00042

Benchmark Results

Case	ENDF/B-VI.5	ENDF/B-VI.5 with ^{238}U ORNL Evaluation
Average	0.99392 ± 0.00042	0.99585 ± 0.00042
An improvement of 193 pcm On average !!		

Gd Evaluation

^{152}Gd, ^{154}Gd, ^{155}Gd, ^{156}Gd, ^{157}Gd, ^{158}Gd, and ^{160}Gd

- **Resolved and Unresolved Resonance Evaluations Revised**
  - **MLBW resonance parameters converted to RM parameters;**
  - **Unresolved resonance evaluation done with SAMMY: Average SLBW parameters obtained.**
  - **SAMMY used to reevaluate the RM parameters.**

- **Resolved and Unresolved Resonance Covariance Evaluation done with SAMMY**
  - **Uncertainty in the Resonance Parameters reported on "Mughabghab's Book" used as input;**
  - **"Typical" data uncertainty on "data" were used. Example: ORELA resolution function, TOF uncertainties, channel widths, jitters, etc.**

- **Use SAMMY Retroactive Scheme to Generate Covariance Data.**

Gd Evaluation
^{152}Gd, ^{154}Gd, ^{155}Gd, ^{156}Gd, ^{157}Gd, ^{158}Gd, and ^{160}Gd

- **NJOY/AMPX were used to process the new evaluation;**
- **Covariance data were processed using the ERROJ code;**
- **ERROJ produces covariance data in the COVERX format;**
- **A fictitious benchmark problem with ^{235}U, ^{27}Al and Gd isotopes were processed with the TSUNAMI code**

Gd Evaluation

^{152}Gd, ^{154}Gd, ^{155}Gd, ^{156}Gd, ^{157}Gd, ^{158}Gd, and ^{160}Gd

- **EXAMPLE**

TSUNAMI OUTPUT

```
-----------------------------
Uncertainty Information
-----------------------------

 the relative standard deviation of k-eff is:
   4.7468  0.0359 percent

  contributions *10**4 by reaction pairs to the
  relative covariances of k-eff

          diagonal                          off-diagonal

al-27  n,gamma  1.7880E  3.3853E-01
                                  al-27  elastic  2.7661E-02  7.8052E-03
                                  al-27  n,n'     3.7241E-06  7.4240E-07
u-235  fission  2.4227E  2.4039E-02
u-235  n,gamma  1.9596E  1.6813E-02
gd-152 n,gamma  1.3633E-01  1.4977E-03
                                  gd-152 elastic  8.2269E-05  1.4698E-05
```

Gd Evaluation

^{152}Gd, ^{154}Gd, ^{155}Gd, ^{156}Gd, ^{157}Gd, ^{158}Gd, and ^{160}Gd


```
al-27  elastic  7.6248E-02  2.6633E-02
                            al-27  n,gamma  2.7661E-02  7.8052E-03
                            al-27  n,n'    -1.0436E-04  9.0945E-04
                            al-27  n,p      3.0675E-06  2.2781E-06
                            al-27  n,alpha  6.0743E-08  3.3522E-08
al-27  n,n'     1.6999E-03  6.0255E-03
                            al-27  elastic -1.0436E-04  9.0945E-04
                            al-27  n,gamma  3.7241E-06  7.4240E-07
                            al-27  n,p      1.6912E-06  3.3714E-07
                            al-27  n,alpha  2.9782E-07  5.9370E-08
al-27  n,n'     1.6999E-03  6.0255E-03
                            al-27  elastic -1.0436E-04  9.0945E-04
                            al-27  n,gamma  3.7241E-06  7.4240E-07
                            al-27  n,p      1.6912E-06  3.3714E-07
                            al-27  n,alpha  2.9782E-07  5.9370E-08

                            al-27  n,2n    -8.4457E-09  1.6854E-09
al-27  n,p      8.7930E-04  1.6291E-05
                            al-27  elastic  3.0675E-06  2.2781E-06
                            al-27  n,n'     1.6912E-06  3.3714E-07
```

Gd Evaluation

^{152}Gd, ^{154}Gd, ^{155}Gd, ^{156}Gd, ^{157}Gd, ^{158}Gd, and ^{160}Gd

al-27 n,alpha 6.8494E-06 1.8273E-07
al-27 n,n' 2.9782E-07 5.9370E-08
al-27 elastic 6.0743E-08 3.3522E-08

gd-152 elastic 1.4253E-06 1.2928E-07
gd-152 n,gamma 8.2269E-05 1.4698E-05
gd-152 n,n' -3.2620E-12 1.9493E-11
gd-152 n,2n -4.6678E-15 2.4573E-15
gd-152 n,alpha 9.0589E-18 0.0000E
gd-152 n,p 1.0529E-18 0.0000E
al-27 n,2n 7.9395E-08 4.0666E-09
al-27 n,n' -8.4457E-09 1.6854E-09
gd-155 n,gamma 1.7760E-08 2.2270E-10
gd-154 n,gamma 5.5774E-09 9.3893E-11
gd-156 n,gamma 4.8276E-09 6.3828E-11
gd-157 n,gamma 3.3695E-09 4.1441E-11
gd-152 n,n' 2.4069E-10 6.3081E-10
gd-152 elastic -3.2620E-12 1.9493E-11
gd-152 n,2n 3.9807E-12 2.2464E-13
gd-152 elastic -4.6678E-15 2.4573E-15

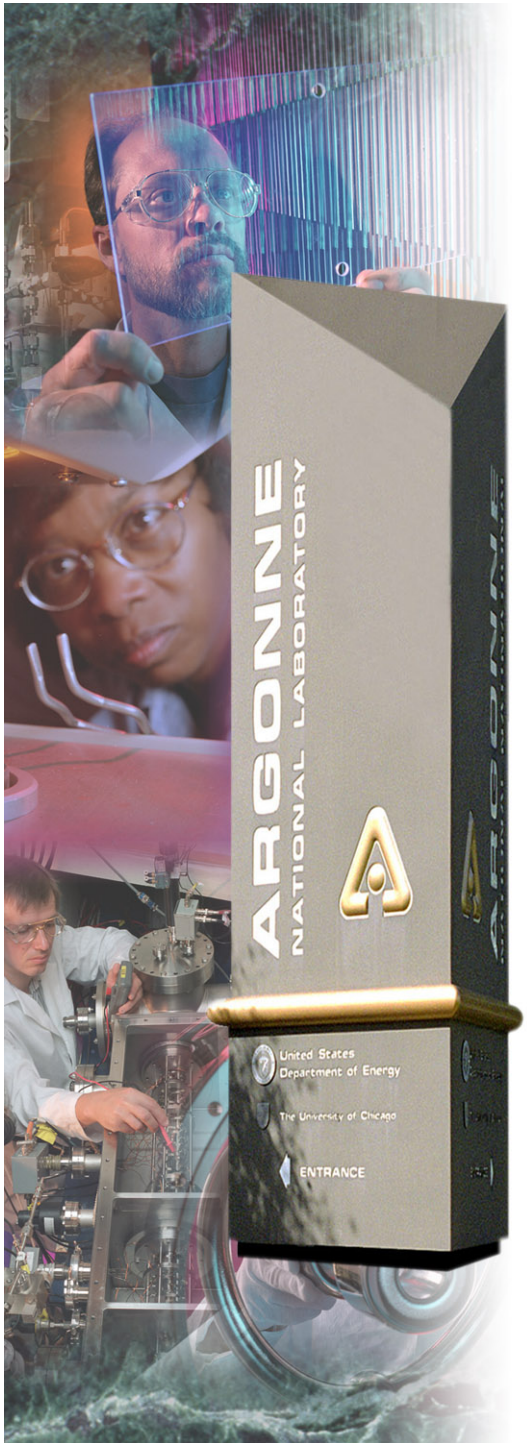
Gd Evaluation

^{152}Gd, ^{154}Gd, ^{155}Gd, ^{156}Gd, ^{157}Gd, ^{158}Gd, and ^{160}Gd


```
gd-157 n,n'     1.9659E-12  6.9852E-13
gd-155 n,n'     1.1530E-12  4.1710E-13
gd-154 elastic  3.3342E-14  7.8187E-15
gd-152 n,alpha  3.2969E-14  0.0000E
                            gd-152 elastic  9.0589E-18  0.0000E
gd-157 elastic  1.9663E-14  6.4717E-15
gd-156 elastic  1.1167E-14  1.8321E-15
gd-155 elastic  9.4478E-15  2.0940E-15
gd-156 n,n'     3.1840E-16  0.0000E
gd-152 n,p      1.8318E-16  0.0000E
                            gd-152 elastic  1.0529E-18  0.0000E
gd-154 n,n'     2.8809E-17  0.0000E

--------------------------
 SAMS execution complete
--------------------------
```

APPENDIX G



The impact of covariance information on criticality safety calculations in the resolved resonance energy range

D. G. Naberejnev, G. Palmiotti and W. S. Yang
Nuclear Engineering Division

NDAG meeting, BNL, Nov. 3, 2003

Argonne National Laboratory

Office of Science
U.S. Department of Energy

A U.S. Department of Energy
Office of Science Laboratory
Operated by The University of Chicago

Introduction

- **A method to estimate group cross-section covariances in the resonance energy range**
- **Numerical results on group cross-sections uncertainties and correlations**
- **Application to the Uranium/Iron Benchmark Assembly**
- **Application to the ZEUS Critical Assembly**
- **Conclusion and future work**

A method to estimate group XS covariances in the resonance energy range

- **Group (G) cross-section variation with the change of parameter** p_i

$$\delta\sigma^G = \sum_i \left(\partial\sigma^G / \partial p_i\right) \delta p_i$$

- **Covariance matrix and packaging of resonances (g)**

$$\left\langle \delta\sigma_x^G, \delta\sigma_x^{G'} \right\rangle = \sum_i \sum_j \frac{\partial\sigma^G}{\partial p_i} \frac{\partial\sigma^{G'}}{\partial p_j} \left\langle \delta p_i, \delta p_j \right\rangle = \sum_g \sum_{g'} \left(\sum_{i\in g} \frac{\partial\sigma^G}{\partial p_i^g} \sum_{j\in g'} \frac{\partial\sigma^{G'}}{\partial p_j^{g'}} \left\langle \delta p_i^g, \delta p_j^{g'} \right\rangle \right)$$

- **Assumptions:**

1) resonances inside packages are 100% correlated: $p_i^g \rightarrow p^g$

2) each resonance in a package contributes equally to the cross-section value in the package: $\sum_{i\in g} \left(\partial\sigma^G / \partial p_i^g\right) \rightarrow \partial\sigma^G / \partial p_g$

- **Result**

$$\left\langle \delta\sigma^G, \delta\sigma^{G'} \right\rangle = \sum_g \sum_{g'} \frac{\partial\sigma^G}{\partial p_g} \frac{\partial\sigma^{G'}}{\partial p_{g'}} \left\langle \delta p_g, \delta p_{g'} \right\rangle$$

Scheme of calculations

- **g=G**

- $\langle \delta p_G, \delta p_{G'} \rangle = \delta p_G \delta p_{G'} C_{p_G p_{G'}}$, **where** $C_{p_G p_{G'}} = 1 - 1.5 * \frac{|\overline{E}_G - \overline{E}_{G'}|}{\varepsilon}$

  Mean group energy

  Total energy interval

- **Final relative covariance matrix** $\left\langle \frac{\delta\sigma^G}{\sigma_x^G}, \frac{\delta\sigma^{G'}}{\sigma_x^{G'}} \right\rangle = \left(\frac{1}{\sigma_x^G} \frac{\Delta\sigma^G}{\Delta p_G} \delta p_G \right) \left(\frac{1}{\sigma_x^{G'}} \frac{\Delta\sigma^{G'}}{\Delta p_{G'}} \delta p_{G'} \right) C_{p_G p_{G'}}$

- **Program:**

  -Reads initial ENDF tape $\Rightarrow$ **processed with NJOY (A)**

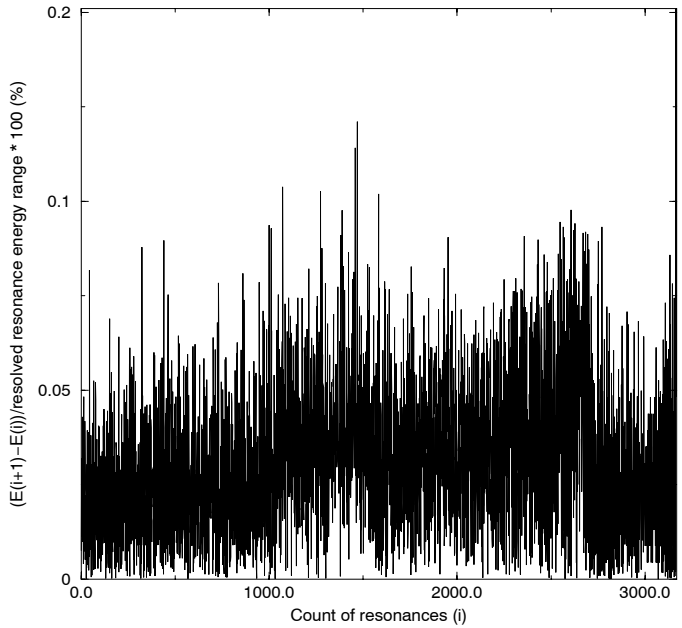
  -Modifies a portion of it as specified my the user

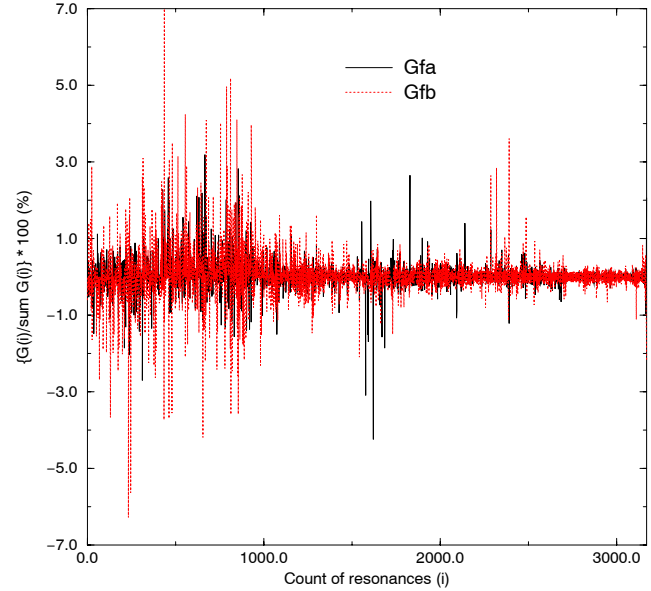
  -Creates a modified tapes (modifications specified by the user) $\Rightarrow$ **processed with NJOY (B)**

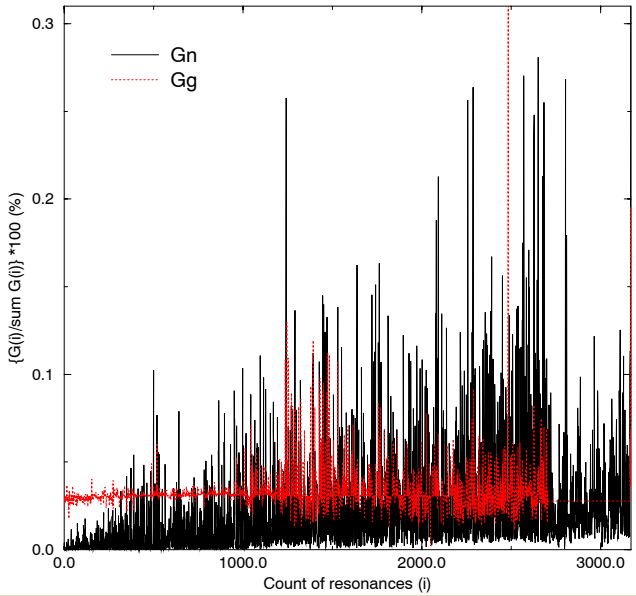
  -Comparison A and B $\Rightarrow$ $\Delta\sigma^G$

  - $\Delta p_G = 0.01\% \overline{p}_G$ $\quad \overline{p}_G = \sum_{i \in G} p_i / n_G$ $\quad \delta p_G = 10\% \overline{p}_G$ $\quad p_G = \Gamma_n$

Assumptions







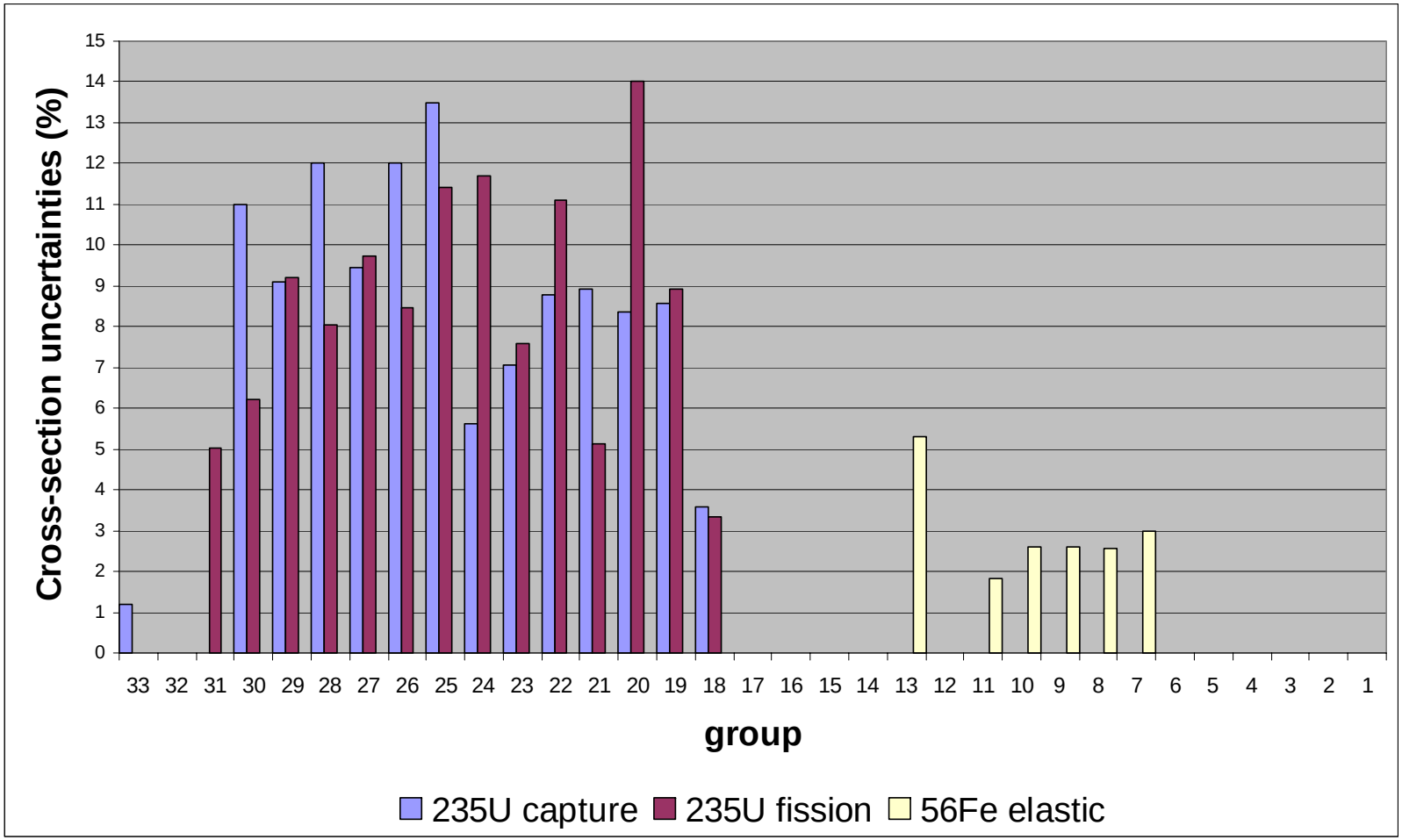
Energy group structure

G	boundaries
1	1.964E+7 to 1.000E+7
2	1.000E+7 to 6.065E+6
3	6.065E+6 to 3.678E+6
4	3.678E+6 to 2.2313E+6
5	2.2313E+6 to 1.353E+6
6	1.353E+6 to 8.208E+05
7	8.208E+5 to 4.9787E+5
8	4.9787E+5 to 3.019E+5
9	3.019E+5 to 1.831E+05
10	1.831E+5 to 1.1109E+5
11	1.1109E+5 to 6.737E+4
12	6.737E+4 to 4.086E+4
13	4.086E+4 to 2.4788E+4
14	2.4788E+4 to 1.503E+4
15	1.503E+4 to 9.118E+03
16	9.118E+3 to 5.5308E+3
17	5.5308E+3 to 3.354E+3
18	3.354E+3 to 2.034E+3
19	2.034E+3 to 1.2341E+3
20	1.2341E+3 to 7.485E+2
21	7.485E+2 to 4.539E+2
22	4.539E+2 to 3.0432E+2
23	3.0432E+2 to 1.486E+2
24	1.486E+2 to 9.166E+1
25	9.166E+1 to 6.7904E+1
26	6.7904E+1 to 4.016E+1
27	4.016E+1 to 2.260E+1
28	2.260E+1 to 1.3709E+1
29	1.3709E+1 to 8.315E+0
30	8.315E+0 to 4.00E+0
31	4.00E+0 to 5.4E-01
32	5.4E-01 to 4.14E-01
33	4.14E-01 to thermal

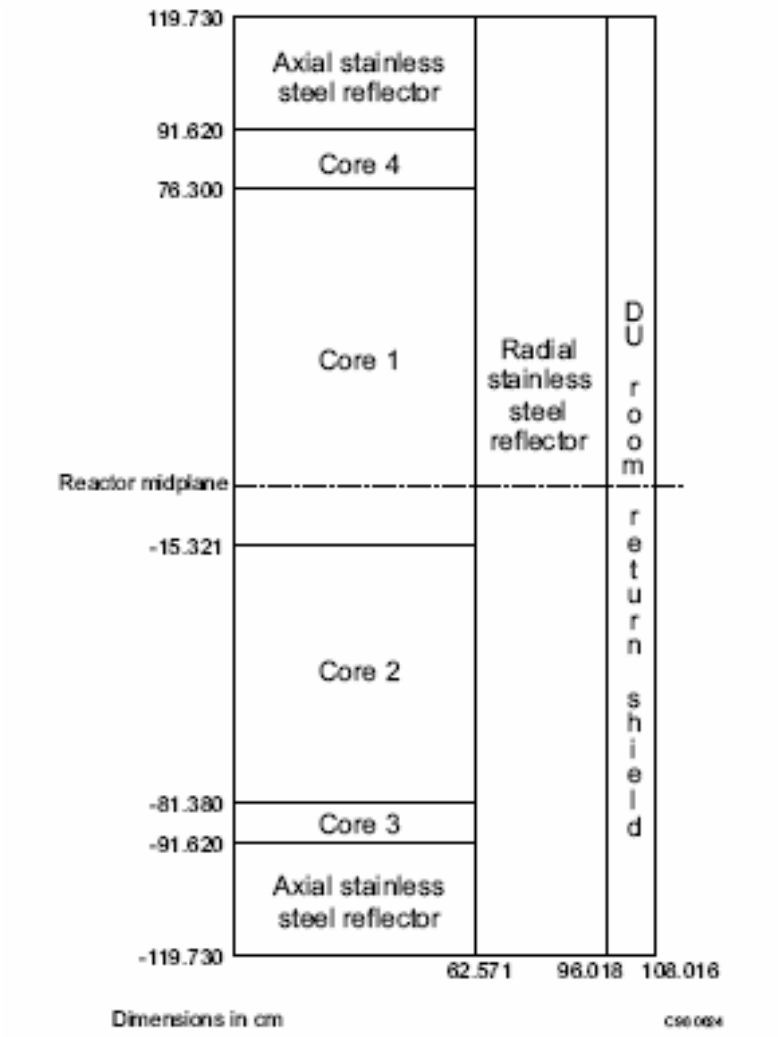
Results : $\left(\frac{1}{\sigma_x^G}\frac{\Delta\sigma_x^G}{\Delta p_G}\delta p_G\right)$ and correlations (variation of Γ_n by +10%)

	^{56}Fe elastic			^{56}Fe elastic	^{235}U fission	^{235}U capture
G	G'	COR	group			
13	13	1.00E+00	33	0	0	0.0121
13	11	9.34E-01	32	0	0	0
11	11	1.00E+00	31	0	0.0503	0
13	10	8.66E-01	30	0	0.0622	0.1100
11	10	9.32E-01	29	0	0.0921	0.0911
10	10	1.00E+00	28	0	0.0804	0.1200
13	9	7.55E-01	27	0	0.0972	0.0944
11	9	8.21E-01	26	0	0.0845	0.1200
10	9	8.89E-01	25	0	0.1140	0.1350
9	9	1.00E+00	24	0	0.1170	0.0563
13	8	5.71E-01	23	0	0.0760	0.0706
11	8	6.37E-01	22	0	0.1110	0.0878
10	8	7.05E-01	21	0	0.0514	0.0891
9	8	8.16E-01	20	0	0.1400	0.0837
8	8	1.00E+00	19	0	0.0892	0.0857
13	7	2.68E-01	18	0	0.0333	0.0359
11	7	3.34E-01	17	0	0	0
10	7	4.01E-01	16	0	0	0
9	7	5.13E-01	15	0	0	0
8	7	6.97E-01	14	0	0	0
7	7	1.00E+00	13	0.0531	0	0
			12	0	0	0
			11	0.0181	0	0
			10	0.0260	0	0
			9	0.0260	0	0
			8	0.0256	0	0
			7	0.0298	0	0
			6	0	0	0
			5	0	0	0
			4	0	0	0
			3	0	0	0
			2	0	0	0
			1	0	0	0

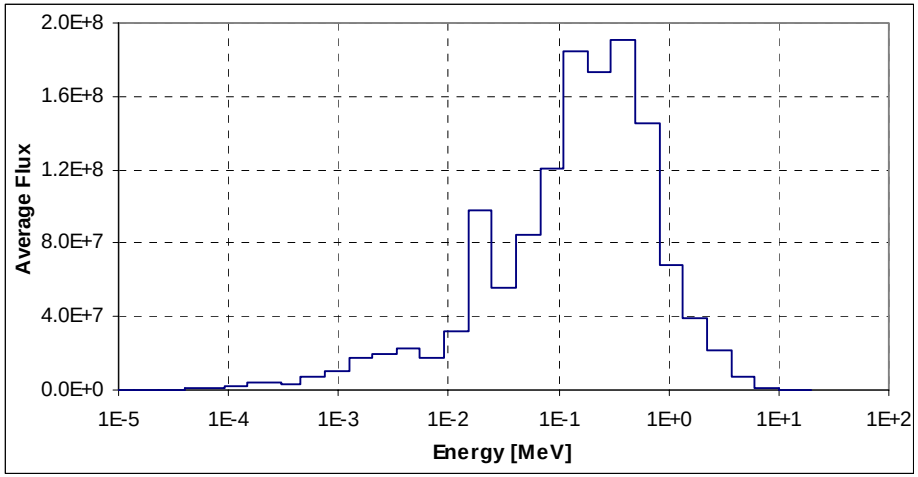
Results : $\left(\frac{1}{\sigma_x^G}\frac{\Delta\sigma_x^G}{\Delta p_G}\delta p_G\right)$ (variation of Γ_n by +10%)



Results: The Uranium/Iron Benchmark Assembly (HEU-MET-INTER-001)



- **235U(93% enriched)/iron cylinder reflected by stainless steel**
- **Simplified R-Z geometry**
- **Transport with BISTRO Sn code**
- **Sensitivity with ERANOS system**



Results: The Uranium/Iron Benchmark Assembly (HEU-MET-INTER-001)

CAPTURE

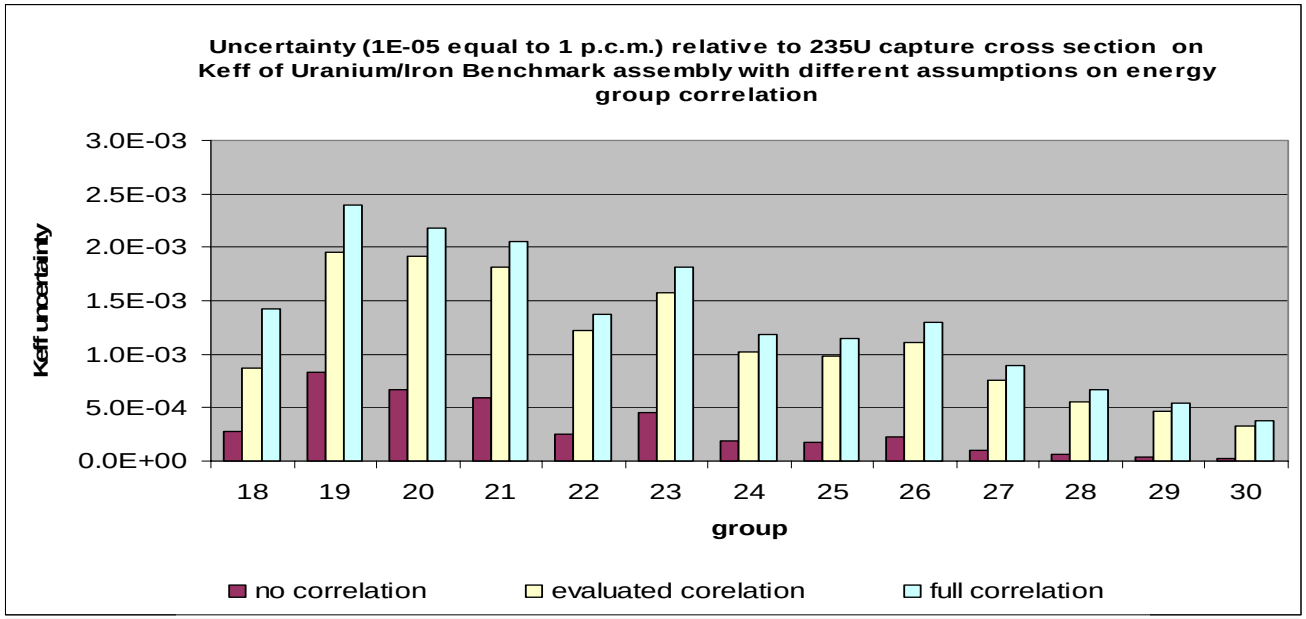
GROUP	No Correlation	Evaluated Correlation	Full Correlation
18	2.74E-04	8.70E-04	1.43E-03
19	8.33E-04	1.95E-03	2.40E-03
20	6.72E-04	1.91E-03	2.18E-03
21	5.94E-04	1.82E-03	2.06E-03
22	2.55E-04	1.22E-03	1.38E-03
23	4.54E-04	1.58E-03	1.82E-03
24	1.84E-04	1.02E-03	1.18E-03
25	1.73E-04	9.77E-04	1.15E-03
26	2.26E-04	1.11E-03	1.30E-03
27	1.03E-04	7.53E-04	8.90E-04
28	5.68E-05	5.59E-04	6.62E-04
29	3.88E-05	4.62E-04	5.47E-04
30	1.91E-05	3.24E-04	3.84E-04
Total	1.41E-03	4.46E-03	5.31E-03

FISSION

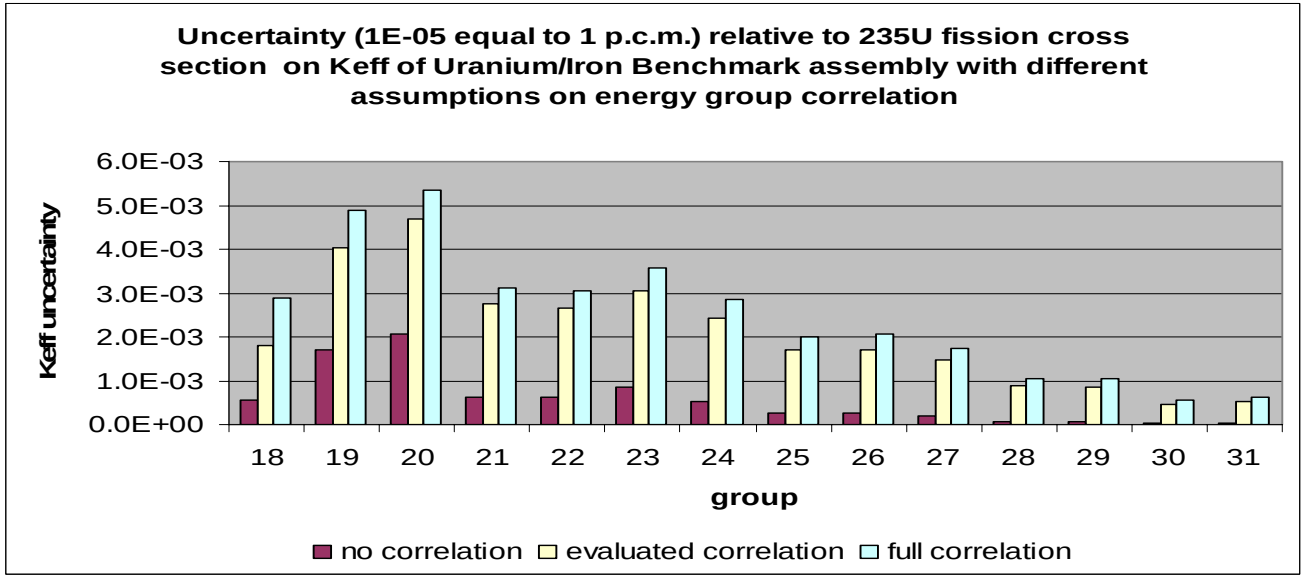
GROUP	No Correlation	Evaluated Correlation	Full Correlation
18	5.48E-04	1.79E-03	2.89E-03
19	1.70E-03	4.02E-03	4.89E-03
20	2.08E-03	4.69E-03	5.33E-03
21	6.37E-04	2.74E-03	3.10E-03
22	6.19E-04	2.67E-03	3.06E-03
23	8.50E-04	3.05E-03	3.56E-03
24	5.31E-04	2.42E-03	2.84E-03
25	2.62E-04	1.70E-03	2.01E-03
26	2.71E-04	1.72E-03	2.05E-03
27	1.95E-04	1.46E-03	1.74E-03
28	7.11E-05	8.82E-04	1.06E-03
29	6.84E-05	8.65E-04	1.04E-03
30	1.96E-05	4.63E-04	5.55E-04
31	2.56E-05	5.29E-04	6.34E-04
Total	3.08E-03	9.01E-03	1.070E-02

Uncertainty(1E-05 equal to 1 p.c.m.) relative to 235U capture and fission cross section on Keff of Uranium/Iron Benchmark assembly with different assumptions on energy group correlation. Total is obtained by statistically combining individual values.

Results: The Uranium/Iron Benchmark Assembly (HEU-MET-INTER-001)

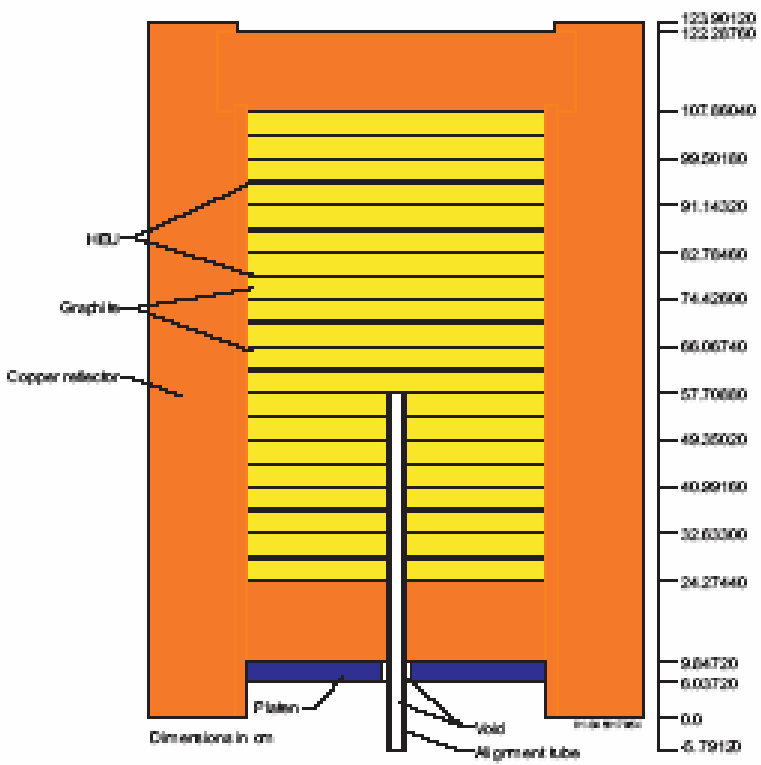


Total	
No Correlation	**1.41E-03**
Evaluated Correlation	**4.46E-03**
Full Correlation	**5.31E-03**

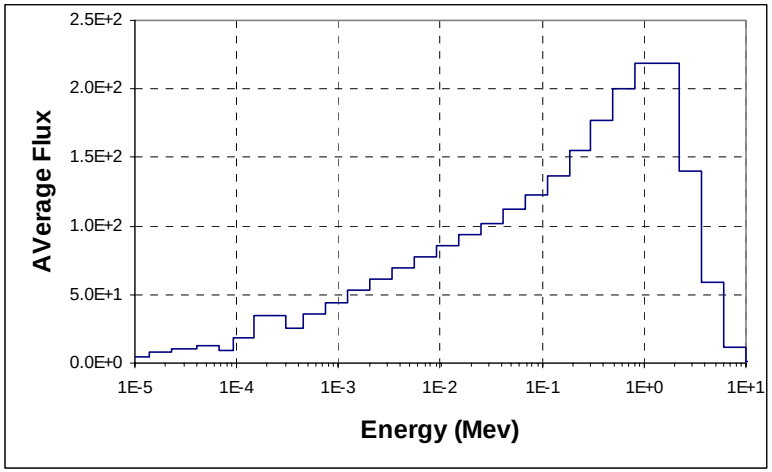


Total	
No Correlation	**3.08E-03**
Evaluated Correlation	**9.01E-03**
Full Correlation	**1.07E-02**

Results: The ZEUS Assembly (HEU-MET-INTER-006)



- A cylindrical core region containing interspersed plates of highly enriched uranium metal and graphite surrounded on all sides by a metallic copper reflector
- Simplified R-Z geometry
- Transport with BISTRO Sn code
- Sensitivity with ERANOS system



Results: The ZEUS Critical Assembly (HEU-MET-INTER-006)

CAPTURE

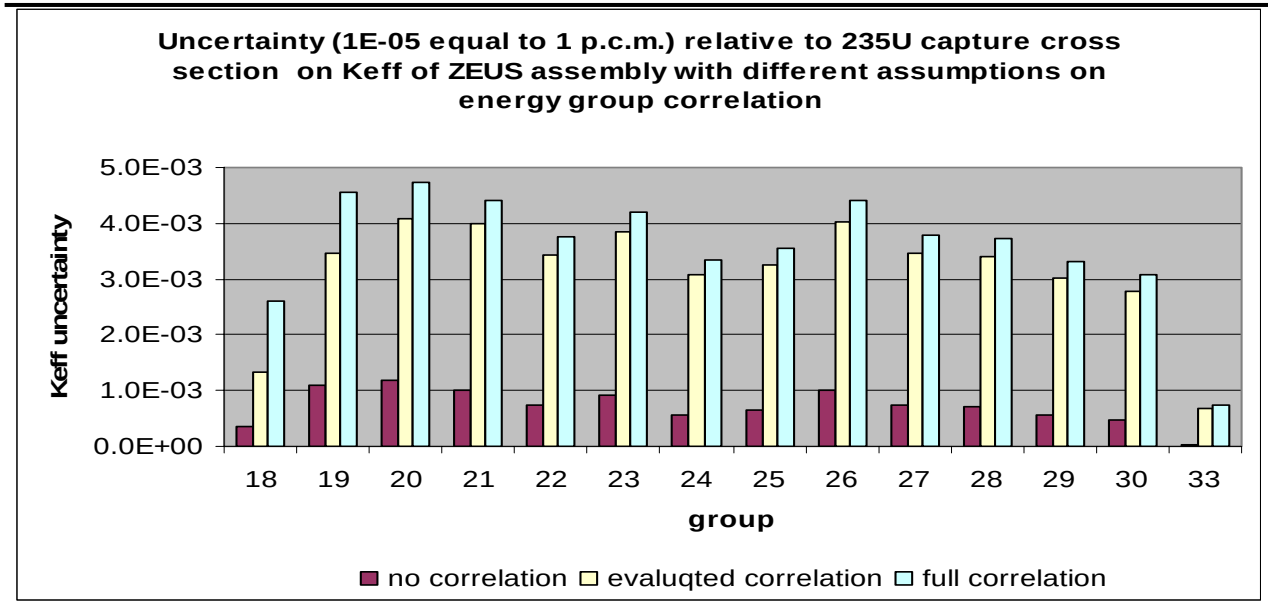
GROUP	No Correlation	Evaluated Correlation	Full Correlation
18	3.42E-04	1.33E-03	2.60E-03
19	1.09E-03	3.47E-03	4.55E-03
20	1.18E-03	4.07E-03	4.72E-03
21	1.02E-03	3.98E-03	4.42E-03
22	7.27E-04	3.43E-03	3.75E-03
23	9.16E-04	3.84E-03	4.19E-03
24	5.76E-04	3.08E-03	3.35E-03
25	6.47E-04	3.25E-03	3.55E-03
26	1.02E-03	4.03E-03	4.42E-03
27	7.48E-04	3.47E-03	3.80E-03
28	7.16E-04	3.39E-03	3.73E-03
29	5.60E-04	3.01E-03	3.31E-03
30	4.80E-04	2.79E-03	3.07E-03
33	2.66E-05	6.66E-04	7.31E-04
Total	2.92E-03	1.223E-02	1.392E-02

FISSION

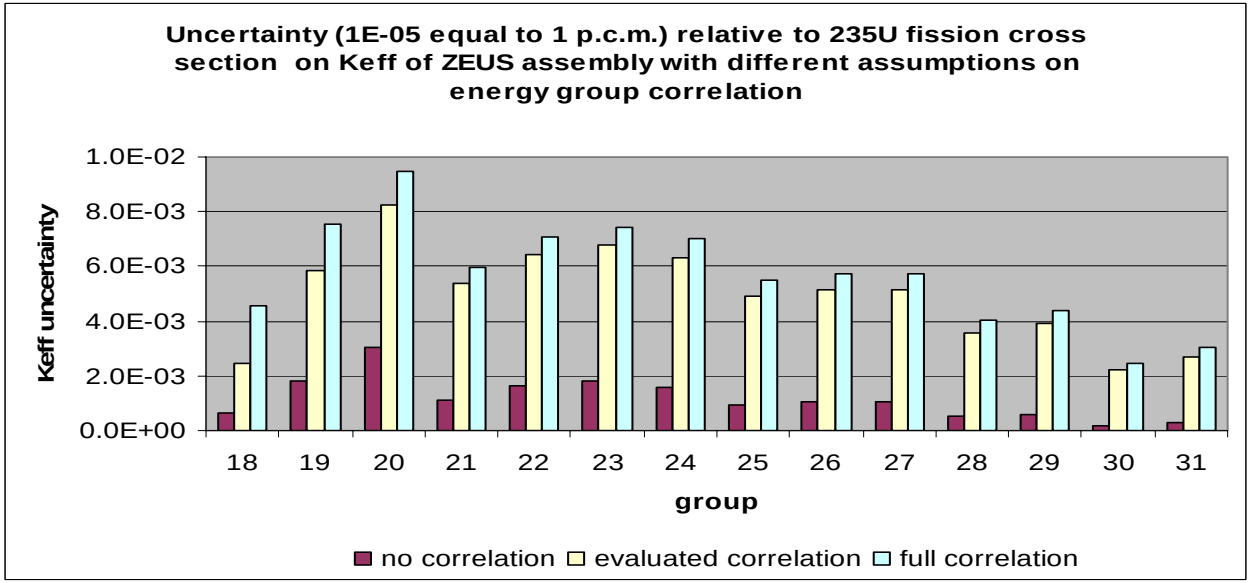
GROUP	No Correlation	Evaluated Correlation	Full Correlation
18	6.57E-04	2.46E-03	4.58E-03
19	1.84E-03	5.85E-03	7.52E-03
20	3.03E-03	8.22E-03	9.46E-03
21	1.14E-03	5.39E-03	5.98E-03
22	1.61E-03	6.41E-03	7.06E-03
23	1.81E-03	6.76E-03	7.45E-03
24	1.59E-03	6.34E-03	7.02E-03
25	9.46E-04	4.93E-03	5.47E-03
26	1.04E-03	5.14E-03	5.73E-03
27	1.04E-03	5.12E-03	5.72E-03
28	5.03E-04	3.59E-03	4.01E-03
29	5.95E-04	3.89E-03	4.36E-03
30	1.87E-04	2.20E-03	2.46E-03
31	2.86E-04	2.71E-03	3.04E-03
Total	5.15E-03	1.951E-02	2.243E-02

Uncertainty(1E-05 equal to 1 p.c.m.) relative to 235U capture and fission cross section on Keff of ZEUS assembly with different assumptions on energy group correlation . Total is obtained by statistically combining individual values.

Results: The ZEUS Critical Assembly (HEU-MET-INTER-006)



Total	
No Correlation	**2.92E-03**
Evaluated Correlation	**1.22E-02**
Full Correlation	**1.39E-02**



Total	
No Correlation	**5.15E-03**
Evaluated Correlation	**1.95E-02**
Full Correlation	**2.24E-02**

Conclusions

- **A method to construct covariances for group XS based on an ad-hoc covariances for resonance parameters was presented**
- **An attempt has been made in order to evaluate the possible impact on multiplicative factor for well studied benchmark problems**
- **The impact was judged to be significant**

  The goal of this study was not to precisely quantify the impact of covariance information in the resonances domain, but to judge how important such an information is for the applications
- **There is a need to confirm these results by using a tool that is based on experimental values and nuclear modeling information**

APPENDIX H

Monte Carlo Benchmark Calculations with Revised ^{235,238}U Evaluations

A. C. (Skip) Kahler
Bettis Atomic Power Laboratory
Bechtel Bettis, Inc.

Presented at the November 4 - 6, 2003
Cross Section Evaluation Working Group (CSEWG) Meeting
held at
Brookhaven National Laboratory

- Bettis has performed continuous energy Monte Carlo eigenvalue calculations for a variety of geometrically simple uranium fueled benchmarks with ENDF/B-VI.8, JEFF-3.0, JENDL-3.3 or alternative (BRC3, KAPL, LANL, Maslov, ORNL) $^{235,238}U$ data sets.
  - ENDF/B, JEFF-3.0 and JENDL-3.3 results were recently published in the NS&E ICSBEP special edition, Nucl. Sci. Eng., 145, 213 (2003).

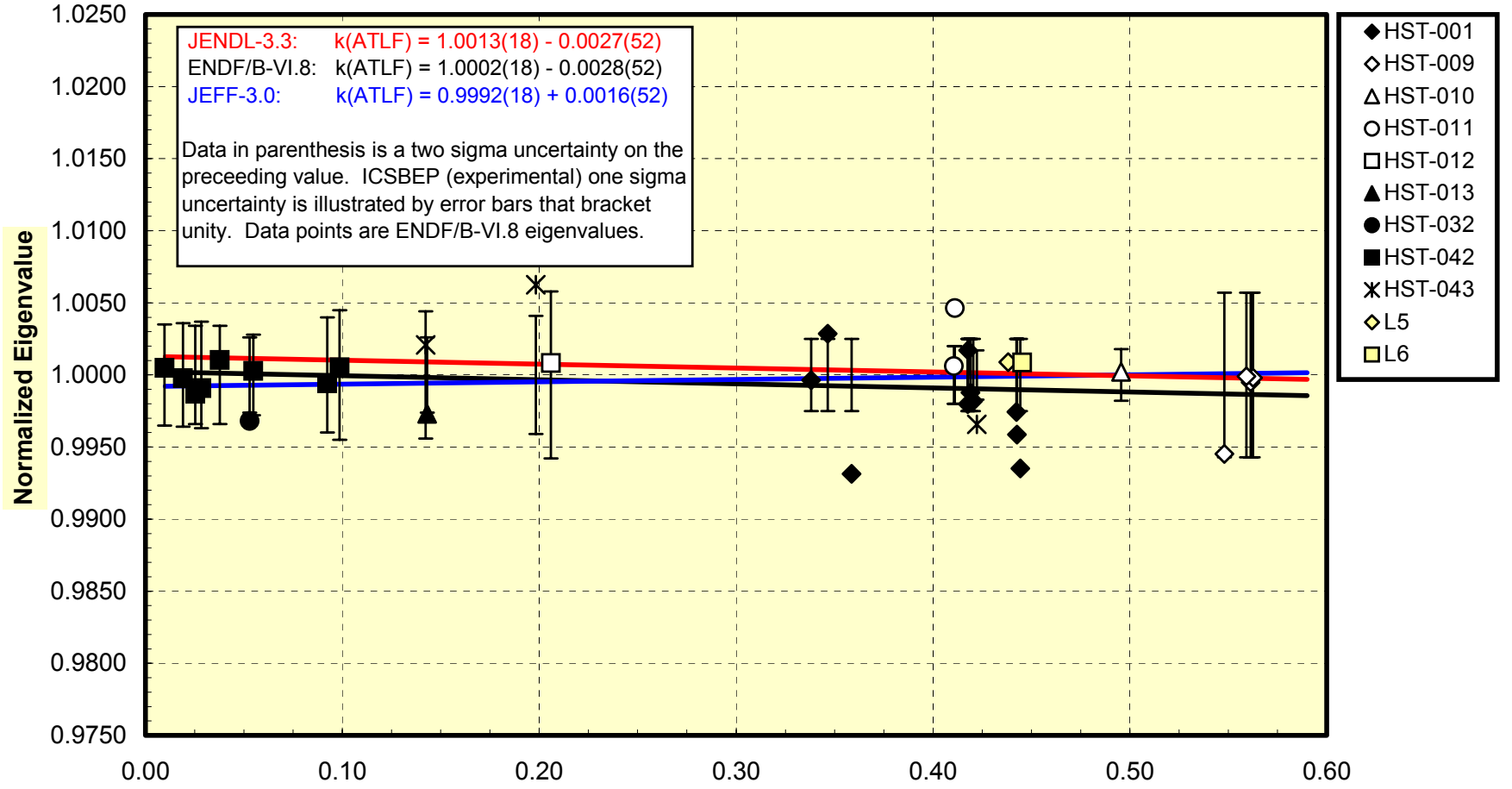
- Eigenvalues presented on subsequent slides are calculated from tracking 50 million neutron histories in ten independent 5 million history jobs.
  - The 95% eigenvalue confidence interval, determined from the variance in the ten independent eigenvalue estimates, is typically less than 0.0005 Δk (i.e., roughly the size of the plot symbol).

- A majority of these calculations use models derived from ICSBEP evaluations, as follows:
  - HEU-SOL-THERM
    - 9 evaluations, 31 critical configurations plus two ORNL experiments (L5, L6) not yet in ICSBEP.
      - 4 evaluations (8 critical configurations) include a H_2O reflector.
  - LEU-COMP-THERM
    - 8 evaluations, 59 critical configurations.
      - 2 evaluations (17 critical configurations) contain metallic reflectors.
  - LEU-SOL-THERM
    - 9 evaluations, 39 critical configurations
      - 4 evaluations (19 critical configurations) include a H_2O reflector.

Selected HEU-SOL-THERM ICSBEP Benchmarks

Benchmark	Description
HEU-SOL-THERM-001 (encompass “RF#”)	Unreflected cylinders (~28cm, ~33cm and ~51cm diameter) containing uranyl nitrate.
HEU-SOL-THERM-009 (Case 3 = ORNL L7)	Water reflected spheres (6.4-liter) containing uranium oxyfluoride.
HEU-SOL-THERM-010	Water reflected sphere (9.7-liter) containing uranium oxyfluoride.
HEU-SOL-THERM-011	Water reflected spheres (17-liter) containing uranium oxyfluoride.
HEU-SOL-THERM-012 (ORNL L10)	Water reflected sphere (91-liter) containing uranium oxyfluoride.
HEU-SOL-THERM-013 (ORNL1)	Unreflected sphere (174-liter) containing uranium nitrate.
HEU-SOL-THERM-032 (ORNL10)	Unreflected sphere (48-inch diameter) containing uranyl nitrate.
HEU-SOL-THERM-042 (encompass ORNL12 through ORNL23)	Unreflected cylinders (~77cm and ~137cm diameter) containing uranium nitrate.
HEU-SOL-THERM-043 (Cases 2 and 3 are ORNL L8 and ORNL L9)	Unreflected spheres (17-liter, 91-liter and 174-liter) containing uranium oxyfluoride solution.
ORNL L5, L6	Unreflected cylinders (13cm diameter) of uranium oxyfluoride. Low H/^{235}U ratios of 27 and 44 lead to large leakage.

HEU-SOL-THERM Eigenvalues



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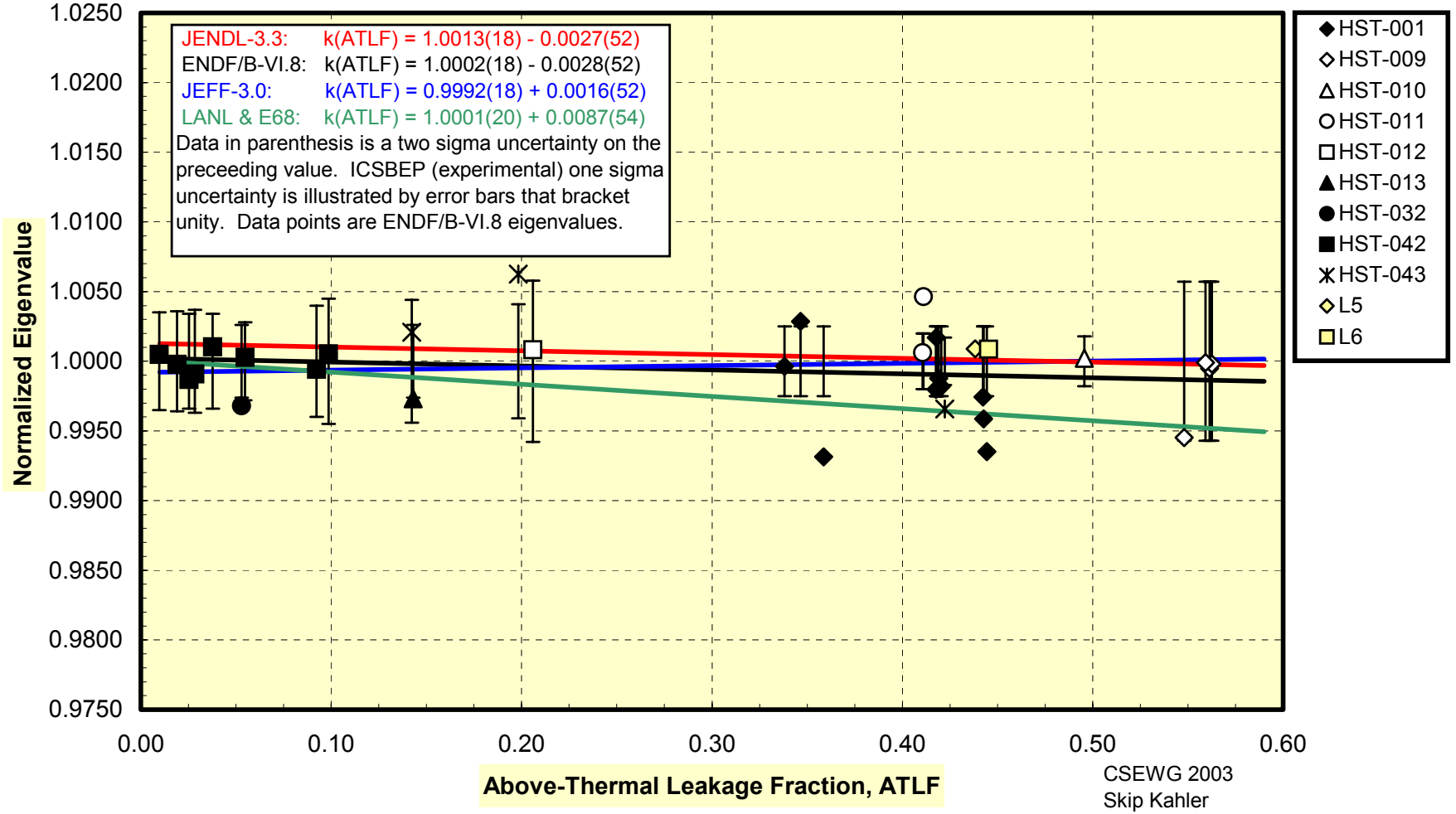
HEU-SOL-THERM Eigenvalues



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HEU-SOL-THERM Observations

- The historical trend in calculated eigenvalue as a function of Above-Thermal Leakage Fraction (ATLF) is statistically insignificant in the latest evaluated cross section libraries.
  - $k_{calc}(ATLF) = 1.0002(18)^{(a)} - 0.0028(52) * ATLF$ (ENDF/B-VI.8)
  - $k_{calc}(ATLF) = 0.9992(18) + 0.0016(52) * ATLF$ (JEFF-3.0)
  - $k_{calc}(ATLF) = 1.0013(18) - 0.0027(52) * ATLF$ (JENDL-3.3)

- The ATLF trend reappears when the latest LANL $^{235,238}U$ data sets are used. The revised ^{235}U fission spectrum is the primary cause for this reappearance.
  - $k_{calc}(ATLF) = 1.0001(20) - 0.0087(54) * ATLF$
  - In the absence of further modifications to eliminate this new trend, the User community will not perceive the latest ^{235}U evaluation to be an improvement over data sets that are already available.

- (a) Read 1.0002(18) as 1.0002 ± 0.0018, where 0.0018 is a two sigma uncertainty on the preceding value.

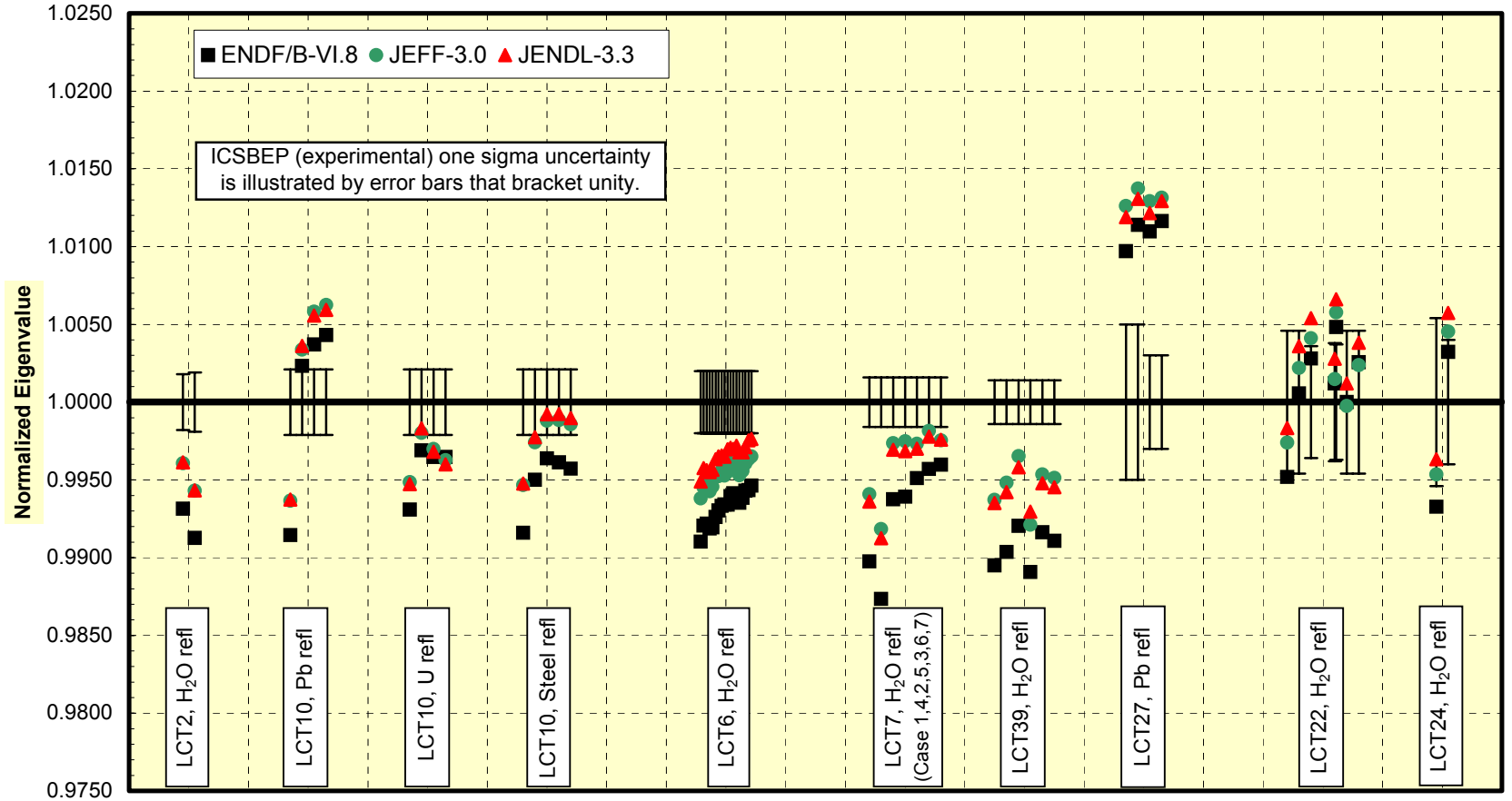
Selected LEU-COMP-THERM ICSBEP Benchmarks

Benchmark	Description
LEU-COMP-THERM-002	Water moderated and reflected U(4.31)O_2 1.415 cm OD fuel rods in 2.54 cm square pitched arrays. Cases 4 and 5 consist of three clusters of 15 x 8 and 13 x 8 rods, respectively.
LEU-COMP-THERM-006	Water moderated and reflected U(2.60)O_2 1.417 cm OD fuel rods in 1.85 cm to 2.29 cm square pitched 15x15 to 21x21 arrays.
LEU-COMP-THERM-007	Water moderated and reflected U(4.74)O_2 0.94 cm OD fuel rods in square and triangular arrays (cases 1 through 7). Pitch varies from 1.26 cm to 2.52 cm.
LEU-COMP-THERM-010	Water moderated U(4.31)O_2 1.415 cm OD fuel rods in 2.54 cm square pitched arrays reflected by two lead, uranium or steel walls at varying distances from the rod clusters. Cases 1 through 13 (mostly 13 x 8 rod clusters with a rod pitch of 2.54 cm) analyzed.
LEU-COMP-THERM-022	Water moderated and reflected U(10)O_2 0.51 cm OD fuel rods in triangular pitched (0.7 cm to 1.85 cm) arrays.
LEU-COMP-THERM-024	Water moderated and reflected U(10)O_2 0.51 cm OD fuel rods in square pitched (0.62 cm and 0.88 cm) arrays.
LEU-COMP-THERM-027	Water moderated and lead reflected U(4.74)O_2 0.94 cm OD fuel rod arrays (1.60 cm square pitch and variable distance to the lead reflector).
LEU-COMP-THERM-039	Water moderated and reflected U(4.74)O_2 0.94 cm OD fuel rods in square 1.26 cm pitch, 21x21 or 22x22 arrays. Arrays contain 1 in *n* missing rods (i.e., water holes), for *n*=5,4,3 and 2.

Selected LEU-SOL-THERM ICSBEP Benchmarks

Benchmark	Description
LEU-SOL-THERM-001	"SHEBA-II". An unreflected UO_2F_2 + H_2O cylindrical (~20 inch diameter) assembly.
LEU-SOL-THERM-002	174-liter spheres of low enriched (4.9%) uranium oxyfluoride solutions. Case 1 is a water reflected solution, case 2 is unreflected.
LEU-SOL-THERM-003	Bare spheres of 10% enriched uranyl nitrate water solutions. Cases 3, 6 and 9 are full spheres with increasing H-to-^{235}U ratio.
LEU-SOL-THERM-004	STACY; Water reflected, 10% enriched uranyl nitrate solution in a 60cm diameter cylindrical tank. Seven cases with varying gU/liter.
LEU-SOL-THERM-007	STACY; Unreflected, 10% enriched uranyl nitrate solution in a 60cm diameter cylindrical tank. Five cases with varying gU/liter.
LEU-SOL-THERM-016	STACY; Water reflected, 10% enriched uranyl nitrate solution in a rectangular (slab, ~28cm x ~69cm) tank. Seven cases with varying gU/liter.
LEU-SOL-THERM-017	STACY; Unreflected, 10% enriched uranyl nitrate solution in a rectangular (slab, ~28cm x ~69cm) tank. Six cases with varying gU/liter.
LEU-SOL-THERM-020	STACY; Water reflected, 10% enriched uranyl nitrate solution in an 80cm diameter cylindrical tank. Four cases with varying gU/liter.
LEU-SOL-THERM-021	STACY; Unreflected, 10% enriched uranyl nitrate solution in an 80cm diameter cylindrical tank. Four cases with varying gU/liter.

LEU-COMP-THERM Eigenvalues



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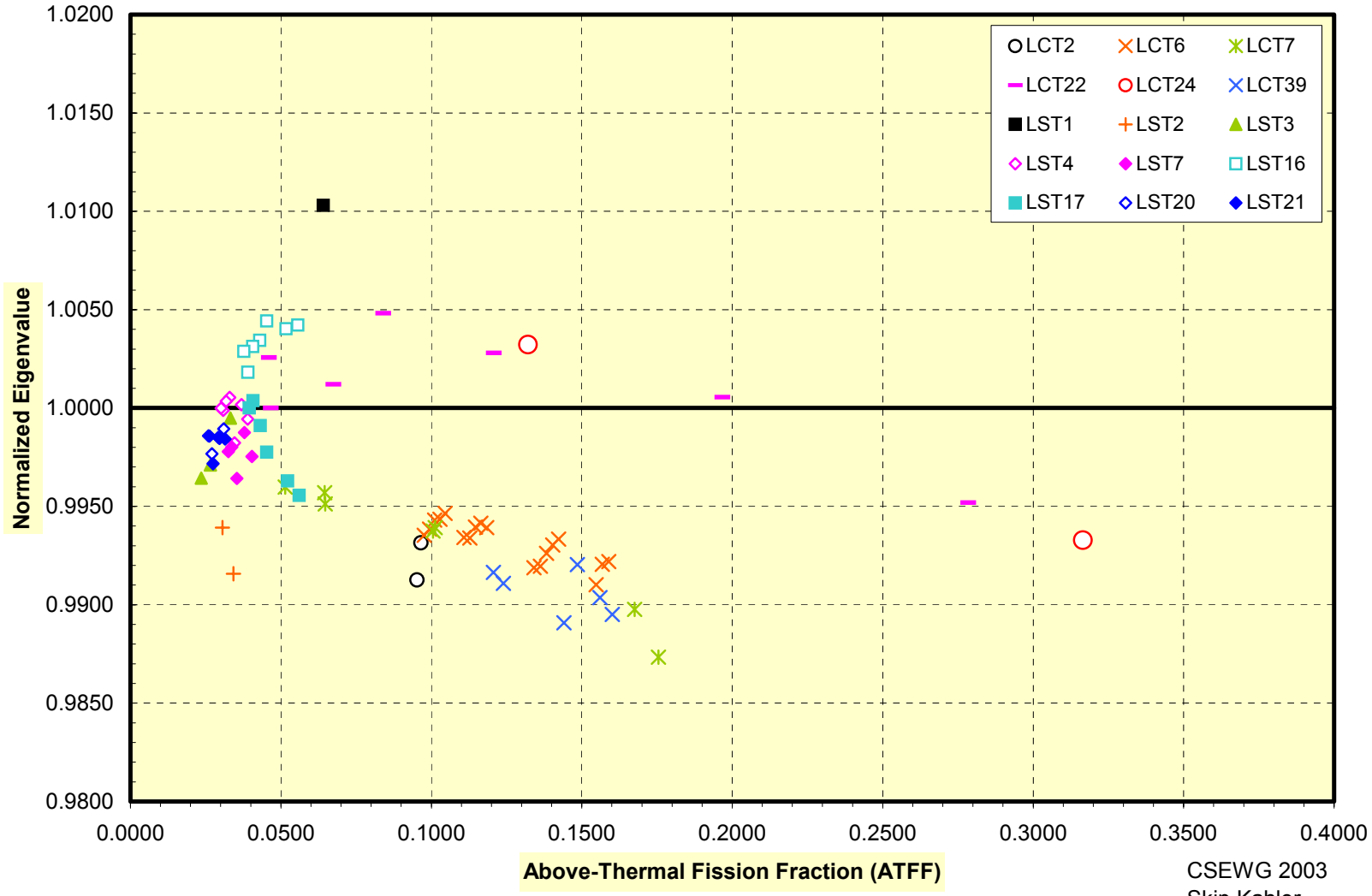
LEU-xxx-THERM Eigenvalues with ENDF/B-VI.8 Cross Sections



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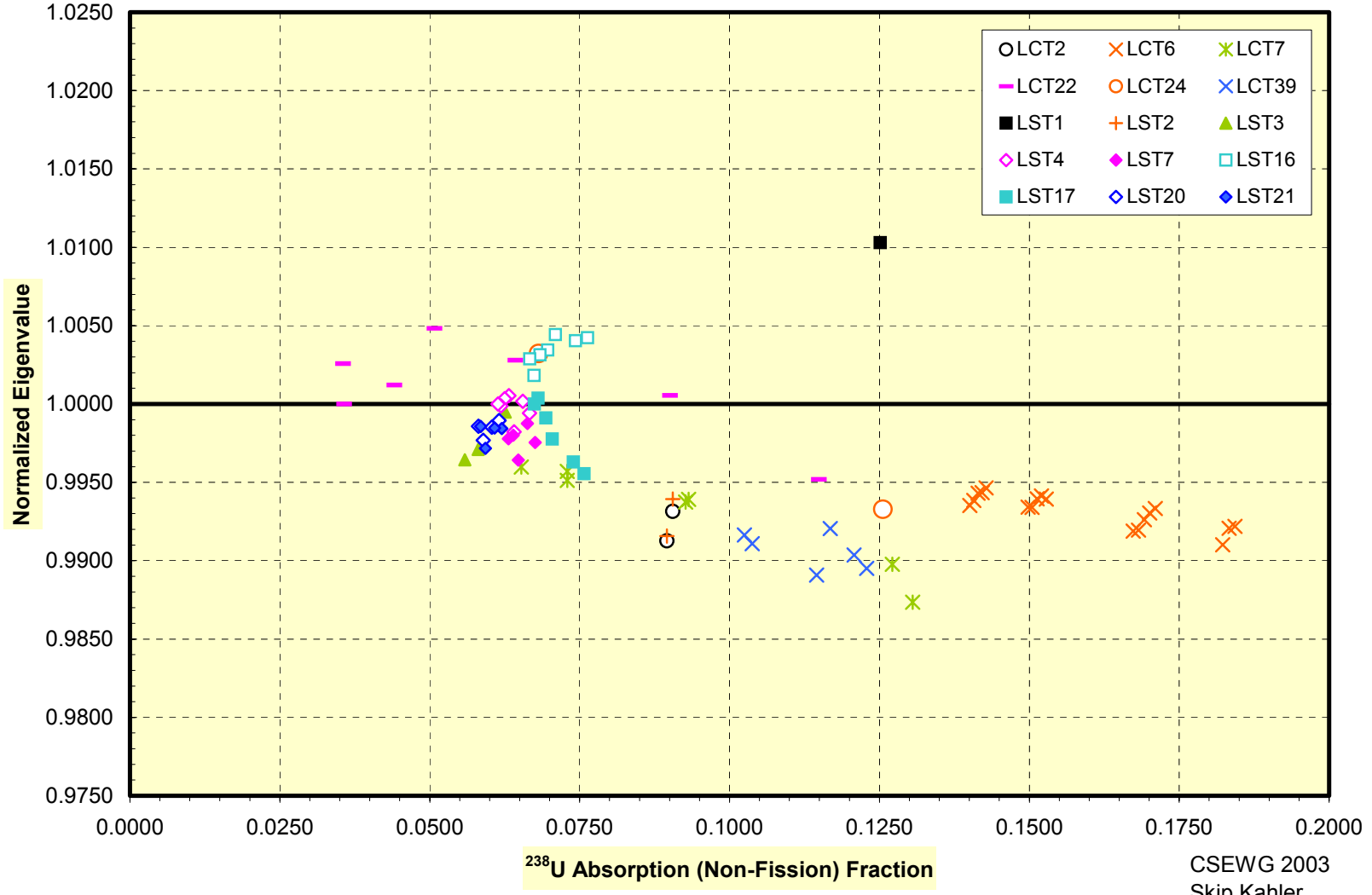
LEU-xxx-THERM Eigenvalues with ENDF/B-VI.8 Cross Sections



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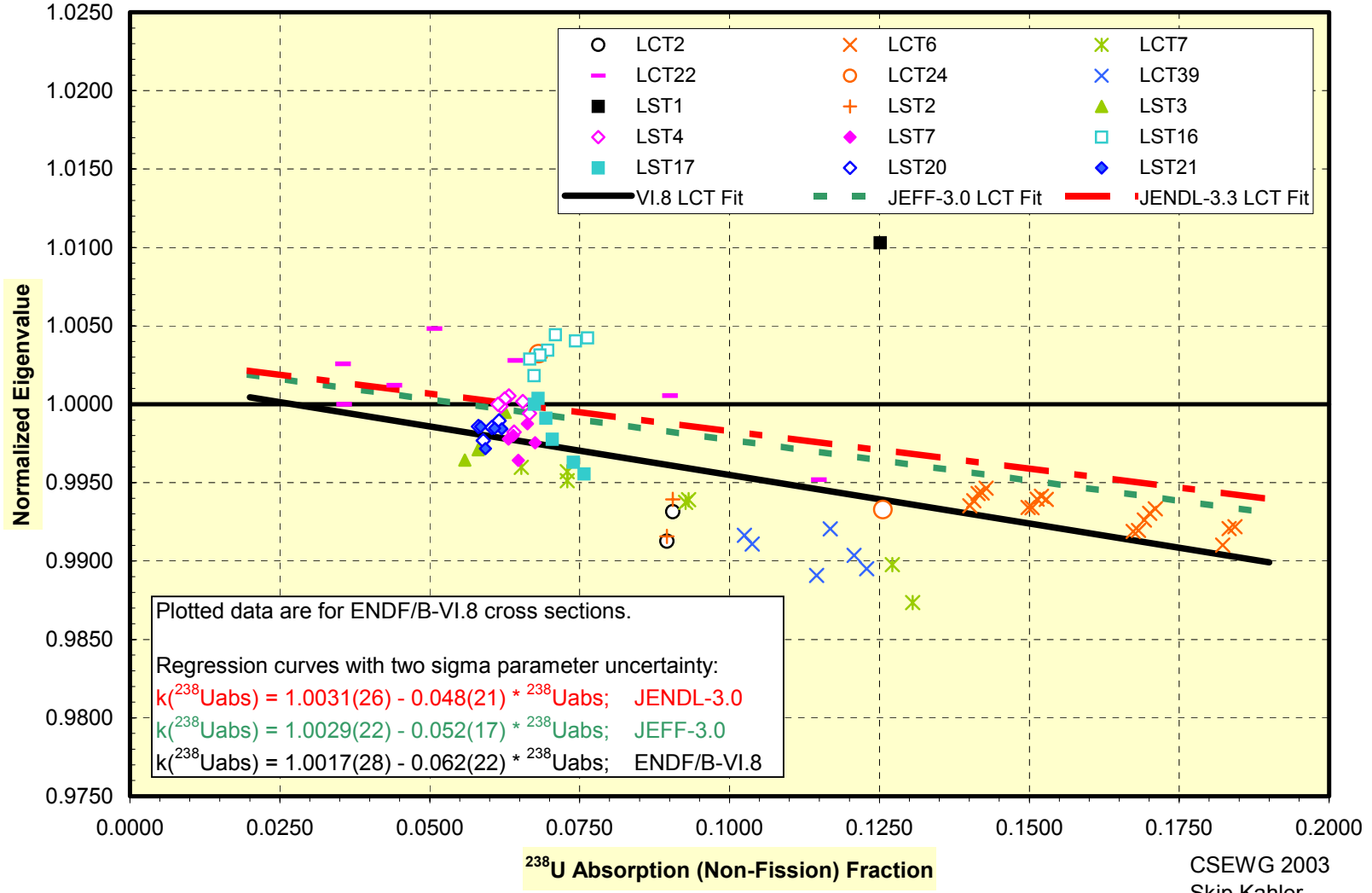
LEU-xxx-THERM Eigenvalues with ENDF/B-VI.8, JEFF-3.0 and JENDL-3.3 Cross Sections



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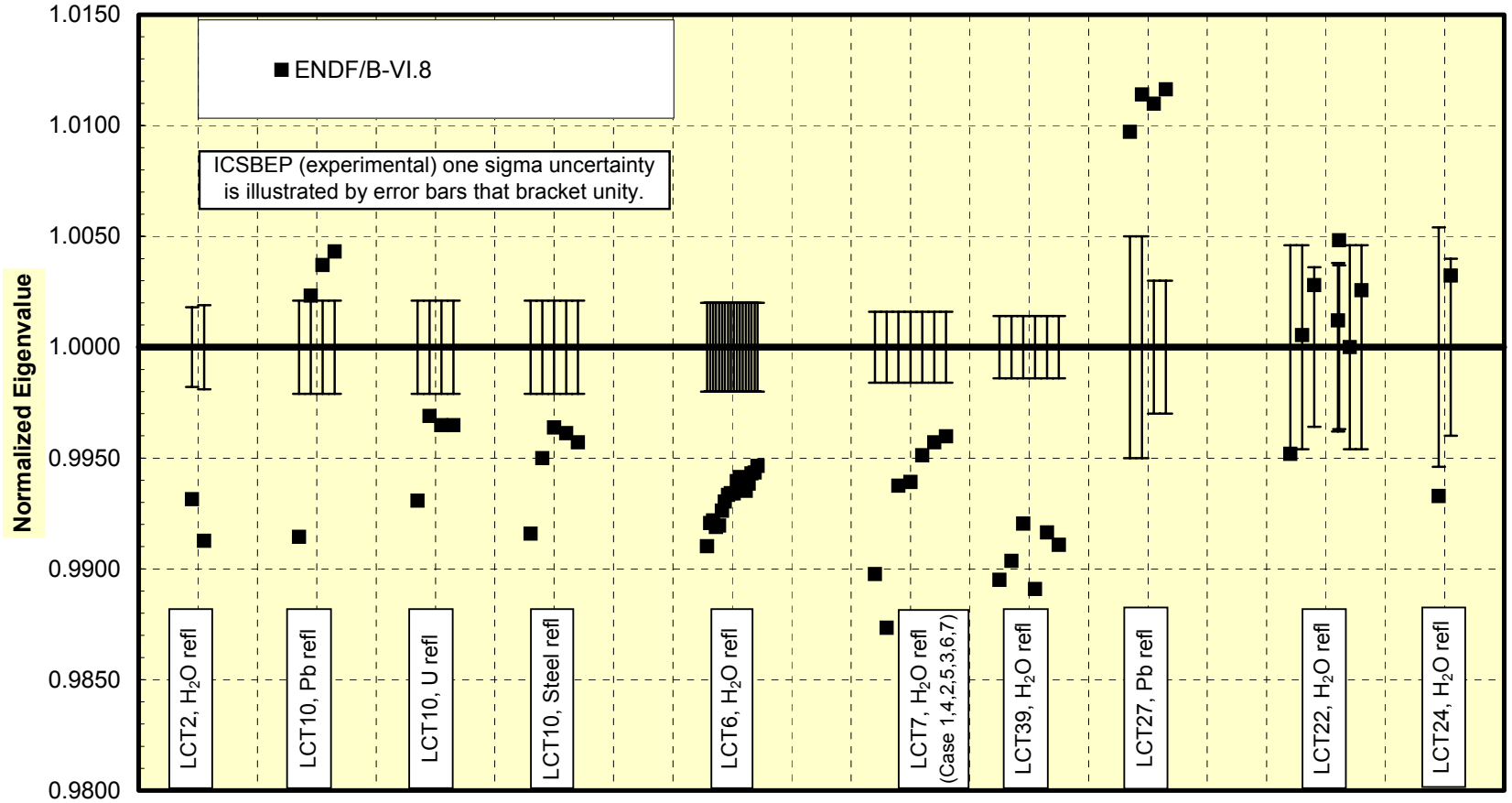
LEU-COMP-THERM Eigenvalues with Selected ^{238}U Data Sets



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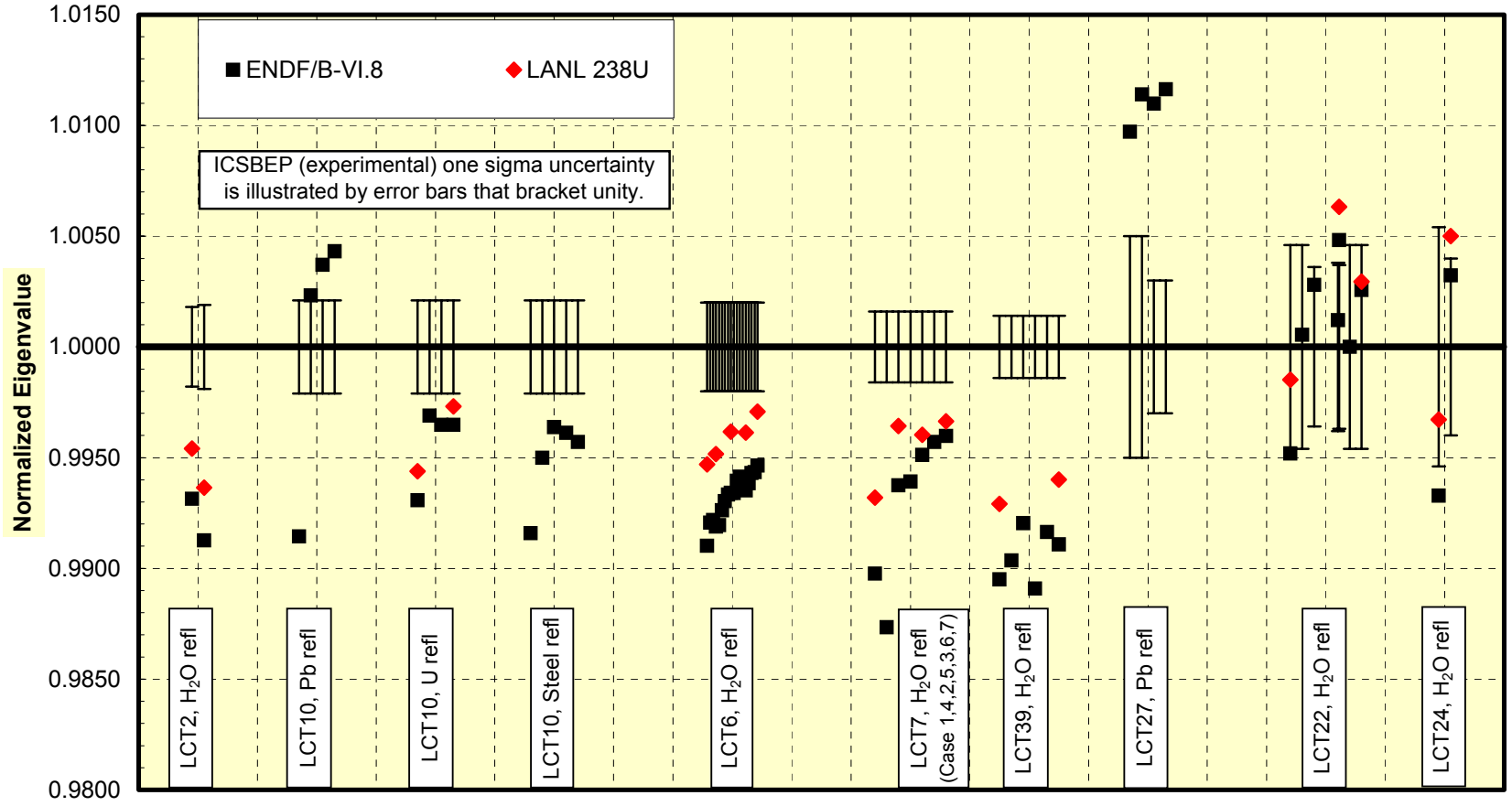
LEU-COMP-THERM Eigenvalues with Selected ^{238}U Data Sets



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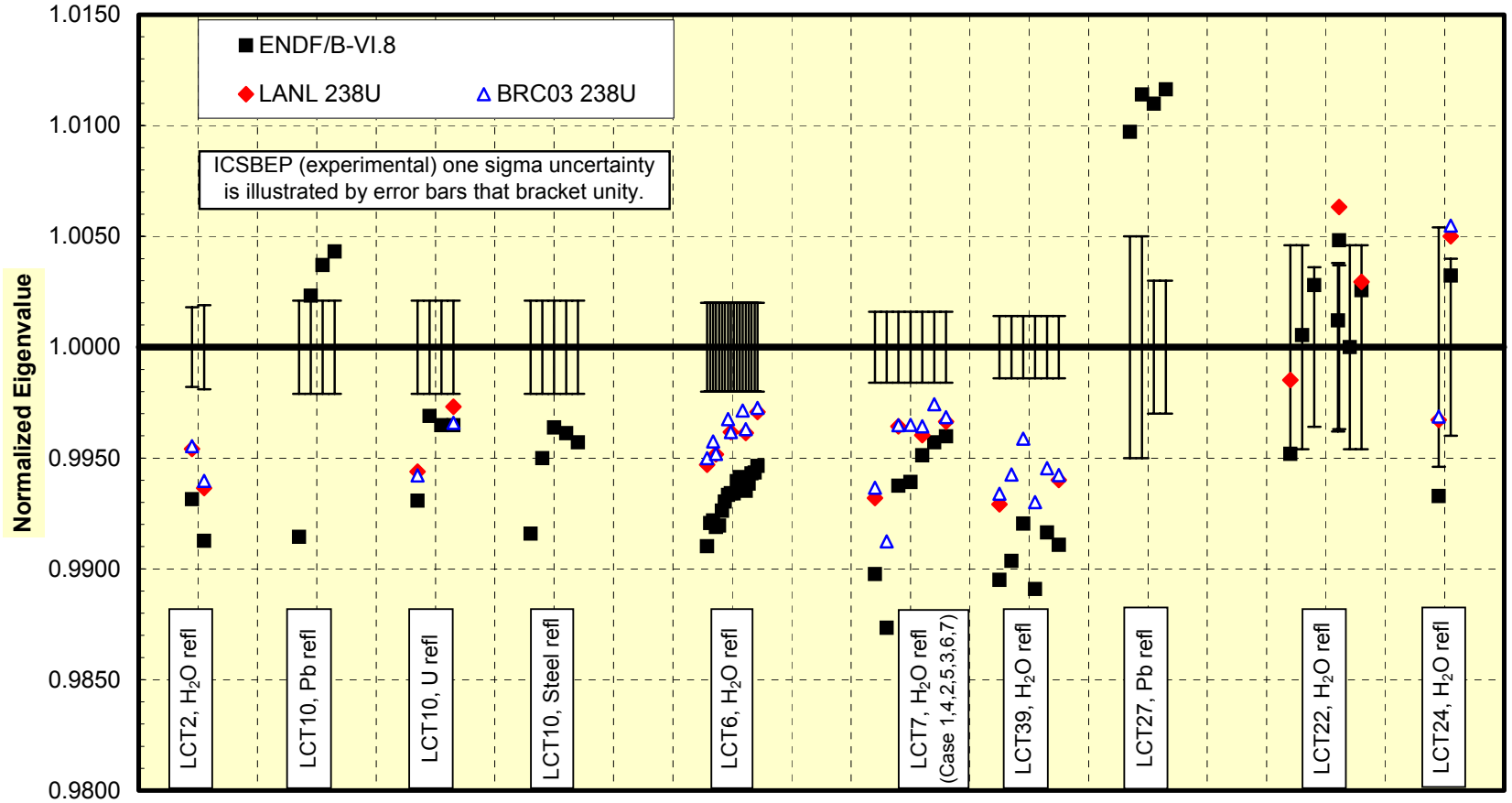
LEU-COMP-THERM Eigenvalues with Selected ^{238}U Data Sets



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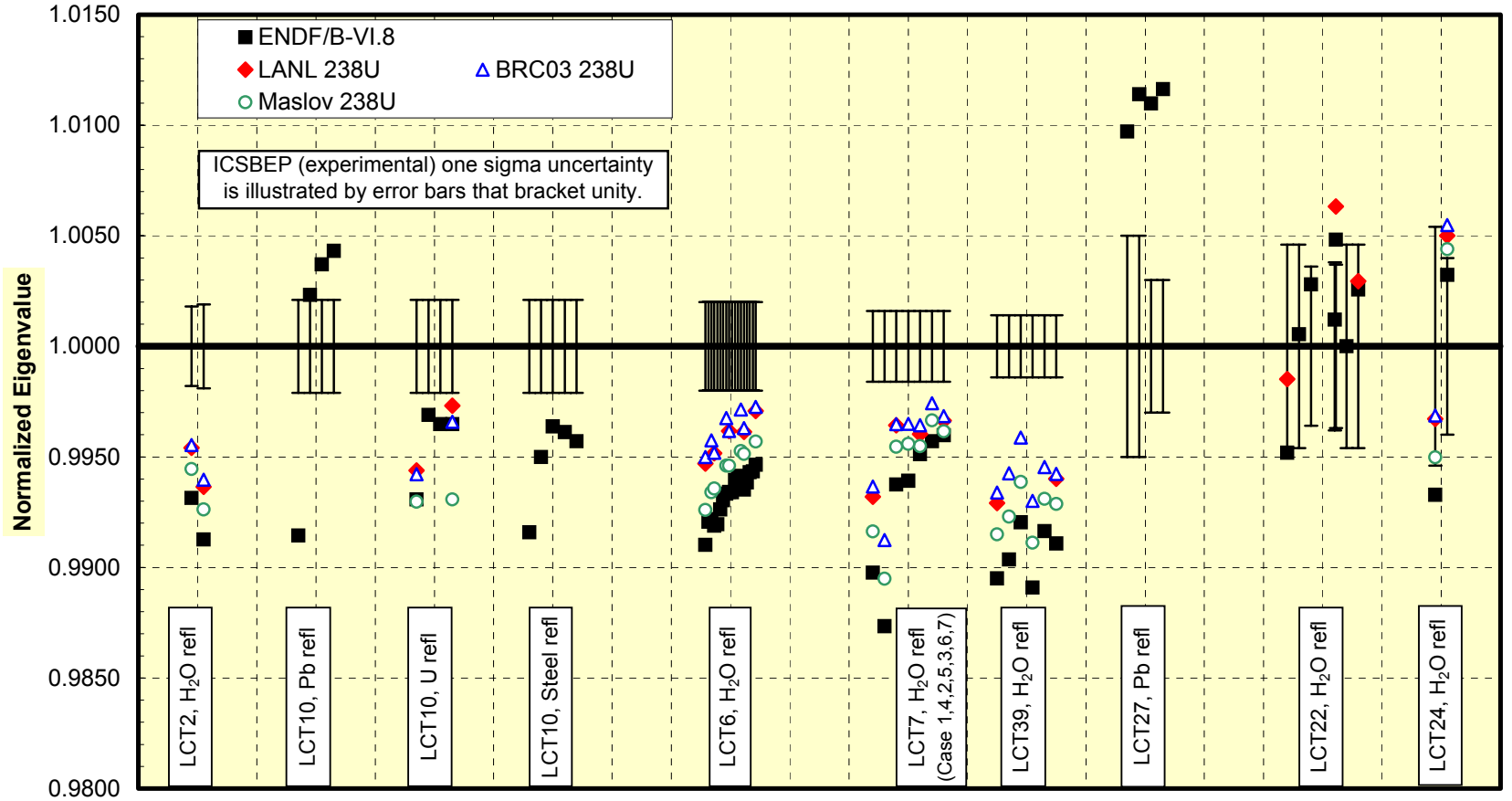
LEU-COMP-THERM Eigenvalues with Selected ^{238}U Data Sets



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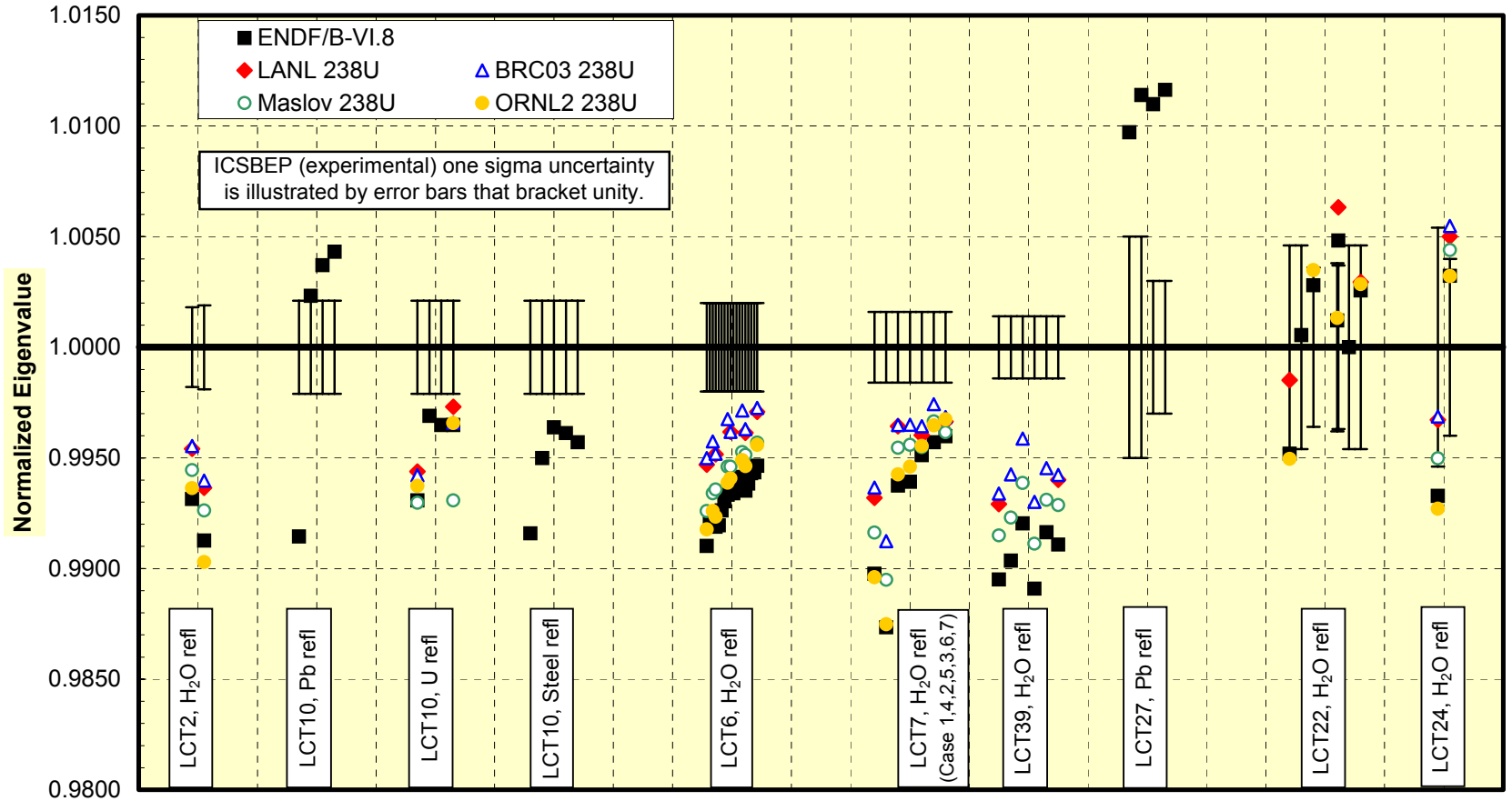
LEU-COMP-THERM Eigenvalues with Selected ^{238}U Data Sets



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LEU-COMP-THERM Observations

- LEU-COMP-THERM eigenvalues are generally low, with a decreasing eigenvalue trend observed as a function of Above Thermal Fission Fraction, ATFF or as a function of ^{238}U absorption fraction.
  - New data sets proposed by BRC03, LANL, Maslov and ORNL all yield larger, but still less than unity, LCT benchmark eigenvalues.
    - The BRC03 and LANL data sets also reduce the magnitude of the eigenvalue trend with ^{238}U capture fraction, but do not eliminate it. Maslov and ORNL changes are too small to draw such a conclusion.
  - Calculations with the KAPL v223 data set, not shown on the previous plots, exhibit an average eigenvalue decrease of ~0.05%, suggesting that the continuum inelastic scattering representation is not a significant contributor to these deficiencies.

Final Observations

- Current (ENDF/B-VI.8, JEFF-3.0, JENDL-3.3) libraries generally perform well for ^{235}U solution systems.
  - HEU-SOL-THERM eigenvalues are virtually unity with no underlying eigenvalue trend as a function of Above-Thermal Leakage Fraction (ATLF).
    - 33 ENDF/B-VI.8 eigenvalues range from 0.9931 to 1.0063, with an average of 0.9994 ± 0.0010 ± 0.0057 (0.0010 and 0.0057 are 95% confidence intervals for the average and for the population, respectively).
  - LEU-SOL-THERM benchmarks, whose ATLF values range from ~0.06 to ~0.24, also exhibit near unity eigenvalues
    - 39 ENDF/B-VI.8 eigenvalues range from 0.9916 to 1.0103, with an average of 0.9994 ± 0.0011 ± 0.0066 (0.0011 and 0.0066 are 95% confidence intervals for the average and for the population, respectively).
- Future data set revisions should strive to maintain these excellent results.

List of Priority Data Needs

K. H. Guber, L. C. Leal

For the Nuclear Data Advisory Group

Al, O, Cl, Si, ^{233}U new evaluations available from ORNL

- O evaluation is ok.
- Al evaluation does not include explicit scattering data, but includes capture.
- Cl evaluation does not include explicit scattering data, but includes capture.
- Si evaluation does not include explicit scattering data, but includes capture.
- ^{233}U evaluation included new fission and total cross section data.

Approach

- Restricted to list of priority data needs from last NDAG meeting.
- Searched CINDA and CSISRS for available data.
- Searched all obtained information for useful experimental data.
- Searched evaluated data files (ENDF and JENDL) for information about the experimental data used.
- Traced the experimental data to the source, i.e. publication or CSISRS in order to evaluated the quality.

^{238}U (n,γ) 0.01keV- 1 MeV

- No suitable high resolution capture data in EXFOR.
- The high resolution Macklin data (150m flight path) is most likely lost.
- Macklin data was taken with scintillation tank, most likely neutron sensitivity problems.
- High resolution transmission data from ORELA (200m) is available and was analyzed by Derrien.

^{155,157}Gd (n,γ) 1ev – 1 keV

- Latest evaluation 1999 does include all available data, resolved (up to ~200eV) and unresolved data (Wisshak et al 1995 from 3keV to 200keV).
- Problems with oxide samples, i.e. probable water content.
- Total CS data is needed for correct evaluation. Correction of the unresolved data? Resolved data is relatively old.
- Resolved (n,γ)-data up to 1 keV is needed.

^{56}Fe (n,tot), (n,γ), scattering 1keV- 1 MeV

- Latest evaluation from 2000. It does include all available high quality data including latest resonance analysis/evaluation by Perey et al. Perey et al. evaluation is based on high quality transmission and scattering data.
- BUT the capture is based on a measurement from GEEL using an incorrect weighting function and possible neutron sensitivity problem with these data. No newer correct capture data was used in the latest evaluation.

Problematic capture data for all others

- Bottom line for most data of the list the used capture data in the evaluated files is insufficient:
  - Measurements performed in many cases with the old capture set up at ORELA, which is known to have had an underestimated neutron sensitivity.
  - Additionally, incorrect weighting function was applied to the data.
  - Most times only capture kernel is given. I.e. the neutron width was guessed, could cause problem with the corrections for experimental effects, like self-shielding and multiple scattering.
- Affected data: ^{63}Cu, ^{65}Cu, ^{90}Zr, ^{91}Zr, ^{92}Zr, ^{94}Zr, ^{40}Ca, ^{44}Ca, ^{50}Cr, ^{52}Cr, ^{53}Cr, ^{58}Ni, ^{60}Ni, ^{140}Ce, ^{142}Ce, ^{54}Fe, ^{57}Fe

Summary

- Many capture data used in the evaluated data files are most likely incorrect. Many of them can be traced down to Macklin et al. obtained with an experimental setup which is known to had problems.
- For a new evaluation high resolution transmission experiments are required.
- Gd data for capture needs to be measured, no resonance data above 200 eV is available.
- Scattering data for all isotopes is very rare.
- No high resolution $^{238}U(n,\gamma)$ data available.
- Many experimental data are not available in CSISRS.

APPENDIX J

National Nuclear Data Center
Progress Report to NCSP Review Meeting

Washington, D.C.
September 25-26, 2003

Nuclear Data Consulting

The NNDC provides advisory services to the NCSP program management on the nuclear data activities funded by the Nuclear Criticality Safety Program. In this capacity, an NNDC representative participates in the semi-annual program review meetings. In the 2003 fiscal, we have participated in the Las Vegas and the present (Washington) meetings. In addition, the NNDC is represented at the semi-annual meetings of the Nuclear Data Coordinating Committee chaired by Richard McKnight that held meetings at BNL and ORNL in the present fiscal year.

ENDF/B-VII

Version VII of the ENDF library is being prepared. So far, evaluations for interaction of neutrons with 23 materials have formally been submitted. In addition, there are 10 evaluations for protons, 3 for tritons, 5 for deuterons, 3 for 3-He and 160 evaluations for photo-nuclear reactions. The latter originate from the IAEA Coordinated Research Project and have been carefully reviewed and corrected at LANL. All charged particle and most of the neutron evaluations were provided by LANL. Other contributors are ORNL and BNL. The evaluations have been collected at BNL NNDC and by the end of September will be posted on the "private" CSEWG web site for review. NNDC is in the process of appointing official reviewers for each evaluation (except of the photo-nuclear evaluations that are considered to be already thoroughly tested).

In order to facilitate review process, the NNDC has prepared a highly automated system aiming to provide reviewers with an easy access to the relevant information. The system is based on the BNL checking codes, preprocessing codes from LLNL, and the ENDF verification package from the IAEA Nuclear Data Section. The numerous codes are linked together through a series of scripts. Plotting capabilities are provided by the powerful ZVView package from the IAEA. User access to the data can be easily obtained via the WWW through a graphical user interface. The ENDF-VII web site containing a list of evaluations and pages for each evaluation is easily prepared with minimal input from the site manager. It is expected that posting this information on the CSEWG-members web site at NNDC will facilitate, standardize and speed up the review process.

The following neutron evaluations of interest to nuclear criticality safety have been processed in FY2003:

^{27}Al, ^{28}Si, ^{99}Tc, ^{153}Eu, ^{157}Gd, ^{208}Pb, ^{232,233,234,235,237,238,239,241}U, ^{237}Np, ^{241}Pu

Evaluation of Fission Product Nuclear Data

The evaluated nuclear data file (ENDF/B-VI), is used by all U.S. nuclear design- and analysis codes. The ENDF/B library has focused on the nuclear performance of nuclear reactors and on national security. Thus nuclides relevant to these applications have received most attention in the past and are in relatively good condition. On the other hand, data for many fission-products and minor actinides that have significant importance for criticality safety, and dose calculations involving long-term discharge fuel, were based on evaluations conducted between 1973 and 1984. To remedy this shortcoming, we are reevaluating the thermal, resonance, and unresolved resonance regions. In this project, emphasis is placed on those isotopes corresponding to elements with Z= 31-68, in FY03 focusing on Z=50-68. All recent measurements in the thermal, resonance and unresolved energy regions are being considered.

The project accomplished the following tasks:

1. Review recent measurements of thermal cross sections and resonance integrals for the purpose of updating previous recommendations, as reported in the BNL-325, Volume 1 compilation (1984).
2. Review recent measurements of neutron resonance parameters, and run physics computer codes that carry out the following functions: calculation of Porter-Thomas distributions for the determination of average neutron spacings, and neutron strength functions; determination of the parity of resonances by the application of Baysian analysis; determination of the resonance contribution to the various thermal capture cross sections and to the resonance integral.
3. Merge the thermal and resonance regions by invoking and determining the parameters of one or two bound levels.
4. Produce resonance parameter files (File 2) in ENDF/B format.
5. Produce a table of average resonance parameters for the description of the unresolved energy region.

The following statistics emerge from this work:

1. 133 new resonance data sets, as well as 150 new thermal data sets, were considered.
2. A total of 100 evaluated files for resonance parameters (File 2) were produced in ENDF/B format.
3. Out of these files, in 38 cases the resonance region was extended to a higher energy than that reported in BNL-325.

4. For the first time, resonance-parameter files are produced for the following nuclides: ^{136}Xe, ^{133g}Ba, ^{153g}Pr, ^{144}Sm, ^{153g}Gd, ^{160g}Tb, ^{166m}Ho.

The high priority fission product nuclides for which resonance region evaluations for the have been completed this year are:

$$^{134,135}\text{Cs}, ^{134}\text{Ba}, ^{149}\text{Sm}, ^{154,155}\text{Eu}, ^{160\text{-}164}\text{Dy}, ^{175,176}\text{Lu}$$

Nuclear Criticality Safety Program Nuclear Data Advisory Group Meeting

Status of Integral Experiments

David K. Hayes

Brookhaven National Laboratory

November 3, 2003

National Nuclear Security Administration

Los Alamos NATIONAL LABORATORY

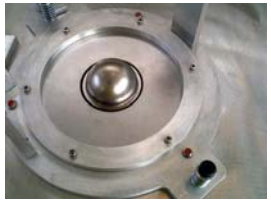
FY03 Accomplishments

- **Experiments proposed for this FY in the 5-Year Plan**

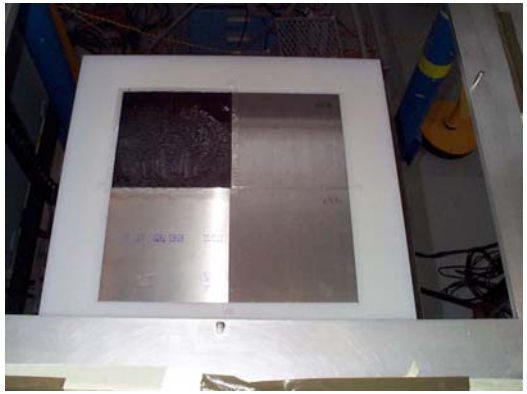
		Status
Z005	Comet/Zeus, Fe^2/HEU/Fe^2	Completed
Z006	Comet/Zeus, Fe^3/HEU/Fe^3	Terminated due to funding shortfall
Z007	Comet/Zeus, Al^1/HEU/Al^1	Moved to FY04
Z008	Comet/Zeus, Al^2/HEU/AL^2	Moved to FY04
NP001	^{237}Np/HEU bare	Completed
NP002	^{237}Np/HEU/NU reflected	Terminated due to funding shortfall
P007	1x1 HEU/Fe/Poly	Completed
P008	2x2 HEU/Fe/Poly	Completed
P009	2x2 HEU/MgO/Poly	Not performed – not enough foils to go critical
SM1	HEU/Graphite	Terminated due to funding shortfall
SM2	HEU/D_2O	Moved to FY04
SUB1	Poly-Reflected Alpha-Phase Pu	Completed

Additional Experiments Added to FY03

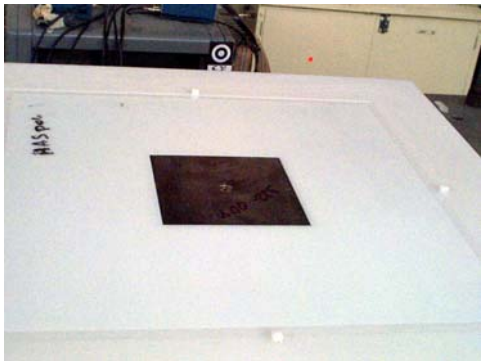
- **2x2 HEU/Al/Poly (now designated as P009)**
- **2x2 HEU/Gd Alloy/Poly (now designated as P010)**
- **Pu (α) BRP Ball/HEU (now designated as P011)**



BRP Ball – in holding fixture and on Planet



2x2 Aluminum in Poly



Gd Alloy sitting in Poly insert. Four HEU foils sit on top of the Poly insert.

Impacts from FY03 Funding Shortfall

- **Some experiments not completed**
  - Three FY03 deliverables not completed
  - Other experiments will have to be pushed back on the schedule

- **Personnel who used to be 100% LACEF**
  - Steven Clement (now supporting TA-18 relocation, AB issues)
  - Derek Dinwiddie (20% LACEF, 80% other programs)
  - Joetta Goda (60% LACEF, 40% other programs)
  - Dave Hayes (now supporting NAP and TA-18 relocation)
  - Peter Jaegers (50% LACEF, 50% other programs)
  - David Loaiza (60% LACEF, 40% other programs)
  - Rene Sanchez (60% LACEF, 40% other programs)
  - Minimal impacts on tech and engineering support

LACEF Input to the NCSP 5-Year Plan

FY 2002 ($k) 1250	FY 2003 ($k) 1300 (actual was $700k)	FY 2004 ($k) 1350	FY 2005 ($k) 1390	FY 2006 ($k) 1430	FY 2007 ($k) 1500
NP001 ^{237}Np critical mass experiment Np/HEU (bare)	NP001 Continue ^{237}Np critical mass experiment Np/HEU bare	NP002 Continue ^{237}Np critical mass experiment Np/HEU/NU reflected	NP004 Np/HEU/Poly	P019 Pu(δ)/HEU	NP005 Np/HEU/Be
P001 1x1 HEU/MgO/Poly	NP002 moved to FY04	NP003 Np/HEU/Be	P016 2x2 Concrete/HEU/Poly	P020 1x1 Pu/SiO_2/Poly	NP006 Np/HEU/W
P002 1x1 HEU/Poly	P007 1x1 HEU/Fe/Poly	NP007 Np/HEU/Steel	P017 1x1 HEU/Al_2O_3/Poly	P021 1x1 Pu/Al/Poly	P022 2x2 Pu/SiO_2/Poly
P003 1x1 HEU/Gd/Poly	P008 2x2 HEU/Fe/Poly	P012 1x1 HEU/CaO/Poly	P018 2x2 HEU/Al_2O_3/Poly	SM5 Pu Reflected D_2O	P023 2x2 Pu/Al/Poly
P004 2x2 HEU/Poly	P009 2x2 HEU/Al/Poly	P013 1x1 HEU/Zr/Poly	SM2 HEU/D_2O	Z010 Initiate ^{239}Pu intermediate energy experiment (if ^{239}Pu available) Graphite^1/Pu/Graphite^1	Z016 SiO_2^2/Pu/SiO_2^2
P005 2x2 HEU/Gd/Poly	P010 2x2 HEU/Gd Alloy/Poly	P014 Component Benchmark	SM3 HEU/Be	Z011 Graphite^2/Pu/Graphite^2	Z017 SiO_2^3/Pu/SiO_2^3
P006 2x2 HEU/Si/Poly	P011 Pu (α) BRP Ball/HEU	P015 1x1 Concrete/HEU/Poly	SM4 Pu Reflected Graphite	Z012 Initiate ^{233}U intermediate energy experiment (if ^{233}U available)	Z018 Fe^1/Pu/Fe^1
Z001 Comet/Zeus, 2 cm Graphite/HEU/2 cm Graphite	SM1 moved to FY04	SM1 HEU/Graphite	SM6 Pu Reflected Be	Z014 SiO_2^2/HEU/SiO_2^2	
Z002 Comet/Zeus, 1 cm Graphite/HEU/1 cm Graphite	SUB1 Pu(α)/Poly Reflected	SUB2 ^{237}Np Bare and Reflected by Cu and HEU	Z008 Comet/Zeus, Al^2/HEU/Al^2	Z015 SiO_2^1/Pu/SiO_2^1	
Z003 Comet/Zeus, all HEU	Z005 Comet/Zeus, Fe^2/HEU/Fe^2	Z006 Comet/Zeus, Fe^3/HEU/Fe^3	Z009 HEU/Gd Alloy (if Gd Alloy becomes available)		
Z004 Fe^1/HEU/Fe^1 (initial configuration)	Z006 moved to FY04	Z007 Comet/Zeus, Al^1/HEU/Al^1	Z013 SiO_2^1/HEU/SiO_2^1		

Completed
Initiated/ongoing
Experiments that will require a change to the AB and nuclear materials not currently available at Los Alamos.
Additional capital funding will be required.

National Nuclear Security Administration

Los Alamos NATIONAL LABORATORY

LACEF Updated Input to the NCSP 5-Year Plan

LACEF Critical Experiments Section of the NCSP 5-Year Plan

FY 2008 ($k) 1550	FY2009 ($k) 1600	FY2010 ($k) 1700
P024 1x1 Pu/MgO/Poly	P029 HEU Reflected Poly	
P025 2x2 Pu/MgO/Poly	P030 HEU Reflected Steel	
P026 HEU bare	P031 HEU Reflected Be	
P027 HEU Reflected NU		
P028 HEU Reflected W		
Z019 $Fe^2/Pu/Fe^2$		
Z020 $Fe^3/Pu/Fe^3$		

Completed
Initiated/ongoing
Experiments that will require approval to a change in the Authorization Basis.
Additional capital funding will be required.

National Nuclear Security Administration

Los Alamos NATIONAL LABORATORY

ICSBEP Benchmark Evaluations Committed to in FY03

- **Committed to in FY03**
  - SUB-PU-MET-FAST-001 — Polyethylene-Reflected Alpha Phase Plutonium Ball
  - HEU-MET-INTER-009 — Zeus Iron (Fe) Core
  - HEU-MET-THERM-012 — HEU/Al/Poly (2x2)
  - HEU-MET-THERM-013 — HEU/Fe/Poly (1x1, 0.015-inch thick and 1/16th-inch thick)
  - HEU-MET-THERM-014 — HEU/SiO_2/Poly (2x2)
  - SPEC-MET-FAST-008 — ^{237}Np/HEU bare

- **As Renegotiated on 1/29/03**
  - SUB-PU-MET-FAST-001 — SUB1, Poly-Reflected Alpha Phase Plutonium Ball
  - HEU-MET-INTER-009 — Z004, Zeus Iron (Fe) Core
  - HEU-MET-THERM-013 — P007, HEU/Fe/Poly (1x1, 0.015-inch thick and 1/16th-inch thick)
  - HEU-MET-THERM-014 — P006, HEU/SiO_2/Poly (2x2)
  - HEU-MET-THERM-016 — P010, HEU/Gd Alloy/Poly
  - SPEC-MET-FAST-008 — NP001, ^{237}Np/HEU bare

    - **HEU-MET-THERM-012 moved to FY04**
    - **HEU-MET-THERM-016 added to FY03**

- **As of the end of September 2003, all ICSBEP evaluation commitments have been submitted for external review**

Proposed ICSBEP Evaluations for Out Years

- **FY04**
  - HEU-MET-INTER-011 — SM1, Special Moderator HEU/Graphite
  - HEU-MET-FAST-072 — Z005/Z006 ZEUS (HEU) Intermediate Energy Spectrum with Iron (Fe)
  - HEU-MET-THERM-012 — P009, Planet Waste Matrix HEU/Al/Poly (2x2 array)
  - HEU-MET-THERM-015 — P007/P008, Planet Waste Matrix HEU-Fe (2x2 array) 15-mil thick iron plates
  - SUB-SPEC-MET-FAST-001 — SUB2, Bare and Cu-reflected Np-237 Spheres
- **FY05**
  - SPEC-MET-FAST-009 — NP001/NP002 Neptunium/HEU Critical (natural uranium reflected)
  - HEU-MET-INTER-010 — Z007/Z008 ZEUS (HEU) Intermediate Energy Spectrum with Aluminum (Al)
  - SPEC-MET-FAST-014 — NP007, Neptunium/HEU Reflected with Steel
  - HEU-MET-THERM-017 — P012, Waste Matrices HEU / Ca / Poly
  - HEU-MET-THERM-018 — P015, Waste Matrices HEU / Concrete / Poly
  - MIX-MET-FAST-013 — P011, Bare Pu(α) / HEU
- **FY06**
  - HEU-MET-INTER-012 — SM2 Special Moderator HEU/D_2O
  - HEU-MET-INTER-009 — ZEUS (HEU) Intermediate Energy Spectrum with Ni-Cr-Mo-Gd Alloy
  - PU-MET-FAST-038 — BRP Ball Experiments Pu/Be
  - HEU-MET-THERM-019 — PO13, Waste Matrices HEU / Zr / Poly (1x1)
  - SPEC-MET-FAST-010 — NP003, Neptunium/HEU/Be Reflected
- **FY07**
  - PU-MET-INTER-003 — SM4/SM6, Pu Reflected with Graphite and Beryllium
  - HEU-MET-INTER-013 — Z013/Z014, ZEUS (HEU) Intermediate Energy Spectrum with SiO_2
  - HEU-MET-INTER-014 — SM3, HEU Reflected by Beryllium
  - HEU-MET-THERM-020 — P016, HEU / Concrete / Poly (2x2)
  - HEU-MET-THERM-021 — P017/P018, HEU / Al_2O_3 / Poly (1x1 and 2x2)
  - SPEC-MET-FAST-011 — NP004, Neptunium/HEU Reflected with Poly
- **FY08**
  - PU-MET-INTER-004 — SM5, Pu Reflected with D_2O
  - MIX-MET-FAST-014 — P019, Pu(δ) /HEU
  - SPEC-MET-FAST-012 — NP006, Neptunium Reflected with Tungsten
  - PU-MET-THERM-002 — P022, Pu / Si / Poly (2x2)
  - PU-MET-THERM-003 — P023, Pu / Al / Poly
  - SPEC-MET-FAST-013 — NP005, Neptunium/HEU Reflected with Beryllium

Current List and Status of Priority Experiments

JUNE 1998 RECOMMENDATIONS FOR PRIORITY OF CRITICAL EXPERIMENTS

PRIORITY	1998 IDENTIFIER	1994 IDENTIFIER	EXPERIMENT DESCRIPTION	RELATIVE PRIORITY	RELATIVE COST	BENEFITS ACRUE TO	STATUS	
1	98-2, 98-4, 98-6, 98-14, 98-28	107, 502i, 603, 609	Intermediate energy spectrum (ZEUS)	HIGH	LOW	DP, EM, MD, RW	Active	
2	98-6, 98-14, 98-2, 98-4	102, 502a, 702, 502g, 303	Fast, intermediate, and thermal energy spectrum with fissile / fissionable material in waste matrices	HIGH	MEDIUM	EM, MD, RW	Active	
3	98-7	206, 207, 102 502a, 702	Reactivity and replacement measurements with SHEBA (CERES, ^{233}U, MOX, etc)	HIGH	LOW	RW, EM, NRC	On hold	
4	98-1	None	Component safety benchmark experiments	HIGH	MEDIUM	DP, DoD	Active	**New for FY03**
5	98-22, 98-3, 98-16, 98-21, 98-23	301, 503, 504	Criticality accident simulation/equipment and methodology qualification	MEDIUM	LOW	DP, EM, MD, RW	Active	**New for FY03**
6	98-8, 98-9, 98-10, 98-13, 98-18, 98-19	601, 605, 605a, 605b, 401	Critical mass measurements and neutron parameters for actinide isotopes	MEDIUM	MEDIUM	DP, EM, MD, RW	Active	
6a			Thermal and intermediate energy experiments with gadoliniated alloy and HEU				Active	**New for FY03**
7	98-8	None	Lattice experiments with MOX fuel pins	MEDIUM	MEDIUM	RW, MD, DP NRC		
8	98-11	707, 304	Special moderators, situations, & anomalies (Be, BeO, D_2O, etc)	MEDIUM	MEDIUM	EM, MD, RW, DP	Active	**New for FY03**
9	98-5, 98-20, 98-21	601, 301, 303	Static benchmark experiments in fissile solutions	HIGH	VERY HIGH	DP, EM		
10	98-27	505, 701	Source jerk, pulsed neutron measurements for subcritical systems	HIGH	MEDIUM	DP, EM, EH, RW, NRC	Active	

Concerns and Emerging Issues

- **Concerns**
  - Limited feedback from AROBCAD and NDAG has been received
    - Two preliminary design packages (PDPs) originally submitted to both groups in late 2002
    - Revised PDPs sent on 3/18/2003
    - Verbal comments on PDPs receive during NCSP technical review in early April 2003
    - Second revision of PDPs (with comments incorporated from NCSP technical review) on 4/11/03.
    - No feedback AROBCAD or NDAG indicating whether or not the experiments are good or bad
- **Emerging Issues**
  - Longer term (FY05 or FY06)
    - LANL RTBF re-organization
      - TA-18 could loose all infrastructure support (currently 4-5 M$ per year) from RTBF
        - » RCTs, industrial hygiene, system engineers, training and QA specialists, security specialists, elec techs, mech techs
        - » Funds would have to be “made up” via programmatic taxes

Changes

- Steven Clement is the Acting N-2 Deputy Group Leader for Operations
- Charlene Cappiello is the Acting LACEF Team Leader
- Job adds for Deputy Group Leader and new LACEF Team Leader will be posted within the month

APPENDIX L

DEPARTMENT OF ENERGY (DOE)
NUCLEAR CRITICALITY SAFETY PROGRAM (NCSP)

Nuclear Data Advisory Group (NDAG)

EXISTING BENCHMARK DATA THAT SUPPORT PRIORITY DATA NEEDS

Prepared By
J. Blair Briggs (INEEL)
Chair, NDAG Working Group on Benchmarking

Revision 1
September 2003

The list of Criticality Safety Priority Needs provided by the Working Group on Data Needs is provided below.

LIST OF CRITICALITY SAFETY PRIORITY DATA NEEDS [a]

Element	Cross Section	Energy Range
Gadolinium (Gd)	(n,γ)	1 ev – 1 keV
Beryllium (Be)	(n,total), (n,n)	1 keV – 1 MeV
Fe, Ni, Cr, Mn, Cl, Cu, Si, Ce, Ca	(n,total), (n,n), (n,γ)	1 keV – 1 MeV
Fe, Ni, Cr, Mn	(n,total), (n,n), (n,γ)	1 keV – 1 MeV
Al, Graphite, O, ZrH_x	(n,total), (n,n)	1 keV – 1 MeV
^{238}U	(n,γ)	TBD
^{233}U	TBD	TBD

a. Taken From Jerry N. McKamy, "List of Priority Data Needs"

In addition to the isotopes and elements listed above, ^{232}Th is also considered in this document.

Lists of existing benchmarks from the "International Handbook of Evaluated Criticality Safety Benchmark Experiments" that support these data needs are provided below. Cerium appears only as an impurity so there are essentially no benchmark data for cerium. The only benchmarks that include significant quantities of calcium are those that are concrete reflected. While the Handbook contains several benchmarks that have concrete reflectors, they could only be considered as secondary benchmarks for calcium. Therefore, the concrete reflected benchmarks are not included in the lists that follow.

Where sufficient data are available in DICE, plots of the fission fraction, capture fraction, and flux versus energy are provided for selected benchmarks that bound the other benchmarks in the same class. Plots for soluble and fixed systems containing gadolinium are provided in Appendix A. Plots for beryllium (do not include data approved in 2003), structural materials (data for Mn are not yet available), graphite, ZrH_x (not yet available), ^{238}U, and ^{233}U are provided in Appendices B through G, respectively.

Table E.2. List of Critical Experiment Needs

1. Data Need #1: Gadolinium

1.1. Existing Critical Experiments with Gadolinium

1.1.1. Critical Experiment: ***HEU-SOL-THERM-014, 015, 016, 017, 018, 019, 025***

1.1.1.1. Description: Tanks filled with uranyl nitrate solution and soluble Gd

Identification	No. of Cases	Fuel Concentration (g/l)	Gadolinium Concentration (g/l)	Percent Absorption In Gadolinium
HEU-SOL-THERM-014	2	67.7 – 68.1	0.1 – 0.193	9.7 – 17.1
HEU-SOL-THERM-015	3	95.2 – 98.8	0.4 – 0.197	12.6 – 23.0
HEU-SOL-THERM-016	2	143.6 – 144.2	0.3 – 0.525	12.8 – 19.9
HEU-SOL-THERM-017	5	186.2 – 196.2	0.298 – 0.79	9.4 – 21.0
HEU-SOL-THERM-018	9	279.6 – 291.3	0.497 – 1.943	8.8 – 26.3
HEU-SOL-THERM-019	2	393.6 – 400	0.647 – 1.16	8.1 – 12.7
HEU-SOL-THERM-025	14	48.7 – 400	0.107 – 10.37	12.2 – 21.2

1.1.1.2. Facility and Date: **IPPE, 1987**

1.1.1.3. Experimental k_{eff} ± Uncertainty

Identification	Benchmark K_{eff}	Benchmark Uncertainty
HEU-SOL-THERM-014	1.0000	0.0052 – 0.0087
HEU-SOL-THERM-015	1.0000	0.0068 – 0.0089
HEU-SOL-THERM-016	1.0000	0.0069 – 0.0079
HEU-SOL-THERM-017	1.0000	0.0047 – 0.0067
HEU-SOL-THERM-018	1.0000	0.0044 – 0.0065
HEU-SOL-THERM-019	1.0000	0.0041 – 0.0067
HEU-SOL-THERM-025	1.000 – 1.0009	0.0043 – 0.0073

1.1.1.4. Calculated k_{eff}

Identification	Code	Library	C/E
HEU-SOL-THERM-014	MCNP	ENDF/B-V	1.0134 – 1.0210
HEU-SOL-THERM-015	MCNP	ENDF/B-V	1.0131 – 1.0154
HEU-SOL-THERM-016	MCNP	ENDF/B-V	1.0100 – 1.0280
HEU-SOL-THERM-017	MCNP	ENDF/B-V	1.0005 – 1.0090
HEU-SOL-THERM-018	MCNP	ENDF/B-V	0.9939 – 1.0272
HEU-SOL-THERM-019	MCNP	ENDF/B-V	0.9999 – 1.0041
HEU-SOL-THERM-025	MCNP	ENDF/B-V	0.9979 – 1.0166

1.1.1.5. Sensitivity of k_{eff} to Gadolinium: Not Available

1.1.1.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
HEU-SOL-THERM-014	94.4 – 94.8	4.8 – 5.2	0.4
HEU-SOL-THERM-015	92.1 – 92.6	6.8 – 7.3	0.6
HEU-SOL-THERM-016	89.0 – 89.9	9.3 – 10.1	0.8 – 0.9
HEU-SOL-THERM-017	85.3 – 87.7	11.2 – 13.5	1.1 – 1.2
HEU-SOL-THERM-018	78.3 – 82.6	15.8 – 19.8	1.6 – 1.8
HEU-SOL-THERM-019	75.9 – 77.6	20.2 – 21.8	2.2 – 2.3
HEU-SOL-THERM-025	95.7 – 81.5	4.0 – 16.9	0.3 – 1.6

1.1.2. Critical Experiment: ***MIX-SOL-THERM-006***

1.1.2.1. Description: Mixed Plutonium - Uranium Nitrate Solution with Dissolved Gd

Identification	No. of Cases	Fuel Concentration (g/l)	Gadolinium Concentration (g/l)	Percent Absorption In Gadolinium
MIX-SOL-THERM-006	6	91	0.03 – 0.712	Not Available

1.1.2.2. Facility and Date: VALDUC, 1986

1.1.2.3. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
MIX-SOL-THERM-006	1.0000	0.0010 – 0.0016

1.1.2.4. Calculated k_{eff}

Identification	Code	Library	C/E
MIX-SOL-THERM-006	KENO	ENDF/B-V 238 Group	0.9991 – 1.0072
	MCNP	ENDF/B-V	0.9986 – 1.0051

1.1.2.5. Sensitivity of k_{eff} to Gadolinium: Not Available

1.1.2.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
MIX-SOL-THERM-006	Not Available	Not Available	Not Available

1.1.3. Critical Experiment: ***HEU-MET-THERM-016***

1.1.3.1. Description: Highly Enriched Uranium Foils with Polyethylene and Ni-Cr-Mo-Gd Alloy Plates (PLANET)

Identification	No. of Cases	Percent Absorption In Gadolinium
HEU-MET-THERM-016	1	Not Available

1.1.3.2. Facility and Date: LACEF, 2002

1.1.3.3. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
HEU-MET-THERM-016	1.0017	0.0017

1.1.3.4. Calculated k_{eff}

Identification	Code	Library	C/E
HEU-MET-THERM-016	MCNP	ENDF/B-VI Rel. 4	1.0019
	MCNP	ENDF/B-V	1.0002

1.1.3.5. Sensitivity of k_{eff} to Gadolinium: Not Available

1.1.3.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
HEU-MET-THERM-016	Not Available	Not Available	Not Available

Critical Experiment: ***LEU-COMP-THERM-005***

1.1.3.7. Description: Low Enriched UO_2 Fuel Rods in H_2O with Dissolved Gd

Identification	No. of Cases	Fuel Rod Enrichment (%)	Gadolinium Concentration (g/l)	Percent Absorption In Gadolinium
LEU-COMP-THERM-005	11	2.35 & 4.31	0.068 – 1.481	Not Available

1.1.3.8. Facility and Date: PNL, 1981

1.1.3.9. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
LEU-COMP-THERM-005	1.0000	0.0020 – 0.0066

1.1.3.10. Calculated k_{eff}

Identification	Code	Library	C/E
LEU-COMP-THERM-005	MONK	ENDF/B-VI Rel. 3	0.9889 – 1.0091

1.1.3.11. Sensitivity of k_{eff} to Gadolinium: Not Available

1.1.3.12. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
LEU-COMP-THERM-005	Not Available	Not Available	Not Available

1.1.4. Critical Experiment: ***LEU-COMP-THERM-035***

1.1.4.1. Description: Low Enriched UO_2 Fuel Rods in H_2O with Dissolved Gd

Identification	No. of Cases	Fuel Rod Enrichment (%)	Gadolinium Concentration (g/l)	Percent Absorption In Gadolinium
LEU-COMP-THERM-035	1	2.6	0.0642	3.1

1.1.4.2. Facility and Date: JAERI, 1988

1.1.4.3. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
LEU-COMP-THERM-035	1.0000	0.0022

1.1.4.4. Calculated k_{eff}

Identification	Code	Library	C/E
LEU-COMP-THERM-035	MCNP	ENDF	0.9982 – 1.0012

1.1.4.5. Sensitivity of k_{eff} to Gadolinium: Not Available

1.1.4.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
LEU-COMP-THERM-035	85.5	8.1	6.3

1.1.5. Critical Experiment: ***LEU-COMP-THERM-052***

1.1.5.1. Description: Low Enriched UO_2 Fuel Rods in H_2O with Dissolved Gd

Identification	No. of Cases	Fuel Rod Enrichment (%)	Gadolinium Concentration (g/l)	Percent Absorption In Gadolinium
LEU-COMP-THERM-052	6	4.738	0.171 – 0.6	11.6 – 20.8

1.1.5.2. Facility and Date: Valduc, 1978

1.1.5.3. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
LEU-COMP-THERM-052	1.0003	0.0023 – 0.0036

1.1.5.4. Calculated k_{eff}

Identification	Code	Library	C/E
LEU-COMP-THERM-052	KENO	ENDF/B-V 238 Group	0.9881 – 1.0005
	MCNP	ENDF/B-V	0.9902 – 1.0006

1.1.5.5. Sensitivity of k_{eff} to Gadolinium: Not Available

1.1.5.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
LEU-COMP-THERM-052	91.9	5.2	2.8

1.1.6. Critical Experiment: ***MIX-MISC-THERM-001 and 004***

1.1.6.1. Description: Mixed Oxide Fuel Pin Lattices in Mixed Plutonium - Uranium Nitrate Solution with Dissolved Gd

Identification	No. of Cases	Fuel Concentration (g/l)	Gadolinium Concentration (g/l)	Percent Absorption In Gadolinium
MIX-MISC-THERM-001	5	461.92 – 466.06	0.49 – 2.16	4.6 – 13.2
MIX-MISC-THERM-004	6	77.6	0.02 – 1.338	Not Available

1.1.6.2. Facility and Date: PNL, 1987 and 1974

1.1.6.3. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
MIX-MISC-THERM-001	1.0000	0.0032 – 0.0060
MIX-MISC-THERM-004	1.0000	0.0027 – 0.0036

1.1.6.4. Calculated k_{eff}

Identification	Code	Library	C/E
MIX-MISC-THERM-001	KENO	ENDF/B-IV 27 Group	0.9983 – 1.0034
	KENO	ENDF/B-V 238 Group	1.0031 – 1.0050
	MCNP	ENDF/B-VI	0.9905 – 1.0071
MIX-MISC-THERM-004	KENO	ENDF/B-V 238 Group	0.9924 – 1.0004
	MCNP	ENDF/B-V	0.9921 – 0.9991
	MCNP	ENDF/B-VI Rel. 6	0.9864 – 0.9907

1.1.6.5. Sensitivity of k_{eff} to Gadolinium: Not Available

1.1.6.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
MIX-MISC-THERM-001	70.6 – 73.9	19.8 – 23.1	6.3 – 6.4
MIX-MISC-THERM-004	Not Available	Not Available	Not Available

1.1.7. Critical Experiment: ***HEU-MET-THERM-010***

1.1.7.1. Description: Polyethylene Reflected And Moderated Highly Enriched Uranium Systems With Gadolinium Foils

Identification	No. of Cases	Gadolinium Foil Thickness (mil)	Percent Absorption In Gadolinium
HEU-MET-THERM-010	2	7.5 and 15.0	3.5 – 3.8

1.1.7.2. Facility and Date: LACEF, 2001

1.1.7.3. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
HEU-MET-THERM-010	1.0030 – 1.0065	0.0070 – 0.0072

1.1.7.4. Calculated k_{eff}

Identification	Code	Library	C/E
HEU-MET-THERM-010	KENO	ENDF/B-V 238 Group	0.9920 – 0.9980
	KENO	ENDF/B-V 44 Group	0.9918 – 0.9971
	MCNP	ENDF/B-V	1.0007 – 1.0053
	MCNP	ENDF/B-VI Rel. 4	1.0031 – 1.0062

1.1.7.5. Sensitivity of k_{eff} to Gadolinium: Not Available

1.1.7.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
HEU-MET-THERM-010	92.6	6.4	1.0

1.1.8. Critical Experiment: ***HEU-COMP-THERM-004***

1.1.8.1. Description: Highly Enriched UO_2 Fuel Rod Lattices in H_2O with Fixed Gadolinium Absorber Rods

Identification	No. of Cases	Percent Absorption In Gadolinium
HEU-COMP-THERM-004	2	2.1 – 3.7

1.1.8.2. Facility and Date: Kurchatov Institute, 1969

1.1.8.3. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
HEU-COMP-THERM-004	1.0000	0.0038 – 0.0039

1.1.8.4. Calculated k_{eff}

Identification	Code	Library	C/E
HEU-COMP-THERM-004	MCNP	ENDF/B-V	0.9904 – 0.9926

1.1.8.5. Sensitivity of k_{eff} to Gadolinium: Not Available

1.1.8.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
HEU-COMP-THERM-004	61.9 – 62.6	32.9 – 33.5	4.5 – 4.6

1.1.9. Critical Experiment: ***HEU-COMP-THERM-016***

1.1.9.1. Description: Impulse Graphite Reactor (IGR) with Gadolinium Absorber Rods

Identification	No. of Cases	Percent Absorption In Gadolinium
HEU-COMP-THERM-016	6	Not Available

1.1.9.2. Facility and Date: Semipalatinsk Nuclear Test Site, 1970 - 1992

1.1.9.3. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
HEU-COMP-THERM-016	1.0000	0.0011

1.1.9.4. Calculated k_{eff}

Identification	Code	Library	C/E
HEU-COMP-THERM-016	MCNP	ENDF/B-VI	1.0033 – 1.0100

1.1.9.5. Sensitivity of k_{eff} to Gadolinium: Not Available

1.1.9.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
HEU-COMP-THERM-016	Not Available	Not Available	Not Available

1.1.10. Critical Experiment: ***HEU-SOL-THERM-044***

1.1.10.1. Description: Highly Enriched Uranyl Nitrate Solution Cylinder Containing Fixed Boraflex + Gadolinium Absorber Rods

Identification	No. of Cases	Percent Absorption In Gadolinium
HEU-SOL-THERM-044	1	Not Available

1.1.10.2. Facility and Date: Rocky Flats CML, 1983 - 1984

1.1.10.3. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
HEU-SOL-THERM-044	0.9984	0.0065

1.1.10.4. Calculated k_{eff}

Identification	Code	Library	C/E
HEU-SOL-THERM-044	MCNP	ENDF/B-VI	1.017

1.1.10.5. Sensitivity of k_{eff} to Gadolinium: Not Available

1.1.10.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
HEU-SOL-THERM-044	Not Available	Not Available	Not Available

1.1.11. Critical Experiment: ***IEU-COMP-THERM-002***

1.1.11.1. Description: Water-Moderated U(17)O2 Annular Fuel Rods With Gadolinium Absorbers In Hexagonal Lattices At Different Temperatures

Identification	No. of Cases	Percent Absorption In Gadolinium
IEU-COMP-THERM-002	2	Not Available

1.1.11.2. Facility and Date: IPPE, 1970 - 1973

1.1.11.3. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
IEU-COMP-THERM-002	1.0017 – 1.0019	0.0044

1.1.11.4. Calculated k_{eff}

Identification	Code	Library	C/E
IEU-COMP-THERM-002	MCNP	ENDF/B-V	0.9980 – 1.0010

1.1.11.5. Sensitivity of k_{eff} to Gadolinium: Not Available

1.1.11.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
IEU-COMP-THERM-002	Not Available	Not Available	Not Available

1.1.12. Critical Experiment: ***LEU-COMP-THERM-036***

1.1.12.1. Description: Low Enriched VVER Lattices with Gadolinium Absorber Rods

Identification	No. of Cases	Fuel Rod Enrichment (%)	Percent Absorption In Gadolinium
LEU-COMP-THERM-036	19	3.6	0.0 – 3.5

1.1.12.2. Facility and Date: KFKI, 1986 - 1990

1.1.12.3. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
LEU-COMP-THERM-036	1.0000	0.0090

1.1.12.4. Calculated k_{eff}

Identification	Code	Library	C/E
LEU-COMP-THERM-036	KENO	ENDF/B-IV 27 Group	0.9885 – 0.9945
	KENO	ENDF/B-V 44 Group	1.0006 – 1.0106

1.1.12.5. Sensitivity of k_{eff} to Gadolinium: Not Available

1.1.12.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
LEU-COMP-THERM-036	81.9 – 82.0	11.1 – 11.4	6.6 – 6.8

2. Data Need #2: Beryllium

2.1. Existing Critical Experiments with Beryllium

2.1.1. Critical Experiment: ***HEU-MET-FAST-017***

2.1.1.1. Description: Beryllium Moderated and Reflected Cylinder of Highly Enriched Uranium Metal

Identification	No. of Cases	Percent Absorption In Beryllium
HEU-MET-FAST-017	1	Not Available

2.1.1.2. Facility and Date: VNIITF, 1984

2.1.1.3. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
HEU-MET-FAST-017	0.9993	0.0014

2.1.1.4. Calculated k_{eff}

Identification	Code	Library	C/E
HEU-MET-FAST-017	MCNP	ENDF/B-V	0.9965

2.1.1.5. Sensitivity of k_{eff} to Beryllium: Not Available

2.1.1.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
HEU-MET-FAST-017	0	18	82

2.1.2. Critical Experiment: ***HEU-MET-FAST-030***

2.1.2.1. Description: Beryllium Moderated Cylinder of Highly Enriched Uranium Metal with Depleted Uranium Reflector

Identification	No. of Cases	Percent Absorption In Beryllium
HEU-MET-FAST-030	1	3.8

2.1.2.2. Facility and Date: VNIITF, 1989

2.1.2.3. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
HEU-MET-FAST-030	1.0000	0.0009

2.1.2.4. Calculated k_{eff}

Identification	Code	Library	C/E
HEU-MET-FAST-030	MCNP	ENDF/B-V	1.0008

2.1.2.5. Sensitivity of k_{eff} to Beryllium: Not Available

2.1.2.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
HEU-MET-FAST-030	0.1	39.5	60.4

2.1.3. Critical Experiment: ***HEU-MET-FAST-038***

2.1.3.1. Description: Beryllium and Beryllium Moderated Cylinder of Highly Enriched Uranium Metal with Depleted Uranium Reflector

Identification	No. of Cases	Percent Absorption In Beryllium
HEU-MET-FAST-038	1	3.4

2.1.3.2. Facility and Date: VNIITF, 1989

2.1.3.3. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
HEU-MET-FAST-038	0.9999	0.0015

2.1.3.4. Calculated k_{eff}

Identification	Code	Library	C/E
HEU-MET-FAST-038	MCNP	ENDF/B-V	1.0012

2.1.3.5. Sensitivity of k_{eff} to Beryllium: Not Available

2.1.3.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
HEU-MET-FAST-038	0.1	38.6	61.3

2.1.4. Critical Experiment: ***HEU-MET-FAST-058***

2.1.4.1. Description: Highly Enriched Uranium Metal Spheres with Beryllium Reflectors (NIMBUS Experiments)

Identification	No. of Cases	Percent Absorption In Beryllium
HEU-MET-FAST-058	5	Not Available

2.1.4.2. Facility and Date: LLNL, 1958 – 1959

2.1.4.3. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
HEU-MET-FAST-058	1.0000	0.002 – 0.004

2.1.4.4. Calculated k_{eff}

Identification	Code	Library	C/E
HEU-MET-FAST-058	MCNP	ENDF/B-VI.2	.9986 – 1.0077
	KENO	ENDF/B-V	.9963 – 1.0038
	VIM	ENDF/B-V	1.0032 – 1.0079
	COG	ENDF/B-VI.7	1.0006 – 1.0069

2.1.4.5. Sensitivity of k_{eff} to Beryllium: Not Available

2.1.4.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
HEU-MET-FAST-058	Not Available	Not Available	Not Available

2.1.5. Critical Experiment:***HEU-MET-THERM-025***

2.1.5.1. Description: Beryllium Moderated Critical Assemblies (CA-1 & 18)

Identification	No. of Cases	Percent Absorption In Beryllium
HEU-MET-THERM-025	2	8.1 – 11.6

2.1.5.2. Facility and Date: ORNL, 1951 – 1954

2.1.5.3. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
HEU-MET-THERM-025	1.0007 – 1.0037	0.0033 – 0.004

2.1.5.4. Calculated k_{eff}

Identification	Code	Library	C/E
HEU-MET-THERM-025	KENO	ENDF/B-V 238 Group	1.011 – 1.016
	MCNP	ENDF/B-V	1.031 – 1.042

2.1.5.5. Sensitivity of k_{eff} to Beryllium: Not Available

2.1.5.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
HEU-MET-THERM-025	50.6 – 81.8	16.6 – 46.3	1.6 – 3.1

2.1.6. Critical Experiment:***HEU-COMP-THERM-010 & HEU-MISC-THERM-002***

2.1.6.1. Description: EBOR Fuel Pins in Water or Uranyl Nitrate

Identification	No. of Cases	Percent Absorption In Beryllium
HEU-COMP-THERM-010	19	0.2 – 0.8
HEU-MISC-THERM-0002	2	0.2

2.1.6.2. Facility and Date: ORNL, 1967

2.1.6.3. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
HEU-COMP-THERM-010	1.0000 – 1.0010	0.0050 – 0.0074
HEU-MISC-THERM-0002	0.9997 – 1.0001	0.0048 – 0.0076

2.1.6.4. Calculated k_{eff}

Identification	Code	Library	C/E
HEU-COMP-THERM-010	KENO	ENDF/B-IV 27 Group	0.9936 – 1.0039
	MCNP	ENDF/B-V	0.9827 – 1.0053
HEU-MISC-THERM-0002	KENO	ENDF/B-IV 27 Group	1.0028 – 1.0050
	MCNP	ENDF/B-V	1.0047 – 1.0050

2.1.6.5. Sensitivity of k_{eff} to Beryllium: Not Available

2.1.6.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
HEU-COMP-THERM-010	67.7 – 92.5	6.3 – 27.8	1.2 – 4.5
HEU-MISC-THERM-0002	90 – 92	6.8 – 8.6	1.2 – 1.4

3. Data Need #3: Steel (Iron, Chromium, Nickel, Manganese)

3.1. Existing Critical Experiments with Steel (Iron, Chromium, Nickel, Manganese)

3.1.1. Critical Experiment: ***HEU-MET-FAST-048***

3.1.1.1. Description: Oil-Reflected Spheres And Hemispheres Of Highly Enriched Uranium (93.1 % 235u) Metal With Oil Or Steel Moderator

Identification	No. of Cases	Percent Absorption In Iron	Percent Absorption In Chromium	Percent Absorption In Nickel	Percent Absorption In Manganese
HEU-MET-FAST-048	7	Not Available	Not Available	Not Available	Not Available

3.1.1.2. Facility and Date: Rocky Flats CML, 1965 - 1967

3.1.1.3. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
HEU-MET-FAST-048	0.9978 – 1.0011	0.0062 – 0.0116

3.1.1.4. Calculated k_{eff}

Identification	Code	Library	C/E
HEU-MET-FAST-048	KENO	ENDF/B-V 238 Group	0.9910 – 1.0019
	MCNP	ENDF/B-V	0.9900 – 1.0018

3.1.1.5. Sensitivity of k_{eff} to Fe, Cr, Ni, Mn: Not Available

3.1.1.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
HEU-MET-FAST-048	17.1 – 22.1	19.9 – 27	50.9 – 63

3.1.2. Critical Experiment: ***HEU-MET-INTER-001***

3.1.2.1. Description: ZPR-9/34 Uranium – Iron Benchmark

Identification	No. of Cases	Percent Absorption In Iron	Percent Absorption In Chromium	Percent Absorption In Nickel	Percent Absorption In Manganese
HEU-MET-INTER-001	1	12.8	0.9	0.8	1.1

3.1.2.2. Facility and Date: ZPR, 1979

3.1.2.3. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
HEU-MET-INTER-001	0.9966	0.0026

3.1.2.4. Calculated k_{eff}

Identification	Code	Library	C/E
HEU-MET-INTER-001	KENO	ENDF/B-V 238 Group	1.0952
	MCNP	ENDF/B-V	0.9953
	MCNP	ENDF/B-VI	1.0163
	MONK	ENDF/B-VI	1.0204
	VIM	ENDF/B-V	0.9917

3.1.2.5. Sensitivity of k_{eff} to Fe, Cr, Ni, Mn: Not Available

3.1.2.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
HEU-MET-INTER-001	0	60	40

3.1.3. Critical Experiment: ***HEU-MET-THERM-013***

3.1.3.1. Description: Highly Enriched Uranium (93% ^{235}U) Metal Foils With Polyethylene and Iron Plates (PLANET)

Identification	No. of Cases	Percent Absorption In Iron	Percent Absorption In Chromium	Percent Absorption In Nickel	Percent Absorption In Manganese
HEU-MET-THERM-013	2	Not Available	Not Available	Not Available	Not Available

3.1.3.2. Facility and Date: LACEF, 2002

3.1.3.3. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
HEU-MET-THERM-013	0.9973 – 1.0006	0.0020 – 0.0022

3.1.3.4. Calculated k_{eff}

Identification	Code	Library	C/E
HEU-MET-THERM-013	MCNP	ENDF/B-V	1.0057 – 1.0081
	MCNP	ENDF/B-VI.4	1.0072 – 1.0116

3.1.3.5. Sensitivity of k_{eff} to Fe: Not Available

3.1.3.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
HEU-MET-THERM-013	Not Available	Not Available	Not Available

3.1.4. Critical Experiment: ***PU-MET-INTER-002***

3.1.4.1. Description: ZPR-6/10 Plutonium, Carbon, Stainless Steel Assembly

Identification	No. of Cases	Percent Absorption In Iron	Percent Absorption In Chromium	Percent Absorption In Nickel	Percent Absorption In Manganese
PU-MET-INTER-001	1	7.7	3.3	2.6	2.7

3.1.4.2. Facility and Date: ZPR, 1981 - 1982

3.1.4.3. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
PU-MET-INTER-001	0.9869	0.0026

3.1.4.4. Calculated k_{eff}

Identification	Code	Library	C/E
PU-MET-INTER-001	KENO	ENDF/B-IV 27 Group	1.0398
	KENO	ENDF/B-V 238 Group	1.0405
	MCNP	ENDF/B-V	1.0171
	MCNP	ENDF/B-VI	1.0381
	MONK	ENDF/B-VI Rel. 3	1.0461
	VIM	ENDF/B-V	0.9993
	VIM	ENDF/B-VI	1.0358

3.1.4.5. Sensitivity of k_{eff} to Fe, Cr, Ni, Mn: Not Available

3.1.4.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
PU-MET-INTER-001	0.0	67.2	32.7

3.1.5. Critical Experiment: ***HEU-COMP-INTER-005***

3.1.5.1. Description: K-Infinity Measurements with Structural Materials

Identification	Material	Percent Absorption Iron	Percent Absorption Chromium	Percent Absorption Nickel	Percent Absorption Manganese
HEU-COMP-INTER-005-1	Ni	1.4	0.5	39.1	0.4
HEU-COMP-INTER-005-2	SST	18.8	7.2	6.7	5.4
HEU-COMP-INTER-005-3	SST	14.7	5.7	5.7	4.0
HEU-COMP-INTER-005-4	Cr	2.9	32.3	1.0	0.9

3.1.5.2. Facility and Date: IPPE (KBR), 1970s & 1980s

3.1.5.3. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
HEU-COMP-INTER-005-1	1.032	0.0040
HEU-COMP-INTER-005-2	1.05	0.0080
HEU-COMP-INTER-005-3	1.03	0.0060
HEU-COMP-INTER-005-4	1.064	0.0018

3.1.5.4. Calculated k_{eff}

Identification	Code	Library	C/E
HEU-COMP-INTER-005-1	KENO	ENDF/B-V 238 Group	0.9504
	KENO	ENDF/B-V 44 Group	0.9515
	MCNP	ENDF/B-V	0.9612
	MCNP	ENDF/B-VI	1.0047
HEU-COMP-INTER-005-2	KENO	ENDF/B-V 238 Group	0.9751
	KENO	ENDF/B-V 44 Group	0.9684
	MCNP	ENDF/B-V	0.9851
	MCNP	ENDF/B-VI	1.0591
HEU-COMP-INTER-005-3	KENO	ENDF/B-V 238 Group	Not Available
	KENO	ENDF/B-V 44 Group	Not Available
	MCNP	ENDF/B-V	0.9599
	MCNP	ENDF/B-VI	1.0141
HEU-COMP-INTER-005-4	KENO	ENDF/B-V 238 Group	0.9021
	KENO	ENDF/B-V 44 Group	0.8273
	MCNP	ENDF/B-V	0.8477
	MCNP	ENDF/B-VI	1.0936

3.1.5.5. Sensitivity of k_{eff} to Fe, Cr, Ni, Mn: Not Available

3.1.5.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
HEU-COMP-INTER-005-1	0.1	71.2	28.7
HEU-COMP-INTER-005-2	0	79.4	20.6
HEU-COMP-INTER-005-3	0	75.2	24.7
HEU-COMP-INTER-005-4	0	82.3	17.7

3.1.6. Critical Experiment: ***HEU-SOL-THERM-038 (Cases 21 – 23)***

3.1.6.1. Description: WINCO Slab Tanks – Two Interacting Tanks Filled with Highly Enriched Uranyl Nitrate Solution with Stainless Steel Absorber and Reflector Plates

Identification	No. of Cases	Percent Absorption In Iron	Percent Absorption In Chromium	Percent Absorption In Nickel	Percent Absorption In Manganese
HEU-SOL-THERM-038	3	3.5 – 5.1	0.0 – 1.2	0.0 – 0.7	0.0 – 0.5

3.1.6.2. Facility and Date: LACEF, 1988

3.1.6.3. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
HEU-SOL-THERM-038	1.0000	0.0025

3.1.6.4. Calculated k_{eff}

Identification	Code	Library	C/E
HEU-SOL-THERM-038	KENO	ENDF/B-V 44 Group	0.9994 – 1.0026
	MCNP	ENDF/B-V	1.0013 – 1.0018
	COG	ENDF/B-VI Rel. 7	0.9981 – 9991

3.1.6.5. Sensitivity of k_{eff} to Fe, Cr, Ni, Mn: Not Available

3.1.6.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
HEU-SOL-THERM-038	76.0 – 76.2	21.4 – 21.7	2.3

3.1.7. Critical Experiment: ***LEU-COMP-THERM-009 (Cases 1 – 4); LEU-COMP-THERM-012 (Case 1); LEU-COMP-THERM-013 (Case 1); LEU-COMP-THERM-016 (Cases 1 – 7); LEU-COMP-THERM-042 (Case 1)***

3.1.7.1. Description: Low Enriched UO_2 Lattice Arrays Separated by Steel Plates

Identification	No. of Cases	% ^{235}U	Percent Absorption In Iron	Percent Absorption In Chromium	Percent Absorption In Nickel	Percent Absorption In Manganese
LEU-COMP-THERM-009	4	4.31	1.0 – 1.4	0.3 – 0.5	0.2 – 0.4	0.1 – 0.2
LEU-COMP-THERM-012	1	2.35	1.0	0.3	0.2	0.1
LEU-COMP-THERM-013	1	4.31	0.7	0.3	0.2	0
LEU-COMP-THERM-016	7	2.35	0.8 – 1.4	0.2 – 0.5	0.2 – 0.3	0.1 – 0.2
LEU-COMP-THERM-042	1	2.35	0.9	0.3	0.2	0.1

3.1.7.2. Facility and Date: PNL, 1977 – 1980

3.1.7.3. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
LEU-COMP-THERM-009	1.0000	0.0021
LEU-COMP-THERM-012	1.0000	0.0034
LEU-COMP-THERM-013	1.0000	0.0018
LEU-COMP-THERM-016	1.0000	0.0031
LEU-COMP-THERM-042	1.0000	0.0016

3.1.7.4. Calculated k_{eff}

Identification	Code	Library	C/E
LEU-COMP-THERM-009	KENO	ENDF/B-V 44 Group	0.9922 – 0.9980
	MCNP	ENDF/B-V	0.9959 – 0.9985
LEU-COMP-THERM-012	KENO	ENDF/B-V 44 Group	0.9864
	MCNP	ENDF/B-V	0.9845
LEU-COMP-THERM-013	MCNP	ENDF/B-V	0.9981
	MCNP	ENDF/B-VI	0.9938
LEU-COMP-THERM-016	KENO	ENDF/B-V 44 Group	0.9933 – 0.9981
	MCNP	ENDF/B-V	0.9923 – 0.9982
LEU-COMP-THERM-042	KENO	ENDF/B-V 44 Group	0.9950
	MCNP	ENDF/B-V	0.9955

3.1.7.5. Sensitivity of k_{eff} to Fe, Cr, Ni, Mn: Not Available

3.1.7.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
LEU-COMP-THERM-009	90.2	5.9 – 6.0	3.8 – 3.9
LEU-COMP-THERM-012	87	7.2	5.8
LEU-COMP-THERM-013	82.2	11.5	6.3
LEU-COMP-THERM-016	91.3 – 91.4	4.5 – 4.6	4.1
LEU-COMP-THERM-042	86.9	7.4	5.7

3.1.8. Critical Experiment: ***LEU-COMP-THERM-051 (Cases 2 – 9)***

3.1.8.1. Description: Water Storage of Power Reactor Fuel with Steel Absorber Plates

Identification	No. of Cases	Percent Absorption In Iron	Percent Absorption In Chromium	Percent Absorption In Nickel	Percent Absorption In Manganese
LEU-COMP-THERM-051	8	3.6	1.5	0.8	0.6

3.1.8.2. Facility and Date: BMW (CX-10), 1978 – 1979

3.1.8.3. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
LEU-COMP-THERM-051	1.0010	0.0019 – 0.0024

3.1.8.4. Calculated k_{eff}

Identification	Code	Library	C/E
LEU-COMP-THERM-051	KENO	ENDF/B-V 44 Group	0.9942 – 0.9964
	MCNP	ENDF/B-V	0.9936 – 0.9964

3.1.8.5. Sensitivity of k_{eff} to Fe, Cr, Ni, Mn: Not Available

3.1.8.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
LEU-COMP-THERM-051	85.6 – 87.1	7.3 – 8.2	5.7 – 6.2

4. Data Need #4: Copper

4.1. Existing Critical Experiments with Copper

4.1.1. Critical Experiment: ***LEU-COMP-THERM-009 (Cases 10 – 13); LEU-COMP-THERM-012 (Case 9); LEU-COMP-THERM-013 (Case 6); LEU-COMP-THERM-06 (Cases 15 – 19); LEU-COMP-THERM-042 (Case 6)***

4.1.1.1. Description: Low Enriched UO_2 Lattice Arrays Separated by Copper Plates

Identification	No. of Cases	Percent Absorption In Copper
LEU-COMP-THERM-009	4	0.4
LEU-COMP-THERM-012	1	0.8
LEU-COMP-THERM-013	1	0.3
LEU-COMP-THERM-016	5	0.8
LEU-COMP-THERM-042	1	0.8

4.1.1.2. Facility and Date: PNL, 1977 - 1980

4.1.1.3. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
LEU-COMP-THERM-009	1.0000	0.0021
LEU-COMP-THERM-012	1.0000	0.0034
LEU-COMP-THERM-013	1.0000	0.0018
LEU-COMP-THERM-016	1.0000	0.0031
LEU-COMP-THERM-042	1.0000	0.0016

4.1.1.4. Calculated k_{eff}

Identification	Code	Library	C/E
LEU-COMP-THERM-009	KENO	ENDF/B-V 44 Group	0.9946 – 0.9989
	MCNP	ENDF/B-V	0.9967 – 0.9992
LEU-COMP-THERM-012	KENO	ENDF/B-V 44 Group	0.9872
	MCNP	ENDF/B-V	0.9835
LEU-COMP-THERM-013	MCNP	ENDF/B-V	0.9971
	MCNP	ENDF/B-VI	0.9932
LEU-COMP-THERM-016	KENO	ENDF/B-V 44 Group	0.9911 – 0.9988
	MCNP	ENDF/B-V	0.9932 – 0.9986
LEU-COMP-THERM-042	KENO	ENDF/B-V 44 Group	0.9966
	MCNP	ENDF/B-V	0.9962

4.1.1.5. Sensitivity of k_{eff} to Copper: Not Available

4.1.1.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
LEU-COMP-THERM-009	90.2	6.0	3.8
LEU-COMP-THERM-012	87.2	7.1	5.8
LEU-COMP-THERM-013	82.2	11.5	6.3
LEU-COMP-THERM-016	91.4	4.5	4.1
LEU-COMP-THERM-042	86.8	7.4	5.7

5. Data Need #5: Aluminum

5.1. Existing Critical Experiments with Aluminum

5.1.1. Critical Experiment: ***LEU-COMP-THERM-009 (Cases 24 – 25)***
LEU-COMP-THERM-016 (Cases 28 – 30)

5.1.1.1. Description: Low Enriched UO_2 Lattice Arrays Separated by Aluminum Plates

Identification	No. of Cases	Percent Absorption In Aluminum
LEU-COMP-THERM-009	4	0.4
LEU-COMP-THERM-016	3	1.0 – 1.1

5.1.1.2. Facility and Date: PNL, 1977

5.1.1.3. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
LEU-COMP-THERM-009	1.0000	0.0021
LEU-COMP-THERM-016	1.0000	0.0031

5.1.1.4. Calculated k_{eff}

Identification	Code	Library	C/E
LEU-COMP-THERM-009	KENO	ENDF/B-V 44 Group	0.9946 – 0.9962
	MCNP	ENDF/B-V	0.9956 – 0.9971
LEU-COMP-THERM-016	KENO	ENDF/B-IV 27 Group	0.9911 – 0.9932
	KENO	ENDF/B-V 44 Group	0.9943 – 0.9995
	MCNP	ENDF/B-V	0.9957 – 0.9970

5.1.1.5. Sensitivity of k_{eff} to Aluminum: Not Available

5.1.1.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
LEU-COMP-THERM-009	90.2	5.9 – 6.0	3.8
LEU-COMP-THERM-016	91.5	4.5	4.0

5.1.2. Critical Experiment: ***LEU-COMP-THERM-061 (Case 9)***

5.1.2.1. Description: Low Enriched UO_2 Lattice Arrays Separated by Aluminum Plates

Identification	No. of Cases	Percent Absorption In Aluminum
LEU-COMP-THERM-061	1	0.6

5.1.2.2. Facility and Date: RRC Kurchatov Institute, 1992 – 1993

5.1.2.3. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
LEU-COMP-THERM-061	1.0005	0.0023

5.1.2.4. Calculated k_{eff}

Identification	Code	Library	C/E
LEU-COMP-THERM-061	KENO	ENDF/B-V 238 Group	0.9992

5.1.2.5. Sensitivity of k_{eff} to Aluminum: Not Available

5.1.2.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
LEU-COMP-THERM-061	83.2	10.9	5.9

5.1.3. Critical Experiment: ***HEU-MET-FAST-055***

5.1.3.1. Description: ZPR-3/23 – Cylindrical Assembly of Highly Enriched Uranium Metal and Aluminum

Identification	No. of Cases	Percent Absorption In Aluminum
HEU-MET-FAST-055	1	0.8

5.1.3.2. Facility and Date: ZPR-3, 1959

5.1.3.3. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
HEU-MET-FAST-055	0.9955	0.0028

5.1.3.4. Calculated k_{eff}

Identification	Code	Library	C/E
HEU-MET-FAST-055	KENO	ENDF/B-V 238 Group	1.0108
	MCNP	ENDF/B-V	1.0003
	MCNP	ENDF/B-VI Rel. 4	1.0059
	MONK	ENDF/B-VI	1.0071
	VIM	ENDF/B-V	1.0022
	VIM	ENDF/B-VI	1.0055

5.1.3.5. Sensitivity of k_{eff} to Aluminum: Not Available

5.1.3.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
HEU-MET-FAST-055	0.0	18.9	81.1

5.1.4. Critical Experiment: ***IEU-MET-FAST-012***

5.1.4.1. Description: ZPR-3/41 – Cylindrical Assembly of 16% Enriched Uranium Metal, Aluminum, and Steel, Reflected by Depleted-Uranium

Identification	No. of Cases	Percent Absorption In Aluminum	Percent Absorption In Iron	Percent Absorption In Chromium	Percent Absorption In Nickel
HEU-MET-FAST-055	1	0.3	0.8	0.2	0.3

5.1.4.2. Facility and Date: ZPR-3, 1955

5.1.4.3. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
HEU-MET-FAST-055	1.0007	0.0027

5.1.4.4. Calculated k_{eff}

Identification	Code	Library	C/E
HEU-MET-FAST-055	KENO	ENDF/B-V 238 Group	1.0101
	MCNP	ENDF/B-V	1.0015
	MONK	ENDF/B-VI Rel. 3	1.0059
	VIM	ENDF/B-V	1.0040
	VIM	ENDF/B-VI Rel. 3	1.0076

5.1.4.5. Sensitivity of k_{eff} to Aluminum: Not Available

5.1.4.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
HEU-MET-FAST-055	0.0	26.7	73.3

6. Data Need #6: Silicon

6.1. Existing Critical Experiments with Silicon

6.1.1. Critical Experiment: ***HEU-MET-THERM-005 (Case 3); HEU-MET-MIXED-005 (Cases 1 – 2); HEU-MET-INTER-005 (Cases 4 – 5)***

6.1.1.1. Description: BFS-1 Highly Enriched Uranium Assembly Moderated by Silicon Dioxide and Polyethylene

Identification	No. of Cases	Percent Absorption In Silicon
HEU-MET-THERM-005	1	8.8
HEU-MET-MIXED-005	2	3.2 – 7.0
HEU-MET-INTER-005	2	1.4 – 2.4

6.1.1.2. Facility and Date: IPPE (BFS-1), 1999

6.1.1.3. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
HEU-MET-THERM-005	1.0012	0.0029
HEU-MET-MIXED-005	1.0003 – 1.0007	0.0027 – 0.0028
HEU-MET-INTER-005	1.0005 – 1.0016	0.0030 – 0.0040

6.1.1.4. Calculated k_{eff}

Identification	Code	Library	C/E
HEU-MET-THERM-005	MCNP	ENDF/B-V	1.0177
	MCNP	ENDF/B-VI	1.0170
	MCNP	ENDF/B-VI Rel. 4	1.0123
HEU-MET-MIXED-005	MCNP	ENDF/B-V	1.0122 – 1.0226
	MCNP	ENDF/B-VI	1.0120 – 1.0231
	MCNP	ENDF/B-VI Rel. 4	1.0044 – 1.0150
HEU-MET-INTER-005	MCNP	ENDF/B-V	1.0142 – 1.0174
	MCNP	ENDF/B-VI	1.0059 – 1.0156
	MCNP	ENDF/B-VI Rel. 4	1.0057 – 1.0110

6.1.1.5. Sensitivity of k_{eff} to Silicon: Not Available

6.1.1.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
HEU-MET-THERM-005	68.4	20.5	11.1
HEU-MET-MIXED-005	35.4 – 47.9	38.3 – 45.9	13.9 – 18.6
HEU-MET-INTER-005	4.4 – 23.8	54.9 – 68.4	21.2 – 27.2

6.1.2. Critical Experiment: ***PU-MET-THERM-001 (Cases 5 – 6); PU-MET-MIXED-001 (Case 1); PU-MET-INTER-001 (Cases 1 – 3)***

6.1.2.1. Description: BFS-1 Highly Enriched Uranium Assembly Moderated by Silicon Dioxide and Polyethylene

Identification	No. of Cases	Percent Absorption In Silicon
PU-MET-THERM-001	2	5.6
PU-MET-MIXED-001	1	2.2
PU-MET-INTER-001	3	1.5 – 1.8

6.1.2.2. Facility and Date: IPPE (BFS-1), 2000

6.1.2.3. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
PU-MET-THERM-001	1.0001 – 1.0003	0.0025
PU-MET-MIXED-001	1.0000	0.0025
PU-MET-INTER-001	1.0002 – 1.0005	0.0025 – 0.0037

6.1.2.4. Calculated k_{eff}

Identification	Code	Library	C/E
PU-MET-THERM-001	MCNP	ENDF/B-V	1.0161 – 1.0177
	MCNP	ENDF/B-VI	1.0103 – 1.0105
	MCNP	ENDF/B-VI Rel. 4	1.0095 – 1.0107
PU-MET-MIXED-001	MCNP	ENDF/B-V	1.0098
	MCNP	ENDF/B-VI	1.0116
	MCNP	ENDF/B-VI Rel. 4	1.0100
PU-MET-INTER-001	MCNP	ENDF/B-V	0.9990 – 1.0056
	MCNP	ENDF/B-VI	1.0095 – 1.0146
	MCNP	ENDF/B-VI Rel. 4	1.0093 – 1.0118

6.1.2.5. Sensitivity of k_{eff} to Silicon: Not Available

6.1.2.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
PU-MET-THERM-001	68.4	20.5	11.1
PU-MET-MIXED-001	34.8	43.8	21.4
PU-MET-INTER-001	70.3 – 70.6	18.1 – 18.2	11.4 – 11.5

6.1.3. Critical Experiment: ***HEU-MET-THERM-001& 014***

6.1.3.1. Description: PLANET Highly Enriched Uranium Assembly Moderated by Silicon Dioxide and Polyethylene

Identification	No. of Cases	Percent Absorption In Silicon
HEU-MET-THERM-001	1	0.5
HEU-MET-THERM-014	1	Not Available

6.1.3.2. Facility and Date: LACEF (PLANET), 1999 and 2002

6.1.3.3. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
HEU-MET-THERM-001	1.0010	0.0060
HEU-MET-THERM-014	0.9939	0.0015

6.1.3.4. Calculated k_{eff}

Identification	Code	Library	C/E
HEU-MET-THERM-001	KENO	ENDF/B-V 238 Group	1.0064
	KENO	ENDF/B-V 44 Group	1.0080
	MCNP	ENDF/B-V	1.0084
	MCNP	ENDF/B-VI Rel. 2	1.0063
	PARTISN	ENDF/B-V 44 Group	0.9980
HEU-MET-THERM-014	KENO	ENDF/B-V 238 Group	1.0104
	MCNP	ENDF/B-V	1.0134
	MCNP	ENDF/B-VI Rel. 4	1.0136

6.1.3.5. Sensitivity of k_{eff} to Silicon: Not Available

6.1.3.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
HEU-MET-THERM-001	91.0	7.8	1.2
HEU-MET-THERM-014	Not Available	Not Available	Not Available

7. Data Need #7: Chlorine

7.1. Existing Critical Experiments with Chlorine

7.1.1. Critical Experiment: ***HEU-SOL-THERM-044 (Cases 2 – 3)***

7.1.1.1. Description: Cylindrical Tank Containing Highly Enriched Uranyl Nitrate Solution with Periodically Distributed PVC Absorber Rods

Identification	No. of Cases	Percent Absorption In Chlorine
HEU-SOL-THERM-044	2	Not Available

7.1.1.2. Facility and Date: Rocky Flats CML, 1983 - 1984

7.1.1.3. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
HEU-SOL-THERM-044	0.9944 – 0.9964	0.0047 – 0.0097

7.1.1.4. Calculated k_{eff}

Identification	Code	Library	C/E
HEU-SOL-THERM-044	MCNP	ENDF/B-VI	0.9987 – 1.0110

7.1.1.5. Sensitivity of k_{eff} to Chlorine: Not Available

7.1.1.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
HEU-SOL-THERM-044	Not Available	Not Available	Not Available

7.1.2. Critical Experiment: ***LEU-COMP-THERM-045 (Cases 6, 18, 19)***

7.1.2.1. Description: Plexiglas Or Concrete-Reflected $U(4.46)_3O_8$ With H/U=0.77 and Interstitial Moderation

Identification	No. of Cases	Percent Absorption In Chlorine
HEU-SOL-THERM-045	3	4.0 – 4.3

7.1.2.2. Facility and Date: Rocky Flats CML, 1978

7.1.2.3. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
HEU-SOL-THERM-045	1.0039 – 1.0041	0.0026 – 0.0031

7.1.2.4. Calculated k_{eff}

Identification	Code	Library	C/E
HEU-SOL-THERM-045	KENO	238 Group ENDF/B-V	1.003 – 1.017

7.1.2.5. Sensitivity of k_{eff} to Chlorine: Not Available

7.1.2.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
HEU-SOL-THERM-045	52.5 – 54.0	41.2 – 42.8	4.8

8. Data Need #8: Graphite

8.1. Existing Critical Experiments with Graphite

8.1.1. Critical Experiment: ***HEU-COMP-INTER-004***

8.1.1.1. Description: HECTOR Intermediate Spectrum Study (HISS) – Infinite System of Highly Enriched Uranium and Graphite

Identification	No. of Cases	Percent Absorption In Graphite
HEU-COMP-INTER-004	1	0.2

8.1.1.2. Facility and Date: Winfrith (HECTOR), 1970

8.1.1.3. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
HEU-COMP-INTER-004	1.0000	0.0040

8.1.1.4. Calculated k_{eff}

Identification	Code	Library	C/E
HEU-COMP-INTER-004	KENO	ENDF/B-V 238 Group	1.0225
	MCNP	ENDF/B-V	1.0277
	MCNP	ENDF/B-VI Rel. 4	1.0129

8.1.1.5. Sensitivity of k_{eff} to Graphite: Not Available

8.1.1.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
HEU-COMP-INTER-004	3.9	90.7	5.5

8.1.2. Critical Experiment: ***HEU-COMP-THERM-002***

8.1.2.1. Description: Graphite and Water Moderated NRX-A3 and NRX-A4 ROVER Assemblies

Identification	No. of Cases	Percent Absorption In Graphite
HEU-COMP-THERM-002	25	0.0 – 0.1

8.1.2.2. Facility and Date: ORNL, 1965 – 1966

8.1.2.3. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
HEU-COMP-THERM-002	1.0008 – 1.0020	0.0043 – 0.0085

8.1.2.4. Calculated k_{eff}

Identification	Code	Library	C/E
HEU-COMP-THERM-002	KENO	ENDF/B-V 44 Group	1.0113 – 1.0245
	MCNP	ENDF/B-V	1.0084 – 1.0208

8.1.2.5. Sensitivity of k_{eff} to Graphite: Not Available

8.1.2.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
HEU-COMP-THERM-002	85.6 – 95.8	3.6 – 12.8	0.5 – 1.6

8.1.3. Critical Experiment: ***HEU-COMP-THERM-016***

8.1.3.1. Description: Impulse Graphite Reactor (IGR)

Identification	No. of Cases	Percent Absorption In Graphite
HEU-COMP-THERM-016	6	Not Available

8.1.3.2. Facility and Date: Semipalatinsk Nuclear Test Site, 1970 - 1992

8.1.3.3. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
HEU-COMP-THERM-016	1.0000	0.0011

8.1.3.4. Calculated k_{eff}

Identification	Code	Library	C/E
HEU-COMP-THERM-016	MCNP	ENDF/B-VI	1.0033 – 1.0100

8.1.3.5. Sensitivity of k_{eff} to Graphite: Not Available

8.1.3.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
HEU-COMP-THERM-016	Not Available	Not Available	Not Available

8.1.4. Critical Experiment: ***HEU-MET-INTER-006***

8.1.4.1. Description: ZEUS: Intermediate-Spectrum Critical Assemblies With A Graphite-HEU Core

Identification	No. of Cases	Percent Absorption In Graphite
HEU-MET-INTER-006	3	0.2

8.1.4.2. Facility and Date: LACEF (COMET), 1999 – 2001

8.1.4.3. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
HEU-MET-INTER-006	0.9976 – 1.0010	0.0008 – 0.0009

8.1.4.4. Calculated k_{eff}

Identification	Code	Library	C/E
HEU-MET-INTER-006	KENO	ENDF/B-V 238 Group	1.0037 – 1.0065
	KENO	ENDF/B-V 44 Group	0.9858 – 0.9907
	MCNP	ENDF/B-V	0.9991 – 0.9998
	MCNP	ENDF/B-VI Rel. 4	0.9972 – 1.0011

8.1.4.5. Sensitivity of k_{eff} to Graphite: Not Available

8.1.4.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
HEU-MET-INTER-006[(a)]	0.4 – 1.4	69.8 – 72.8	25.7 – 29.8

(a) Only Cases 1 and 2 are included in these data

8.1.5. Critical Experiment: ***LEU-COMP-THERM-060***

8.1.5.1. Description: Graphite Moderated and Reflected RBMK

Identification	No. of Cases	Percent Absorption In Graphite
LEU-COMP-THERM-060	10	Not Available

8.1.5.2. Facility and Date: RRC KI, 1982 – 1988

8.1.5.3. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
LEU-COMP-THERM-060	0.9891 – 1.0043	0.0029

8.1.5.4. Calculated k_{eff}

Identification	Code	Library	C/E
LEU-COMP-THERM-060	MCNP	ENDF/B-V	0.990 – 0.993

8.1.5.5. Sensitivity of k_{eff} to Graphite: Not Available

8.1.5.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
LEU-COMP-THERM-060	Not Available	Not Available	Not Available

8.1.6. Critical Experiment: ***MIX-MET-FAST-008; MIX-MET-INTER-001***

8.1.6.1. Description: ZEBRA – K-Infinity Measurements with Mixed Plutonium–Uranium and Graphite

Identification	No. of Cases	Percent Absorption In Graphite
MIX-MET-FAST-008	3	0
MIX-MET-INTER-001	2	0

8.1.6.2. Facility and Date: Winfrith (ZEBRA), 1987

8.1.6.3. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
MIX-MET-FAST-008	0.973 – 1.001	0.0023 – 0.0045
MIX-MET-INTER-001	0.972 – 0.971	0.0042 – 0.0063

8.1.6.4. Calculated k_{eff}

Identification	Code	Library	C/E
MIX-MET-FAST-008	MCNP	ENDF/B-V	0.9520 – 0.9877
	MCNP	ENDF/B-V Subgroups + Continuous	0.9634 – 0.9995
MIX-MET-INTER-001	MCNP	ENDF/B-V	0.9658 – 0.9724
	MCNP	ENDF/B-V Subgroups + Continuous	0.9852 – 0.9862

8.1.6.5. Sensitivity of k_{eff} to Graphite: Not Available

8.1.6.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
MIX-MET-FAST-008	0	27.7 – 45.6	54.4 – 72.3
MIX-MET-INTER-001	0 – 0.2	52.9 – 57.5	42.5 – 46.9

8.1.7. Critical Experiment: ***PU-COMP-INTER-001***

8.1.7.1. Description: HECTOR Intermediate Spectrum Study (HISS) – Infinite System of Plutonium and Graphite

Identification	No. of Cases	Percent Absorption In Graphite
PU-COMP-INTER-001	1	0.2

8.1.7.2. Facility and Date: Winfrith (HECTOR), 1970

8.1.7.3. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
PU-COMP-INTER-001	1.0000	0.011

8.1.7.4. Calculated k_{eff}

Identification	Code	Library	C/E
PU-COMP-INTER-001	KENO	ENDF/B-V 238 Group	0.9981
	MCNP	ENDF/B-V	1.0016
	MCNP	ENDF/B-VI Rel. 4	1.0105

8.1.7.5. Sensitivity of k_{eff} to Graphite: Not Available

8.1.7.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
PU-COMP-INTER-001	1	88.6	10.4

8.1.8. Critical Experiment: ***PU-MET-INTER-002***

8.1.8.1. Description: ZPR-6/10 – Cylinder of Plutonium, Carbon, and Stainless Steel

Identification	No. of Cases	Percent Absorption In Graphite
PU-MET-INTER-002	1	0

8.1.8.2. Facility and Date: ZPR-6, 1981

8.1.8.3. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
PU-MET-INTER-002	0.9869	0.0026

8.1.8.4. Calculated k_{eff}

Identification	Code	Library	C/E
PU-MET-INTER-001	KENO	ENDF/B-IV 27 Group	1.0398
	KENO	ENDF/B-V 238 Group	1.0405
	MCNP	ENDF/B-V	1.0171
	MCNP	ENDF/B-VI	1.0381
	MONK	ENDF/B-VI Rel. 3	1.0461
	VIM	ENDF/B-V	0.9993
	VIM	ENDF/B-VI	1.0358

8.1.8.5. Sensitivity of k_{eff} to Graphite: Not Available

8.1.8.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
PU-MET-INTER-002	0.0	67.2	32.7

9. Data Need #9: ZrH_x

9.1. Existing Critical Experiments with Zirconium Hydride

9.1.1. Critical Experiment: ***HEU-COMP-MIXED-003; HEU-COMP-MIXED-004***

9.1.1.1. Description: Highly Enriched Uranium Dioxide with Zirconium Hydride Moderator

Identification	No. of Cases	Reflector	Percent Absorption In Zirconium	Percent Absorption In Hydrogen
HEU-COMP-MIXED-003	6	Beryllium	2.7 – 3.0	4.4 – 6.8
HEU-COMP-MIXED-004	5	Sand & Water	2.6 – 2.7	5.0 – 5.6

9.1.1.2. Facility and Date: RRC Kurchatov Institute, 1993 - 1994

9.1.1.3. Experimental $k_{eff} \pm$ Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
HEU-COMP-MIXED-003	1.0000	Not Available
HEU-COMP-MIXED-004	1.0000	0.0004

9.1.1.4. Calculated k_{eff}

Identification	Code	Library	C/E
HEU-COMP-MIXED-003	MCNP	ENDF/B-V	0.9920 – 0.9971
HEU-COMP-MIXED-004	Not Available	Not Available	Not Available

9.1.1.5. Sensitivity of k_{eff} to ZrH_x: Not Available

9.1.1.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
HEU-COMP-MIXED-003	44.8 – 50.9	36.2 – 40.0	13.0 – 15.2
HEU-COMP-MIXED-004	46.0 – 47.8	37.9 – 39.6	14.2 – 14.5

9.1.2. Critical Experiment: ***IEU-COMP-THERM-003***

9.1.2.1. Description: TRIGA MARK II Reactor: U(20) – Zirconium Hydride Fuel Rods In Water With Graphite Reflector

Identification	No. of Cases	Reflector	Percent Absorption In Zirconium	Percent Absorption In Hydrogen
IEU-COMP-THERM-003	2	Water	Not Available	Not Available

9.1.2.2. Facility and Date: Ljubljana, Slovenia, TRIGA Mark II Reactor, 1991

9.1.2.3. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
IEU-COMP-THERM-003	1.0006 – 1.0046	0.0056

9.1.2.4. Calculated k_{eff}

Identification	Code	Library	C/E
IEU-COMP-THERM-003	KENO	ENDF/B-V 44 Group	1.0022 – 1.0023
	MCNP	ENDF/B-V	0.9988 – 0.9996

9.1.2.5. Sensitivity of k_{eff} to ZrH_x: Not Available

9.1.2.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
IEU-COMP-THERM-003	Not Available	Not Available	Not Available

10. Data Need #10: ^{238}U

10.1. Existing Critical Experiments with ^{238}U

10.1.1. Critical Experiment: ***IEU-MET-FAST-007***

10.1.1.1. Description: Big Ten: Large, Mixed-Uranium-Metal Cylindrical Core With 10% Average ^{235}U Enrichment, Surrounded By A Thick ^{238}U Reflector

Identification	No. of Cases	Reflector	Percent Absorption In ^{238}U
IEU-MET-FAST-007	1	^{238}U	39

10.1.1.2. Facility and Date: LACEF, 1971

10.1.1.3. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
IEU-MET-FAST-007	0.9948	0.0013

10.1.1.4. Calculated k_{eff}

Identification	Code	Library	C/E
IEU-MET-FAST-007	KENO	ENDF/B-IV 27-Group	1.0039
	KENO	ENDF/B-V 238-Group	1.0125
	MCNP	ENDF/B-V	1.0084

10.1.1.5. Sensitivity of k_{eff} to ^{238}U: Not Available

10.1.1.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
IEU-MET-FAST-007	0.0	19.6	80.4

10.1.2. Critical Experiment: ***IEU-MET-FAST-010***

10.1.2.1. Description: The U9 Benchmark Assembly: A Cylindrical Assembly Of U Metal (9% ^{235}U) With A Thick Depleted-Uranium Reflector

Identification	No. of Cases	Reflector	Percent Absorption In ^{238}U
IEU-MET-FAST-010	1	^{238}U	42.7

10.1.2.2. Facility and Date: ZPR-6/9, 1980

10.1.2.3. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
IEU-MET-FAST-010	0.9954	0.0024

10.1.2.4. Calculated k_{eff}

Identification	Code	Library	C/E
IEU-MET-FAST-010	KENO	ENDF/B-IV 27-Group	1.0060
	KENO	ENDF/B-V 238-Group	1.0105
	MCNP	ENDF/B-V	1.0052
	MCNP	ENDF/B-VI	1.0147
	MONK	ENDF/B-VI Rel. 3	1.0123
	VIM	ENDF/B-V	1.0098

10.1.2.5. Sensitivity of k_{eff} to ^{238}U: Not Available

10.1.2.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
IEU-MET-FAST-010	0.0	22.7	77.3

10.1.3. Critical Experiment: ***LEU-COMP-THERM-033 (Cases 1 –4, 23 – 25)***

10.1.3.1. Description: Reflected and Unreflected Assemblies of 2 And 3%-Enriched Uranium Fluoride in Paraffin

Identification	No. of Cases	Reflector	Percent Absorption In ^{238}U
LEU-COMP-THERM-033	7	Unreflected or Paraffin	35.5 – 37.0

10.1.3.2. Facility and Date: ORNL, 1958 - 1968

10.1.3.3. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
LEU-COMP-THERM-033	1.000 – 1.0014	0.0073

10.1.3.4. Calculated k_{eff}

Identification	Code	Library	C/E
LEU-COMP-THERM-033	KENO	ENDF/B-V 44-Group	1.0003 – 1.0034

10.1.3.5. Sensitivity of k_{eff} to ^{238}U: Not Available

10.1.3.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
LEU-COMP-THERM-033	83.7 – 85.1	8.8 – 9.7	6.1 – 6.6

10.1.4. Critical Experiment: ***MIX-MET-FAST-008 and MIX-MET-INTER-001***

10.1.4.1. Description: K-Infinity Experiments In Fast/Intermediate Neutron Spectra For Various Fissile Materials

Identification	**No. of Cases**	**Percent Absorption In ^{238}U**
MIX-MET-FAST-008	5	44.9 – 59.3
MIX-MET-INTER-001	2	52.0 – 53.2

10.1.4.2. Facility and Date: WINFRITH (ZEBRA), 1967 - 1969

10.1.4.3. Experimental k_{eff} ± Uncertainty:

Identification	**Benchmark K_{eff}**	**Benchmark Uncertainty**
MIX-MET-FAST-008	0.973 – 1.03	0.0025 – 0.0069
MIX-MET-INTER-001	0.971 – 0.992	0.0042 – 0.0063

10.1.4.4. Calculated k_{eff}

Identification	**Code**	**Library**	**C/E**
MIX-MET-FAST-008	MCNP	ENDF/B-V	0.9520 – 0.9902
	MCNP	ENDF/B-V Subgroup + Continuous	0.9634 – 0.9995
MIX-MET-INTER-001	MCNP	ENDF/B-V	0.9658 – 0.9724
	MCNP	ENDF/B-V Subgroup + Continuous	0.9862 – 0.9852

10.1.4.5. Sensitivity of k_{eff} to ^{238}U: Not Available

10.1.4.6. Spectral index

Identification	**Percent Fission Below 0.625 eV**	**Percent Fission Between 0.625 eV – 100 keV**	**Percent Fission Above 100 keV**
MIX-MET-FAST-008	0.0	25 – 45.6	54.4 – 75.0
MIX-MET-INTER-001	0.0 – 0.2	52.9 – 57.5	42.5 – 46.9

11. Data Need #11: ^{233}U

11.1. Existing Critical Experiments with ^{233}U

11.1.1. Critical Experiment: ***U233-MET-FAST-001, 002, 003, 004, 005, 006***

11.1.1.1. Description: Metallic ^{233}U Spheres

Identification	No. of Cases	Reflector
U233-MET-FAST-001	1	None
U233-MET-FAST-002	2	^{235}U
U233-MET-FAST-003	2	Natural U
U233-MET-FAST-004	2	Tungsten
U233-MET-FAST-005	2	Beryllium
U233-MET-FAST-006	1	Natural U

11.1.1.2. Facility and Date: LACEF, 1958 - 1964

11.1.1.3. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
U233-MET-FAST-001	1.0000	0.001
U233-MET-FAST-002	1.0000	0.001 – 0.0011
U233-MET-FAST-003	1.0000	0.001
U233-MET-FAST-004	1.0000	0.0007 – 0.0008
U233-MET-FAST-005	1.0000	0.003
U233-MET-FAST-006	1.0000	0.0014

11.1.1.4. Calculated k_{eff}

Identification	Code	Library	C/E
U233-MET-FAST-001	MCNP	ENDF/B-V	0.997
U233-MET-FAST-002	MCNP	ENDF/B-V	0.9969 – 1.0007
U233-MET-FAST-003	MCNP	ENDF/B-V	0.9956 – 1.0003
U233-MET-FAST-004	MCNP	ENDF/B-V	1.0049 – 1.0076
U233-MET-FAST-005	MCNP	ENDF/B-V	0.9948 – 0.9971
U233-MET-FAST-006	MCNP	ENDF/B-V	1.0023

11.1.1.5. Sensitivity of k_{eff} to ^{233}U: Not Available

11.1.1.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
U233-MET-FAST-001	0	3.1	96.9
U233-MET-FAST-002	0	3.4 – 3.7	96.3 – 96.6
U233-MET-FAST-003	0	3.6 – 4.3	95.7– 96.4
U233-MET-FAST-004	0	3.7 – 4.7	95.4 – 96.3
U233-MET-FAST-005	0	5.7 – 9.7	90.3 – 94.3
U233-MET-FAST-006	0	6.3	93.7

11.1.2. Critical Experiment: ***U233-SOL-INTER-001, U233-SOL-MIXED-001, U233-SOL-THERM-011, U233-SOL-MIXED-002, U233-SOL-THERM-015***

11.1.2.1. Description: Uranyl-Fluoride (^{233}U) Solutions In Spherical Stainless Steel Vessels With Reflectors Of Be, Ch_2 And Be-Ch_2 Composites

Identification	No. of Cases	Uranium Concentration (g/l)
U233-SOL-INTER-001	29	567.2 – 866.0
U233-SOL-MIXED-001	3	567.2 – 749.0
U233-SOL-MIXED-002	5	446.9
U233-SOL-THERM-011	1	567.2
U233-SOL-THERM-015	26	124.4 – 446.9

11.1.2.2. Facility and Date: LLNL, Late 1950s

11.1.2.3. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
U233-SOL-INTER-001	1.0000	0.0046 – 0.0098
U233-SOL-MIXED-001	1.0000	0.0028 – 0.0053
U233-SOL-MIXED-002	1.0000	0.0050 – 0.0099
U233-SOL-THERM-011	1.0000	0.0061
U233-SOL-THERM-015	1.0000	0.0023 – 0.0079

11.1.2.4. Calculated k_{eff}

Identification	Code	Library	C/E
U233-SOL-INTER-001	MCNP	ENDF/B-VI	0.9757 – 0.9953
	VIM	ENDF/B-V	0.9813 – 1.0029
	KENO	ENDF/B-IV 27-Group	0.9992 – 1.0208
U233-SOL-MIXED-001	MCNP	ENDF/B-VI	0.9782 – 0.9932
	VIM	ENDF/B-V	0.9811 –0.9984
	KENO	ENDF/B-IV 27-Group	0.9994 – 1.0146
U233-SOL-MIXED-002	MCNP	ENDF/B-VI	0.9700 – 0.9887
	COG	ENDF/B-V	0.9686 – 0.9877
	KENO	ENDF/B-V 238-Group	0.9669 – 0.9870
U233-SOL-THERM-011	MCNP	ENDF/B-VI	0.9858
	VIM	ENDF/B-V	0.9934
	KENO	ENDF/B-IV 27-Group	1.0109
U233-SOL-THERM-015	MCNP	ENDF/B-VI	0.9754 - 9985
	COG	ENDF/B-VI	0.9748 – 0.9988
	KENO	ENDF/B-V 238-Group	0.9726 – 0.9987

11.1.2.5. Sensitivity of k_{eff} to ^{233}U: Not Available

11.1.2.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
U233-SOL-INTER-001	28.0 – 45.0	50.3 – 64.4	4.1 – 7.7
U233-SOL-MIXED-001	45.5 – 49.8	45.6 – 49.2	4.3 – 5.3
U233-SOL-MIXED-002	48.3 – 49.4	47.1 – 47.7	3.5 – 4.0
U233-SOL-THERM-011	51.8	44.2	4.1
U233-SOL-THERM-015	50.0 – 79.2	19.8 – 46.6	1.0 –3.7

Critical Experiment: U233-SOL-THERM-001, U233-SOL-THERM-008

11.1.2.7. Description: Unreflected Spheres of ^{233}U Nitrate Solution

Identification	No. of Cases	Uranium Concentration (g/l)
U233-SOL-THERM-001	5	17.1 – 19.8
U233-SOL-THERM-008	1	13.2

11.1.2.8. Facility and Date: ORNL, 1950 - 1960

11.1.2.9. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
U233-SOL-THERM-001	0.9998 – 1.0006	0.0031 – 0.0033
U233-SOL-THERM-008	1.0006	0.0029

11.1.2.10. Calculated k_{eff}

Identification	Code	Library	C/E
U233-SOL-THERM-001	MCNP	ENDF/B-V	1.0000 – 1.0013
	KENO	ENDF/B-IV 27-Group	1.0013 – 1.0024
	TWODANT	ENDF/B-IV 27-Group	1.0009 – 1.0020
U233-SOL-THERM-008	MCNP	ENDF/B-VI	0.9979
	KENO	ENDF/B-IV 27-Group	0.9976
	ONEDANT	ENDF/B-IV 27-Group	0.9976

11.1.2.11. Sensitivity of k_{eff} to ^{233}U: Not Available

11.1.2.12. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
U233-SOL-THERM-001	94.0 – 94.8	5.0 – 5.8	0.2
U233-SOL-THERM-008	95.5	4.3	0.2

11.1.3. Critical Experiment: ***U233-SOL-THERM-002, 003, 004, 005, 012, 13***

11.1.3.1. Description: Reflected Spheres of ^{233}U Nitrate Solution

Identification	No. of Cases	Uranium Concentration (g/l)	Reflector
U233-SOL-THERM-002	17	34 – 385	Paraffin
U233-SOL-THERM-003	10	33.6 – 609.4	Paraffin
U233-SOL-THERM-004	8	169.1 – 385.8	Paraffin
U233-SOL-THERM-005	2	50.0 – 63.3	Water
U233-SOL-THERM-012	8	47.8 – 200.2	Water
U233-SOL-THERM-013	21	44.6 – 203.8	None

11.1.3.2. Facility and Date: ORNL, 1952 – 1953, and 1966 - 1968

11.1.3.3. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
U233-SOL-THERM-002	1.0004	0.0087
U233-SOL-THERM-003	0.9991 – 1.0018	0.0087 – 0.0151
U233-SOL-THERM-004	1.0020 – 1.0051	0.0086 – 0.0105
U233-SOL-THERM-005	1.0060 – 1.0072	0.0020 – 0.0026
U233-SOL-THERM-012	0.9987 – 1.0000	0.0011 – 0.0071
U233-SOL-THERM-013	0.9992 – 0.9996	0.0020 – 0.0077

11.1.3.4. Calculated k_{eff}

Identification	Code	Library	C/E
U233-SOL-THERM-002	MCNP	ENDF/B-V	0.9856 – 1.0157
	KENO	ENDF/B-V 238 Group	0.9750 – 1.0072
	KENO	ENDF/B-IV 27-Group	0.9939 – 1.0239
U233-SOL-THERM-003	VIM	ENDF/B-VI	1.027 – 1.0217
	MONK	ENDF/B-VI Rel. 3	1.0014 – 1.0206
	MCNP	ENDF/B-V	1.0040 – 1.0282
	KENO	ENDF/B-V 238-Group	0.9932 – 1.0163
	KENO	ENDF/B-IV 27-Group	1.0143 –1.0384
U233-SOL-THERM-004	MCNP	ENDF/B-V	0.9853 – 1.0086
	KENO	ENDF/B-V 238-Group	0.9765 – 0.9997
	KENO	ENDF/B-IV 27-Group	1.0111 – 1.0267
U233-SOL-THERM-005	VIM	ENDF/B-VI	0.9944 – 0.9955
	MONK	ENDF/B-VI Rel. 3	0.9927 – 0.9941
	MCNP	ENDF/B-V	0.9988
	VIM	ENDF/B-V	0.9991 – 1.0017
	KENO	ENDF/B-IV 27-Group	1.0047 – 1.0072
U233-SOL-THERM-012	MCNP	ENDF/B-V	1.0010 – 1.0116
	KENO	ENDF/B-V 238-Group	0.9938 – 1.0033
U233-SOL-THERM-013	MCNP	ENDF/B-V	0.9968 – 1.0250
	MCNP	ENDF/B-VI Rel. 6	0.9929 – 1.0203
	MCNP	ENDF/B-VI Rel. 6 Prob.	0.9921 – 1.0202
	KENO	ENDF/B-V 238-Group	0.9895 – 1.0170

11.1.3.5. Sensitivity of k_{eff} to ^{233}U: Not Available

11.1.3.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
U233-SOL-THERM-002	63.7 – 92.5	5.8 – 33.5	0.3 – 2.8
U233-SOL-THERM-003	54.5 – 92.7	7.0 – 41.5	0.3 – 4.0
U233-SOL-THERM-004	63.8 – 79.1	19.3 – 33.4	1.2 – 2.8
U233-SOL-THERM-005	88.9 – 90.5	9.0 – 10.6	0.4 – 0.5
U233-SOL-THERM-012	75.8 – 90.8	8.8 – 22.7	0.4 – 1.5
U233-SOL-THERM-013	Not Available	Not Available	Not Available

11.1.4. Critical Experiment: ***U233-SOL-THERM-006***

11.1.4.1. Description: Arrays of Cylinders Containing of ^{233}U Nitrate Solution

Identification	No. of Cases	Uranium Concentration (g/l)
U233-SOL-THERM-006	17	203.8 – 342.8

11.1.4.2. Facility and Date: ORNL, 1966 - 1968

11.1.4.3. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
U233-SOL-THERM-006	1.0000	0.0028 – 0.0035

11.1.4.4. Calculated k_{eff}

Identification	Code	Library	C/E
U233-SOL-THERM-006	KENO	ENDF/B-V 238 Group	0.9895 – 1.0020

11.1.4.5. Sensitivity of k_{eff} to ^{233}U: Not Available

11.1.4.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
U233-SOL-THERM-006	55.0 – 75.0	23.4 – 41.7	1.6 – 3.3

11.1.5. Critical Experiment: ***U233-SOL-THERM-014***

11.1.5.1. Description: Arrays of Cylinders Containing of ^{233}U Nitrate Solution

Identification	No. of Cases	Uranium Concentration (g/l)
U233-SOL-THERM-014	16	312.6– 332.5

11.1.5.2. Facility and Date: PNL, 1966

11.1.5.3. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
U233-SOL-THERM-014	0.9965 – 1.0000	0.0054 – 0.0126

11.1.5.4. Calculated k_{eff}

Identification	Code	Library	C/E
U233-SOL-THERM-014	KENO	ENDF/B-V 238 Group	0.9782 – 1.0108
	MCNP	ENDF/B-V	0.9859 – 1.0197
	MCNP	ENDF/B-VI Rel. 2	0.9805 – 1.0159

11.1.5.5. Sensitivity of k_{eff} to ^{233}U: Not Available

11.1.5.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
U233-SOL-THERM-014	58.9 – 72.5	25.5 – 38.1	2.0 – 2.9

12. Data Need #12: ^{232}Th

12.1. Existing Critical Experiments with ^{232}Th

12.1.1. Critical Experiment: ***HEU-MET-FAST-068, HEU-MET-INTER-008, IEU-COMP-FAST-002, IEU-COMP-INTER-001, IEU-COMP-THERM-005***

12.1.1.1. Description: Assemblies of High and Intermediate Enriched Uranium with Thorium and Polyethylene

Identification	No. of Cases	Name	Percent Absorption in Thorium
HEU-MET-FAST-068	1	KBR-22	Not Available
HEU-MET-INTER-008	1	KBR-23	Not Available
IEU-COMP-FAST-002	1	KBR-18 (k_∞)	48.4
IEU-COMP-INTER-001	2	KBR-19 & 20 (k_∞)	40.7 – 45.8
IEU-COMP-THERM-005	1	KBR-21 (k_∞)	39.6

12.1.1.2. Facility and Date: IPPE (KBR), 1990 - 1997

12.1.1.3. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
HEU-MET-FAST-068	1.0001	0.0041
HEU-MET-INTER-008	1.0008	0.0036
IEU-COMP-FAST-002	0.969	0.005
IEU-COMP-INTER-001	0.98 – 1.014	0.003 – 0.006
IEU-COMP-THERM-005	0.964	0.012

12.1.1.4. Calculated k_{eff}

Identification	Code	Library	C/E
HEU-MET-FAST-068	MCNP	ENDF/B-V	1.0063
	MCNP	ENDF/B-VI	1.0017
	MCNP	ENDF/B-VI Rel. 4	1.0058
HEU-MET-INTER-008	MCNP	ENDF/B-V	1.0112
	MCNP	ENDF/B-VI	1.0086
	MCNP	ENDF/B-VI Rel. 4	1.0107
IEU-COMP-FAST-002	MCNP	ENDF/B-V	1.0213
	MCNP	ENDF/B-VI Rel. 2	1.0186
	MCNP	ENDF/B-VI Rel. 4	1.0263
IEU-COMP-INTER-001	MCNP	ENDF/B-V	1.0018 – 1.0021
	MCNP	ENDF/B-VI Rel. 2	0.9964 – 0.9982
	MCNP	ENDF/B-VI Rel. 4	0.9959 – 1.0118
IEU-COMP-THERM-005	MCNP	ENDF/B-V	0.9619
	MCNP	ENDF/B-VI Rel. 2	0.9582
	MCNP	ENDF/B-VI Rel. 4	0.9599

12.1.1.5. Sensitivity of k_{eff} to ^{232}Th: Not Available

12.1.1.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
HEU-MET-FAST-068	Not Available	Not Available	Not Available
HEU-MET-INTER-008	Not Available	Not Available	Not Available
IEU-COMP-FAST-002	0	36.6	63.4
IEU-COMP-INTER-001	0.2 – 21.4	56.5 – 63.6	15.0 – 43.2
IEU-COMP-THERM-005	68.4	25.7	6.0

12.1.2. Critical Experiment: ***LEU-COMP-THERM-060***

12.1.2.1. Description: Graphite Moderated and Reflected RBMK

Identification	No. of Cases	Percent Absorption in Thorium
LEU-COMP-THERM-060	10	Not Available

12.1.2.2. Facility and Date: RRC KI, 1982 – 1988

12.1.2.3. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
LEU-COMP-THERM-060	0.9991 – 1.0042	0.0029 – 0.0030

12.1.2.4. Calculated k_{eff}

Identification	Code	Library	C/E
LEU-COMP-THERM-060	MCNP	ENDF/B-V	0.991 – 0.994

12.1.2.5. Sensitivity of k_{eff} to ^{232}Th: Not Available

12.1.2.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
LEU-COMP-THERM-060	Not Available	Not Available	Not Available

12.1.3. Critical Experiment: ***U233-COMP-THERM-001 & HEU-COMP-THERM-015***

12.1.3.1. Description: LWBR SB Core Experiments

Identification	No. of Cases	Percent Absorption in Thorium
U233-COMP-THERM-001	5	Not Available
HEU-COMP-THERM-015	3	Not Available

12.1.3.2. Facility and Date: Bettis Atomic Power Laboratory (BAPL), 1960s

12.1.3.3. Experimental k_{eff} ± Uncertainty:

Identification	Benchmark K_{eff}	Benchmark Uncertainty
U233-COMP-THERM-001	0.9995 – 1.0015	0.0024 – 0.0028
HEU-COMP-THERM-015	1.0006 – 1.0015	0.0026 – 0.0028

12.1.3.4. Calculated k_{eff}

Identification	Code	Library	C/E
U233-COMP-THERM-001	MCNP	ENDF/B-V	0.996 – 1.001
	KENO	238 Group ENDF/B-V	0.988 – 1.002
HEU-COMP-THERM-015	MCNP	ENDF/B-V	0.996 – 0.997
	KENO	238 Group ENDF/B-V	0.983 – 1.004

12.1.3.5. Sensitivity of k_{eff} to ^{232}Th: Not Available

12.1.3.6. Spectral index

Identification	Percent Fission Below 0.625 eV	Percent Fission Between 0.625 eV – 100 keV	Percent Fission Above 100 keV
U233-COMP-THERM-001	Not Available	Not Available	Not Available
HEU-COMP-THERM-015	Not Available	Not Available	Not Available

APPENDIX A

SUMMARY OF GADOLINIUM BENCHMARK DATA

Soluble Gadolinium Benchmark Data

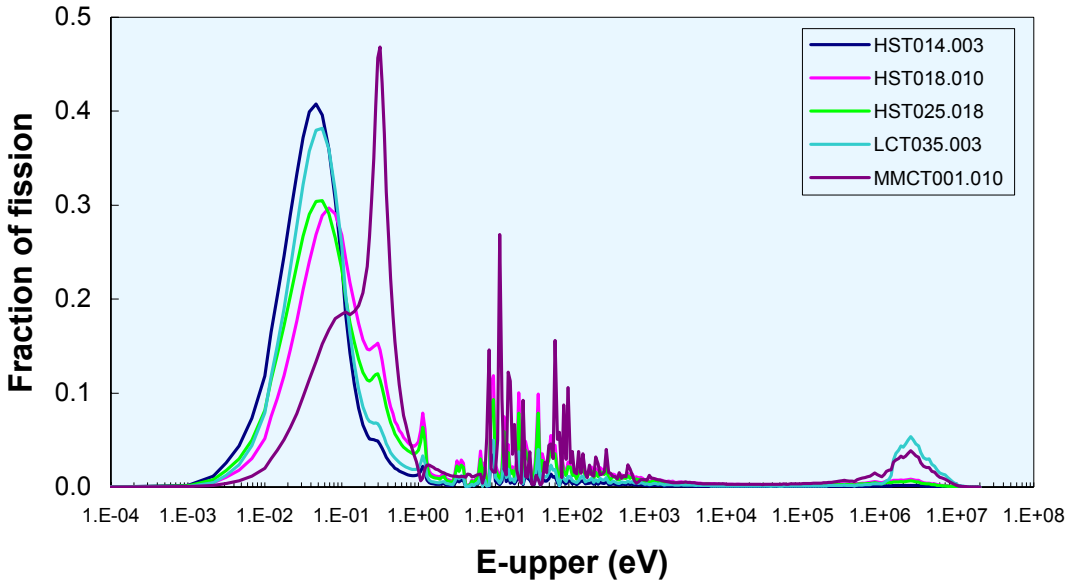


Figure A-1. Summary of Soluble Gadolinium Benchmark Fission Distribution Data

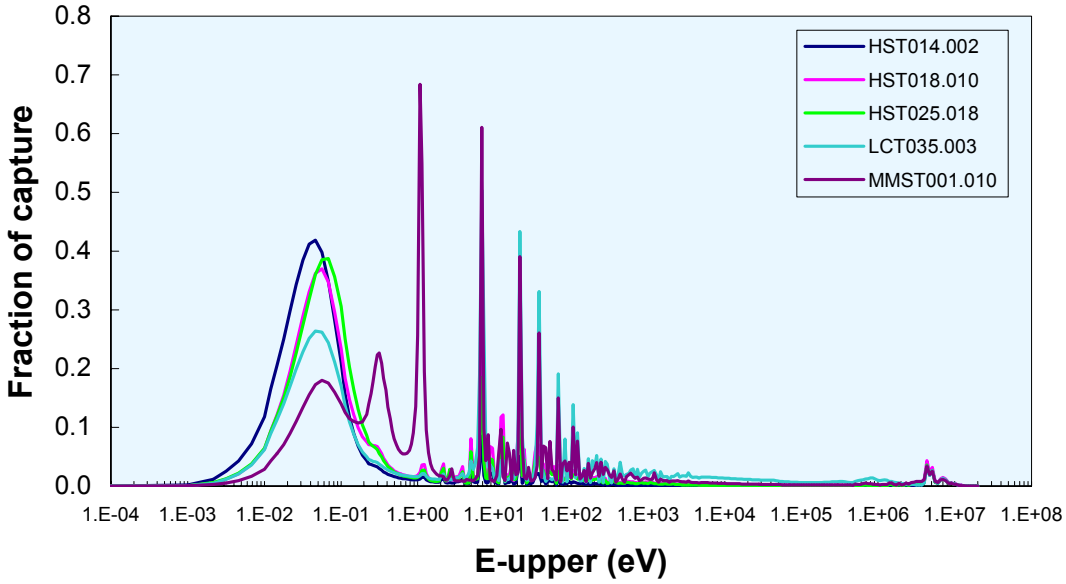


Figure A-2. Summary of Soluble Gadolinium Benchmark Capture Distribution Data

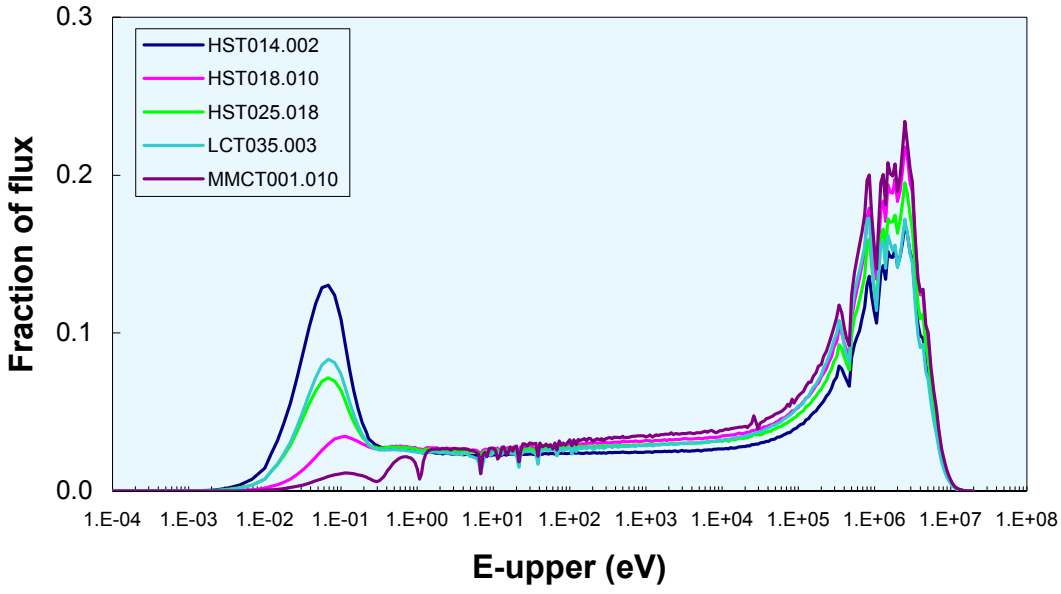


Figure A-3. Summary of Soluble Gadolinium Benchmark Flux Distribution Data

Fixed Gadolinium Benchmark Data

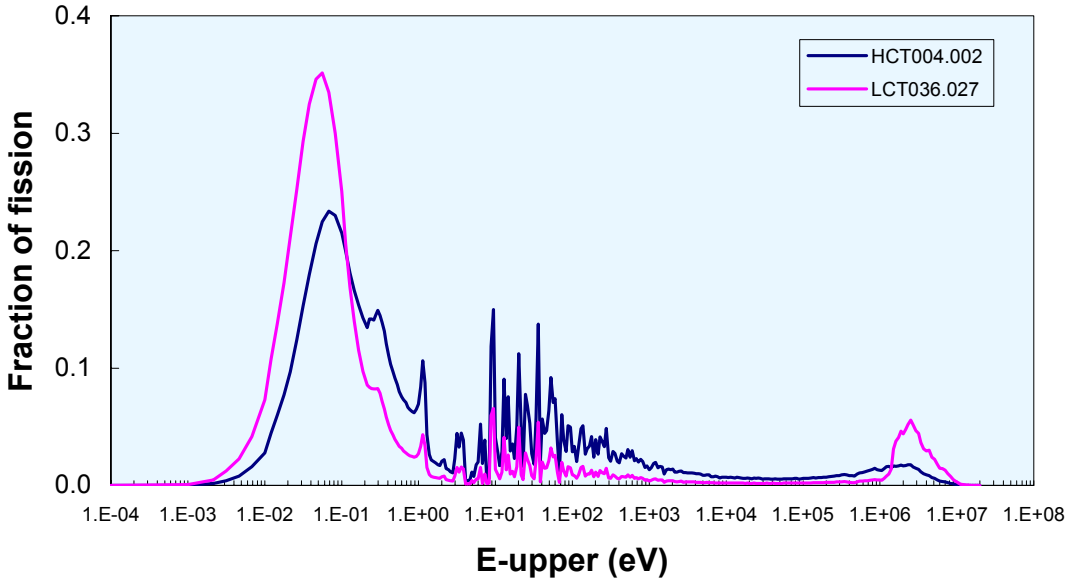


Figure A-4. Summary of Fixed Gadolinium Benchmark Fission Distribution Data

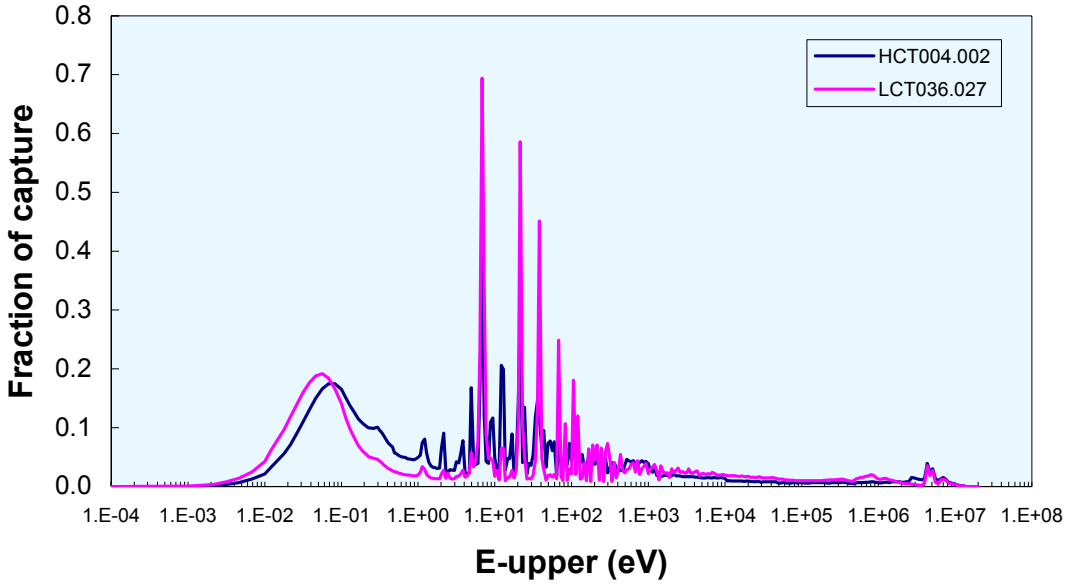


Figure A-5. Summary of Fixed Gadolinium Benchmark Capture Distribution Data

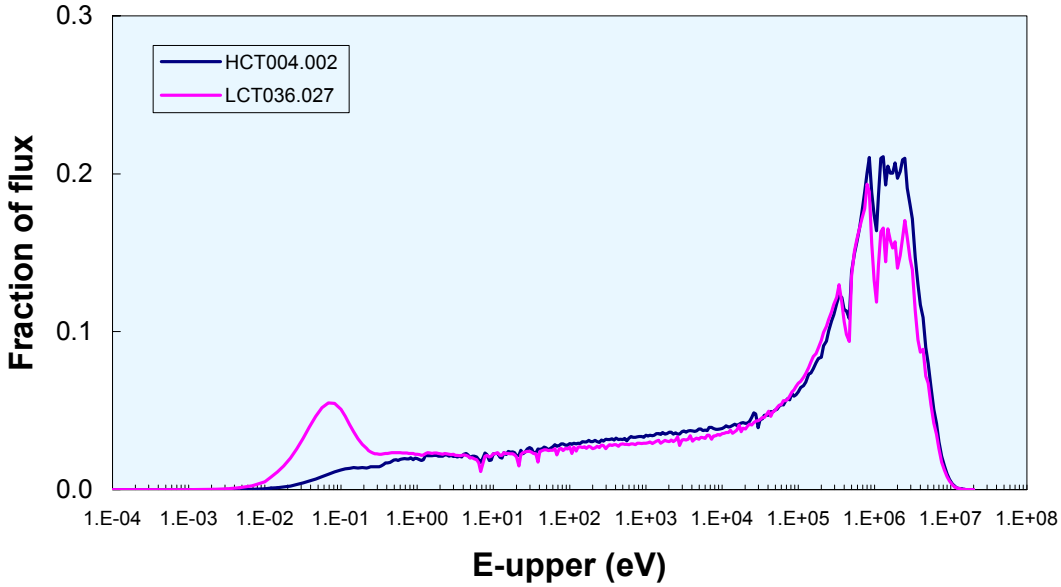


Figure A-6. Summary of Fixed Gadolinium Benchmark Flux Distribution Data

APPENDIX B

SUMMARY OF BERYLLIUM BENCHMARK DATA

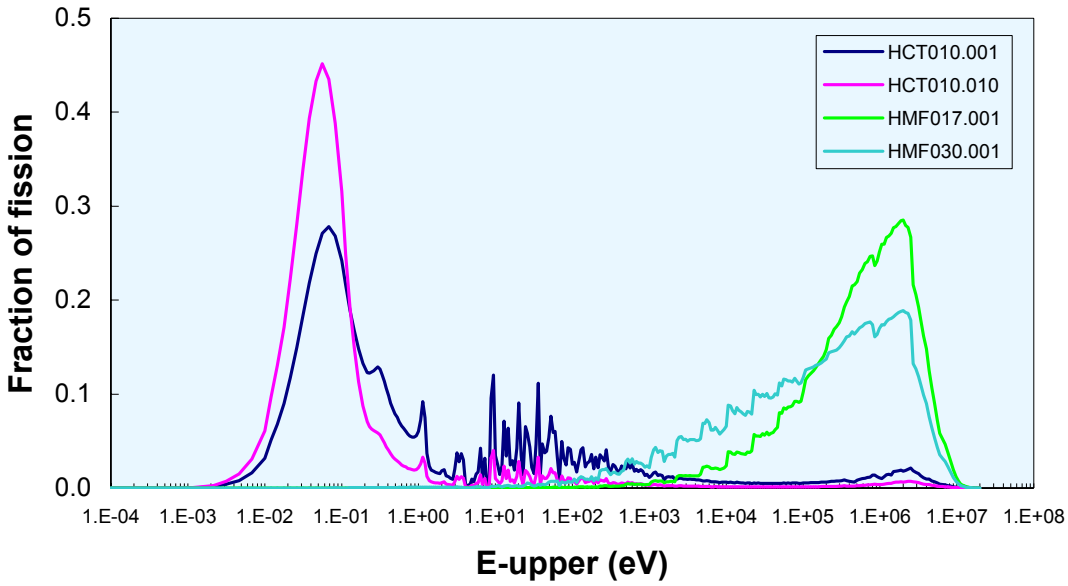


Figure B-1. Summary of Beryllium Benchmark Fission Distribution Data

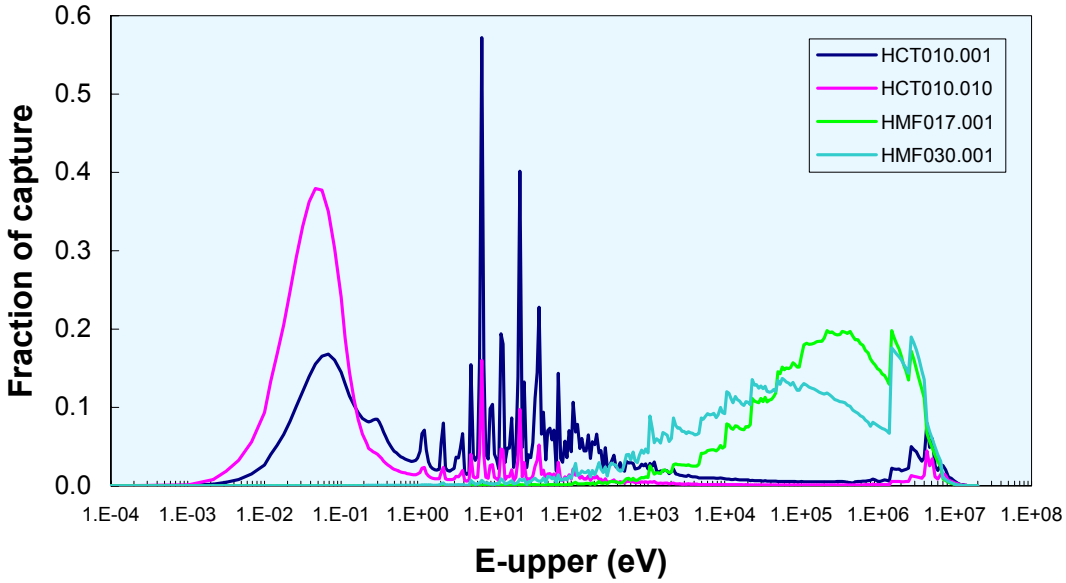


Figure B-2. Summary of Beryllium Benchmark Capture Distribution Data

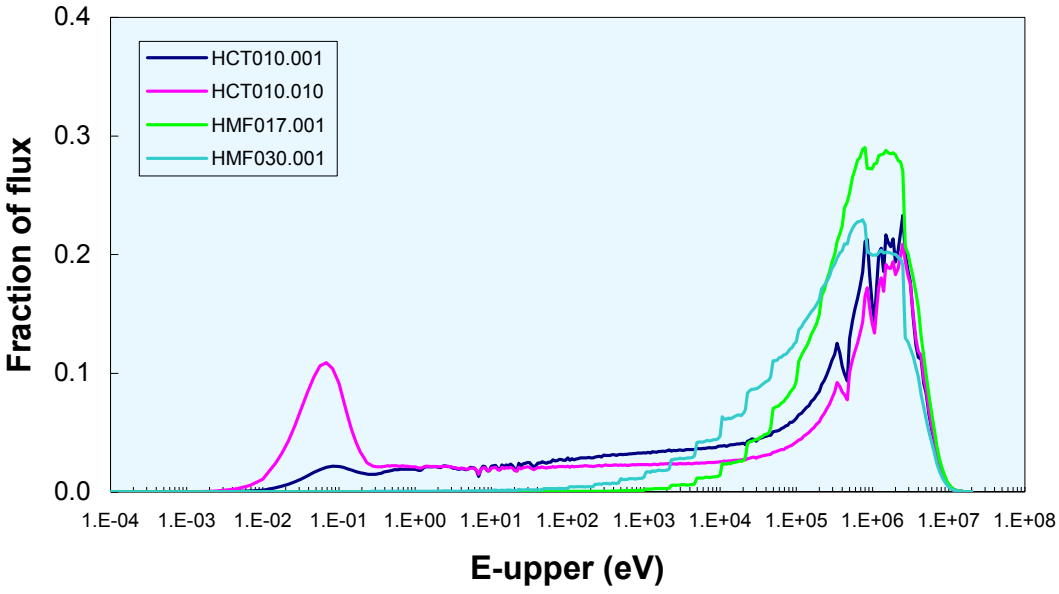


Figure B-3. Summary of Beryllium Benchmark Flux Distribution Data

APPENDIX C

SUMMARY OF STRUCTURAL MATERIAL BENCHMARK DATA

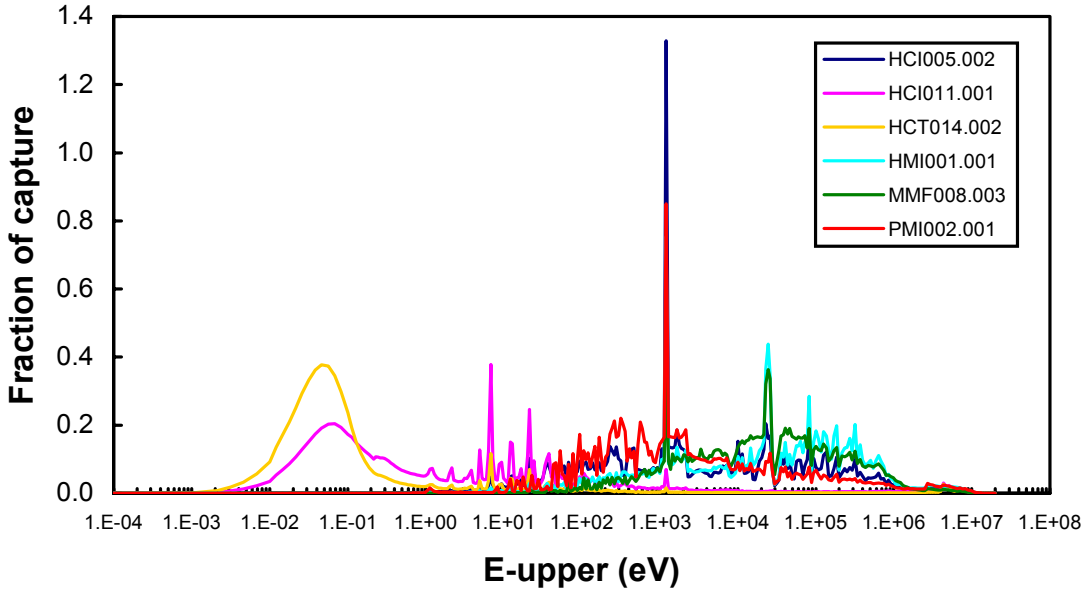


Figure C-1. Summary of Iron Benchmark Capture Distribution Data (Cases > 5% capture in iron)

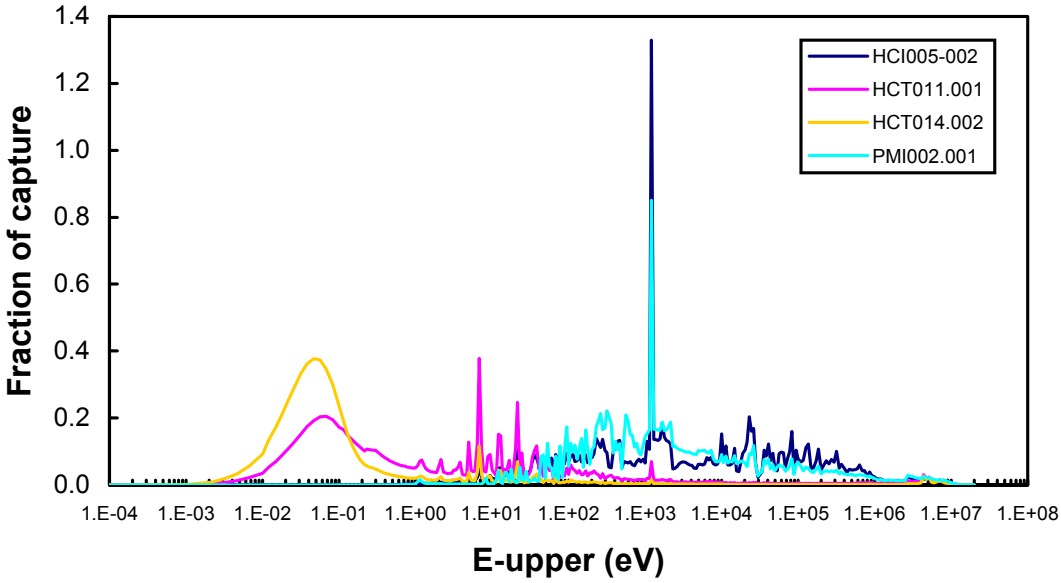


Figure C-2. Summary of Chromium Benchmark Capture Distribution Data (Cases with > 3% capture in chromium)

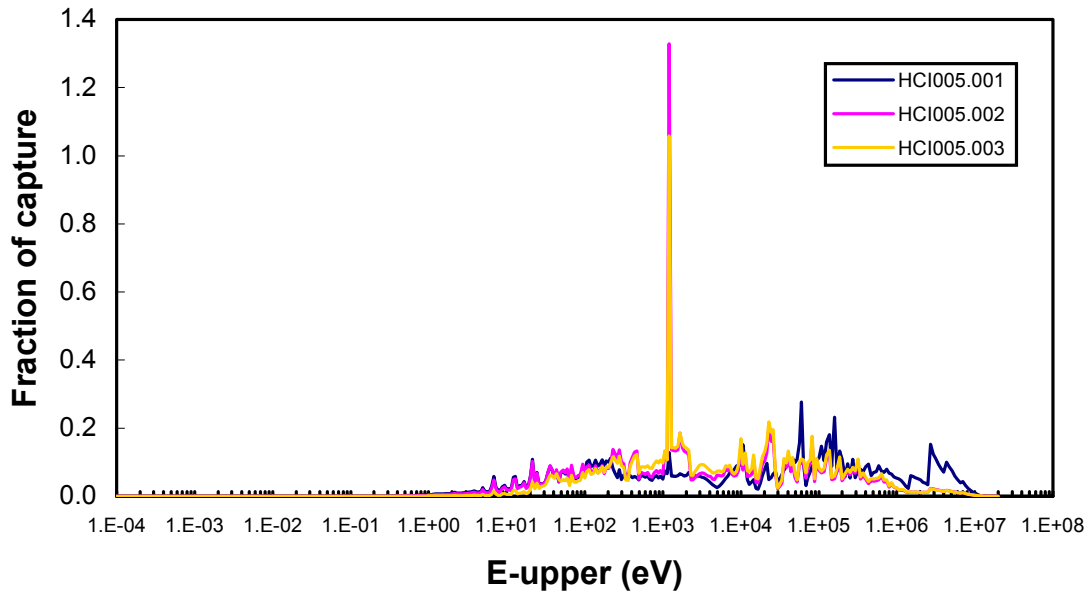


Figure C-3. Summary of Nickel Benchmark Capture Distribution Data (Cases with > 3% capture in chromium)

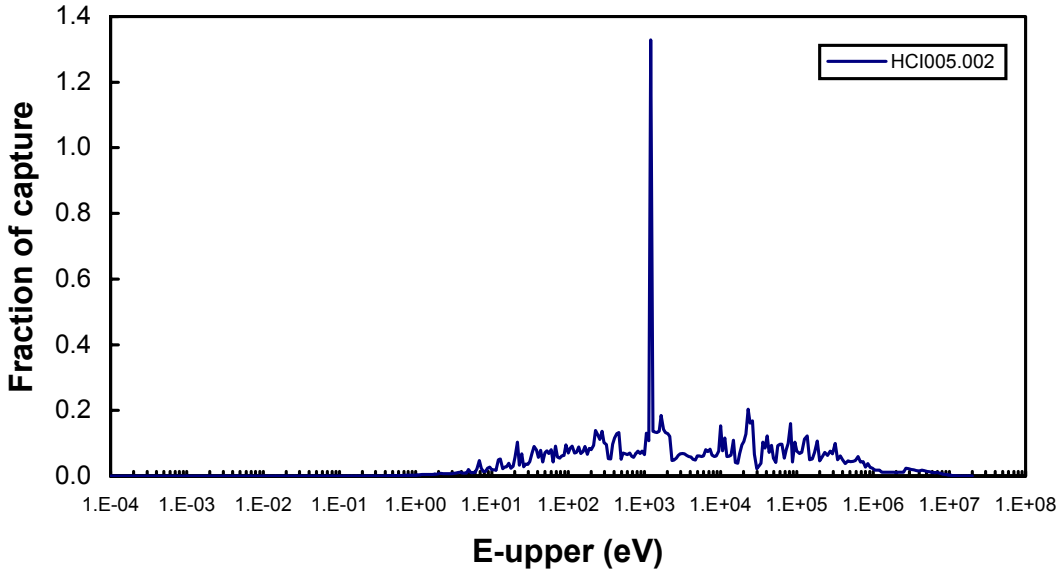


Figure C-4. Summary of Manganese Benchmark Capture Distribution Data

APPENDIX D

SUMMARY OF GRAPHITE BENCHMARK DATA

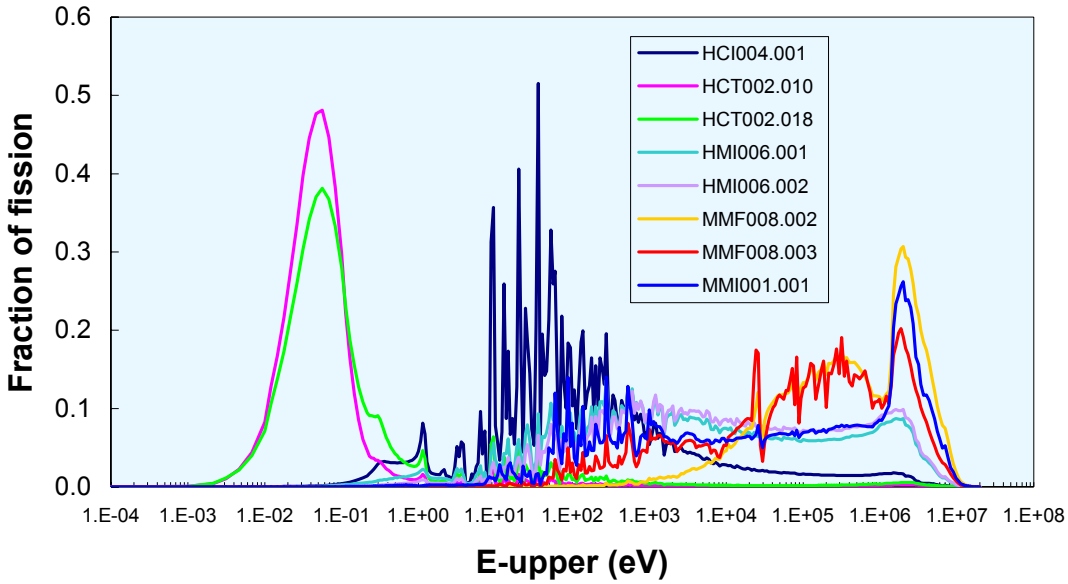


Figure D-1. Summary of Graphite Benchmark Fission Distribution Data

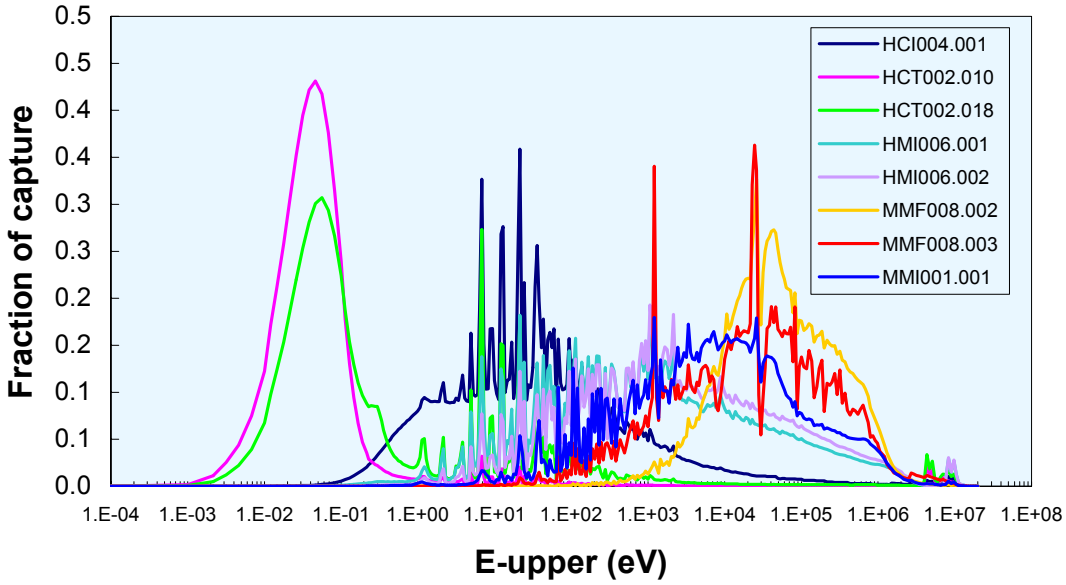


Figure D-2. Summary of Graphite Benchmark Capture Distribution Data

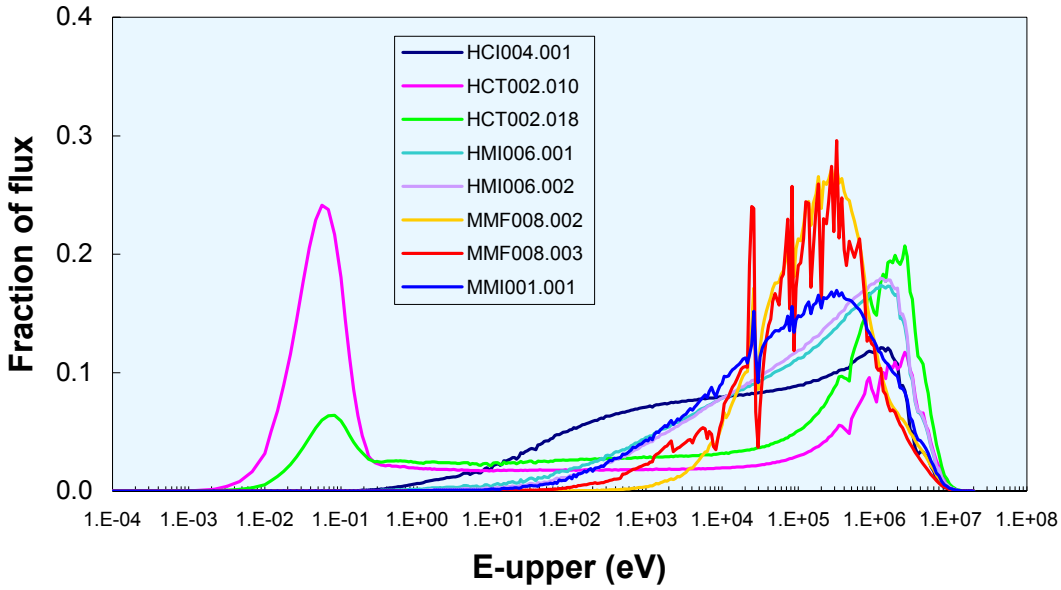


Figure D-3. Summary of Graphite Material Benchmark Flux Distribution Data

APPENDIX E

SUMMARY OF ZrH_x BENCHMARK DATA

Data Not Yet Available in DICE

Figure E-1. Summary of ZrH_x Benchmark Fission Distribution Data

Data Not Yet Available in DICE

Figure E-2. Summary of ZrH_x Benchmark Capture Distribution Data

Data Not Yet Available in DICE

Figure E-3. Summary of ZrH_x Benchmark Flux Distribution Data

APPENDIX F

SUMMARY OF ^{238}U BENCHMARK DATA

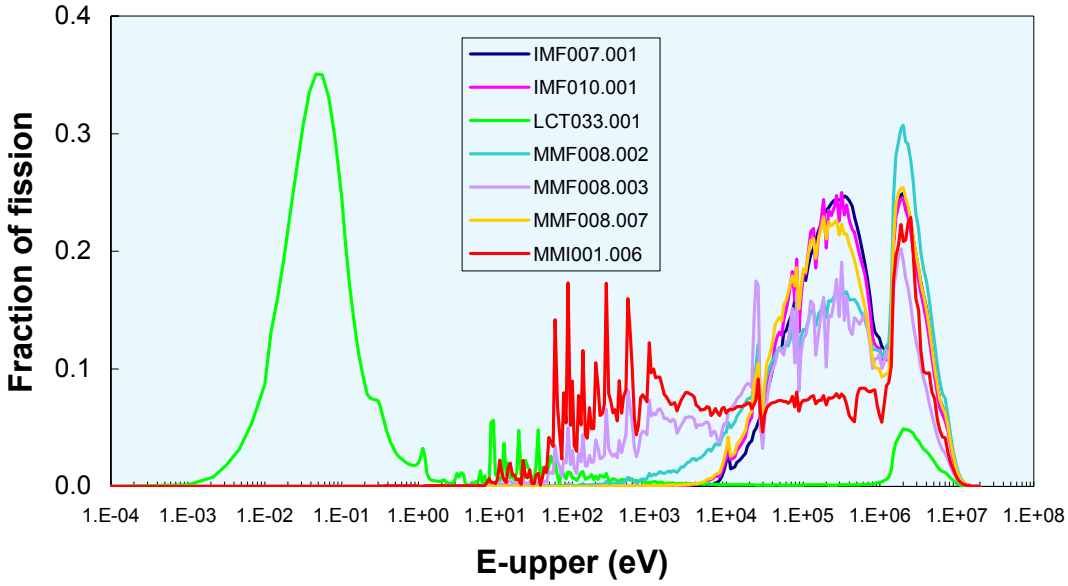


Figure F-1. Summary of ^{238}U Benchmark Fission Distribution Data

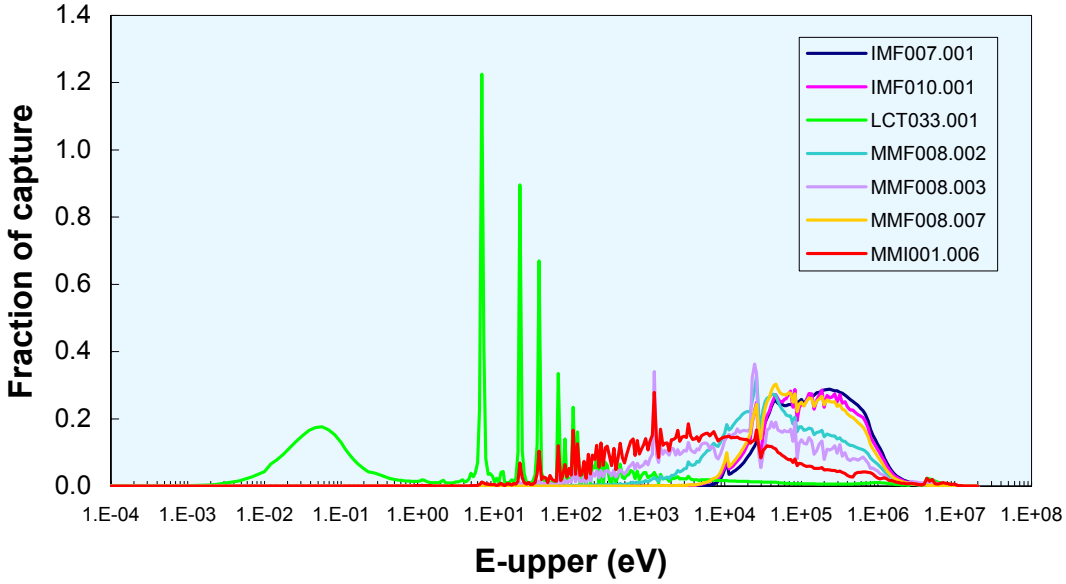


Figure F-2. Summary of ^{238}U Benchmark Capture Distribution Data

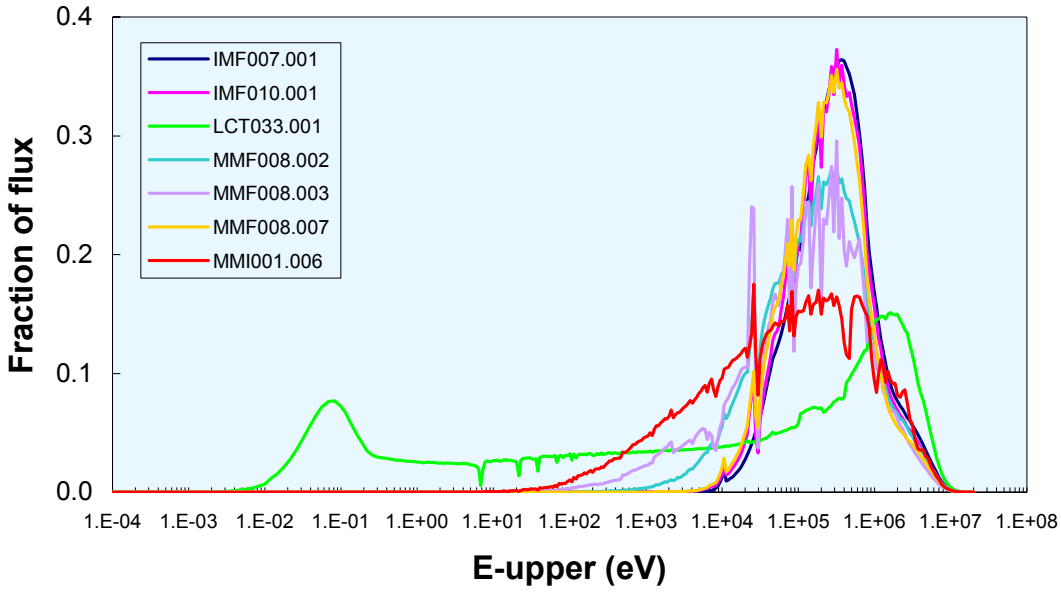


Figure F-3. Summary of ^{238}U Benchmark Flux Distribution Data

APPENDIX G

SUMMARY OF ^{233}U BENCHMARK DATA

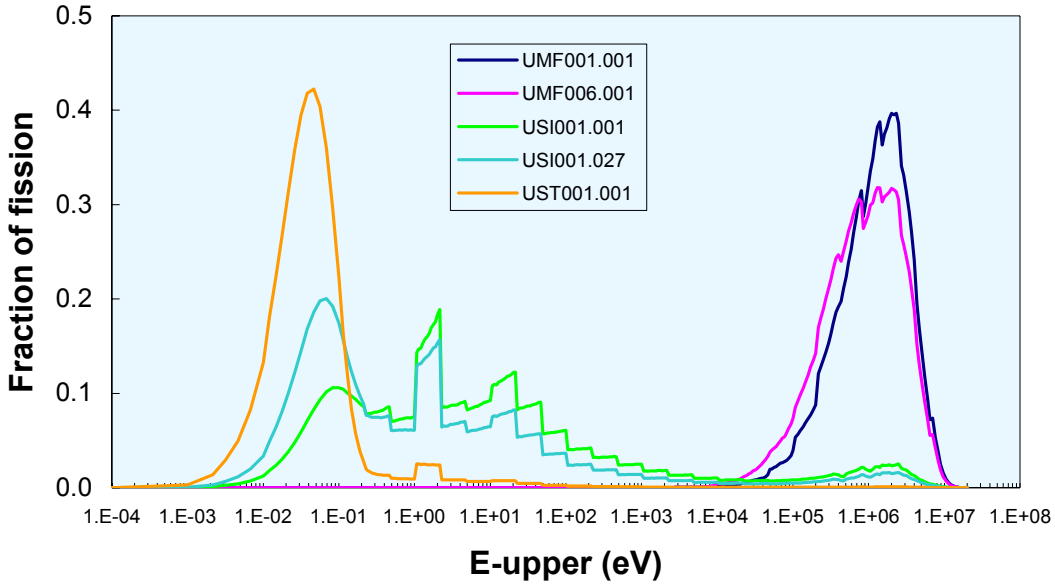


Figure G-1. Summary of ^{233}U Benchmark Fission Distribution Data

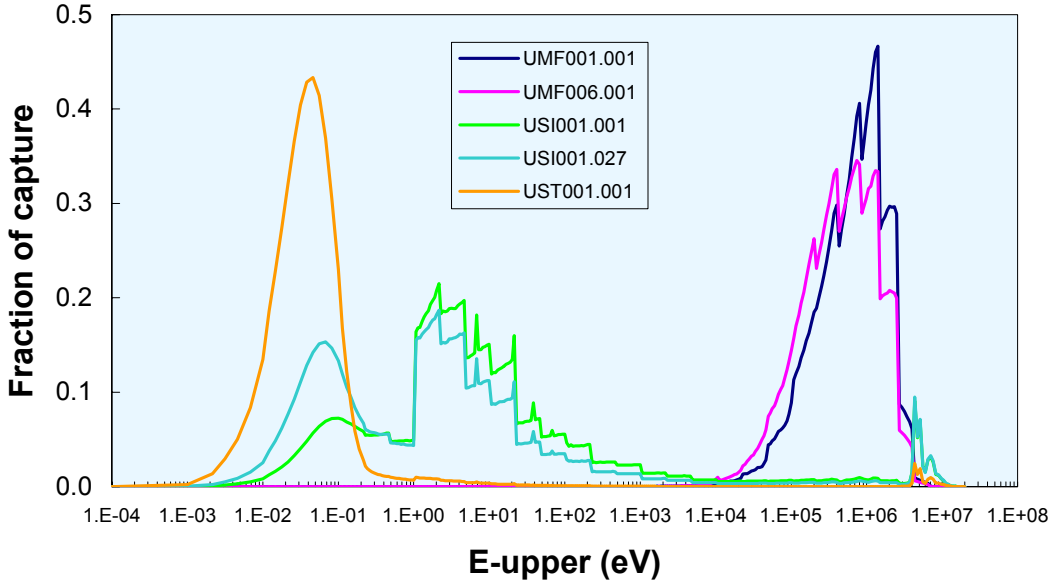


Figure G-2. Summary of ^{233}U Benchmark Capture Distribution Data

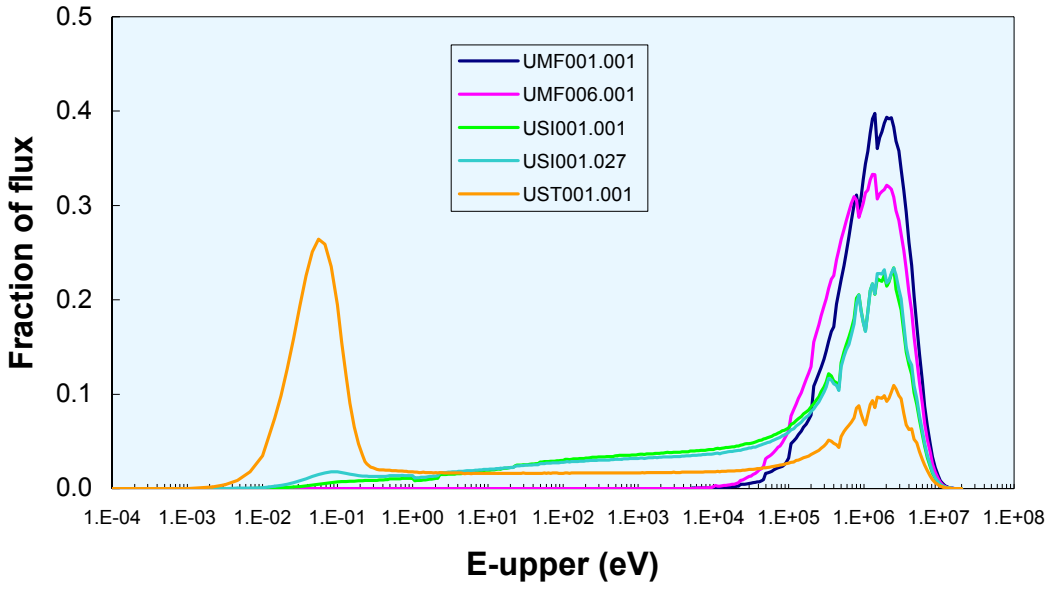


Figure G-3. Summary of ^{233}U Benchmark Flux Distribution Data

UT-BATTELLE

AROBCAD Activities having Potential Interest to NDAG

3 November 2003

CM Hopper

Brookhaven National Laboratory

S/U Analyses

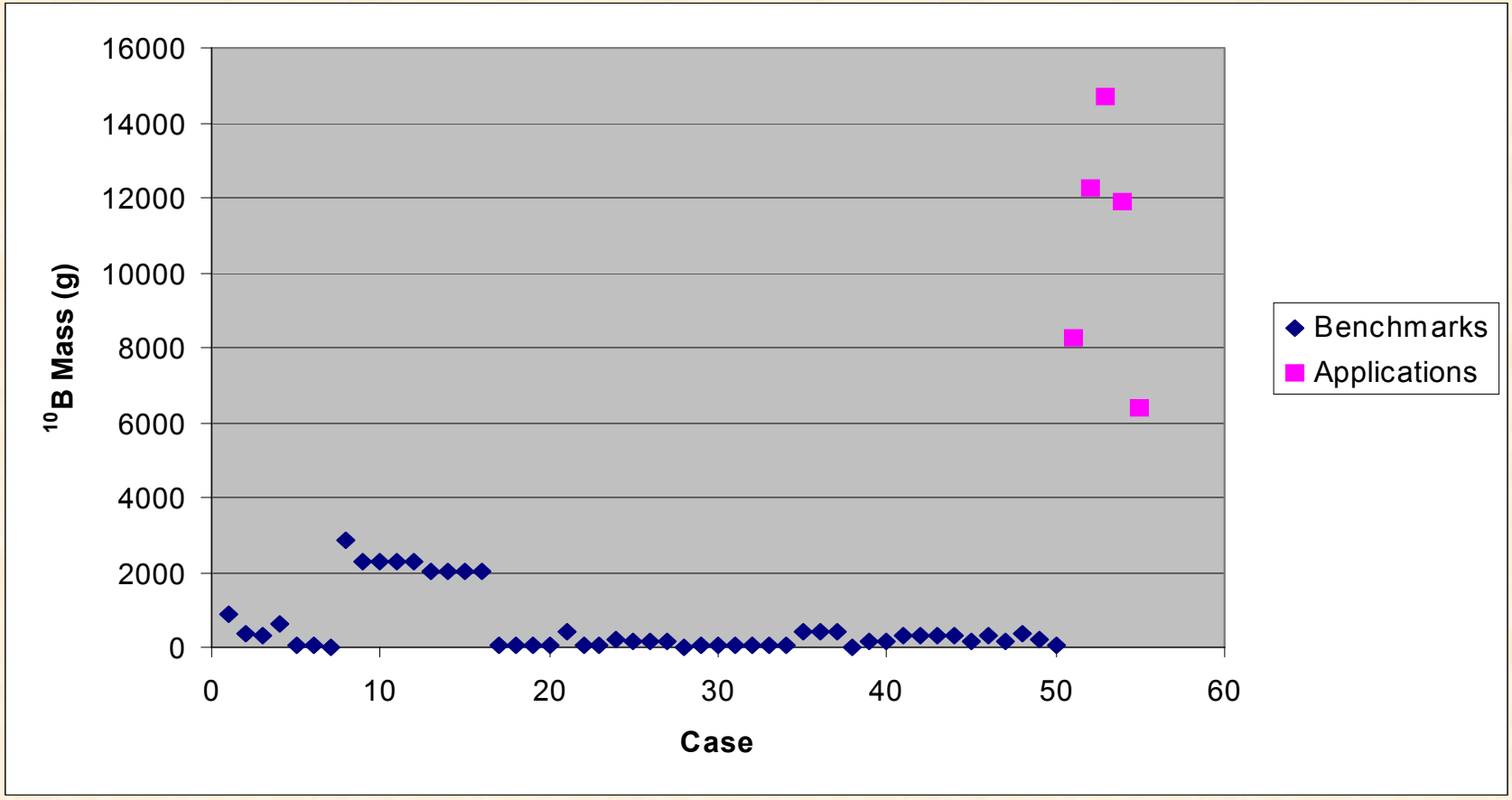
- Related NRC funded work for ^{10}Boron use in shipping casks
- Dilution experiments (C & Fe) for waste exemptions (DOE & NRC issues)
- NASA/DOE Space Fission Power Systems
- Storage of ^{237}Np at Oak Ridge Y-12 Plant

Related NRC funded work for ^{10}Boron use in shipping casks

1. Holtec HI-STAR MPC-24 cask, which holds 24 fresh PWR assemblies. It contains 0.055-in-thick Boral plates. The model used contained 4.2% enriched fuel.
2. Holtec HI-STAR MPC-68 cask, which holds 68 fresh BWR assemblies. It contains 0.08-in-thick Boral plates. The model used contained 4.2% enriched fuel.
3. General Atomic GA-4 cask, which holds 4 PWR assemblies. The boron is present as B4C pellets in horizontal tubes inside a cruciform stainless steel fuel support structure. The model used contained 3% enriched fresh fuel.
4. GBC-32 cask, which is a computational benchmark cask and holds 32 PWR assemblies. The model contains 0.101-in-thick Boral plates. The fuel has been burned for 45GWD/MTU, and cooled for 5 years.
5. OECD computational benchmark cask, which holds 21 PWR assemblies. The model contains 0.08-in-thick borated steel with 1 wt% boron. The fuel has been burned for 30 GWD/MTU, and cooled for 1 year.

Related NRC funded work for ^{10}Boron use in shipping casks (cont)

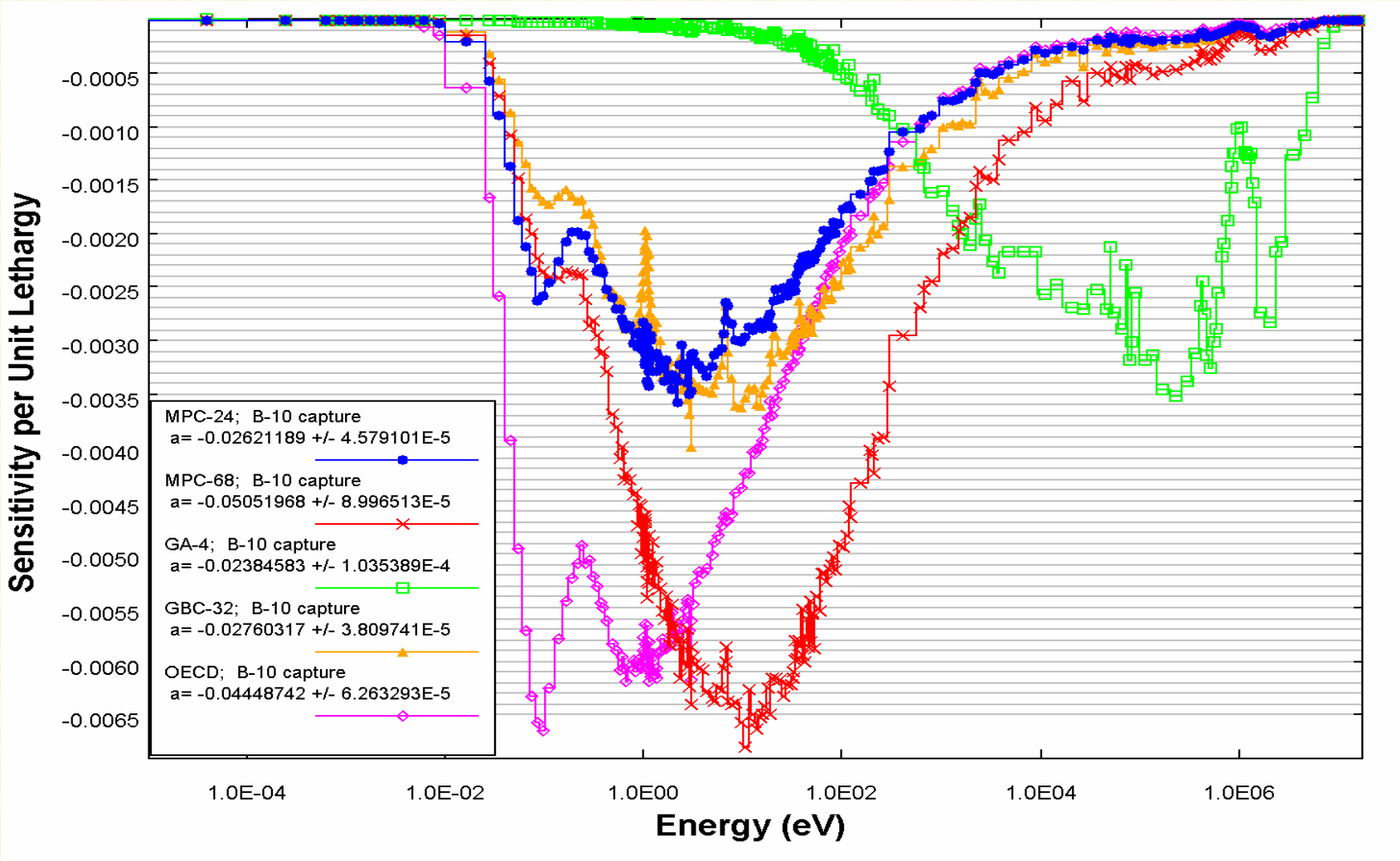
^{10}B mass for all benchmarks and applications



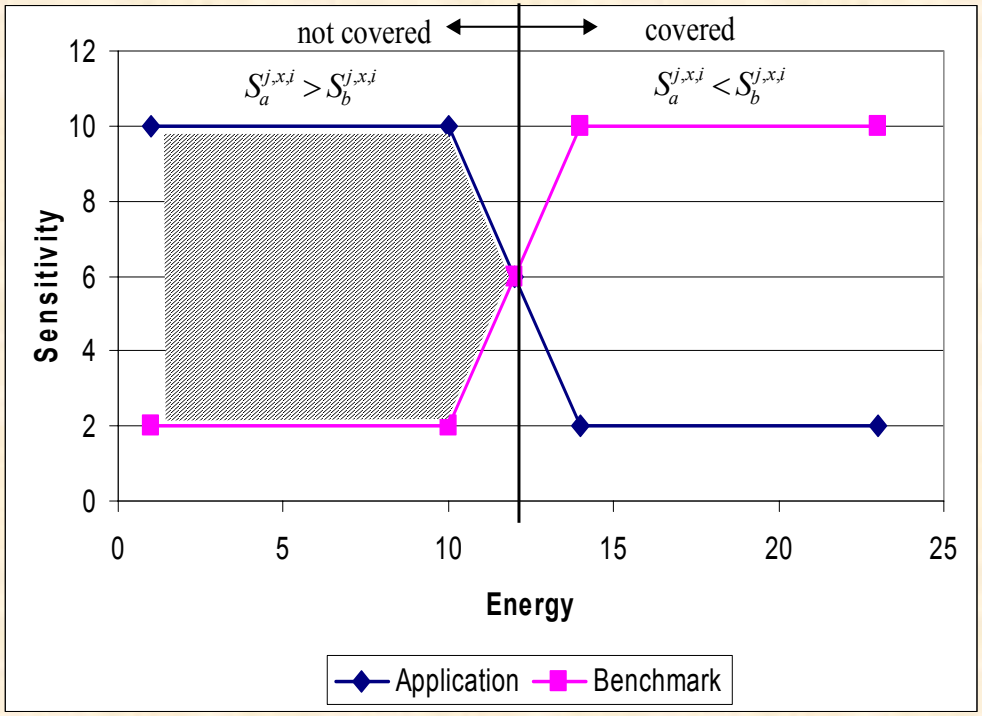
Nuclear fuel cask model parameters

Cask	Calculated $k_{eff} \pm \sigma$	EALF (eV)	^{10}B Capture sensitivity	^{10}B Form	Total ^{10}B (kg)	^{10}B Surface density (at/cm^2)
MPC-24	0.9458 ± 0.0005	2.257E−01	−2.62E−02	Boral	~12	1.216E21
MPC-68	0.9349 ± 0.0005	2.775E−01	−5.05E−02	Boral	~15	1.658E21
GA-4	0.9221 ± 0.0005	4.572E−01	−2.38E−02	B_4C	~8	4.750E22
GBC-32	0.8941 ± 0.0004	2.474E−01	−2.76E−02	Boral	~12	1.688E21
OECD	1.1303 ± 0.0005	6.311E−02	−4.45E−02	Borated steel	~6	3.918E20

^{10}B Capture Sensitivity per Unit Lethargy vs Neutron Energy



Approach to developing “penalty” for ignorance



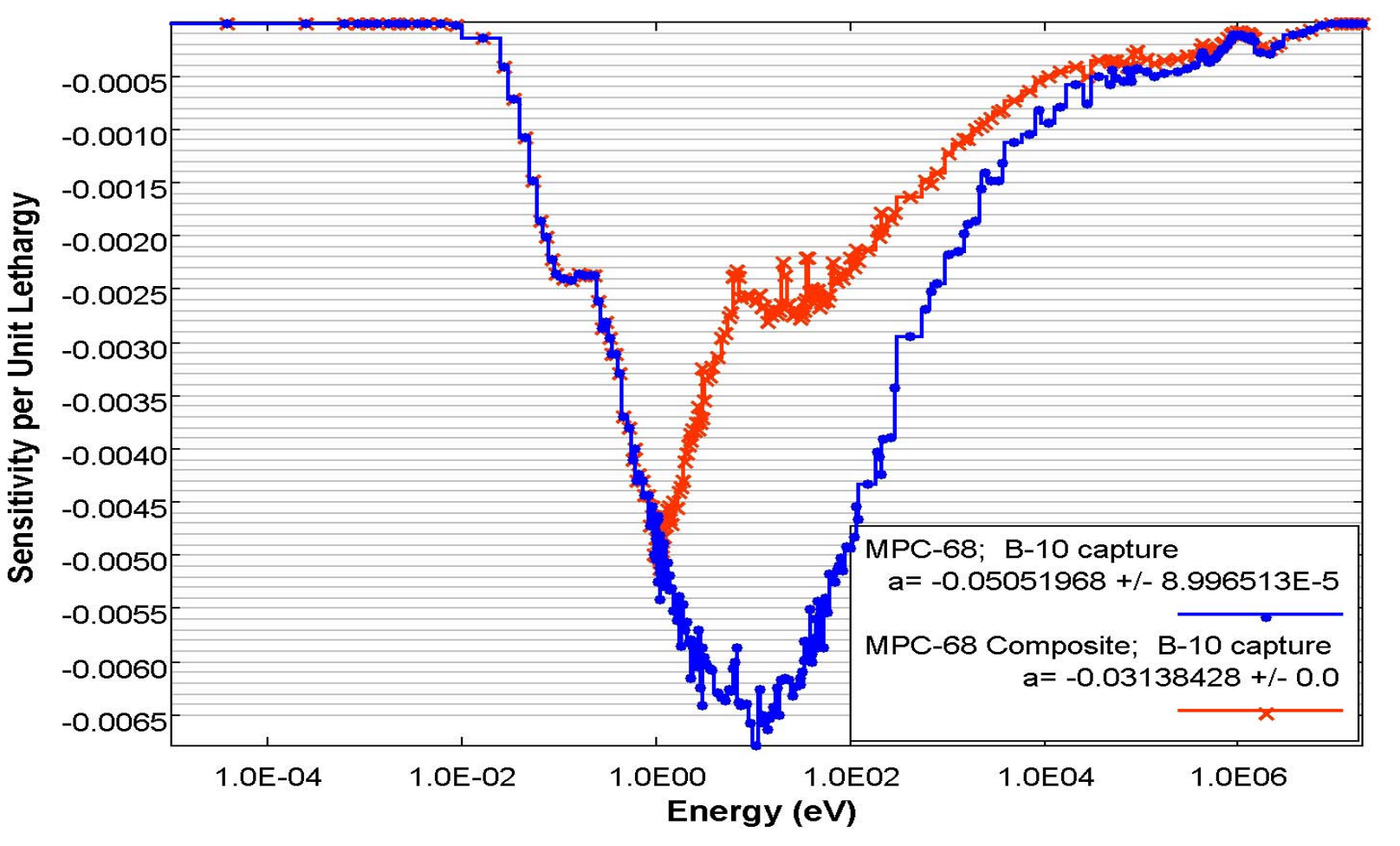
$$g_{j,x} = 1 - g'_{j,x} = 1 - \frac{\sum_i Z_{ab}^{j,x,i}}{\sum_i S_a^{j,x,i}}$$

Non-Coverage by number of groups

Cask	Number of groups without coverage	Sum of minimum sensitivity differences in 10B capture (Z_a)
MPC-24	28	–4.76E–04
MPC-68	179	–1.91E–02
GA-4	69	–1.69E–02
GBC-32	72	–1.65E–03
OECD	107	–5.73E–03
MPC-68m*	83	–3.16E–03
GA-4m*	14	–1.81E–04

* Modified by reducing boron concentration.

Comparison of MPC-68 10B capture sensitivity profile and composite profile for MPC-68



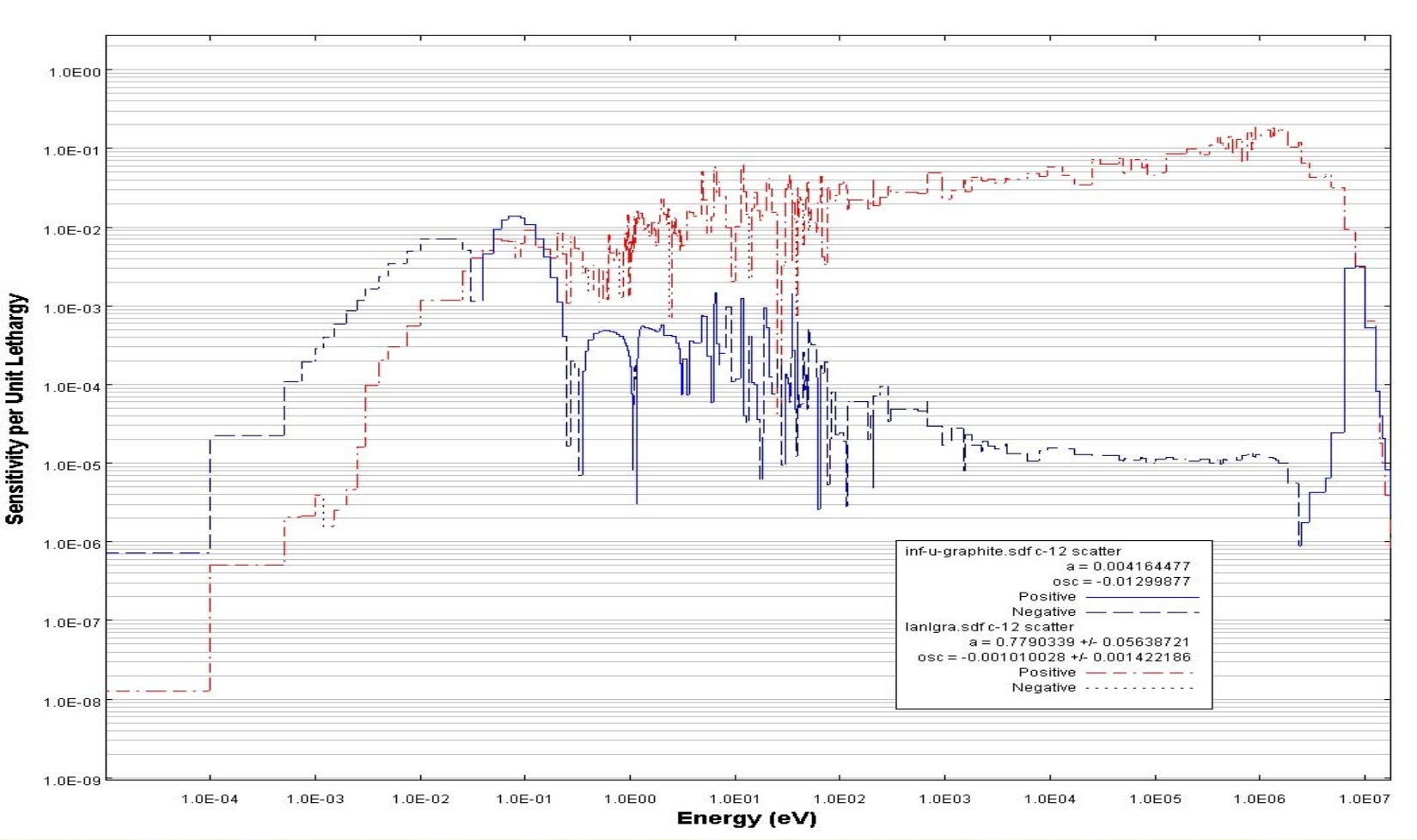
Final results

Cask	Calculated k_{eff}	Maximum g for ^{10}B capture	Sum of minimum sensitivity differences in 10B capture ($Z_a^{j,x}$)	Penalty ($\%\Delta k_{eff}/k_{eff}$) due to ^{10}B capture only	Adjusted k_{eff}
MPC-24	0.9458	0.79	–4.76E–04	0.0001	0.9458
MPC-68	0.9349	0.46	–1.91E–02	0.0044	0.9349
GA-4	0.9221	0.29	–1.69E–02	0.0077	0.9221
GBC-32	0.8941	0.73	–1.65E–03	0.0009	0.8941
OECD	1.1303	0.75	–5.73E–03	0.0013	1.1303
MPC-68m*	1.0012	0.83	–3.16E–03	0.0010	1.0012
GA-4m*	0.9959	0.86	–1.81E–04	0.0000	0.9959

* Modified by reducing boron concentration.

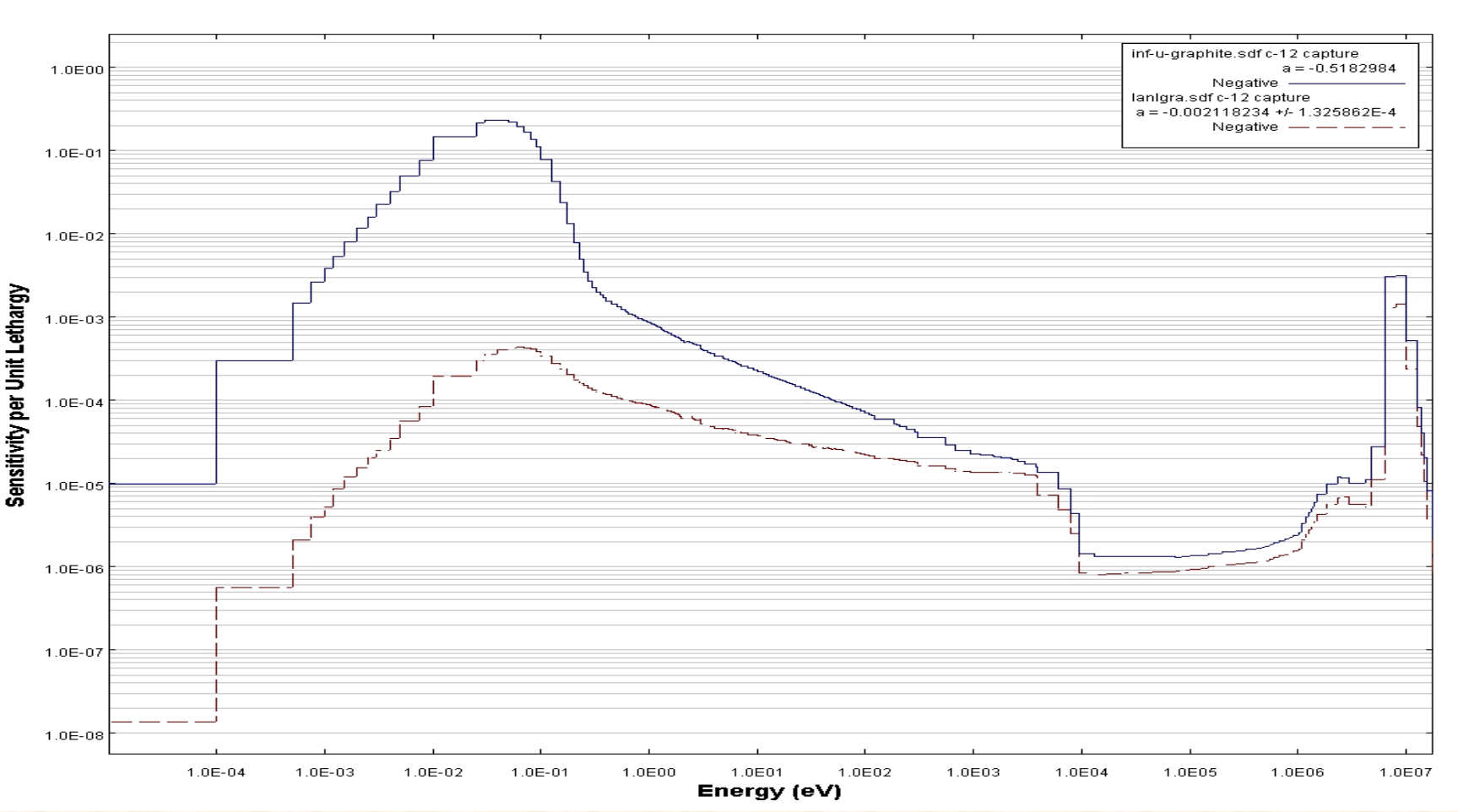
Dilution experiments for waste exemptions (DOE & NRC issues)

- **10CFR71 Carbon Scatter considerations**



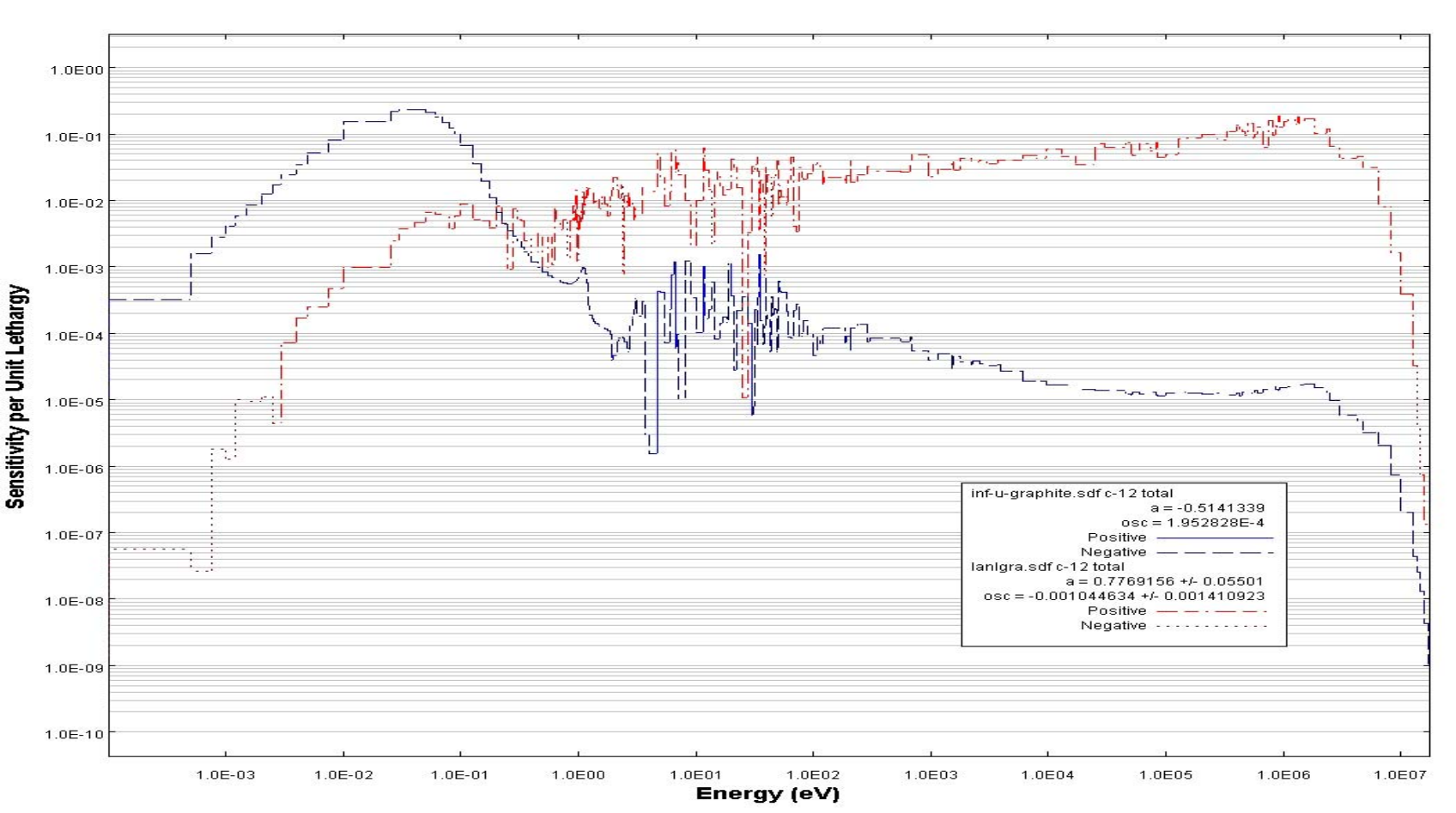
Dilution experiments for waste exemptions (DOE & NRC issues) (cont.)

- **10CFR71 Carbon Capture considerations**



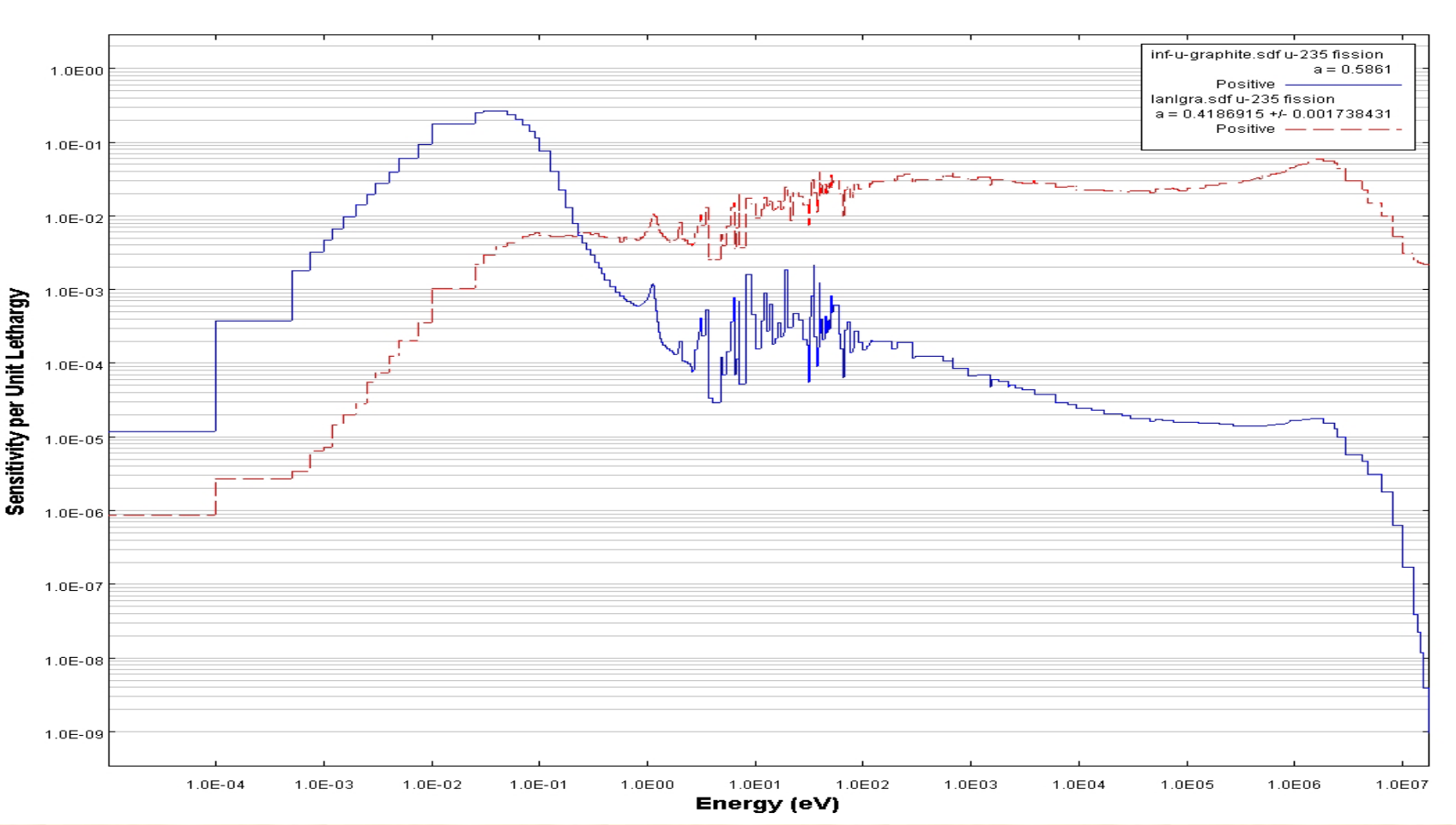
Dilution experiments for waste exemptions (DOE & NRC issues) (cont.)

- **10CFR71 Carbon Total considerations**



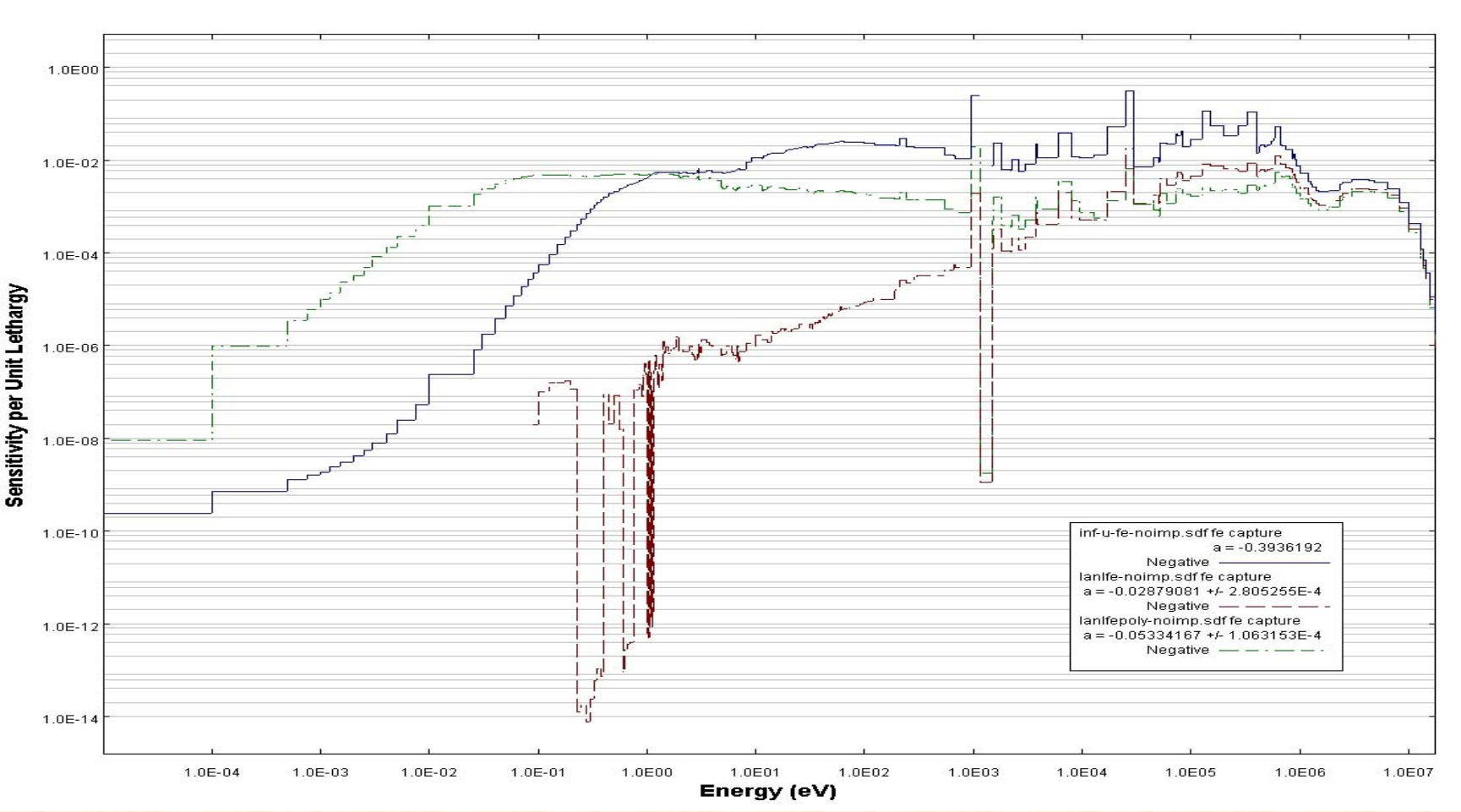
Dilution experiments for waste exemptions (DOE & NRC issues) (cont.)

- 10CFR71 ^{235}U fission in Carbon considerations



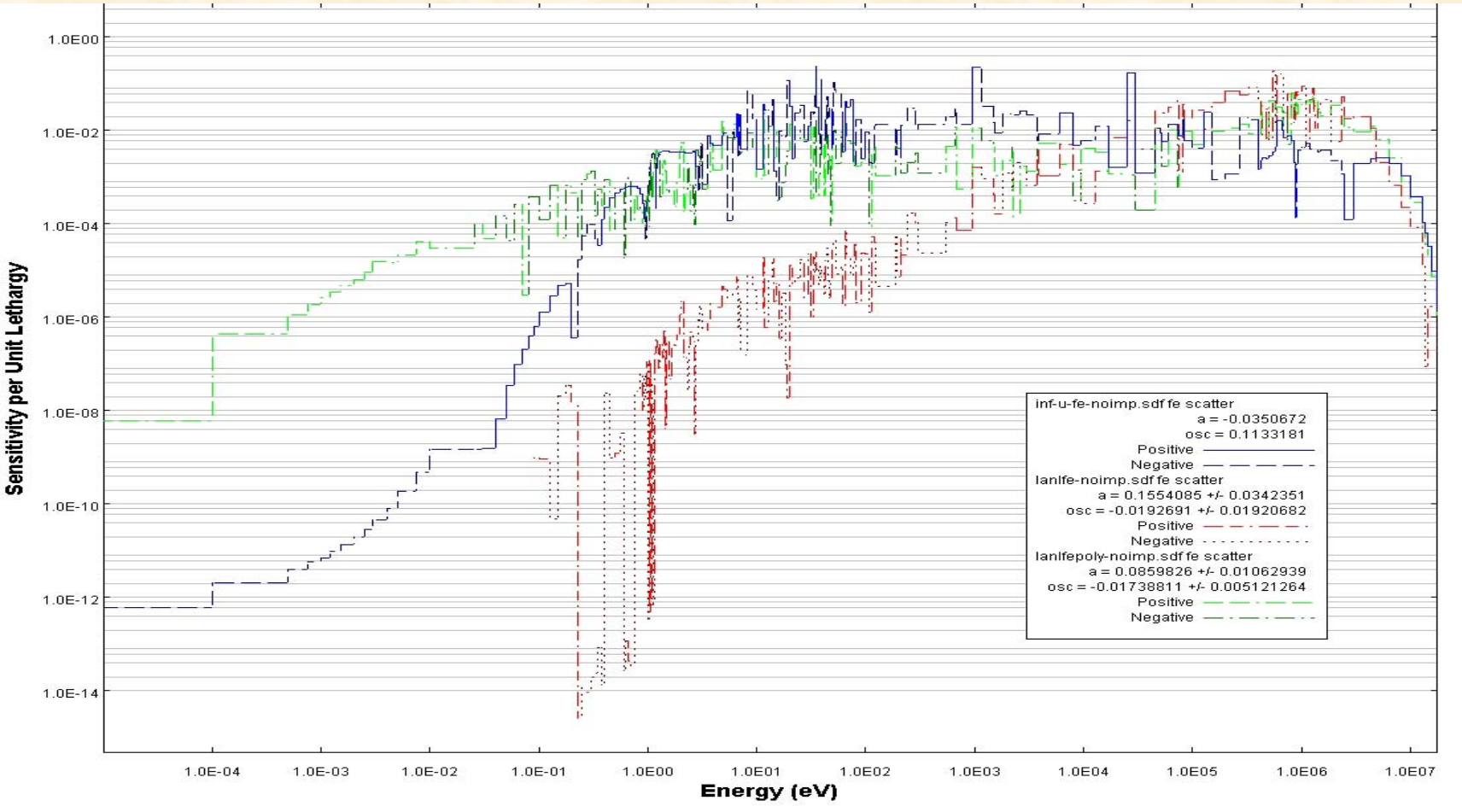
Dilution experiments for waste exemptions (DOE & NRC issues) (cont.)

- **Fe Capture considerations**



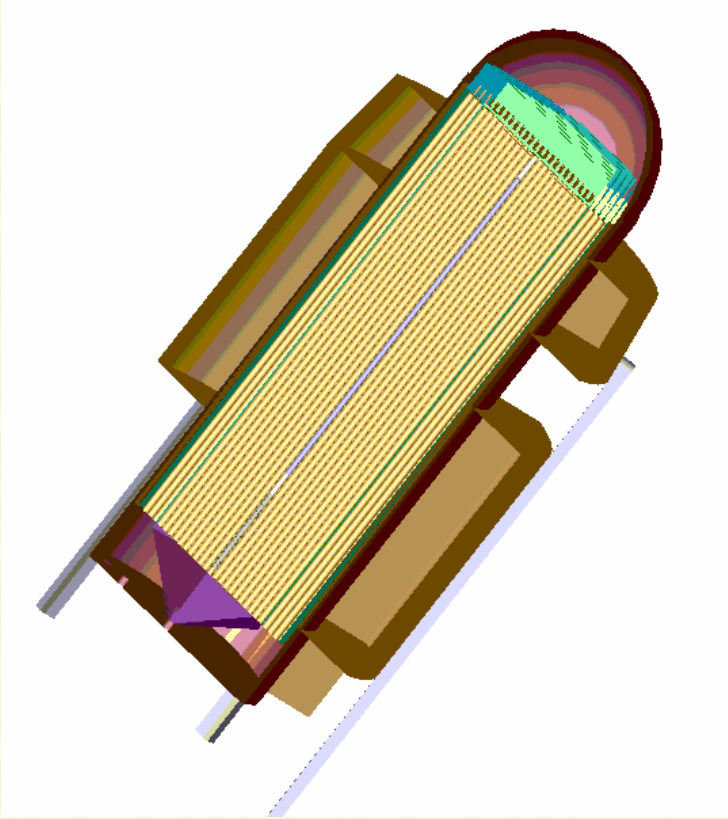
Dilution experiments for waste exemptions (DOE & NRC issues) (cont.)

- **Fe Scatter considerations**

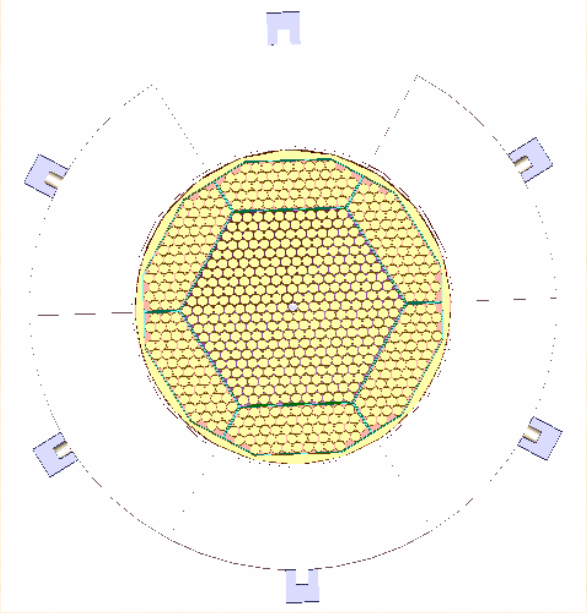


NASA/DOE Space Fission Power Systems

Vertical



Plan

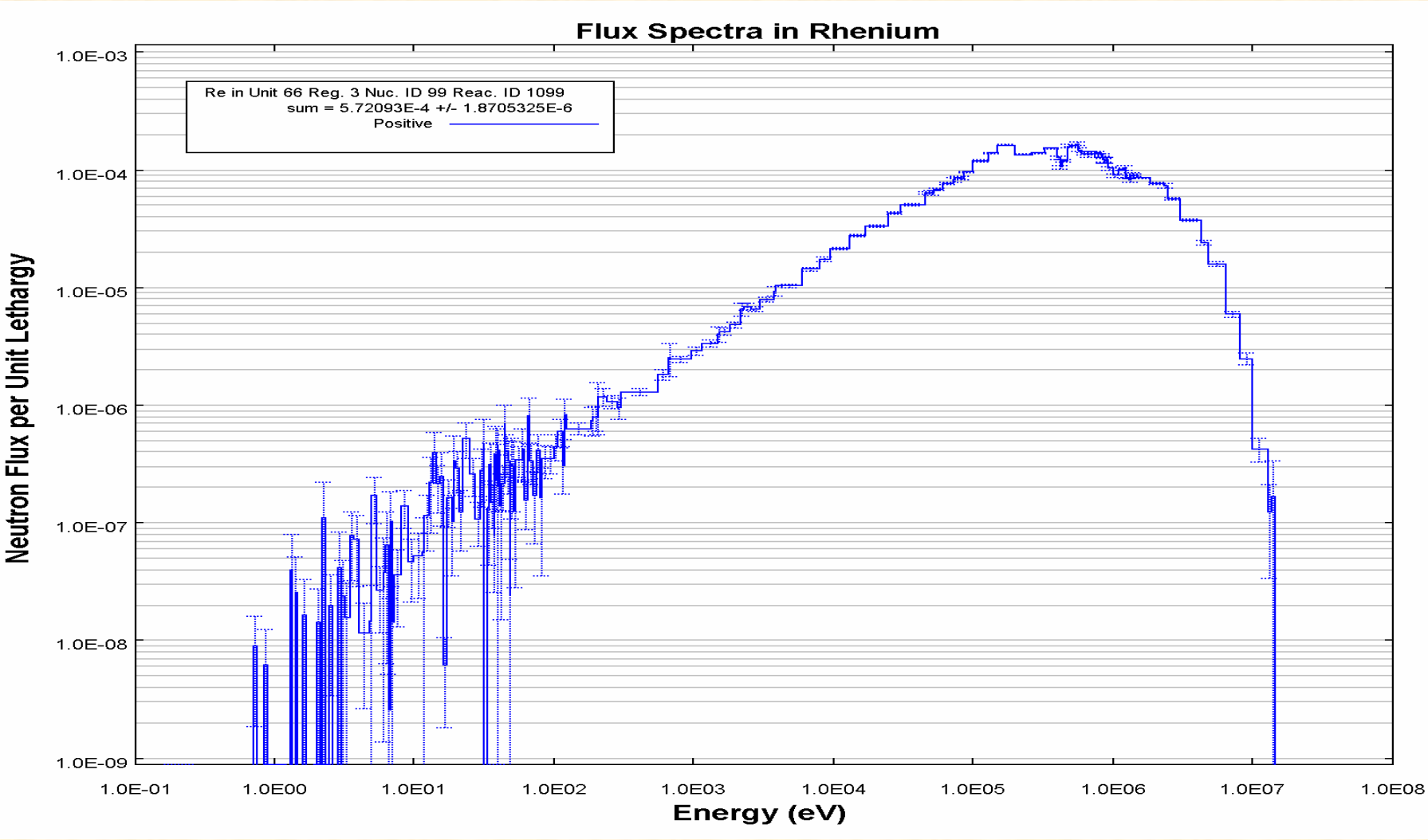


NASA/DOE Space Fission Power Systems (cont.)

	Material in Coolant Channels	
	Lithium	Water
k_{eff}	1.03242 ± 0.00049	1.04124 ± 0.00056
Re-185 n, γ sensitivity	-2.2684E-02	-3.4975E-02
Re-187 n, γ sensitivity	-2.9796E-02	-4.3754E-02
Nb-93 n, γ sensitivity	-1.1695E-02	-1.1947E-02
U-235 n, γ sensitivity	+7.409E-02	+9.745E-02
N-14 n, γ sensitivity	-1.738E-02	-1.551E-02

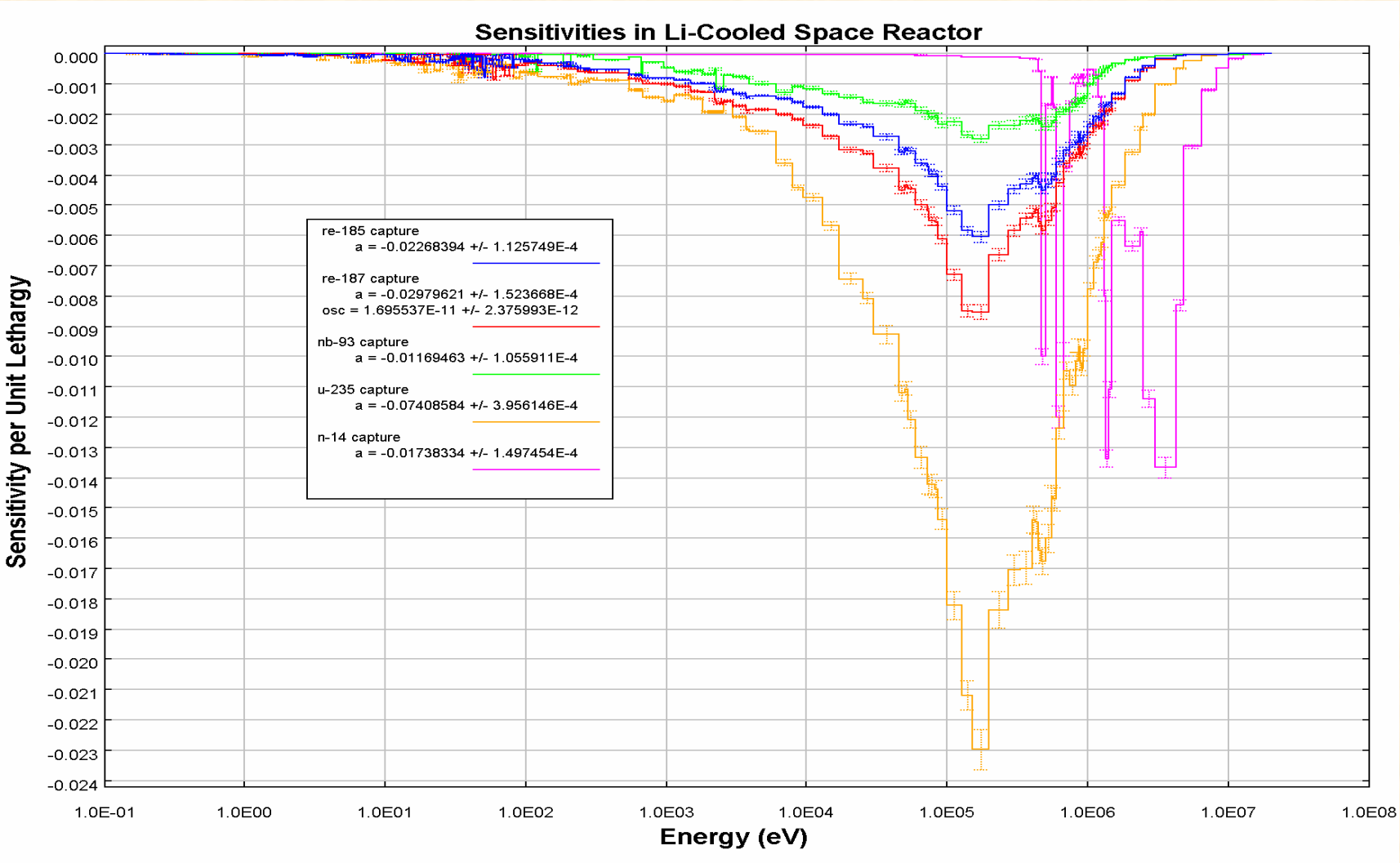
NASA/DOE Space Fission Power Systems (cont.)

- Lithium coolant



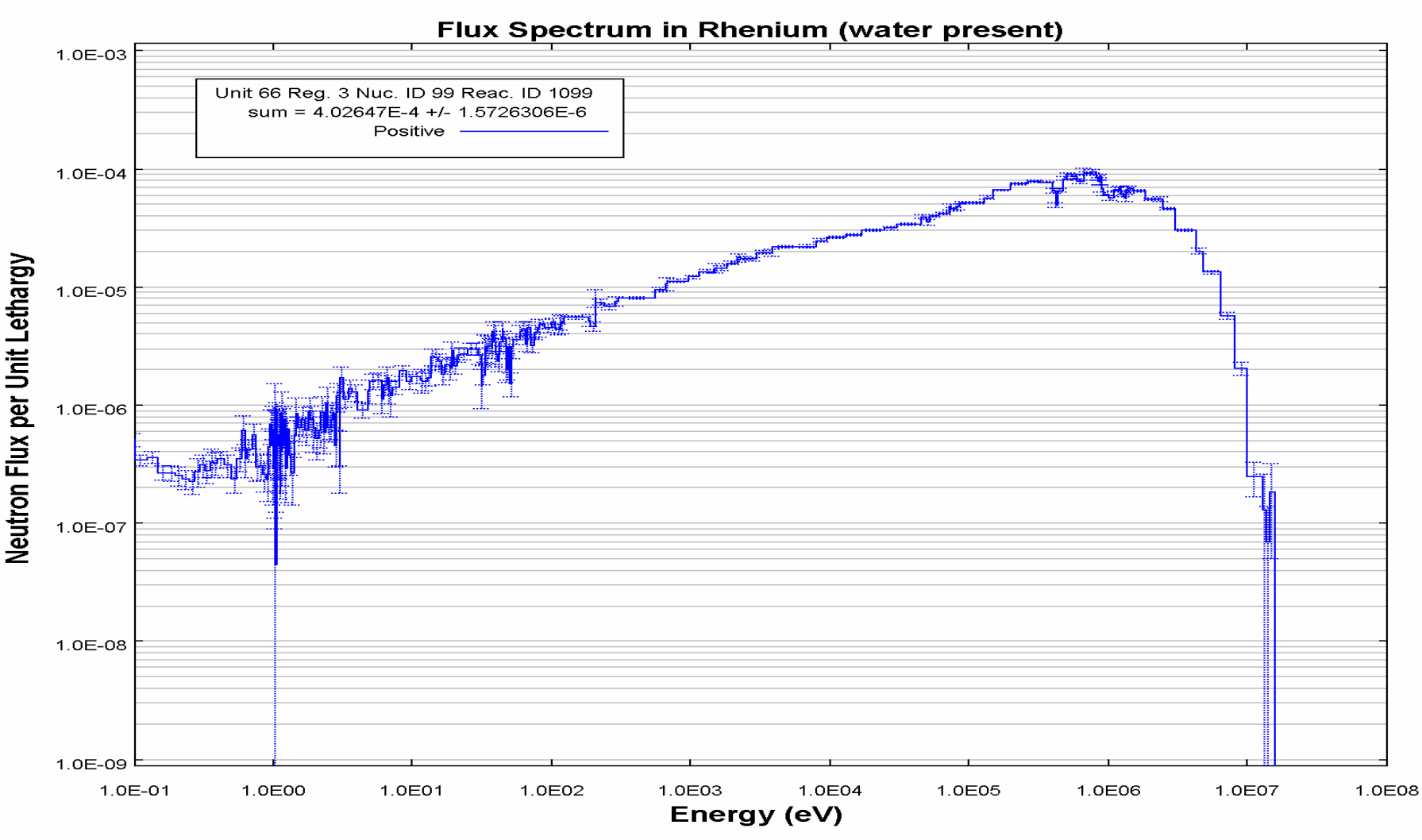
NASA/DOE Space Fission Power Systems (cont.)

- Lithium coolant



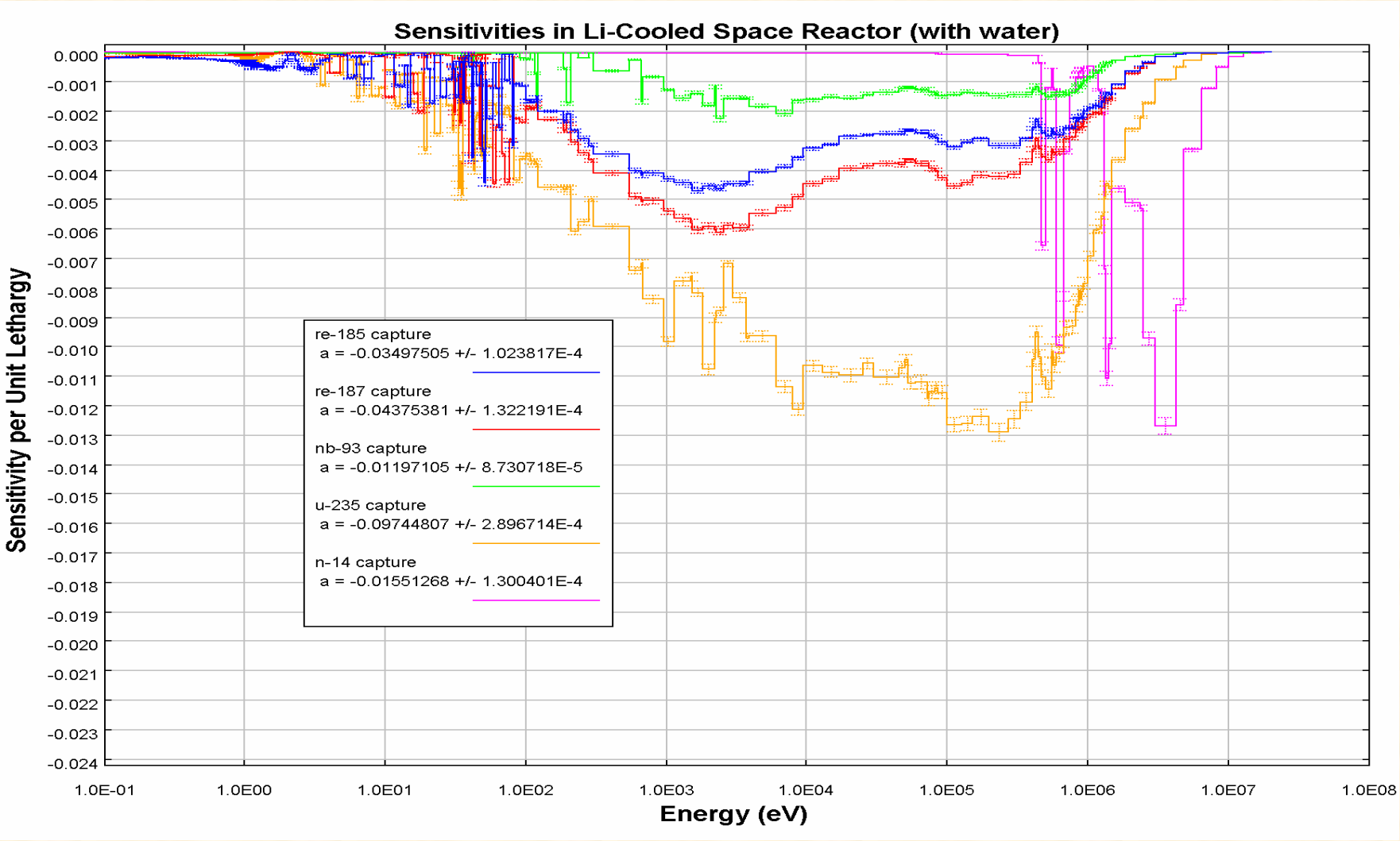
NASA/DOE Space Fission Power Systems (cont.)

- Water coolant

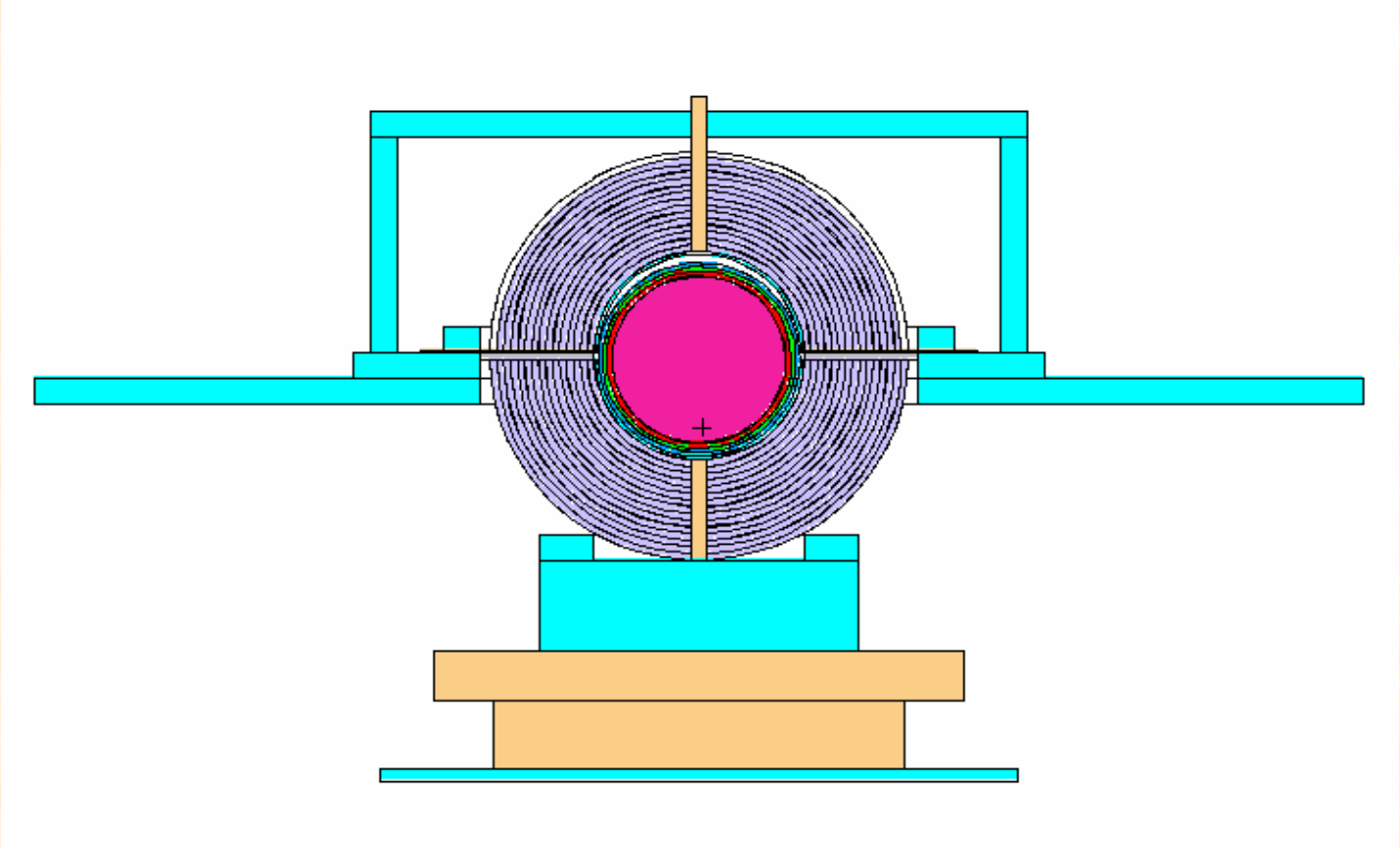


NASA/DOE Space Fission Power Systems (cont.)

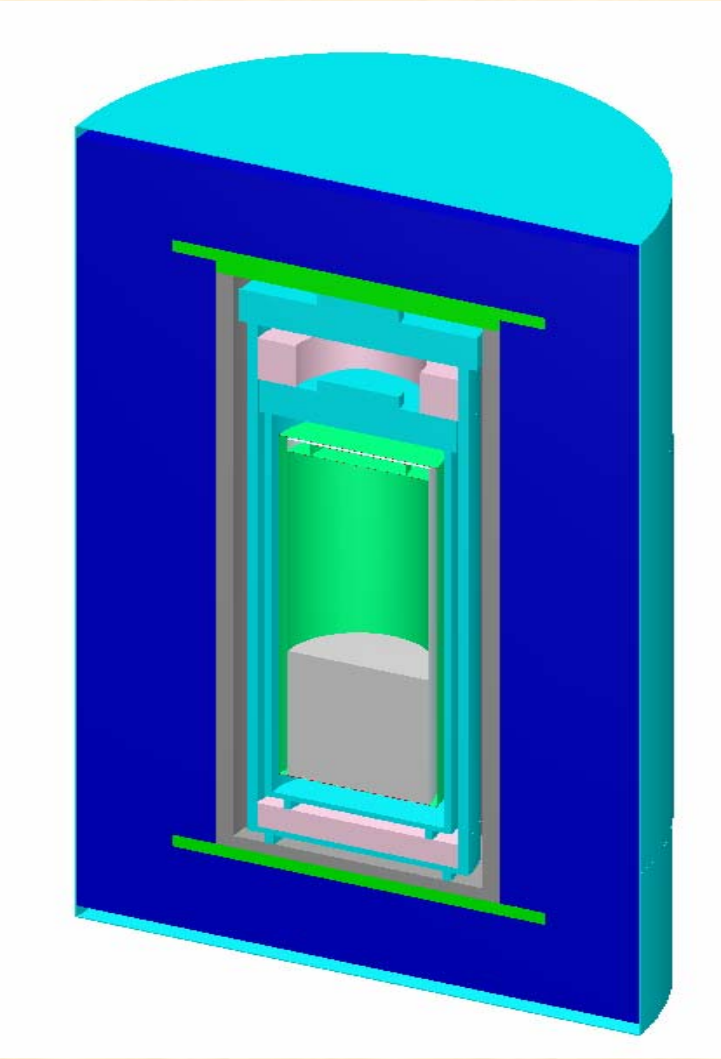
- Water coolant



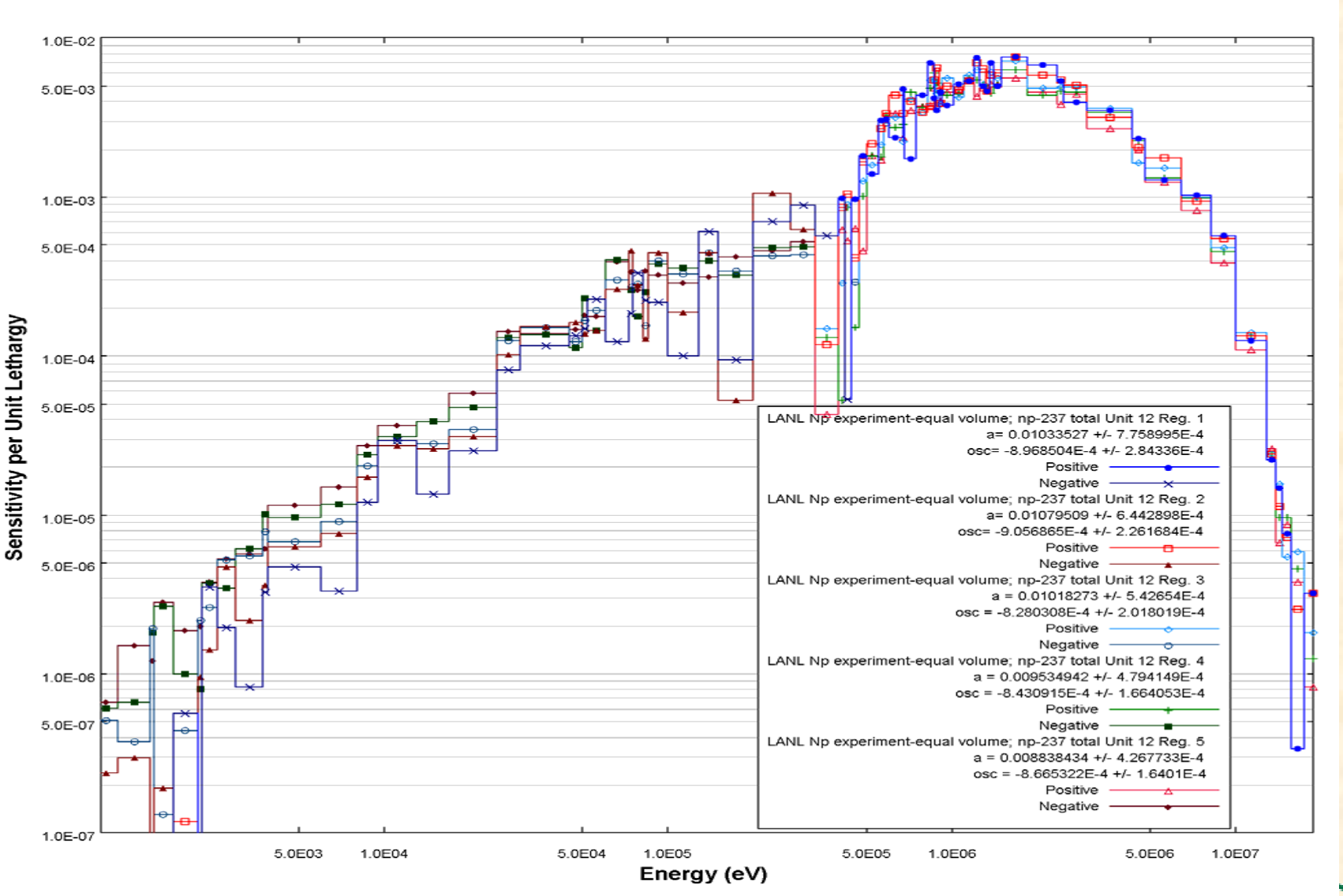
Storage of ^{237}Np at Oak Ridge Y-12 Plant
LANL ^{237}Np Ball Experiment



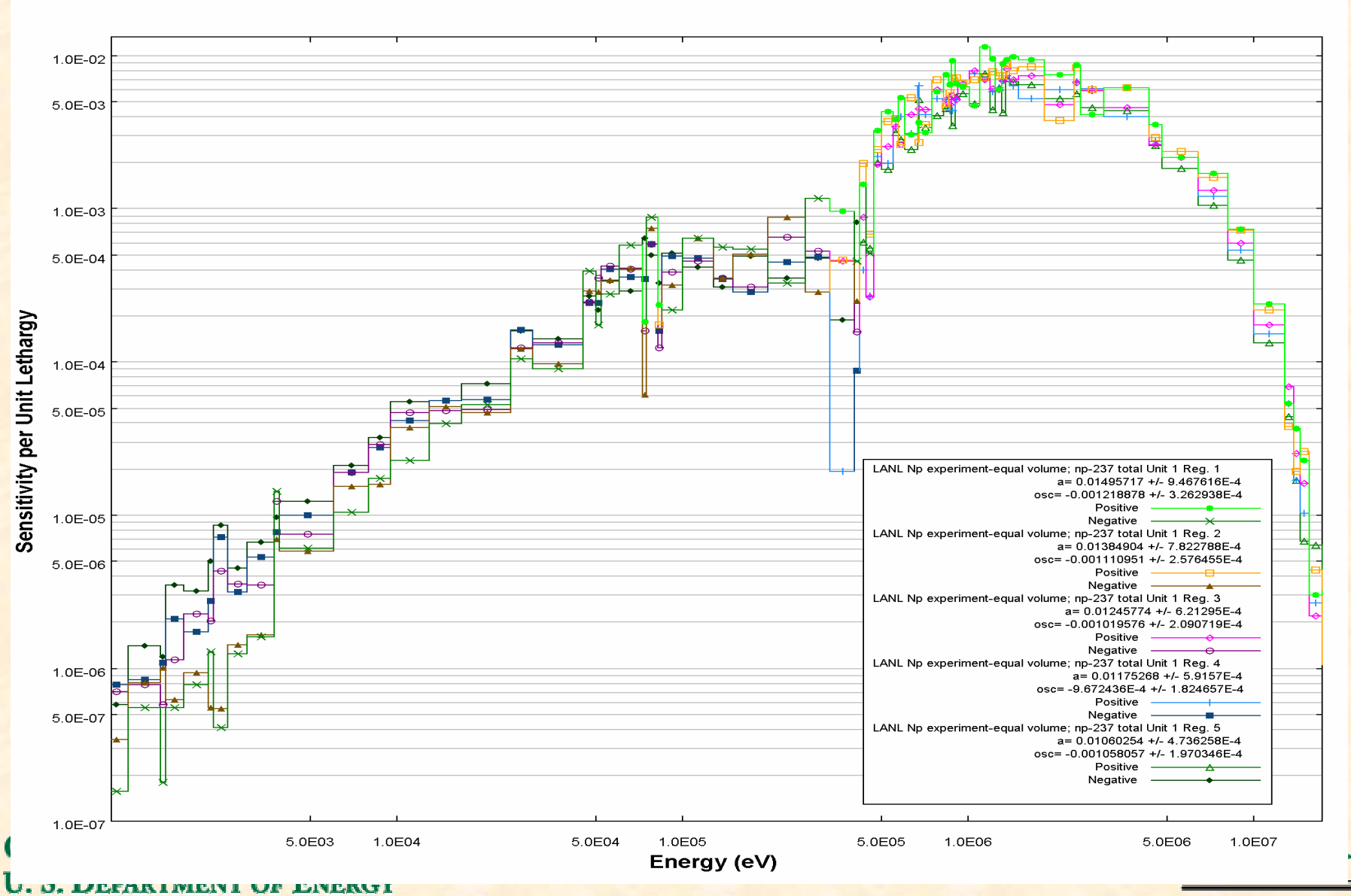
Storage of 237Np at Oak Ridge Y-12 Plant
9975 Shipping/Storage Container



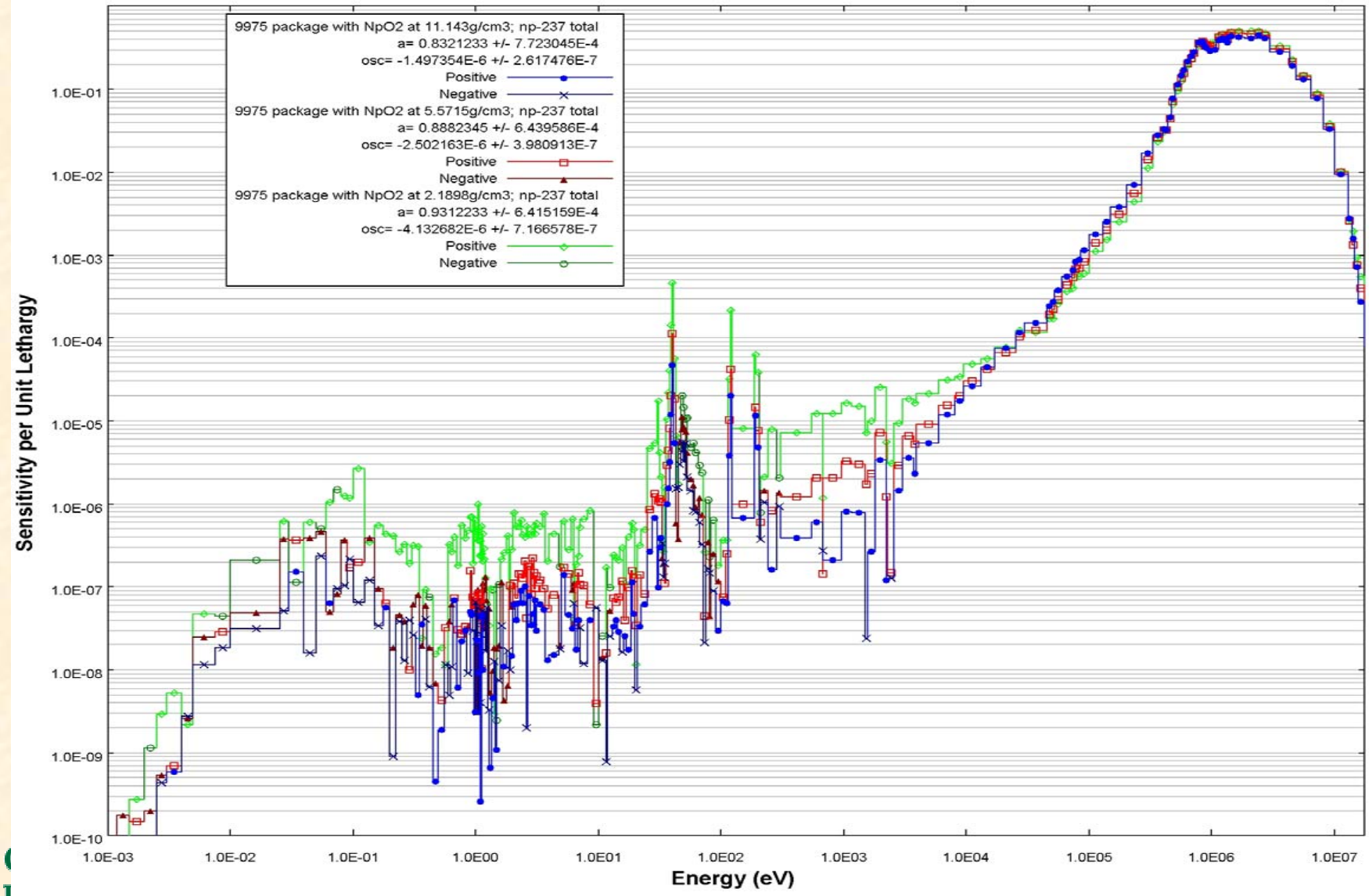
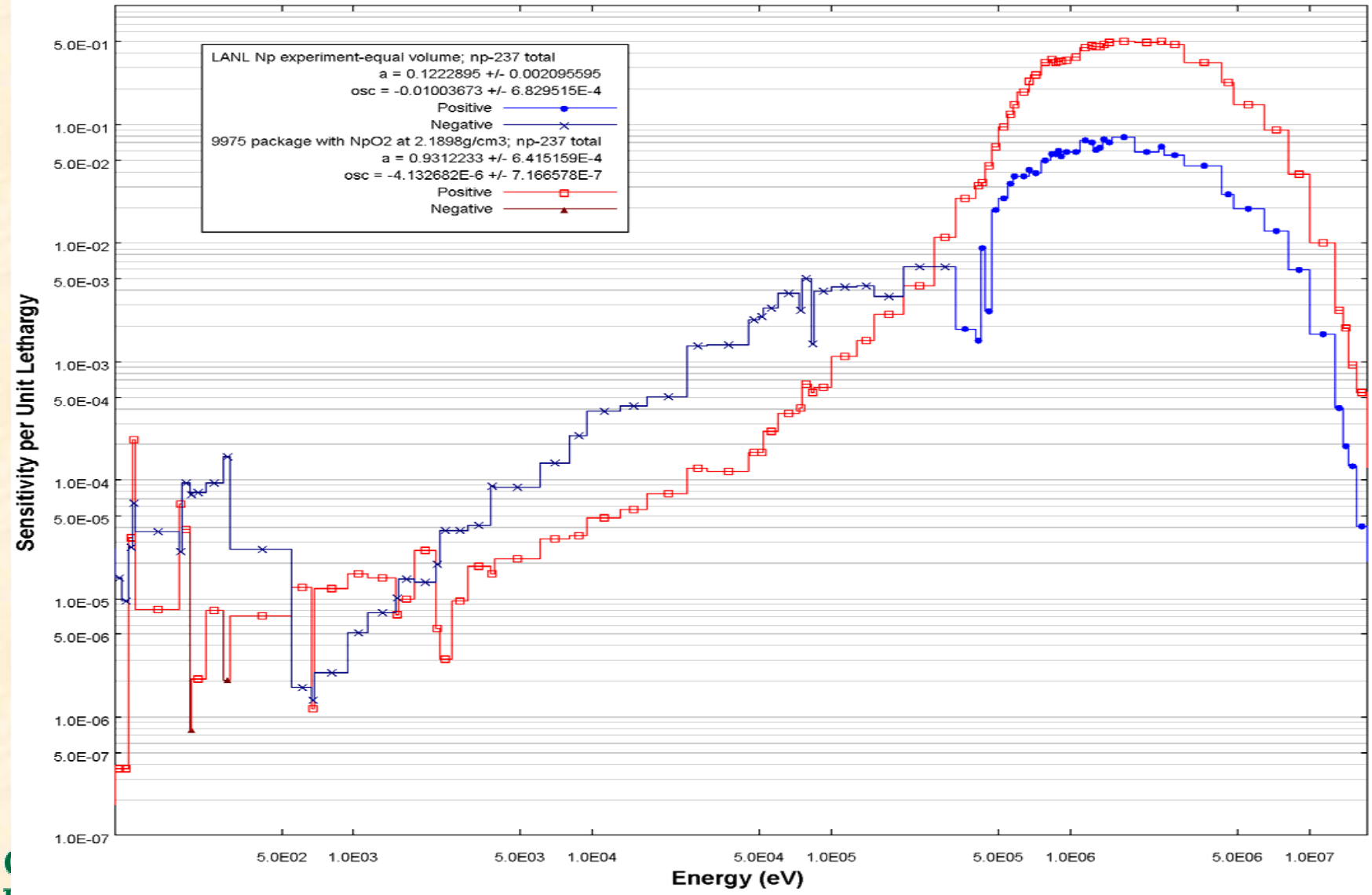
Storage of 237Np at Oak Ridge Y-12 Plant
9975 Shipping/Storage Container



Storage of 237Np at Oak Ridge Y-12 Plant
9975 Shipping/Storage Container



Storage of 237Np at Oak Ridge Y-12 Plant
9975 Shipping/Storage Container



Storage of 237Np at Oak Ridge Y-12 Plant
9975 Shipping/Storage Container