INL/EXT-18-51518
Revision 0

A Proposed Path Forward for Transportation of High-Assay Low-Enriched Uranium

Josh Jarrell

September 27, 2018

Idaho National Laboratory

The INL is a U.S. Department of Energy National Laboratory
operated by Battelle Energy Alliance

INL/EXT-18-51518
Revision 0

A Proposed Path Forward for Transportation of High-Assay Low-Enriched Uranium

Josh Jarrell

September 2018

Idaho National Laboratory
Idaho Falls, Idaho 83415

http://www.inl.gov

Prepared for the
U.S. Department of Energy
Office of Nuclear Energy
Under DOE Idaho Operations Office
Contract DE-AC07-05ID14517

SUMMARY

Many of the advanced reactors currently being designed will use high-assay low-enriched uranium (HALEU) as the reactor fuel. HALEU is fuel that is enriched to 5–20% uranium-235. With the change to higher enriched material, the industry will have new challenges regarding the development and regulatory approval of enrichment and fuel fabrication facilities and suitable transportation packages to support the economic use of HALEU materials. One area of concern relates to ensuring sub-criticality of the material during transportation as identified by the Nuclear Energy Institute (NEI). To evaluate the relevant work, expertise, and industry perspectives on HALEU, a workshop was organized to share relevant experience and insights into HALEU transportation, handling, and management.

At the workshop, held August 30 and 31, 2018, NEI and industry provided the following recommendations to the Department of Energy (DOE) and the national lab complex.

- DOE and the lab complex should communicate and educate the Nuclear Regulatory Commission (NRC) on criticality issues related to HALEU.
- Idaho National Laboratory (INL) should support work needed to certify package design(s) for the transportation of HALEU.
    - An amendment of the Certificate of Compliance of an existing package could be used for the shipment of commercial quantities.
    - DOE could provide funding to package designer(s) for analysis and engineering work for a package to be submitted to NRC for approval.
- INL should provide the expected amount of impurities (either a specific number or a range) that will be present in recycled naval fuel.
- In the longer term, DOE and the lab complex should increase the availability of criticality benchmark data to further reduce conservatism in package design.

In addition, a couple of key takeaways were identified, including the following:

- Although the labs can provide additional criticality experiments, industry has enough data to license facilities, overpacks, and cylinders. Validation from additional critical experiments to establish less uncertainty in the benchmarks will be helpful.
- A collective effort from industry is needed to express consistency on how much information exists or is needed related to criticality.

Based on interactions with industry, DOE, and national laboratories, large-volume transportation of fresh HALEU appears to be feasible from a criticality perspective. Specifically, an initial review of applicable criticality benchmark

experiments identified numerous applicable experiments. However, a more thorough review of a realistic transportation package design is suggested. Therefore, the following next steps are proposed:

1. Evaluate a large-volume package with uranium dioxide enriched to 20%. The GNF-A NPC package was specifically proposed as a potentially viable option.
2. Based on the results of (1), determine if additional package designs and fuel material should be evaluated.
3. Based on the results of (1), determine if additional criticality experiments would be beneficial to improve the margins for criticality due to uncertainties.
4. Determine DOE transportation needs (packages sizes, shielding requirements, handling/operational requirements, and timing of availability) related to HALEU.
5. Continue to interface with NEI and interested industry companies to determine the appropriate time to engage the NRC.

CONTENTS

FIGURES

ACRONYMS

DOE	Department of Energy
HALEU	high-assay low-enriched uranium
HEU	highly-enriched uranium
INL	Idaho National Laboratory
LEU	low-enriched uranium
NE	Office of Nuclear Energy
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory

A Proposed Path Forward for Transportation of High-Assay Low-Enriched Uranium

1. INTRODUCTION

Many of the advanced reactors currently being designed will use high-assay low-enriched uranium (HALEU) as the reactor fuel [NEI 2018b]. HALEU is fuel that is enriched to 5–20% uranium-235. With the change to higher enriched material, the industry will face new challenges regarding the development and regulatory approval of enrichment and fuel fabrication facilities and suitable transportation packages to support the economic use of HALEU materials. One area of concern relates to ensuring sub-criticality of the material during transportation as identified by the Nuclear Energy Institute (NEI) [NEI 2018a].

To evaluate the relevant work, expertise, and industry perspectives on HALEU, a workshop was organized to share relevant experience and insights into HALEU transportation, handling, and management.

2. AUGUST 2018 WORKSHOP

The Idaho National Laboratory (INL)/NEI Invitation-Only Technical Workshop on Transportation of High-Assay Low-Enriched Uranium was hosted at NEI in Washington, D.C. August 30 and 31, 2018. It brought together a range of industry participants, national laboratories, and Department of Energy (DOE) representatives. The primary objective of this workshop was to advise DOE on the gaps related to transportation of HALEU and licensing support activities. The fundamental goal was to ensure that transportation and handling of HALEU at associated fuel cycle facilities does not delay the deployment of advanced reactors. A summary of the workshop is included in Appendix A. Of particular interest are the recommendations from NEI and industry participants for DOE/INL:

- DOE and the lab complex should communicate and educate the NRC on criticality issues related to HALEU.
- INL should support work needed to certify package design for the transportation of HALEU.
  - An amendment of the COC of an existing package could be used for the shipment of commercial quantities.
  - DOE could provide funding to package designer(s) for analysis and engineering work for a package to be submitted to NRC for approval.
- INL should provide the expected amount of impurities (either a specific number or a range) that will be present in recycled naval fuel.
- In the longer term, DOE and the lab complex should increase the availability of criticality benchmark data (i.e., by performing, sponsoring, or data mining additional criticality benchmarks) to further reduce conservatism in package design.

3. PACKAGE DESIGNS

Current package designs are generally divided into two groups: (1) large packages designed for less than 5% enriched material or (2) smaller packages designed for up to 100% enriched material. For example, 2,277 kg of UF_6 currently can be transported in Type 30B packages at up to 5% enrichment (as illustrated in Figure 1), while only 24.9 kg of UF_6 can be transported in Type 5A/B packages at up to 100% enrichment [ANSI 2012].

Figure 1. Example of Model 30B UF6 Cylinder [Appendix A].

Another example is the DOE-certified ES-3100 package, which has been design to hold 24 kg of UO_2 as illustrated in Figure 2.

Figure 2. ES-3100 contents [Appendix A].

For fresh fuel packaging, GNF has a package, the GNF-A NPC, currently designed to move 5% enriched UO_2, U_3O_8, UO_x, and other uranium materials as shown in Figure 3.

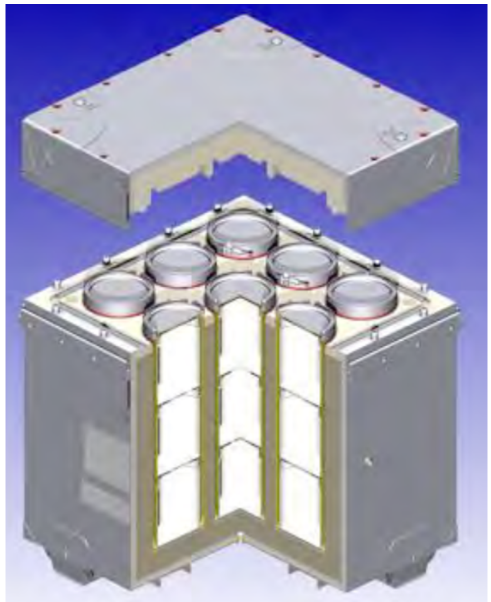


Figure 3. GNF-A NPC package [Appendix A].

In addition, Daher-TLI is in the process of developing a package based on the 30B package. It could accommodate 20% enriched UF_6 and is called the 30B-20. It is being developed with a goal to transport up to 1,600 kg of UF_6 enriched to 20% as illustrated in Figure 4.

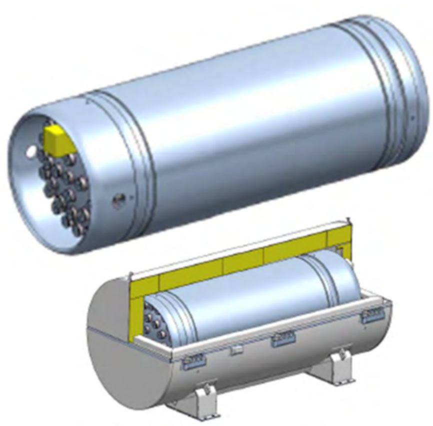


Figure 4. DAHER-TLI conceptual design of the 30B-20 cylinder for UF_6 [Appendix A].

4. APPLICABILITY OF CRITICAL EXPERIMENTS

To date, there have been over 5,000 approved International Criticality Safety Benchmark Evaluation Project (ICSBEP) criticality benchmarks, though most uranium experiments are done with less than 5% enriched or greater than 20% enriched material. This potential lack of experiments in the 5–20% enriched range may increase the needed conservatism in package design. As such, Oak Ridge National Laboratory (ORNL) performed a set of initial analyses to explore the number of applicable experiments and found 376 ICSBEP experiments using uranium with 5–25% enrichment.

The applicability of experiments is not solely dependent on enrichment, but must also take materials, configuration, and design into account. To determine how similar the application and the critical experiment models are, sensitivity/uncertainty tools in the TSUNAMI/SCALE software package were used to compare each application/experiment pair. This approach produced a correlation coefficient (ck) for each application pair, which ranged from 0 to 1. A high ck value (near 1) for an application pair indicates that both models have similar sensitivities to the same nuclear data and, consequently, should have similar biases. Conversely, a low ck value (near 0) indicates that the two systems differ significantly and may have significantly different biases.

For the initial analysis, the ES-4100 package was evaluated with 20% enriched UF6. This package allows 1 kg of U-235 in each of the four containment vessels and has a B_4C poison element in the central location as illustrated in Figure 5.

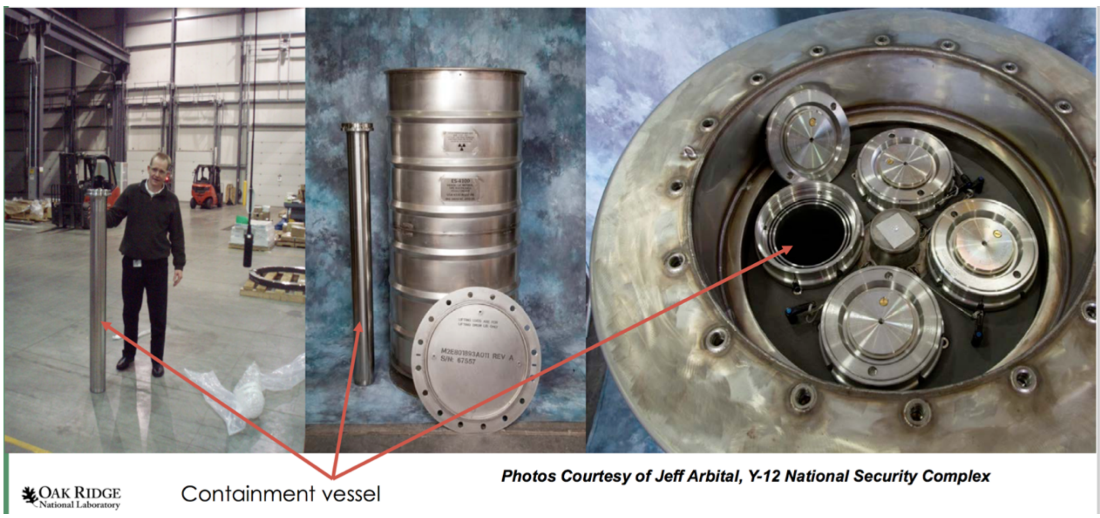


Figure 5. ES-4100 package [Appendix A].

The initial results indicated a large number of applicable experiments as illustrated in Figure 6.

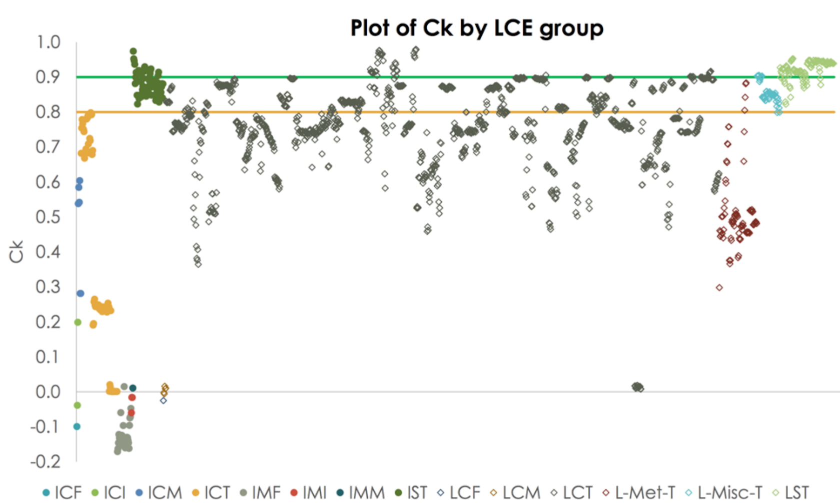


Figure 6. Plot of ck when comparing criticality experiments with the ES-4100 package with 20% enriched UF_6 [Appendix A].

Of the 1,584 evaluated experiments, 173 had ck above 0.9 and 698 had ck above 0.8. Therefore, initial results imply that a significant number of experiments will be applicable to a transportation package with HALEU. However, there are some questions that will need to be confirmed, including:

- Do larger-volume packages with more reactive configurations have similar numbers of applicable benchmarks?
- Do other fuel forms (e.g., UO_2, U_3O_8, TRISO-based fuels, and metallic fuels) have similar numbers of applicable benchmarks?
- For larger packages, what are the biases and uncertainties due to nuclear data?

5. RECOMMENDATIONS

Based on communication with industry, DOE, and national laboratories, large-volume transportation of fresh HALEU appears to be feasible from a criticality perspective. Specifically, an initial review of applicable criticality benchmark experiments identified numerous applicable experiments. In fact, the handling requirements driven by the material characteristics may be a more pressing concern than the transportation issues. However, for completeness, a more thorough review of a realistic transportation package design is suggested. Therefore, the following next steps related to the transportation of HALEU are proposed:

1. Evaluate a large-volume package with uranium dioxide enriched to 20%. The GNF-A NPC package was specifically proposed as a potentially viable option.
2. Based on the results of (1), determine if additional package designs and fuel material should be evaluated.
3. Based on the results of (1), determine if additional criticality experiments would be beneficial to improve the margins for criticality due to uncertainties.
4. Determine DOE transportation needs (packages sizes, shielding requirements, handling/operational requirements, and timing of availability) related to HALEU.
5. Continue to interface with NEI and interested industry companies to determine the appropriate time to engage the NRC.

6. REFERENCES

ANSI 2012 — *ANSI N14.1: Packaging of Uranium Hexafluoride for Transport*, American National Standards Institute, 2012.

NEI 2018a — *Addressing the Challenges with Establishing the Infrastructure for the front-end of the Fuel Cycle for Advanced Reactors, NEI White Paper*, January 2018. https://www.nei.org/CorporateSite/media/filefolder/resources/reports-and-briefs/white-paper-advanced-fuel-cycle-infrastructure-201801.pdf

NEI 2018b — *NEI February 22, 2018 Statement on HALEU* https://www.nei.org/news/2018/revamp-of-fuel-industry-support-advanced-reactors

Appendix A

Meeting Summary

SUBJECT: INL-NEI Invitation-Only Technical Workshop on Transportation of High Assay Low-Enriched Uranium

ORGANIZER: INL and NEI

AUTHOR: Gordon Petersen (INL)

DATE: August 30th and August 31st

PURPOSE: The primary objective of this workshop will be to advise DOE on the gaps related to transportation of HALEU and licensing support activities. The goal is to ensure that transportation and handling of HALEU at associated fuel cycle facilities does not delay the ability of advanced reactors to be deployed.

OVERVIEW: The meeting started with lunch provided by NEI. Everett Redmond from NEI then began the meeting by announcing safety procedures and letting all the attendees introduce themselves. He then went over the mission statement of the NEI Fuels Task Force and the letter sent to Secretary Perry by NEI specifiying the amount of HALEU needed over the next ten years. Josh Jarrell from INL took over and introduced the goals of the meeting and reiterated some of the questions Everett proposed. Over the next day, presentations were given by industry, national laboratories, and the NRC. Each presentation concluded with time to ask questions and have discussions. The first day concluded with a discussion in preparation for the NRC visit led by Nima Ashkeboussi. The second day was led off with a presentation from the NRC followed by discussion. Next the labs and industry continued presenting topics related to the capabilities and needs related to HALEU management. The second day concluded with a DOE perspective given by John Herczeg, industry/NEI recommendations for DOE led by Nima, and a wrap up of action items led by Josh. The following notes provide a short overview of the presentations given.

Industry provided information from an enrichment, licensing, and transportation perspective:

1. Capabilities exist for enrichment up to 20% (Melissa Mann/URENCO)
    a. Imperative to develop fuel cycle with consortium (fabricators, convertors, enrichers, reactor operators, transporters, etc.) approach for licensing framework
    b. Questions remain concerning transforming Cat III facility into Cat II facility and transportation off site
    c. Suggests engaging NRC and ANSI/ASTM standards now
2. Experience in licensing facilities with enrichments greater than 5.0 wt.% U^{235} and have transportation packages that can be amended for HALEU (Lon Paulson/GNF)
    a. GNFA Wilmington fuel fabrication facility
    b. Model RAJ-II Type B fissile package will require SAR update to transport HALEU
    c. Model NPC Type A fissile package will require SAR update to transport HALEU
    d. Licensing a new package takes 42 weeks minimum for NRC review, but start to finish takes ~5 years
3. Packages for shipping 20% enriched materials (Andy Langston/DAHER-TLI)
    a. Majority of DOE 20% enriched fuel shipped in drum type packages (Versa-Pac)
    b. Currently Versa-Pac is under NRC amendment application for 1S/2S cylinder
    c. 30B cylinder design up to 20% UF_6 enrichment currently under development
        i. 1600 kg
        ii. 30B-20 can be operated and handled in same way as 30B cylinder

        iii. Licensing overpack and cylinder with French, German, and NRC.
    d. Package for 5B/A cylinders under development
        i. VP-55XL is an enhanced version of the TLI's NRC approved VP-55
4. Licensing transport overpacks and packages with NRC (Rick Migliore/TN Americas)
    a. Little concern in ability to license/certify package
    b. Industry is not in position to create criticalitybenchmarks
    c. More concerned with licensing and packaging on the SNF side after the fuel is removed from the reactor

The labs presented on the following capabilities:

1. Nuclear Data and Benchmarking Program (Brad Rearden/ORNL)
    a. High uncertainties in cross sections with-in intermediate and high energy ranges
    b. Cross cutting program can support the needs of advance reactors
        i. Use correlation coefficients in trending analyses to determine cross section sensitivities
        ii. Perform gap analyses for non LWRs
    c. Mine existing experiments to determine similarities
2. INL could bridge material gap for 10 years (Monica Regalbuto/INL)
    a. Naval reactor fuel, EBR-II, and ZPPR plates can be available for downblending
    b. Issues may exist with uncertainties and dose of U-234
3. Nuclear Criticality Safety Program (Doug Bowen/ORNL)
    a. National Criticality Experiments Research Center (NCERC) best for 20% enrichment experiments
    b. Experiments are expensive and time consuming to setup and perform
        i. Cost→ \$425k-\$2.1M
        ii. Time frame→24-54 months
4. Validation discussion (John Scaglione/ORNL)
    a. Some techniques do not need experiments but can instead use physics-based solution
    b. Criticality validation process for ES-4100 package
        i. Requires detailed knowledge of the application system
        ii. Used similarity assessment to find how similar experiments were to target (Ck value)
        iii. Over 175 relevant experiments with C_k over 0.9 and just under 700 with Ck over 0.8, when considering HALEU UF_6 in the ES-4100 package. Therefore, optimism that experiments exist to defend future package designs for HALEU transport.

The NRC's also gave a short presentation followed by a discussion (Drew Barto/NRC)

1. Stressed the lack of information from >5% x <19.75% enrichment
2. Explained difficulty in changing existing regulation, especially regarding moderator exclusion for >5% enriched UF_6.
3. Gave timeline for expected review
    a. Complete entire process from day of acceptace of application to certifying in 7.4 months for 80% of transportation reviews and 2 years for all transportation reviews

ACTION ITEMS/IMPORTANT TAKE-AWAYS

1. DOE is committed to transportation of material regardless of form, and NEI will be be the focal point for prioritization of different strategies.
2. Although the labs can provide additional criticality experiments, industry has enough data to license facilities, overpacks, and cylinders. Validation to find more critical experiments to establish less uncertainty in the benchmarks will be helpful.

3. A collective effort from industry is needed to express consistency on how much information exists or is needed in regards to criticality.
4. NEI will change HALEU white paper concerning criticality.
5. NRC needs to validate methodology is applicable at >5% enriched.
6. NRC already has group that meets bi-weekly concerning HALEU.
    a. It will be very difficult and time-consuming to change NRC regulations

INDUSTRIES REQUEST FOR NEI, DOE, LAB COMPLEXES

1. DOE and the lab complex should communicate and educate the NRC on criticality issues related to HALEU.
2. INL should support work needed to certify package design for the transportation of HALEU.
    a. Suggest amending the COC of an existing package used for the shipment of commercial quantities.
    b. Suggest DOE provide funding to package designer(s) for analysis and engineering work for a package to be submitted to NRC for approval.
3. INL should provide specific, or a range, on the expected impurities that will be present in recycled naval fuel.
4. In the longer term, DOE and the lab complex should increase the availabilitiy of criticality benchmark data (i.e., by performing, sponsoring, or data mining additional criticality benchmarks) to further reduce conservatism in package design.

ATTACHMENTS

- Part I: Agenda
- Part II: Attendee List
- Part III: Presentations

Agenda
INL-NEI Technical Workshop on Transportation of High Assay Low-Enriched Uranium
August 30-31, 2018

Nuclear Energy Institute,
1201 F Street NW, Suite 1100
Washington, DC 20004
Room Clean Air A-B

August 30

12:00 pm	Lunch / Introductions / Goals of Meeting – Josh Jarrell, INL and Everett Redmond, NEI
1:00 pm	Validation and role of critical experiments and nuclear data – Brad Rearden, ORNL
1:45 pm	Potential for material recovery and form/dose/isotope concerns – Monica Regalbuto, INL
2:15 pm	Potential for enrichment up to 20% - Melissa Mann, URENCO
2:45 pm	Break
3:00 pm	Evaluation of HALEU fabrication issues – Lon Paulson, GEH/GNF
3:30 pm	HALEU UF6 transportation issues – Andy Langston, DAHER-TLI
4:00 pm	Cask supplier perspective – Rick Migliore, TN Americas
4:15 pm	Criticality sensitivity analysis – Brad Rearden, ORNL
4:45 pm	Discussion and preparation for NRC Visit – Nima Ashkeboussi, NEI
5:15 pm	Adjourn

August 31

8:30 am	NRC perspective – Drew Barto, NRC
9:30 am	Break
9:40 am	Criticality facilities and cost and process for new criticality experiments – Doug Bowen (ORNL)
10:10 am	Validation discussion: How important are material forms to establishing applicability of criticality experiments? Is there "common ground"? If not, what forms should be the focus? – John Scaglione, ORNL
10:30 am	Example validation process: How important is the application model (e.g., transportation package design, fuel form, and materials) and how are the criticality experiments used to establish appropriate bias and uncertainty? – John Scaglione, ORNL
11:00 am	DOE perspective – John Herczeg, DOE
11:10 am	Industry/NEI Needs and Recommendations for DOE – Nima Ashkeboussi, NEI
11:40 am	Wrap up /Action Items – Josh Jarrell, INL
12:00 pm	Adjourn

Part II

HALEU Tech Workshop

Thursday Aug. 30 and 31, 2018
12:00PM- 5:00PM 8:00AM-1:00PM

Last Name	First Name	Company
Blanton	Paul	SRNL
Bowen	Doug	ORNL
Bowers	Harlan	X-Energy
Caponiti	Alice	DOE
Carmichael	Ben	Southern Nuclear
Cummings	Kris	Curtiss Wright
Duskas	Andrea	DOE
Gillespie	Mary	DOE
Gresham	Jim	Westinghouse
Griffith	Andy	DOE
Hackett	Micah	Kairos
Herczeg	John	DOE
Jarrell	Josh	INL
Kliewer	Rod	Framatome
Knauf	Florie	Centrus
Krich	Rod	X-Energy
Kucukboyaci	Vefa	Westinghouse
Lane	Carol	X-Energy
Langston	Andy	Daher-TLI
Lell	Richard	ANL
Lutchenkov	Dmitiri	MPR
Mann	Melissa	URENCO
Migliore	Rick	Orano
Moss-Herman	Cheryl	DOE
Pappano	Pete	X-Energy
Paulson	Lon	GNF
Petersen	Gordon	INL
Rearden	Brad	ORNL
Regalbuto	Monica	INL
Scaglione	John	ORNL
Schilthelm	Steve	BWXT
Scott	C. Tyler	Westinghouse
Stucker	Dave	Westinghouse
Tweardy	Matt	NNSA
Waud	Brian	NNSA
Welling	Craig	DOE

Part III

NUCLEAR ENERGY INSTITUTE

Everett Redmond, Ph.D.
Nuclear Energy Institute

HALEU WORKSHOP

NEI FUELS TASK FORCE

- **Mission:** Lead industry efforts in identifying and resolving regulatory and policy issues for the development of the nuclear fuel supply chain for advanced reactors with an emphasis on challenges related to the utilization of high assay low enriched uranium.

INDUSTRY NEEDS

- Values in MTU
- Current fleet uses about 2000 MTU/year
- Letter to Secretary Perry July 5, 2018
- Data from eight companies
- Not all ARs or advanced fuels need HALEU

Year	Total	Cumulative
2018	0.026	0.026
2019	1.506	1.532
2020	2.21	3.7
2021	4.2	7.9
2022	3.7	11.6
2023	18.8	30.4
2024	10.3	40.7
2025	12.4	53.1
2026	57.4	110.5
2027	73.6	184.1
2028	108.1	292.2
2029	111.8	404.0
2030	185.5	589.5

NEI

QUESTIONS TO CONSIDER

- Will the fuel cycle process be similar to current fleet?

Mining → Conversion → Enrich → Fab → Reactor

- What differences might exist – material form, etc.?
- Should the task force engage publicly with NRC on the issues from this workshop?
- What other topics should the task force tackle?

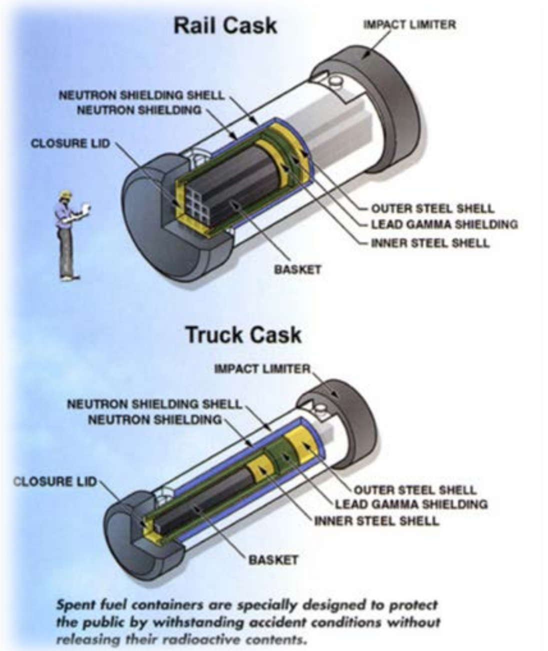
Transportation of HALEU Workshop – Introduction and Background

Josh Jarrell
Used Fuel Relationship Manager, INL
Josh.Jarrell@inl.gov
208-526-1614

Purpose of this workshop

- Develop and collect industry input and recommendations for future HALEU transportation needs
  - Avoid transportation delaying deployment of advanced reactors/fuels
- Focus is on large volume shipments of materials
  - Criticality is expected to be most challenging design aspect
  - Applicable to handling and storage of material at other facilities
- INL will be providing a “path forward” report to DOE by the end of September
  - Recommendations from this workshop will be included in this report

Depleted UF_6 storage cylinder (48” diameter)
http://web.ead.anl.gov/uranium/guide/prodhand/sld038.cfm



Type B packages for spent fuel
https://www.nrc.gov/reading-rm/doc-collections/fact-sheets/transport-spenfuel-radiomats-bg.html

Agenda

August 30

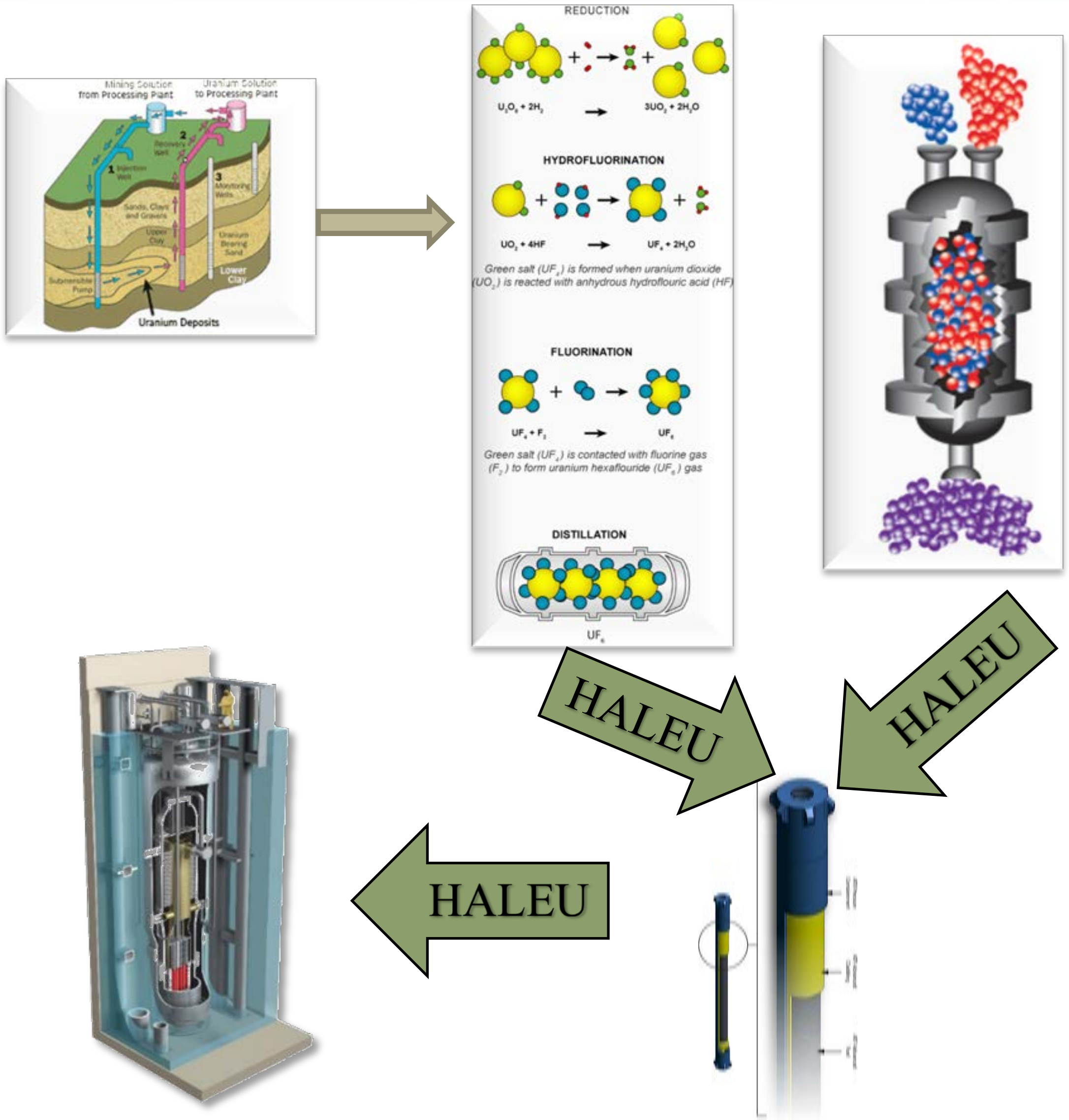
12:00 pm	Lunch / Introductions / Goals of Meeting – Josh Jarrell, INL and Everett Redmond, NEI
1:00 pm	Validation and role of critical experiments and nuclear data – Brad Rearden, ORNL
1:45 pm	Potential for material recovery and form/dose/isotope concerns – Monica Regalbuto, INL
2:15 pm	Potential for enrichment up to 20% - Melissa Mann, URENCO
2:45 pm	Break
3:00 pm	Evaluation of HALEU fabrication issues – Lon Paulson, GEH/GNF
3:30 pm	HALEU UF6 transportation issues – Andy Langston, DAHER-TLI
4:00 pm	Cask supplier perspective – Rick Migliore, TN Americas
4:15 pm	Criticality sensitivity analysis – Brad Rearden, ORNL
4:45 pm	Discussion and preparation for NRC Visit – Nima Ashkeboussi, NEI
5:15 pm	Adjourn

August 31

8:30 am	NRC perspective – Drew Barto, NRC
9:30 am	Break
9:40 am	Criticality facilities and cost and process for new criticality experiments – Doug Bowen (ORNL)
10:10 am	Validation discussion: How important are material forms to establishing applicability of criticality experiments? Is there "common ground"? If not, what forms should be the focus? – John Scaglione, ORNL
10:30 am	Example validation process: How important is the application model (e.g., transportation package design, fuel form, and materials) and how are the criticality experiments used to establish appropriate bias and uncertainty? – John Scaglione, ORNL
11:00 am	DOE perspective – John Herczeg, DOE
11:10 am	Industry/NEI Needs and Recommendations for DOE – Nima Ashkeboussi, NEI
11:40 am	Wrap up /Action Items – Josh Jarrell, INL
12:00 pm	Adjourn

Fuel Cycle Infrastructure for Advanced Reactors

- High assay low enriched uranium (HALEU) feed stock
  - Develop a domestic capability to enrich Uranium between 5% and 20%
  - Downblending current and/or recovered HEU in the federal complex
- HALEU Fuel Fabrication
  - Multiple fuel form options (metallic, oxide, nitride, etc.)
- HALEU Transportation
  - HALEU as UF_6 to fuel fabrication facility
  - HALEU fuel to reactor facility



Large volume shipments are anticipated

- By 2023, almost 20 MTUs of HALEU may be needed (NEI July 5, 2018 letter to Secretary of Energy)
  - Current UF_6 package (5A or 5B) hold **~24.9 kg of UF_6** or ~16.8 kg of 20% enriched uranium
  - Current UF_6 packages for 5% enriched (30B) hold **~2277 kg of UF_6**
  - DOE certified package (ES-3100) holds **~24 kg of UO_2** or ~21 kg of 20% enriched uranium
- 20 MTUs of UF_6 HALEU would require:
  - **~1191** 5A package shipments
  - **~13** 30B packages shipments (assuming 20% enriched was allowable)
- 20 MTUs of UO_2 HALEU would require:
  - **~953** ES-3100 package shipments

Figure 1. ES-3100 Container

ES-3100 Container
Deployment and Operation of the ES-3100 Type B Shipping Container, PVP2006-ICPVT-11, July 2006
https://www.osti.gov/servlets/purl/974250

Table 1 - Standard UF_6 Cylinder Data

Model Number	Nominal Diameter (in)	Material of Construction	Minimum Volume Ft^3	Approximate Tare Weight (Without Valve Protector) (lb)	Maximum Enrichment Wt% ^{235}U	Maximum Fill Lmit (lb UF_6)
1S	1.5	Nickel or Nickel-copper alloy[a]	0.0053	1.75	100	1.0[b]
2S	3.5	Nickel or Nickel-copper alloy[a]	0.0254	4.2	100	4.9[b]
5A	5	Nickel or Nickel-copper alloy[a]	0.284	55	100	54.9[b]
5B	5	Nickel	0.284	55	100	54.9[b]
8A	8	Nickel or Nickel-copper alloy[a]	1.319	120	12.5	255[b]
12A[c]	12	Nickel	2.38	185	5	460[b]
12B	12	Nickel or Nickel-copper alloy[a]	2.38	185	5	460[b]
30B[d]	30	Steel	26	1400	5[e]	5020[b]
48A[f]	48	Steel	108.9	4500	4.5[e]	21030[b]
48X	48	Steel	108.9	4500	4.5[e]	21030[b]
48F[f]	48	Steel	140	5200	4.5[e]	27030[b]
48Y	48	Steel	142.7	5200	4.5[e]	27560[b]
48T[g]	48	Steel	107.2	2450	1	20700[b]
48O[g]	48	Steel	135	2650	1	26070[h]
48OM[g] Allied	48	Steel	140	3050	1	27030[h]
48OM[g]	48	Steel	135	2650	1	26070[h]
48H[g]	48	Steel	140	3250	1	27030[h]
48HX[g] 48G	48	Steel	139	2650	1	26840[h]

[a] For example, Monel or the equivalent.

[b] Fill limits are based on 250°F maximum UF_6 temperature (203.3 lb UF_6 per ft^3), certified minimum internal volumes for all cylinders, and a minimum cylinder ullage of 5%. These operating limits apply to UF_6 with a minimum purity of 99.5%. More restrictive measures are required if additional impurities are present. This maximum temperature shall not be exceeded. It should be noted that initial cylinder heating may result in localized pressures above a normal UF_6 vapor pressure. This may be evidenced by an audible bumping similar to a water hammer.

[c] This cylinder is presently in service. New procurement should be model 12B.

[d] This cylinder replaces the Model-30A cylinder, which has a fill limit of 4950 pounds.

[e] These maximum enrichments require moderation control equivalent to a UF_6 purity of 99.5%. Without moderation control the maximum permissible enrichment is 1.0 wt% ^{235}U.

[f] Cylinder 48A and 48F are identical to 48X and 48Y, respectively, except that the volumes are not certified.

[g] This cylinder is similar to design to the 48G in that their design conditions are based on 100 psig at 235°F.

[h] Fill limits are based on 250°F maximum UF_6 temperature and minimum UF_6 purity of 99.5%. The allowable fill limit for tails UF_6 with a minimum UF_6 purity of 99.5% may be higher but shall not result in a cylinder ullage or less than 5% when heated to the cylinder design temperature of 235°F based on the actual certified volume.

Consider these questions:

1. What form(s) of HALEU should be considered in this scope (e.g., UF_6, oxide, metal)?
   - How important are the forms to additional criticality experiments? Is there “common ground” regardless of HALEU form?
   - What are the potential material pathways and transportation needs of (a) sources to (b) finished forms that need to be scoped out? E.g., HALEU UF_6 transported to a fuel fabrication facility, converted to metal fuel, which is subsequently transported to reactor site.

2. For a given form, what does an economic transportation package design look like? Truck cask? Rail cask? Amount of material? Construction materials? Absorber materials?
   - Are there licensed/certified packages that are suitable (domestic and international packages)?
   - Are there licensed/certified/designed packages that could be the basis for an economic HALEU package?

Consider these questions:

3. Are the current criticality benchmark experiments sufficient to justify certification of packages and licensing of facilities by the NRC?
   - If so, how much conservatism in the calculations do the current experiments cause? Can we quantify the volume-reduction / cost implications of this conservatism?
   - If not, what experiments should be proposed?
   - Can the existing critical experiment facilities perform the necessary benchmarks?
   - What changes to the existing safety bases need to be made, how long will that take, and how long is it good for?

4. What are the roles and responsibilities of the nuclear utilities, fuel fabricators, reactor developers, transportation package vendors, DOE, and the DOE laboratory complex?

U.S. DEPARTMENT OF ENERGY | Office of NUCLEAR ENERGY

Clean. Reliable. Nuclear.

Validation and Role of Critical Experiments and Nuclear Data

Presented by:
Bradley T. Rearden, Ph.D.
National Technical Director
Nuclear Data and Benchmarking Program

Presented to:
Technical Workshop on Transportation of High Assay Low-Enriched Uranium
August 30-31, 2018
Nuclear Energy Institute
August 30, 2018

Nuclear Data and Benchmarking Program

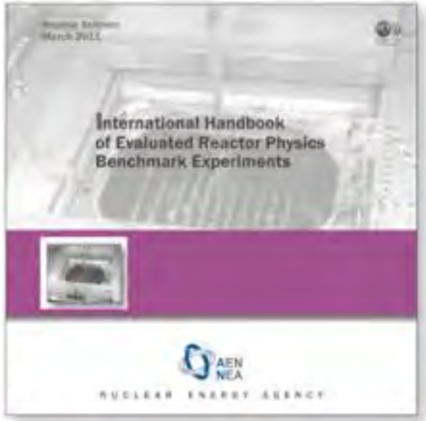
U.S. DEPARTMENT OF ENERGY | Office of Nuclear Energy

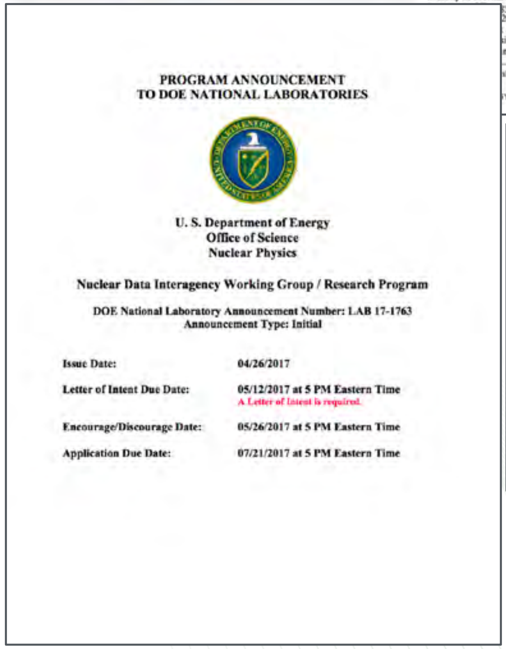
- New Nuclear Energy Enabling Technology (NEET) Crosscutting Program
- Partner with industry, NRC, and other programs to:
  - Identify priority needs for nuclear data and benchmarking
  - Perform new data measurements and evaluations
  - Support integral experiments and handbooks
  - Participate in application benchmark studies


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

FISSION REACTORS

INTRODUCTION



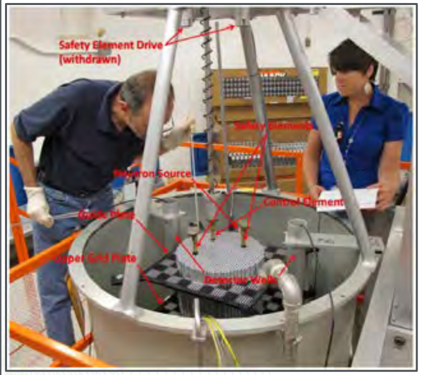

PROGRAM ANNOUNCEMENT
TO DOE NATIONAL LABORATORIES

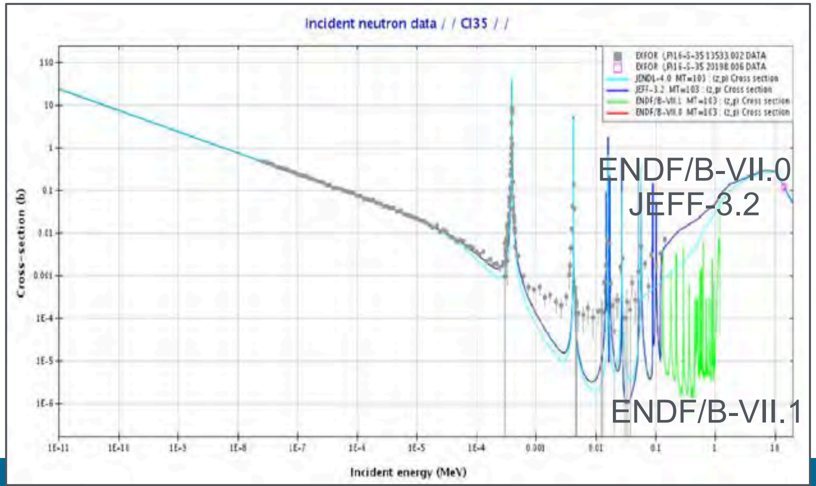
U. S. Department of Energy
Office of Science
Nuclear Physics

Nuclear Data Interagency Working Group / Research Program

DOE National Laboratory Announcement Number: LAB 17-1763
Announcement Type: Initial

Issue Date:	04/26/2017
Letter of Intent Due Date:	05/12/2017 at 5 PM Eastern Time A Letter of Intent is required.
Encourage/Discourage Date:	05/26/2017 at 5 PM Eastern Time
Application Due Date:	07/21/2017 at 5 PM Eastern Time





Abbreviated advanced reactor technology matrix (1/2)

Reactor Type	Companies Red = NRC Priority	Licensing action expected	Fuel / Enrichment	Thermal spectrum	Fast Spectrum	Coolant	Radial core expansion	Flowing Fuel	Fuel Form	Control elements
HPR	Oklo	2019	~20%		✓	Sodium heat pipes	✓		Metallic Castings	External drums
	Westinghouse (eVinci)	2019	19.75%	Thermal/ Epithermal		Sodium heat pipes (dual condenser)			Oxide	External drums
SFR	TerraPower (TWR)		~20%		✓	Sodium	✓		Metallic Rods	Internal rods
	GE PRISM		~20%		✓	Sodium	✓		Metallic Rods	Internal rods
LFR	Westinghouse		15-20%		✓	Lead	✓		Oxide/ Nitride	Internal rods
HTGR	X-energy (Xe-100)	2020s	15.5%	✓		Helium		Pebbles	TRISO	External rods
	Areva (SC-HTGR)		~20%	✓		Helium			TRISO	Internal rods
FHR	Kairos	2020s	~17%	✓		FLiBe		Pebbles	TRISO	External rods

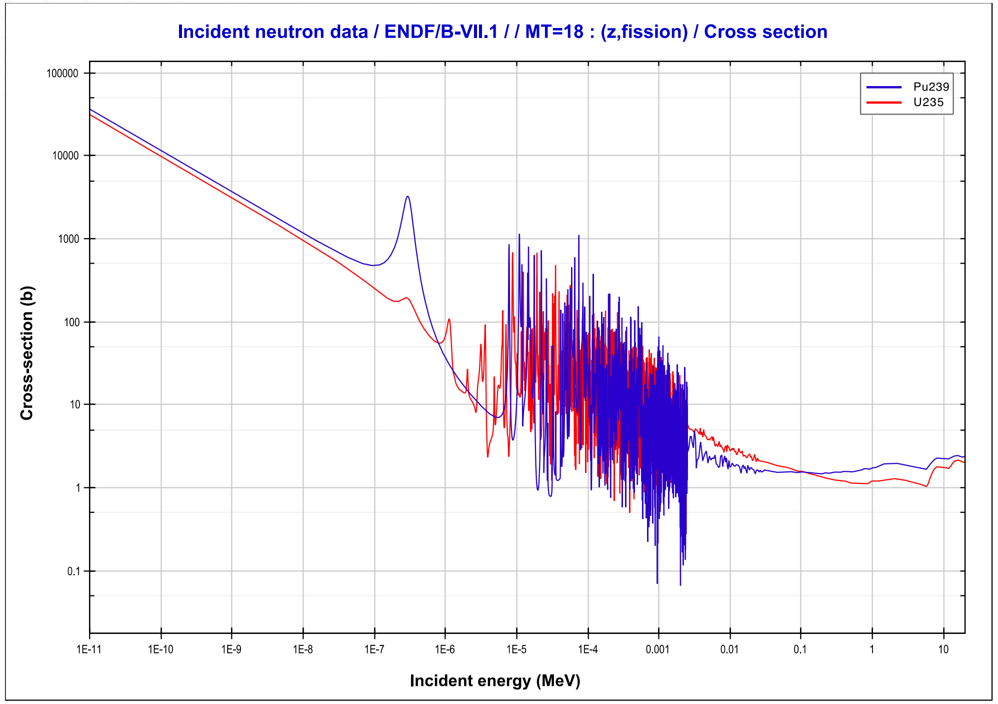
Abbreviated advanced reactor technology matrix (2/2)

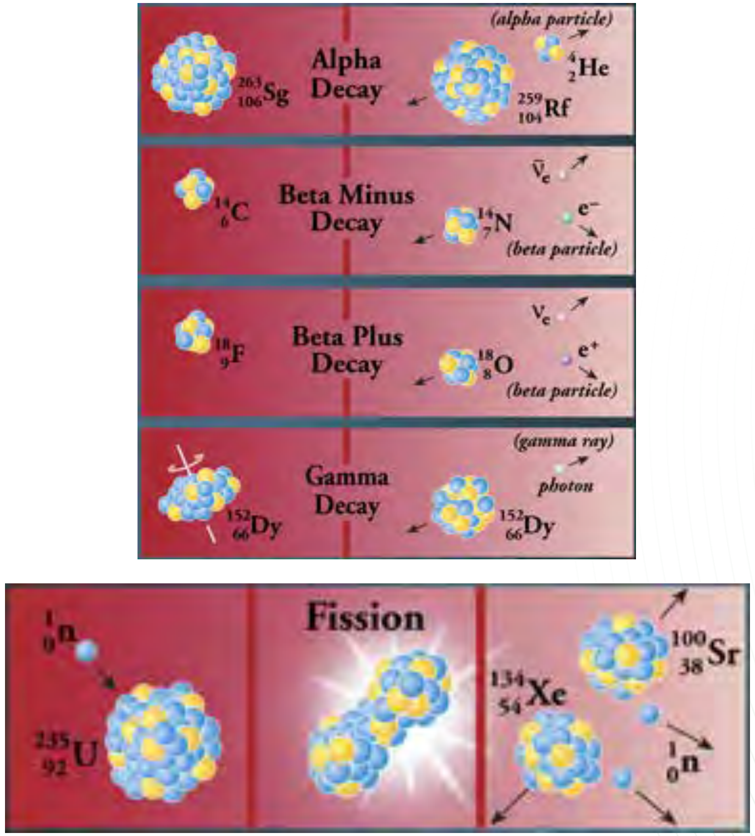
Reactor Type	Companies Red = NRC Priority	Licensing action expected	Fuel / Enrichment	Thermal spectrum	Fast Spectrum	Coolant	Radial core expansion	Flowing Fuel	Fuel Form	Control elements
MSR	Terrestrial Energy (IMSR)	2019	~5%	✓		Proprietary		Salt	Molten Salt	Internal rod
	Transatomic	2020s	~5%	Thermal/ Epithermal		FLiBe		Salt	Molten Salt	Internal ZrH moderating rods
	TerraPower (MCFR)	2020s	~20%		✓	Chloride salt		Salt	Molten Salt	External rods?
	Elysium		~20%		✓	Chloride salt		Salt	Molten Salt	
	FLiBe Energy		Thorium	✓		FLiBe		Salt	Molten Salt	Internal rods

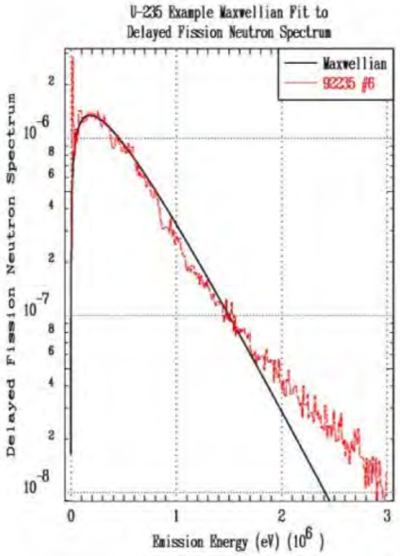
Send updates to Brad - reardenb@ornl.gov

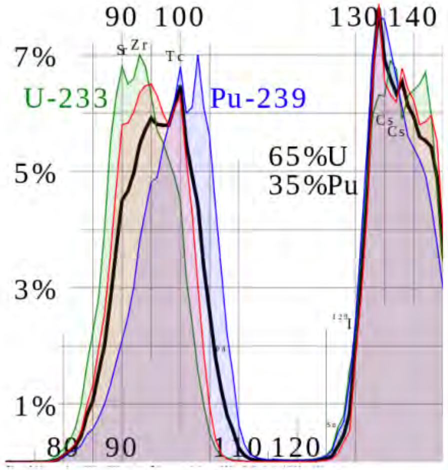
Nuclear data is of fundamental importance in nuclear science and engineering

Neutronics calculations rely on nuclear data for criticality, reactivity, power distributions, depletion, decay heat, and more.

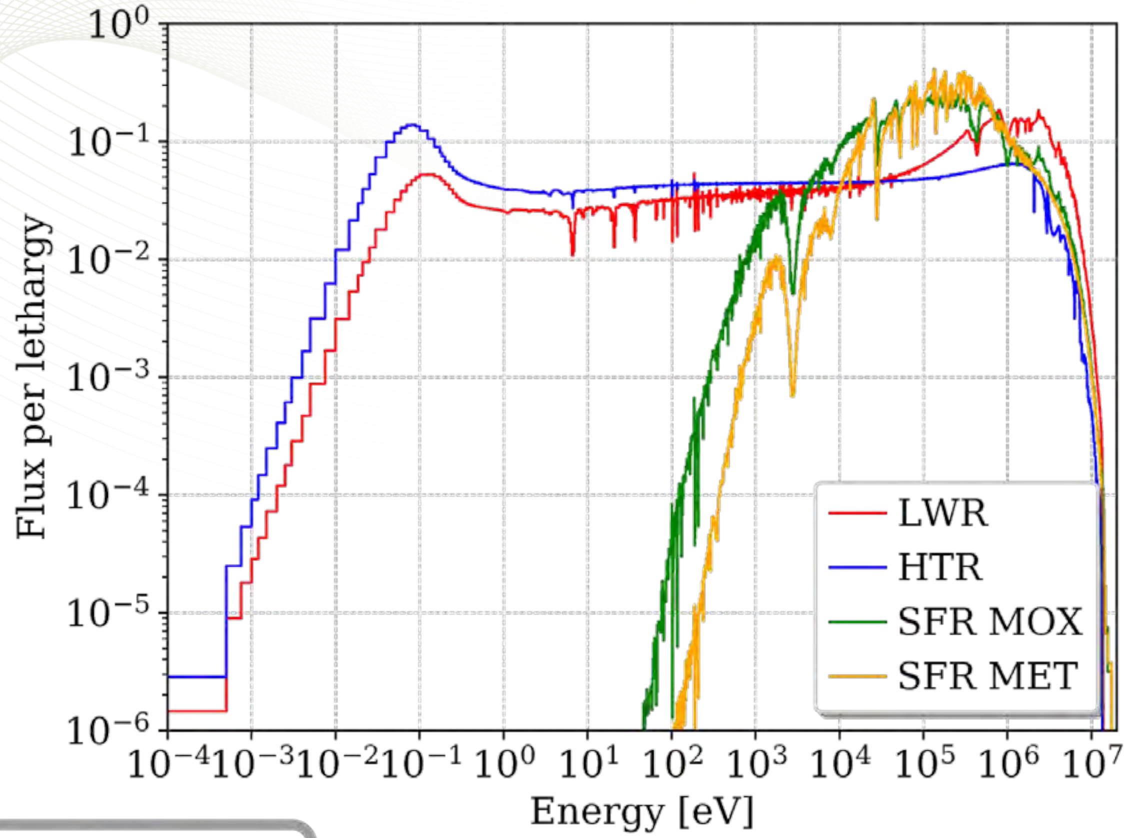


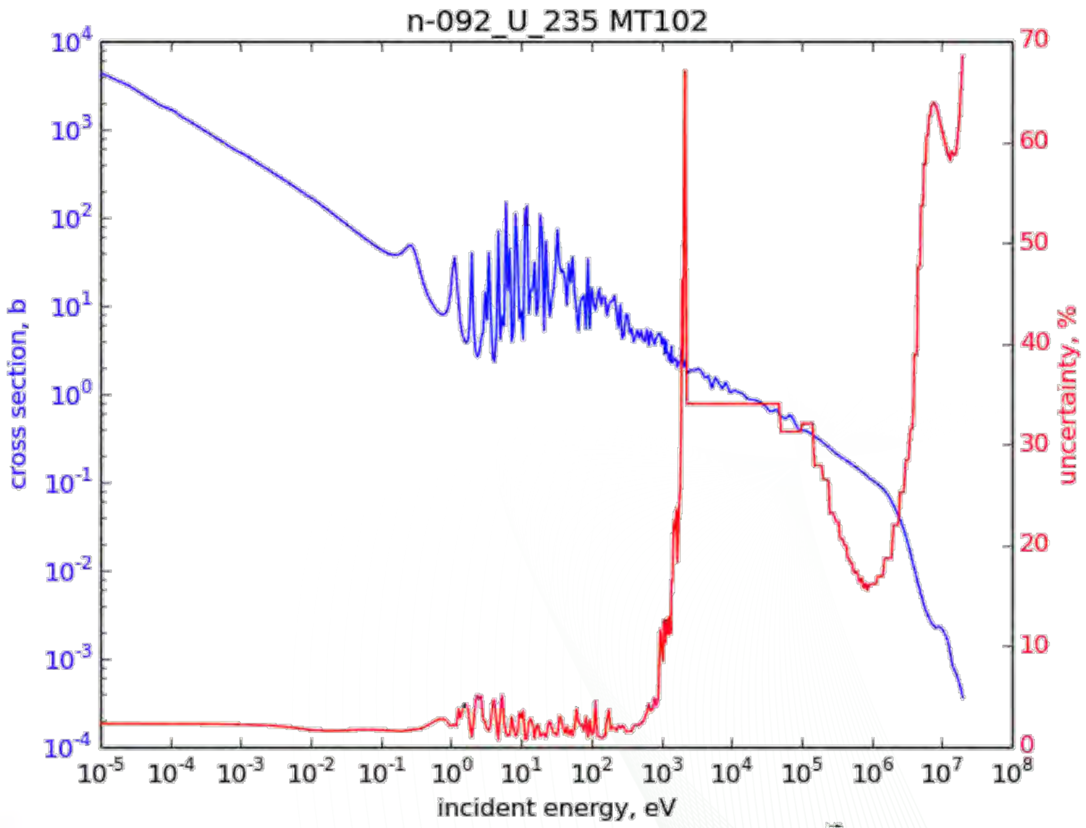


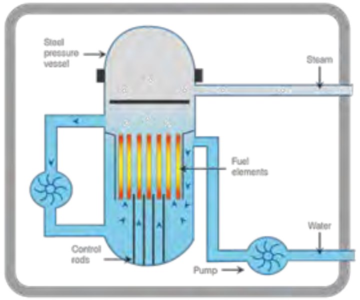


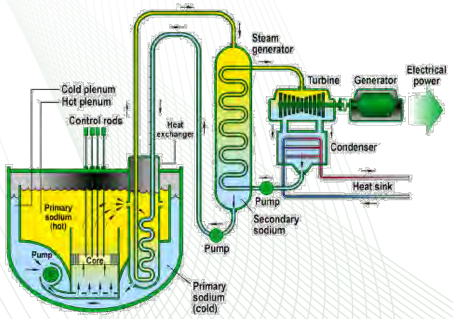


Different reactor designs have different nuclear data needs

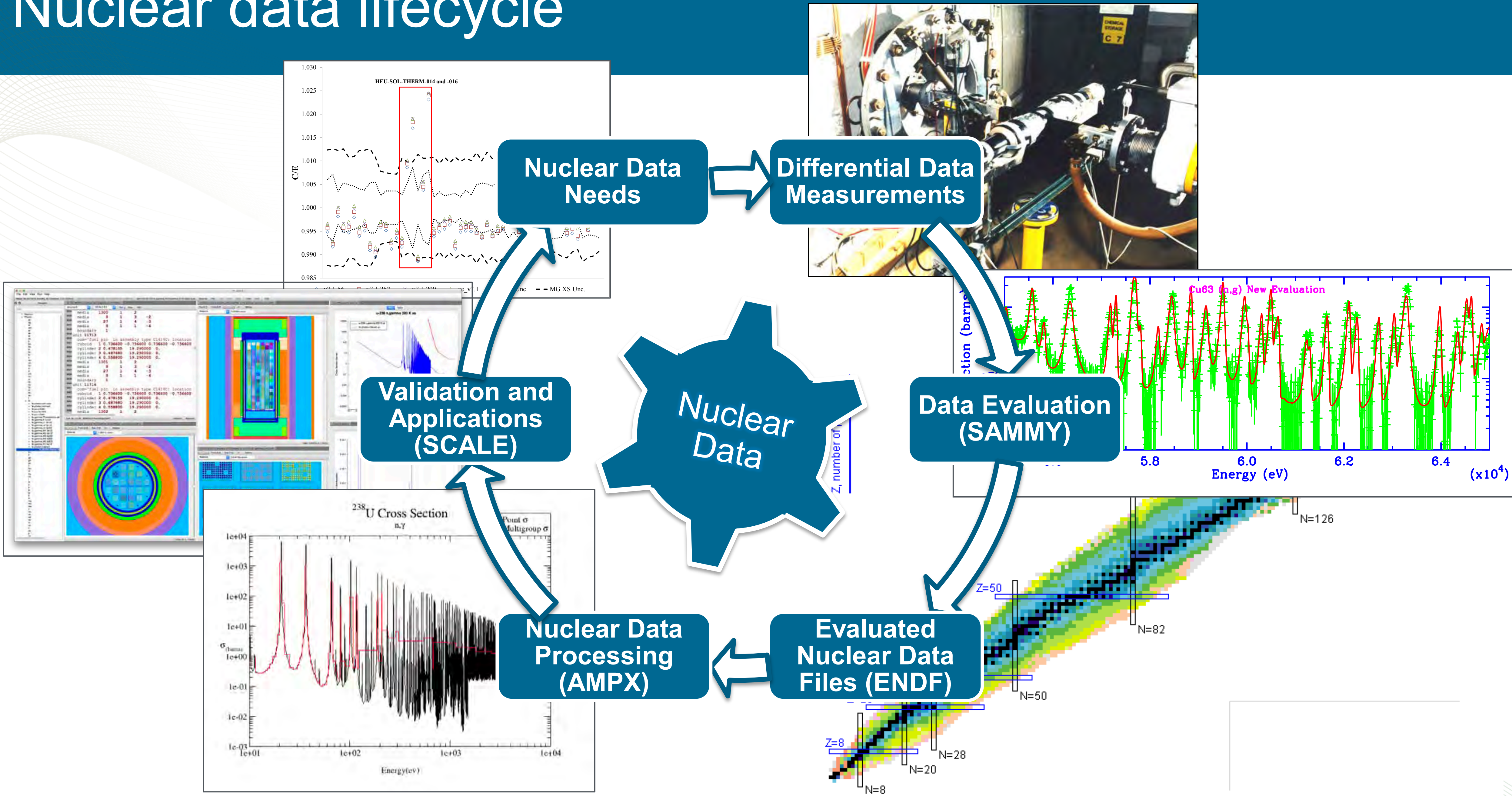




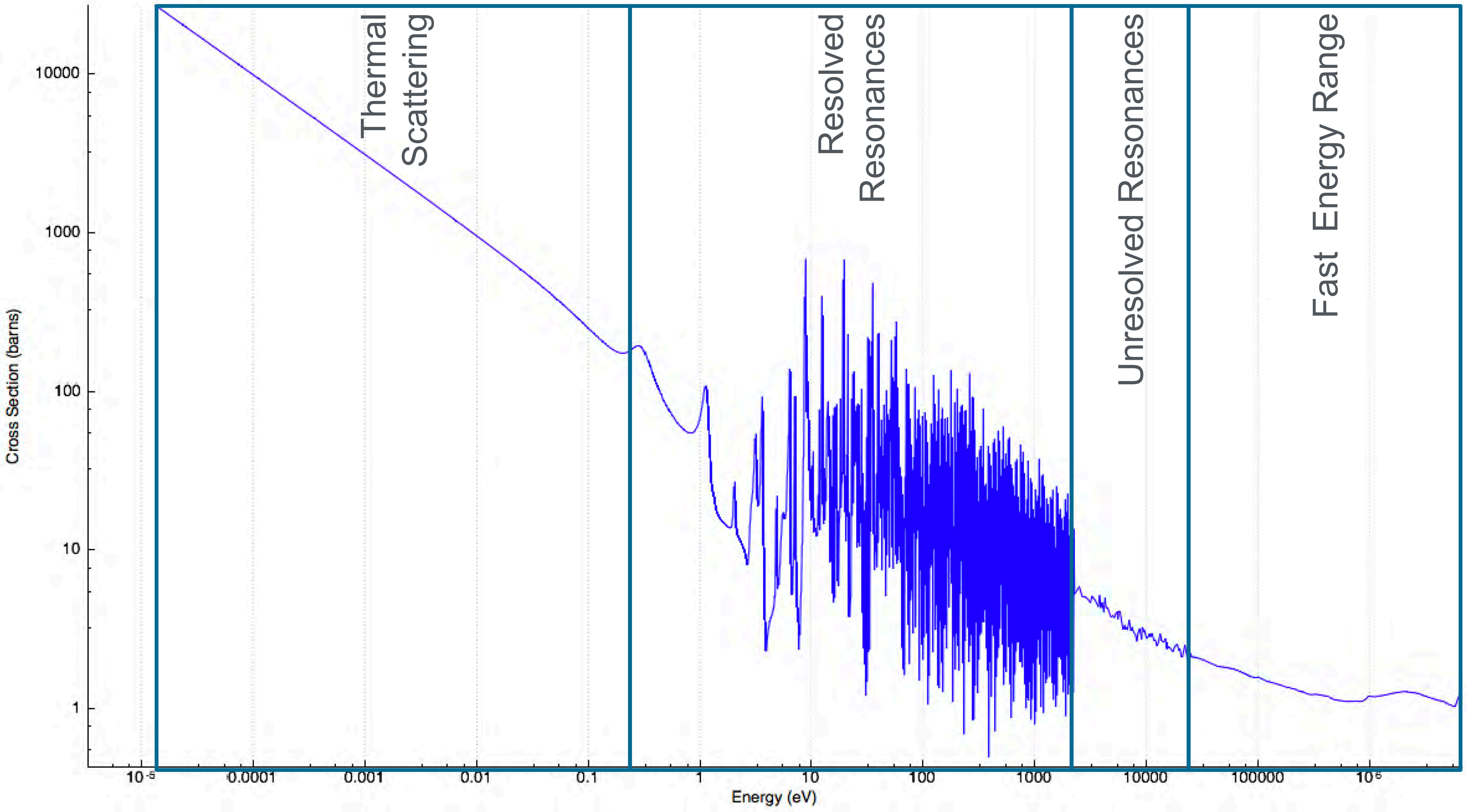




Nuclear data lifecycle



Cross section components:
Typically generated separately, then combined for distribution

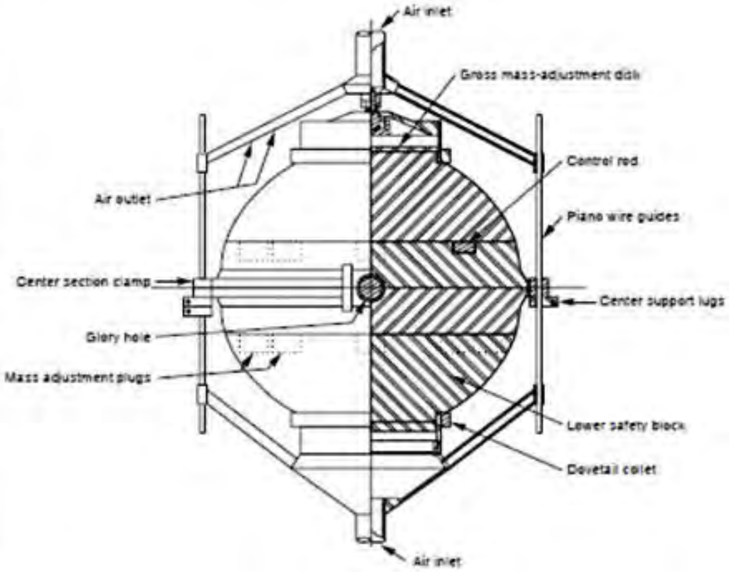


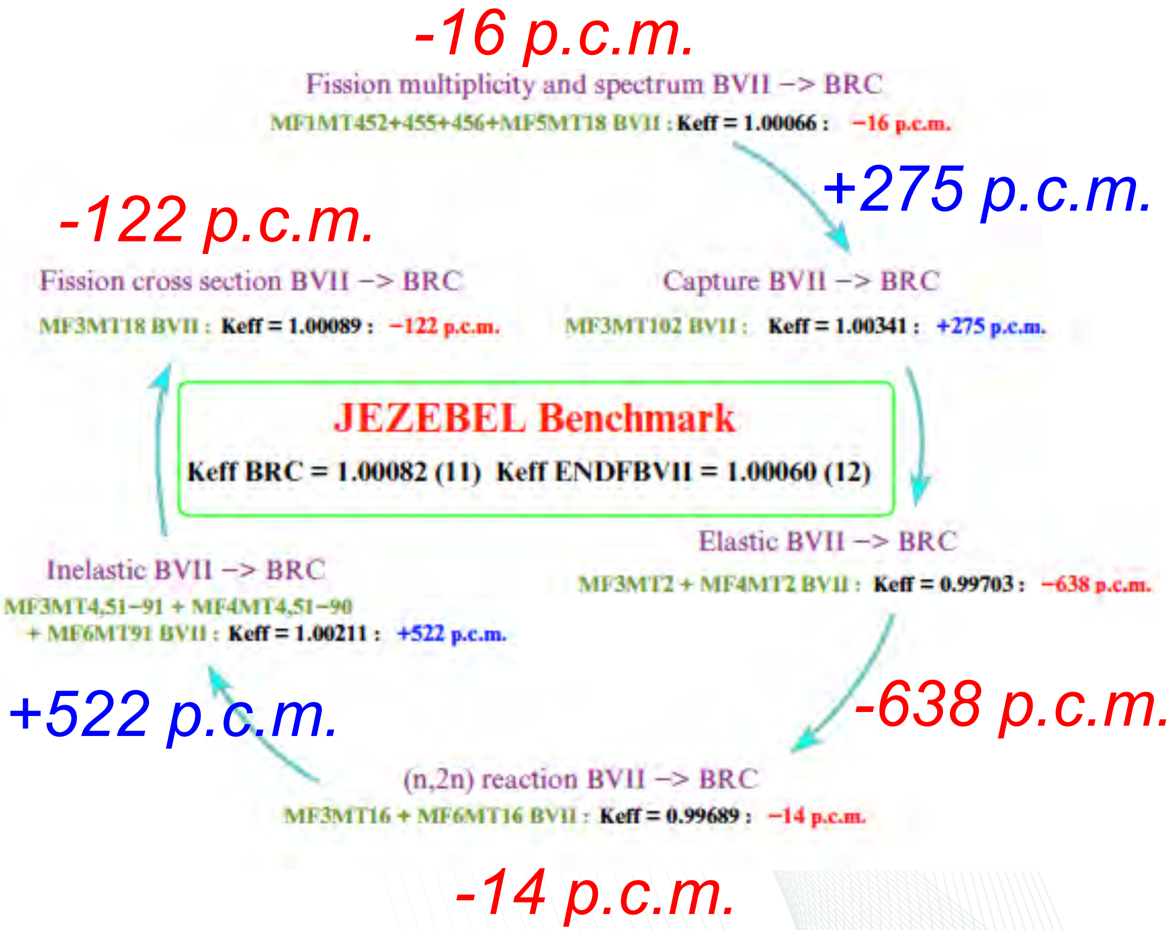
How are these "general purpose" libraries generated?

- A specific program (DOE-SC, NNSA/NCSP, NNSA/NA-22, DOD, international participant) funds an update in a nuclear data evaluation
  - New differential physics experiments
  - Data processing
  - Comparison to and **optimization with applications in their interest**
- National Nuclear Data Center - Cross Section Evaluation Working Group (CSEWG)
  - Updates are exchanged through a beta repository for ENDF and reviewed by a global team
  - Meets twice annually, with participation from IAEA, OECD/NEA, and others to review proposed updates
  - If changes benefit, or do not disrupt, applications of interest to these teams, the new evaluation is approved
- Until now, no official representation for Nuclear Energy applications

Compensating Errors in the Jezebel k_{eff}

- Eric Bauge* reported on an analysis where components of the Bruyères-le-Châtel (BRC) ^{239}Pu evaluation were replaced with those from ENDF/B-VII.1. At each step in the replacement process, k_{eff} of the Jezebel critical assembly was computed. While both the BRC and ENDF/B-VII.1 give the same k_{eff} for Jezebel, they do so for very different reasons. This replacement study shows how different parts of the evaluation substantially shift the reactivity of Jezebel. We do not know if either evaluation is "correct" but both get the "correct" answer.





*E. Bauge *et al.*, Eur. Phys. J. A (2012) 48: 113

We do not know if either evaluation is "correct" but both get the "correct" answer.

Generation of Cu evaluation for ENDF/B-VIII.0

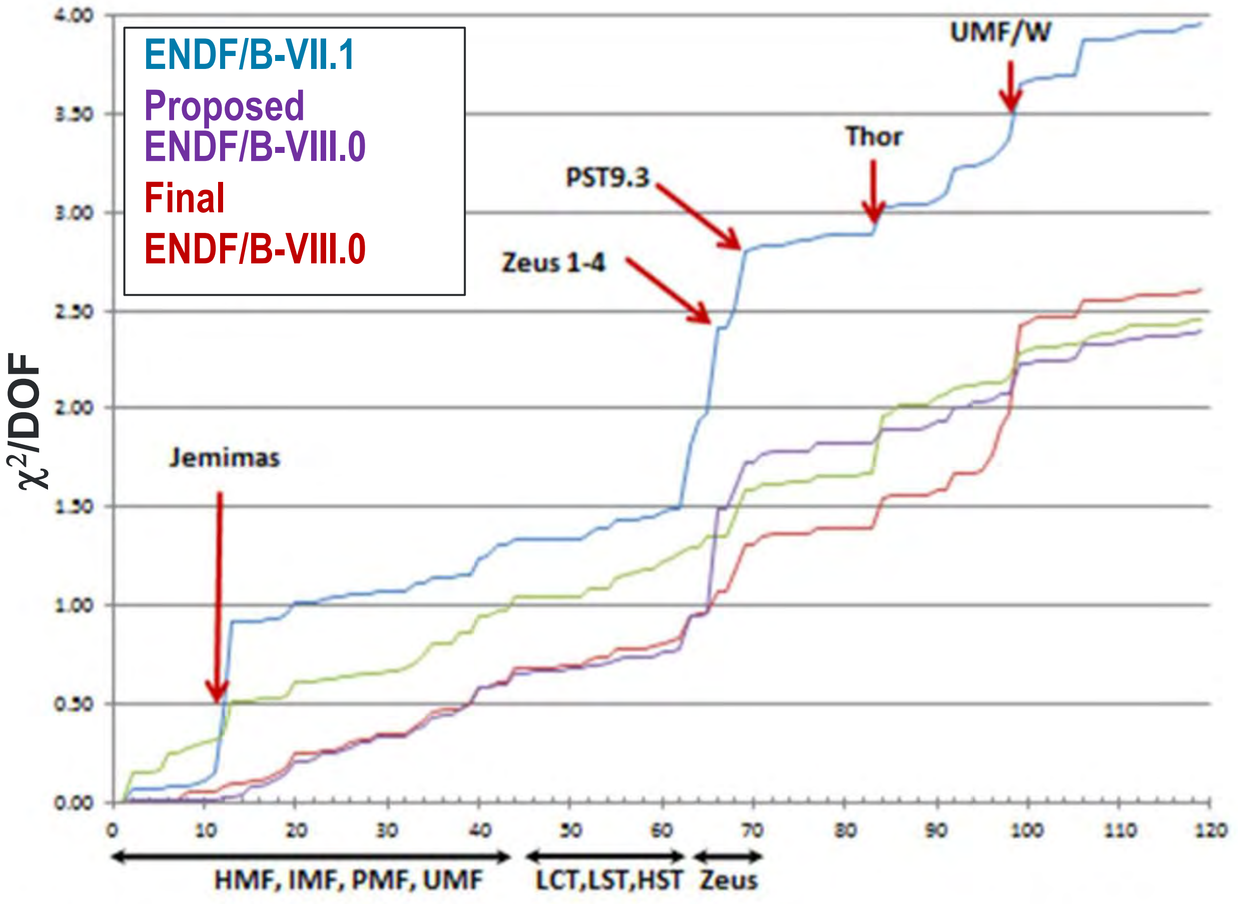
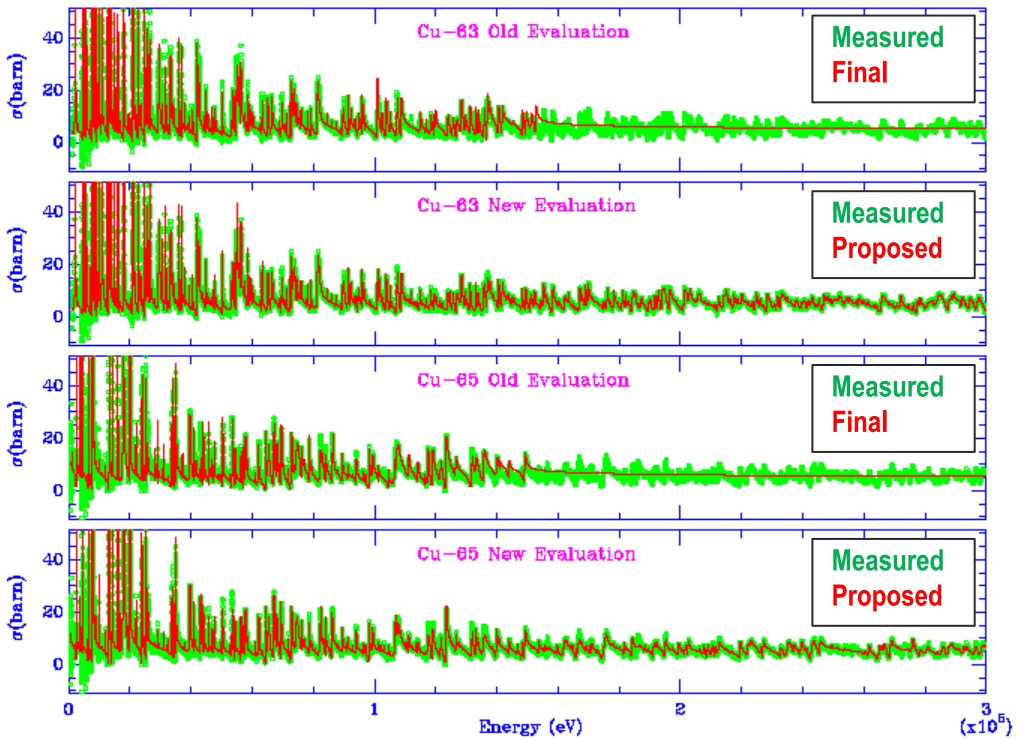
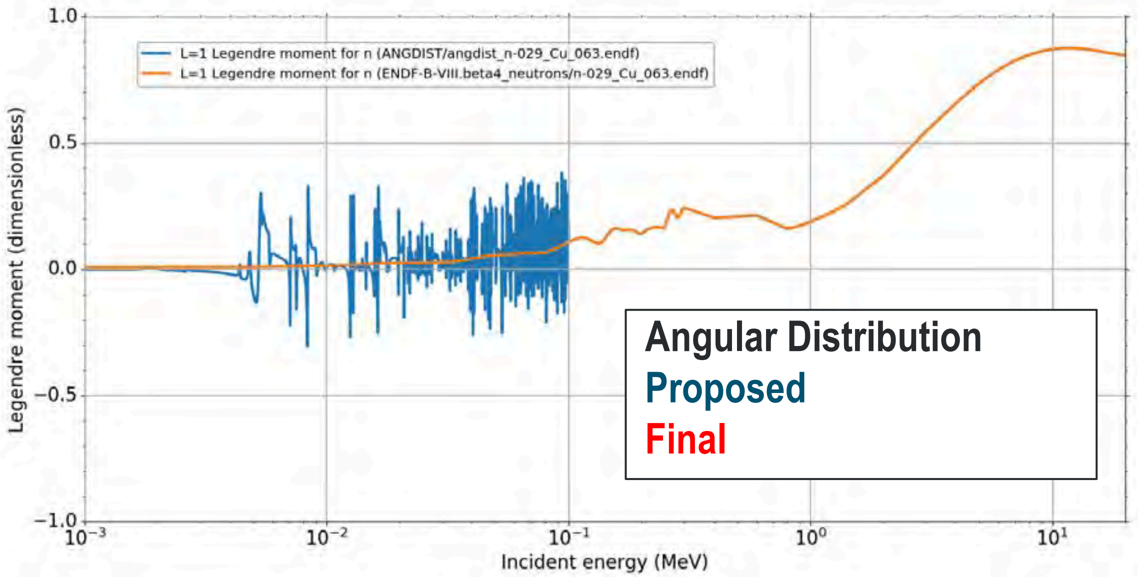


Figure 1: Cumulative Chi^2/DoF for the LANL suite of 119 benchmarks with different libraries.

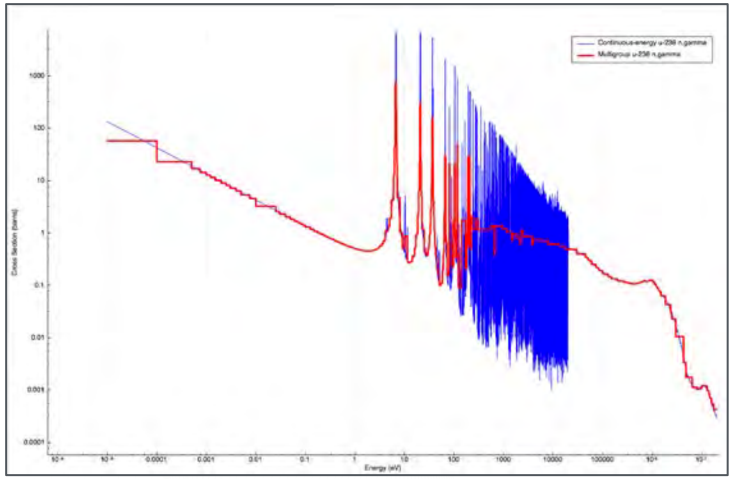


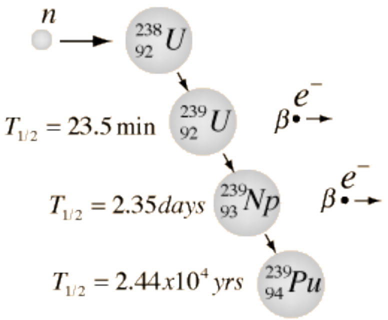


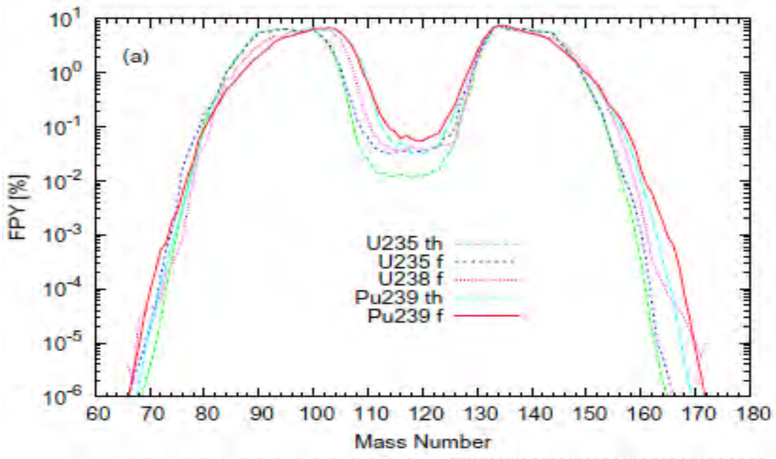
V. Sobes - ORNL

Nuclear data for activation, depletion, and decay

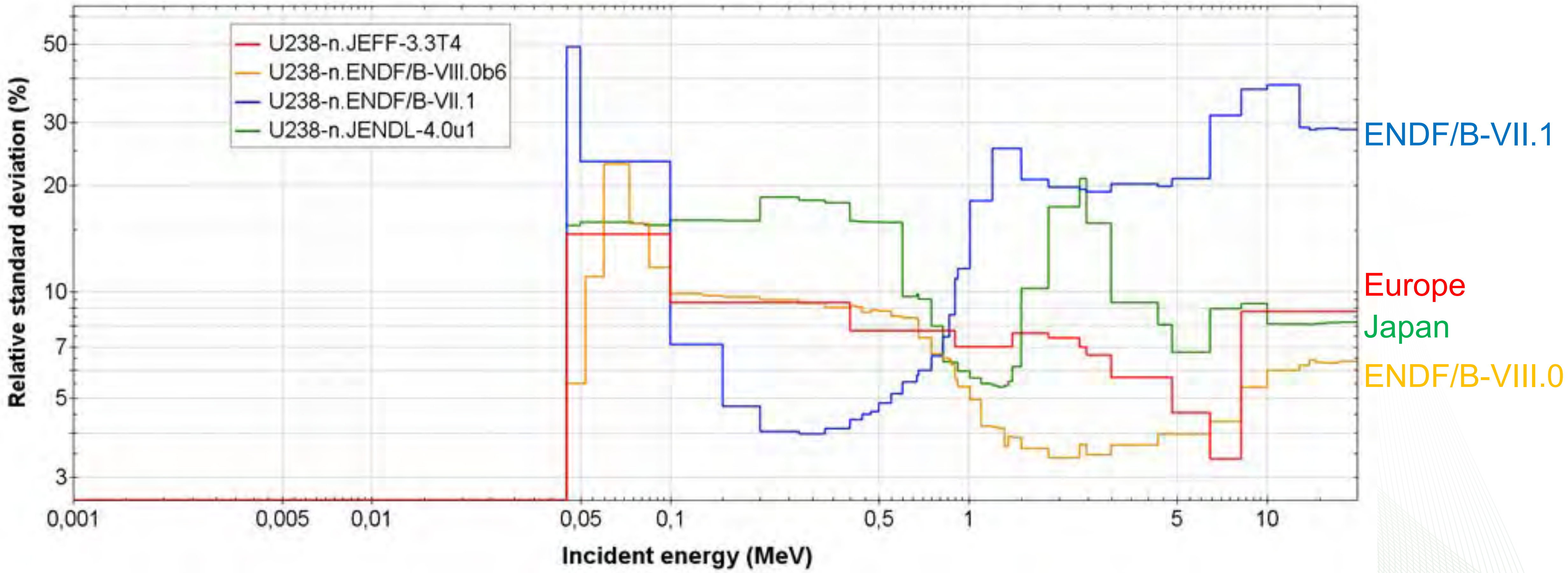
- **Decay data**
  - ENDF/B-VII.1
  - Natural isotopic abundances (NIST database)
  - ICRP 72 inhalation dose coefficients, EPA Report 12 on external exposure
- **Neutron reaction cross section data**
  - JEFF 3.1/A special purpose activation file
  - ENDF/B-VII.0, -VII.1
- **Fission product yields:** ENDF/B-VII.0
- **Photon emission line-energy data**
  - Evaluated Nuclear Structure Data Files (ENSDF)
  - ENDF/B-VII.1
- **Neutron emission libraries**
  - SOURCES 4C code
  - Spontaneous fission decay and delayed neutron data
  - Alpha stopping powers, (α,n) cross sections, excitation levels







^{238}U inelastic scattering cross section uncertainty differences between international libraries



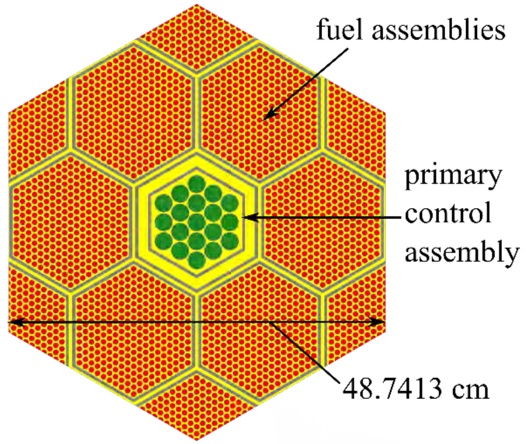
OECD Nuclear Energy Agency Uncertainty Analysis in Modeling sodium fast reactor study with ENDF/B-VII.1 uncertainties

CE TSUNAMI: nominal values and uncertainties

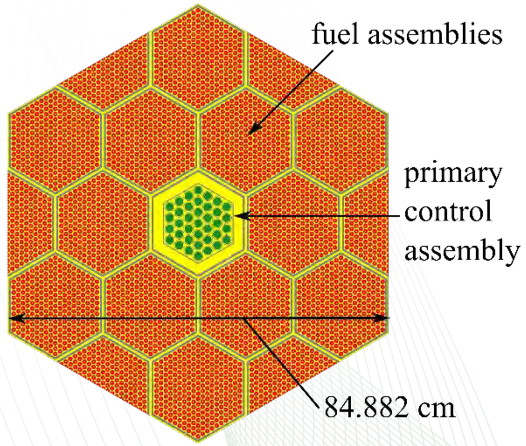
	MET1000		MOX3600	
	nominal	uncertainty	nominal	uncertainty
Eigenvalue	1.0841(1)	1.49(1)%	1.0771(1)	1.52(1)%
CR worth	12081(11) pcm	2.81(1)%	4973(11) pcm	2.67(1)%

CE TSUNAMI: Top 3 contributors

MET1000		MOX3600	
Eigenvalue	CR worth	Eigenvalue	CR worth
U-238 inel.	U-238 inel.	U-238 inel.	U-238 inel.
Fe-56 inel.	Fe-56 inel.	U-238 cap.	Na-23 el.
Na-23 el.	Na-23 el.	Pu-239 cap.	U-238 chi



MET1000



MOX3600

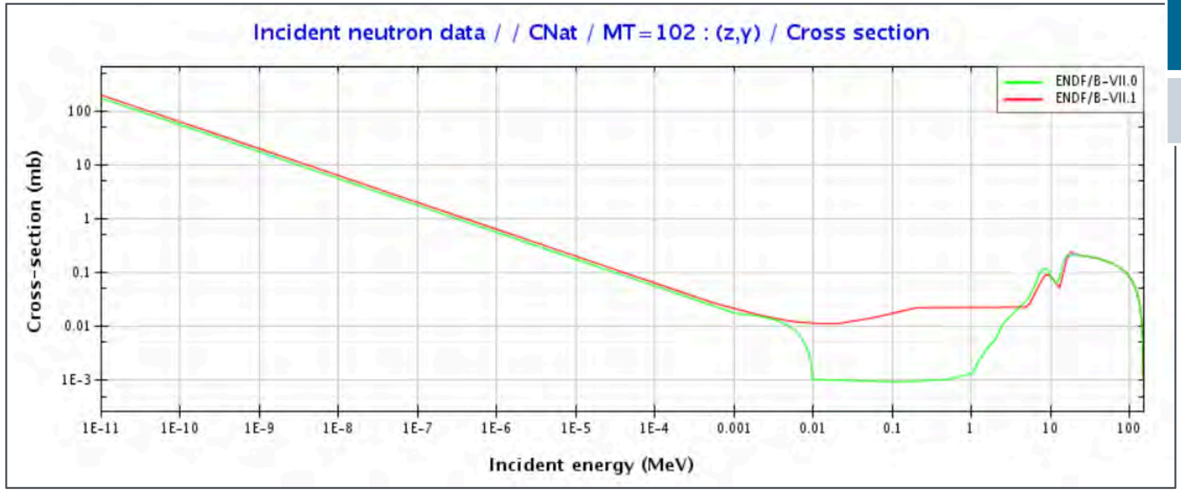
Recent nuclear data developments of interest to the advanced reactor community

Changes in graphite data

ENDF/B-VII.0 (2006) to ENDF/B-VII.1 (2011)

- Capture cross section increased from 3.36 mb to 3.86 mb: ~1,000 pcm

HTTR loading	ENDF-VII.0 C/E	ENDF-VII.1 C/E
Initial criticality	1.0165	1.0011
Full core	1.0097	1.0015

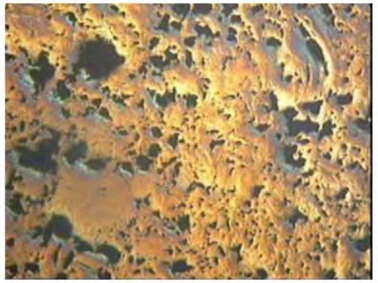


ENDF/B-VIII.0 (2018)

- New evaluations for thermal scatter based on molecular dynamics models from North Carolina State
- Includes data for crystalline and reactor-processed graphite

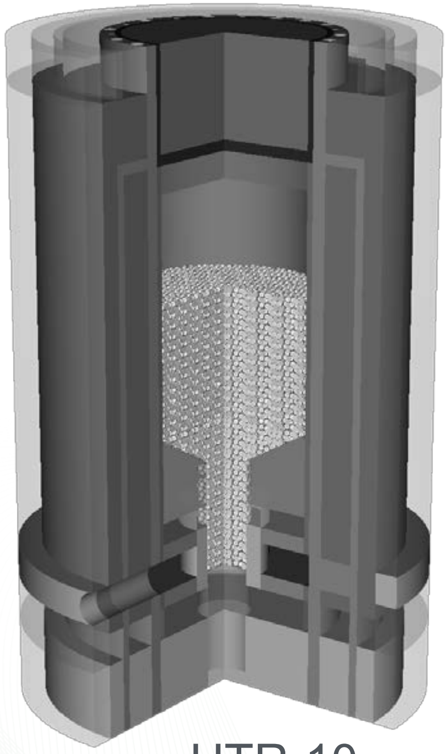
HTR-10 Configuration	ENDF-VII.1 C/E	ENDF-VIII.0 C/E
First core	1.00267	1.00582

Ideal Graphite
Density = 2.25 g/cm^3



Nuclear Graphite
Density = 1.5 – 1.8 g/cm^3

A. Hawari
NC State



HTR-10
Benchmark

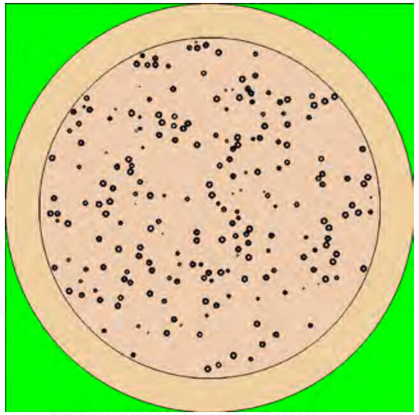
HTR-10 pebble: KENO-VI eigenvalue comparison

Library	Code	XS lib	k_∞	Δk (pcm)
ENDF/B-VII.1	KENO	CE	1.6770(4)	(ref)
ENDF/B-VIII.0	KENO	CE	1.6722(4)	−438(57)

Note: The 1σ statistical uncertainties are given in parentheses.

Replace individual nuclides in ENDF/B-VII.1 calculation by ENDF/B-VIII.0 data:

Basis: ENDF 7.1	Δk to all ENDF 7.1 (pcm)
But: graphite from ENDF 8.0	**−7**
But: ^{235}U from ENDF 8.0	**−702**
But: ^{238}U from ENDF 8.0	**239**
All ENDF 8.0	**−438**

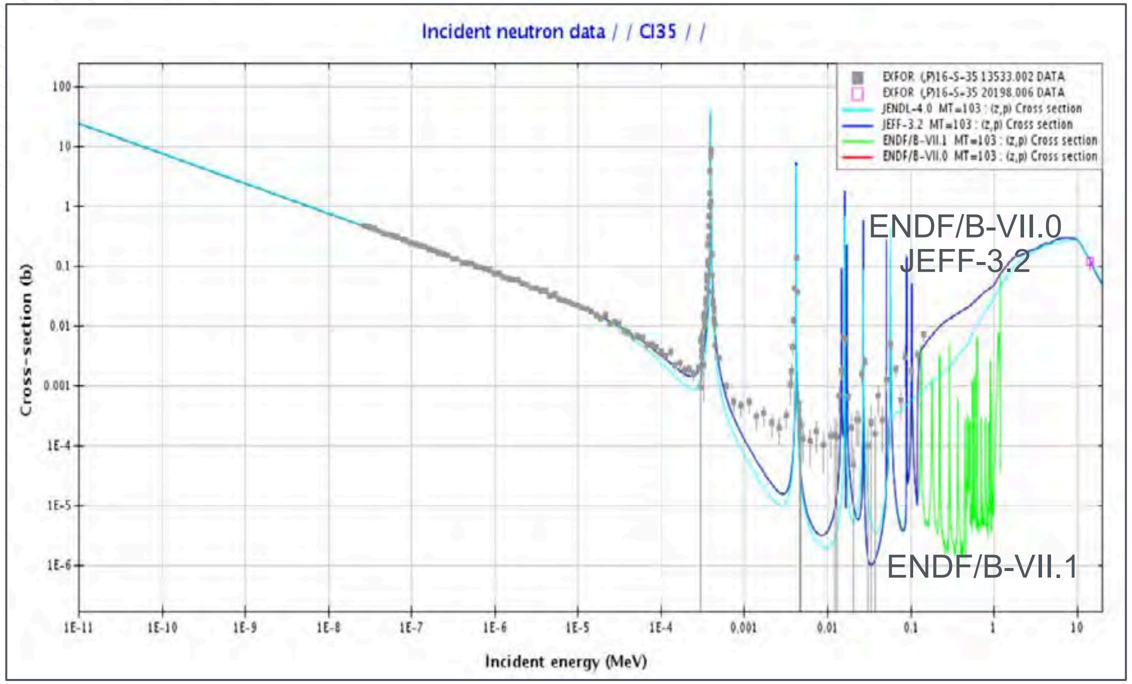


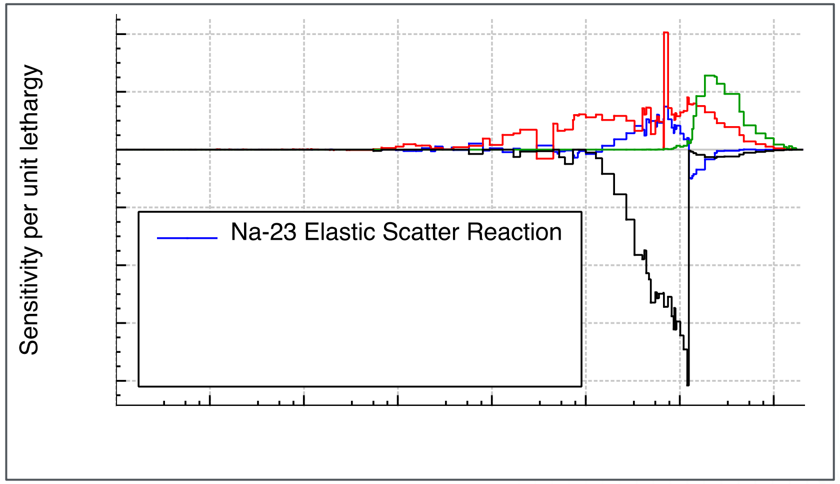
HTR-10 fuel pebble

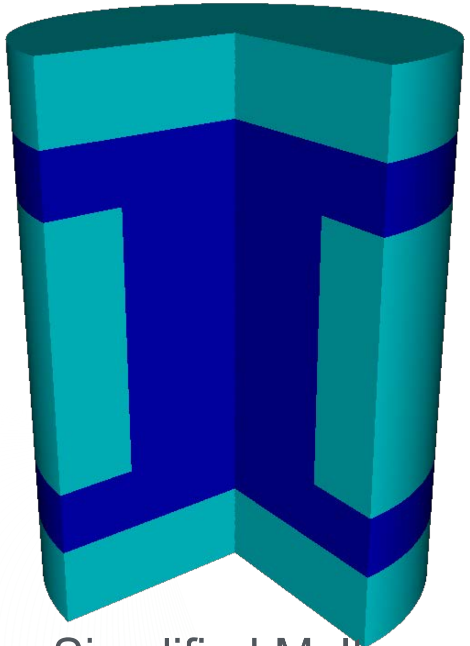
- Differences between ENDF/B-VII.0 and VII.1: carbon capture
- Differences between ENDF/B-VII.1 and VIII.0: ^{235}U and ^{238}U

Changes in ^{35}Cl(n,p) cross section from ENDF/B-VII.0 to VII.1

Data Library	k_{eff}
ENDF/B-VII.0	1.02993 ± 0.00002
ENDF/B-VII.1	1.04924 ± 0.00002







Simplified Molten Chloride Fast Reactor

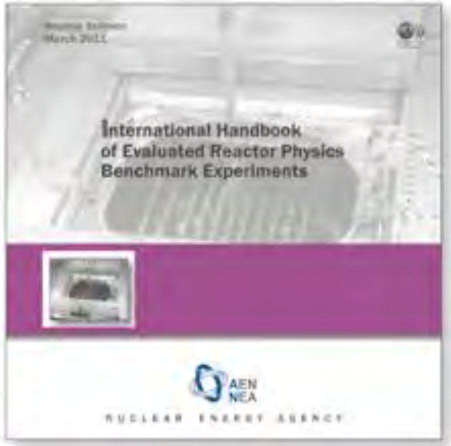
No data for FLiBe / FLiNaK thermal scattering

Reaction	Sensitivity
Cl-35 (n,p) Capture Reaction	-0.958
Pu-239 Nu-bar	0.603
U-238 Nu-bar	0.281
Na-23 Elastic Scatter Reaction	0.114

Validation of methods and nuclear data for advanced applications

International benchmark evaluation projects

- Programmatic support for US leadership of the following projects:
  - International Criticality Safety Benchmark Evaluation Project (ICSBEP)
  - International Reactor Physics Benchmark Evaluation Project (IRPhEP)
- Handbooks generated by these projects provide thousands of benchmark experiments from dozens of countries with an assessment of data integrity, quantification of experimental uncertainties, and thorough technical review with established deployment process
- Strong collaborations have been implemented with the Organisation for Economic Cooperation and Development (OECD) Nuclear Energy Agency (NEA)



ICSBEP

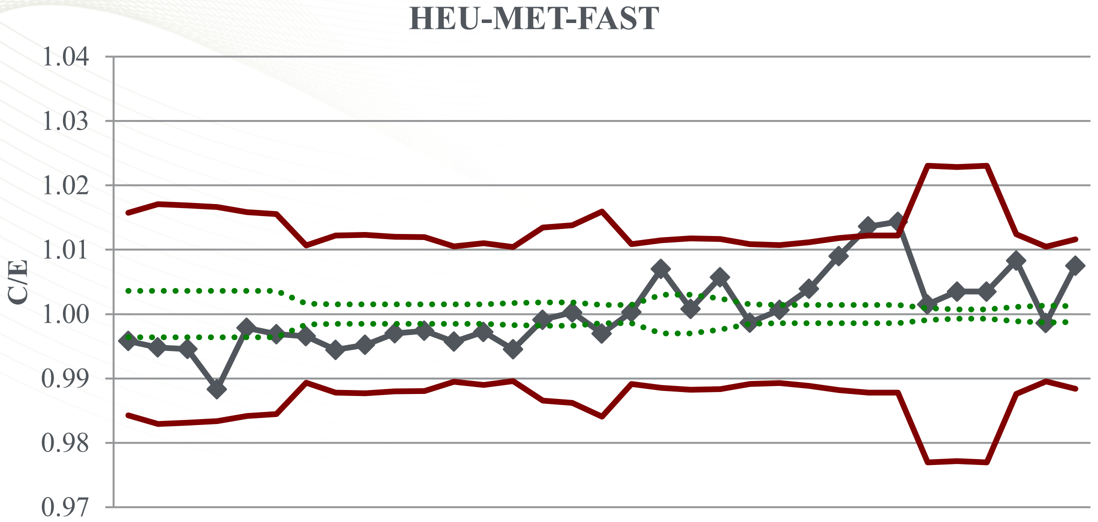
- **22 contributing Countries**
- **~69,000 pages**
- **>5,000 approved benchmarks**

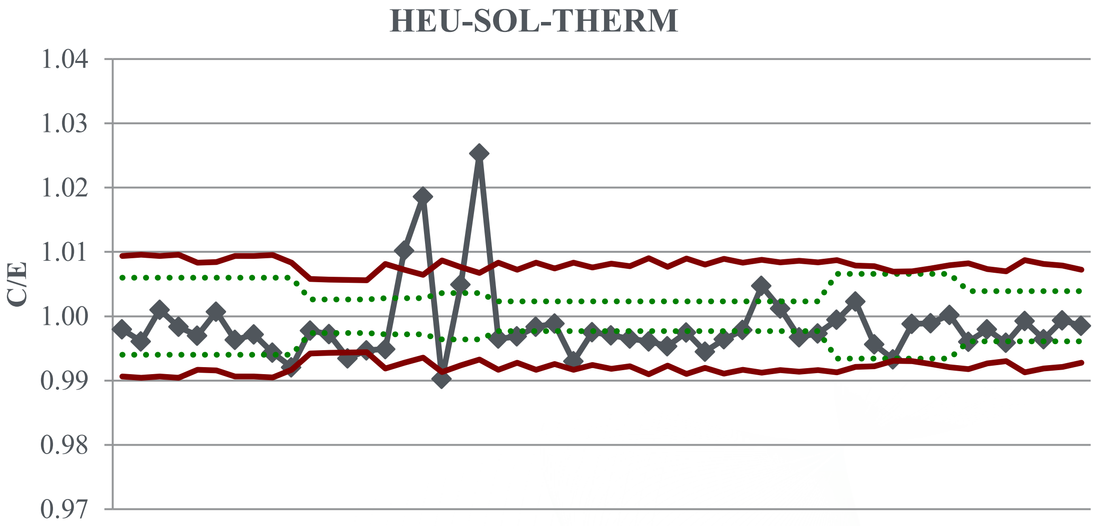
IRPhEP

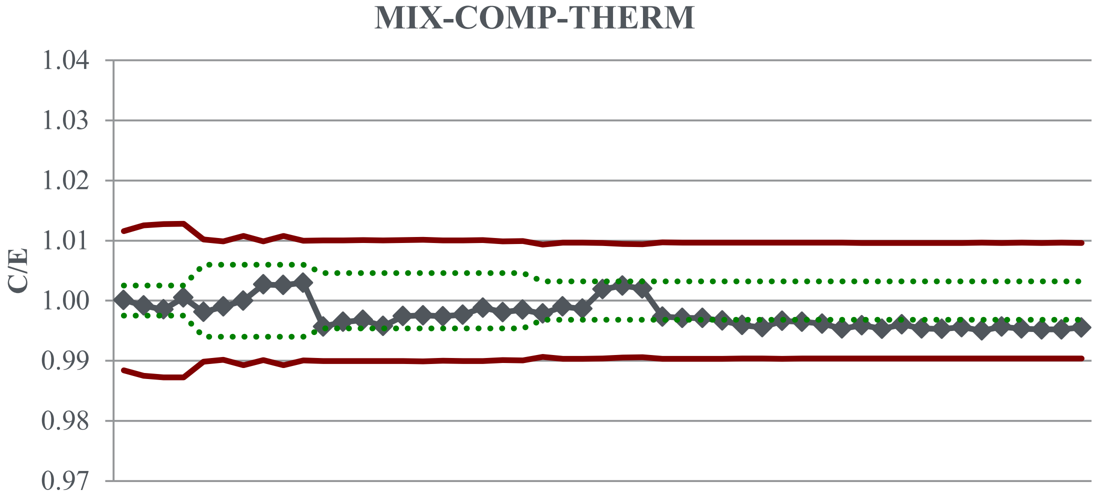
- **21 contributing countries**
- **50 reactor facilities**
- **147 approved benchmarks**

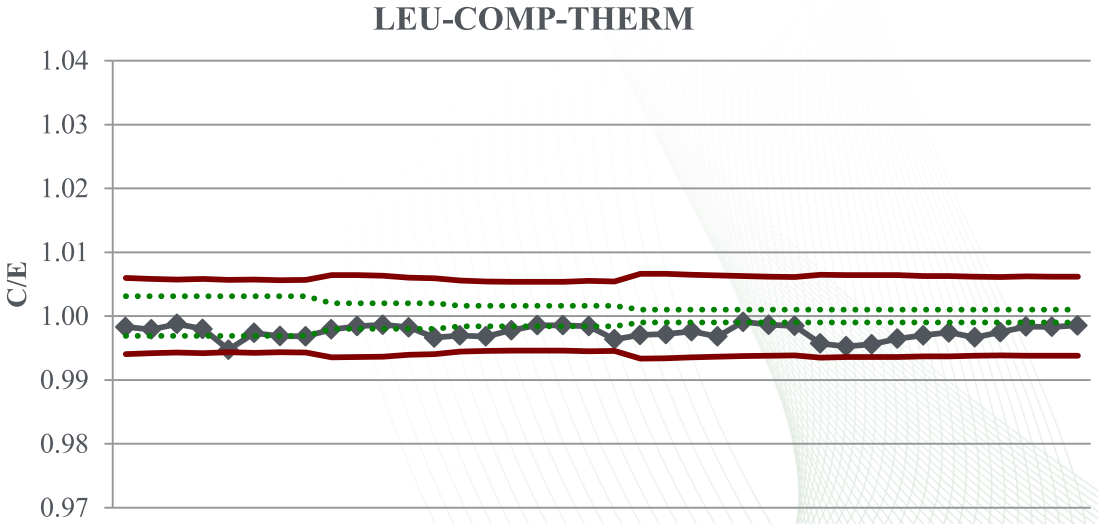
Computational bias for critical benchmarks

Computational Bias
Experimental Uncertainty
Cross-section Uncertainty





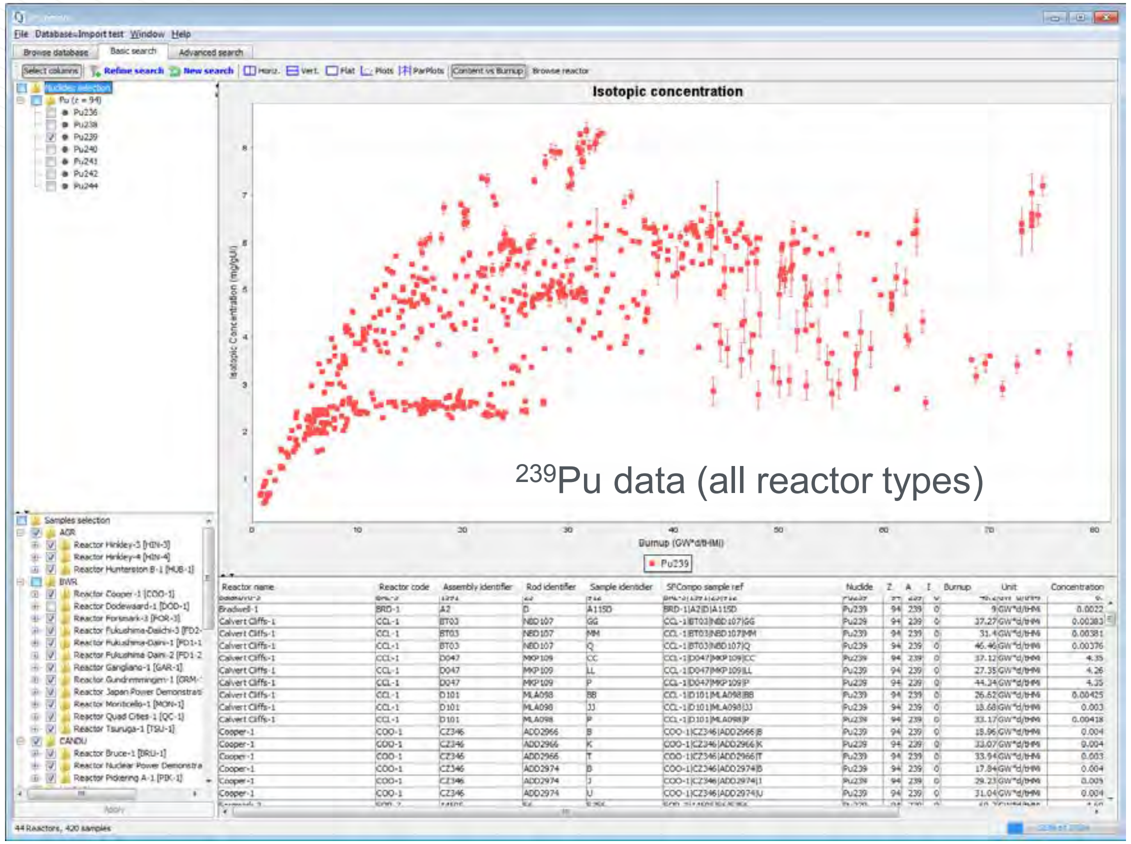




International Spent Nuclear Fuel Database SFCOMPO 2.0 provides a central repository of destructive assay data

Modern database of measured fuel compositions was expanded as part of a multi-year international collaboration. ORNL has coordinated this effort through the OECD/NEA Expert Group on Assay Data for Spent Fuel since 2009.

http://www.oecd-nea.org/sfcompo/



^{239}Pu data (all reactor types)

- Databases maintained by OECD Nuclear Energy Agency Data Bank include:
  - ICSBEP (Criticality safety database)
  - IRPhEP (Reactor physics database)
  - SFCOMPO (Spent fuel composition and decay heat database)
- Data for PWR, BWR, AGR, MAGNOX, CANDU, RBMK, VVER-440, VVER-1000 fuels
- 44 reactors, 118 assemblies, 91 isotopes important to fuel cycle safety and WM
- 750 samples > 22,000 measurements
- Data essential for code validation and uncertainty analysis, integral nuclear data testing -- Energy and Security applications

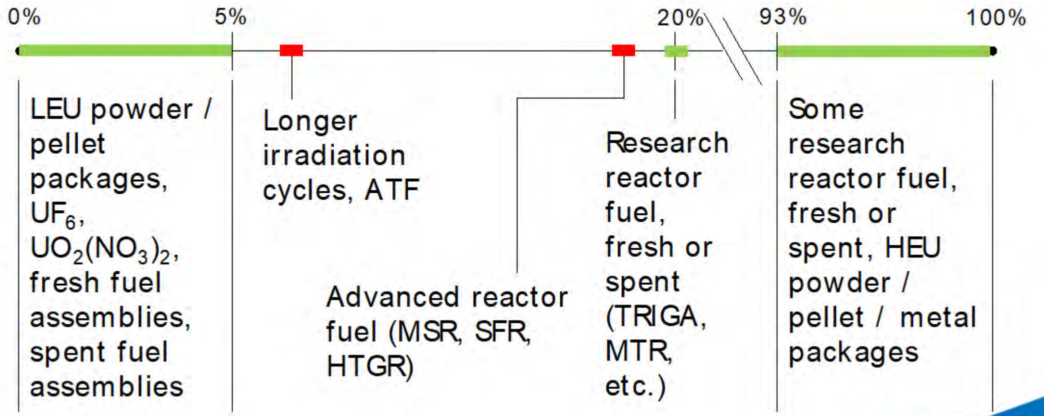
5% < Hi-assay LEU < 20%

NRC/NMSS perspectives on high assay fuel

> 5.0 Weight Percent

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

Code Validation:



NMSS Division of Spent Fuel Management

6

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

NRC Regulatory Perspective on Criticality Safety in Fissile Material Transportation and Spent Fuel Storage

Drew Barto
Criticality Shielding and Risk Assessment Branch
Division of Spent Fuel Management
US NRC

American Nuclear Society Winter Meeting
Washington, DC
October 30, 2017

NMSS Division of Spent Fuel Management

1

Part 71/72 Interface

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

High-Capacity PWR Cask Criticality Safety Criteria:

Storage:
- < 5.0% Initial enrichment
- Minimum soluble boron during loading

Transportation:
- < 5.0% Initial enrichment
- > 45 GWd/MTU burnup
- Cooling time
- Limits on irradiation parameters:
  - Soluble boron
  - Specific power
  - Moderator temp.
  - Fuel temp.

NMSS Division of Spent Fuel Management

7

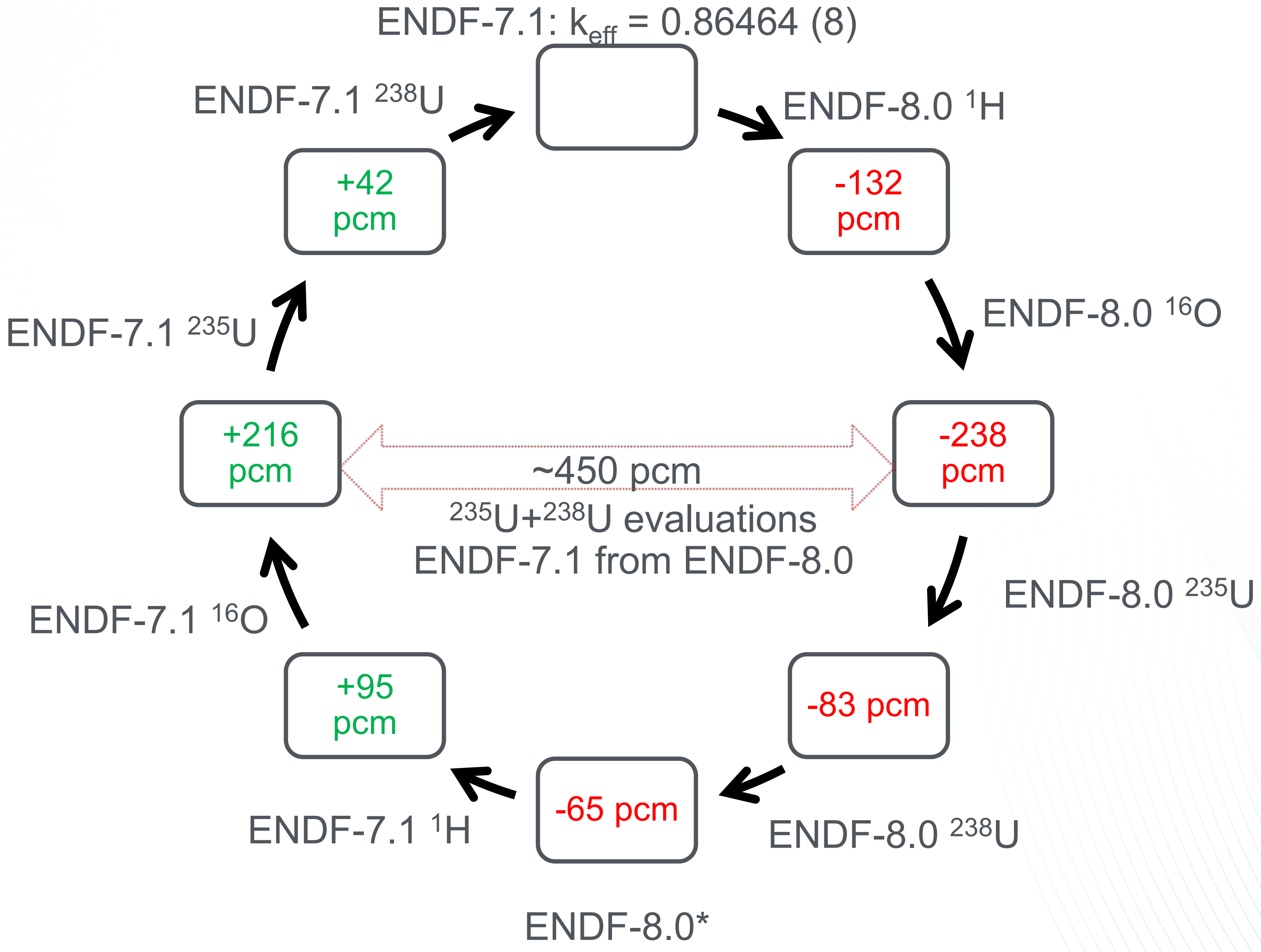
Example criticality validation process using the ES-4100 package

Containment vessel

Photos Courtesy of Jeff Arbital

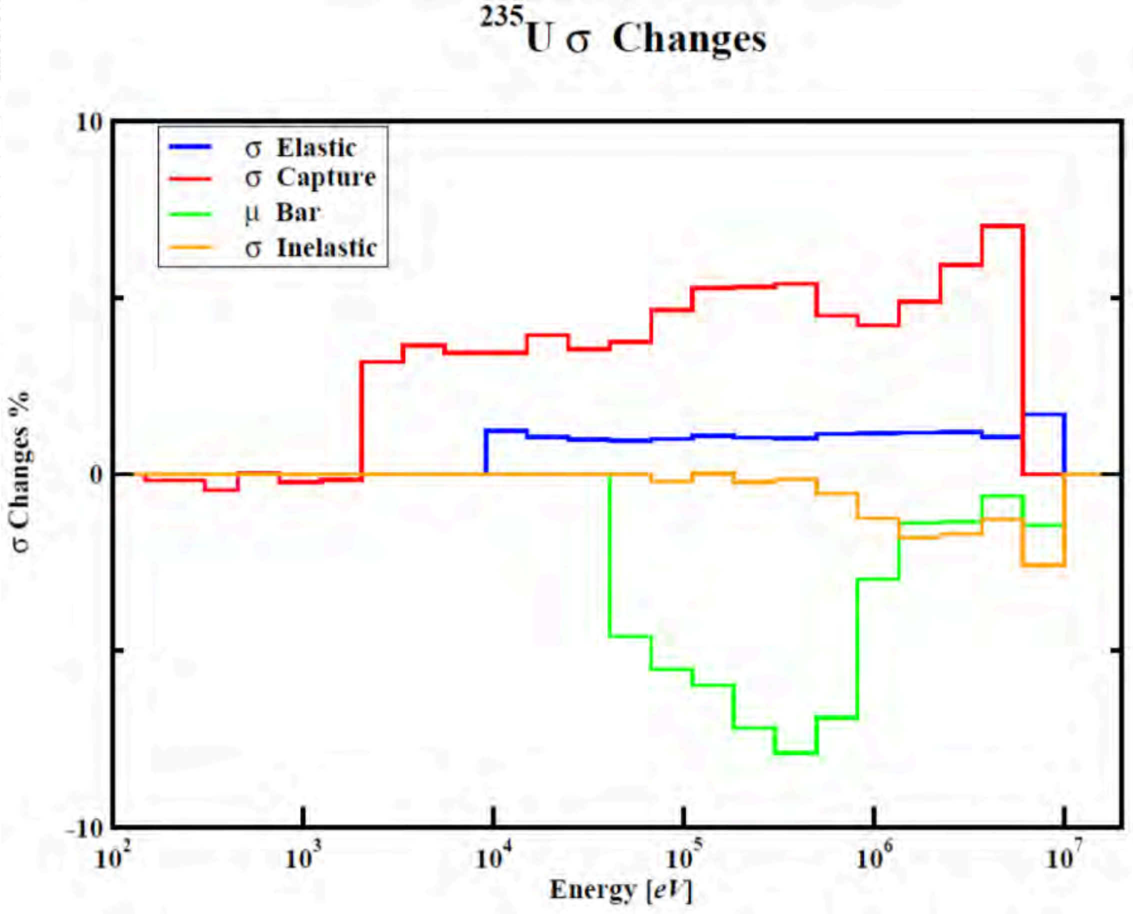
Y-12 National Security Complex

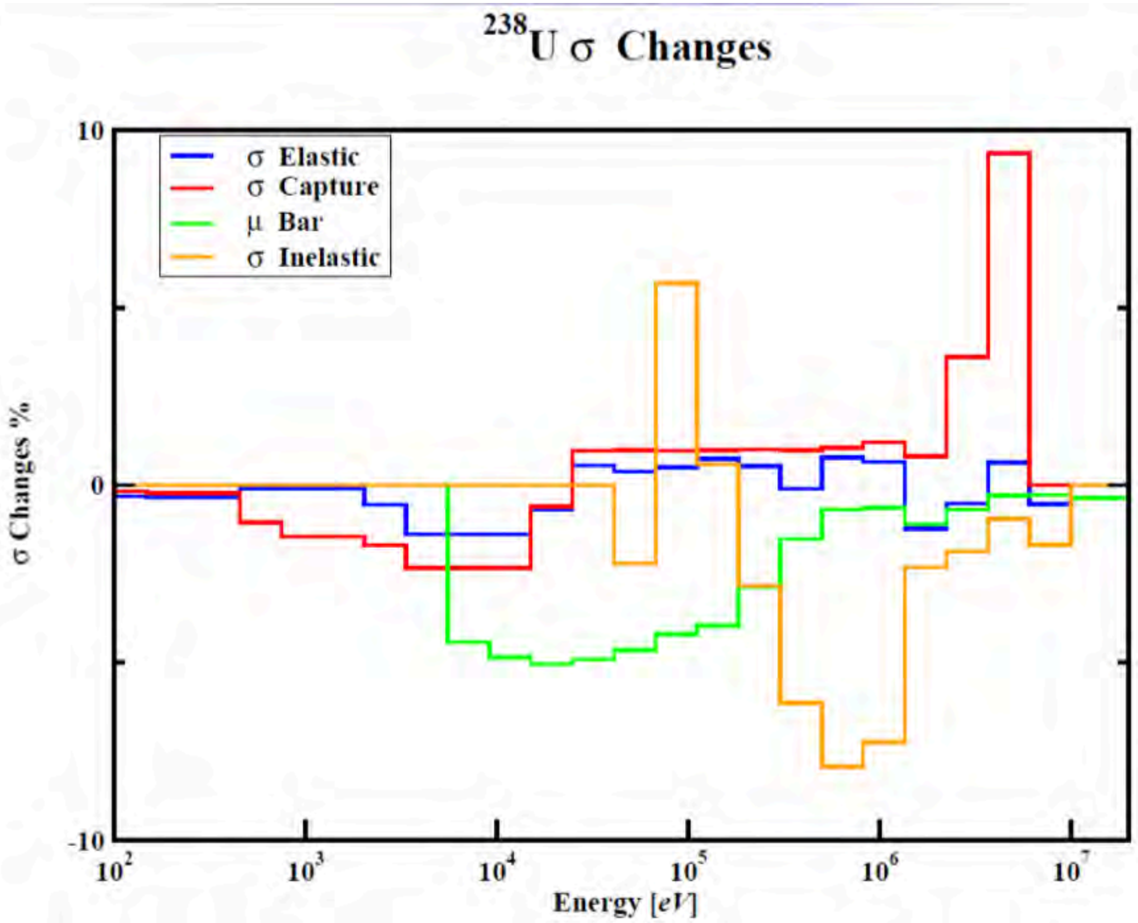
ES-4100 w/ 20 w/o UF_6 study:
Counteracting errors in ENDF/B-VII.1 – ENDF/B-VIII.0



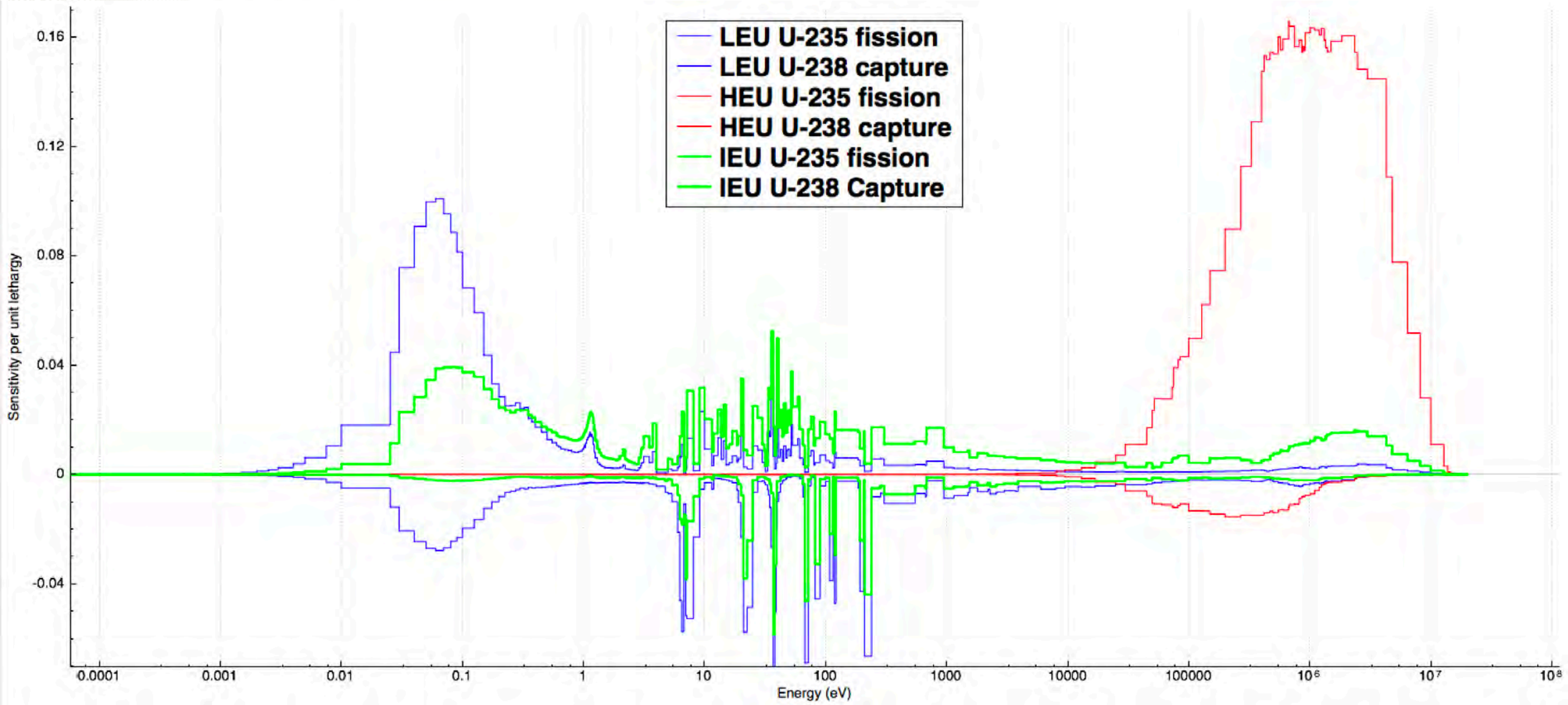
Cross section changes ENDF/B-VII.1 – ENDF/B-VIII.0

OECD/NEA SG-46





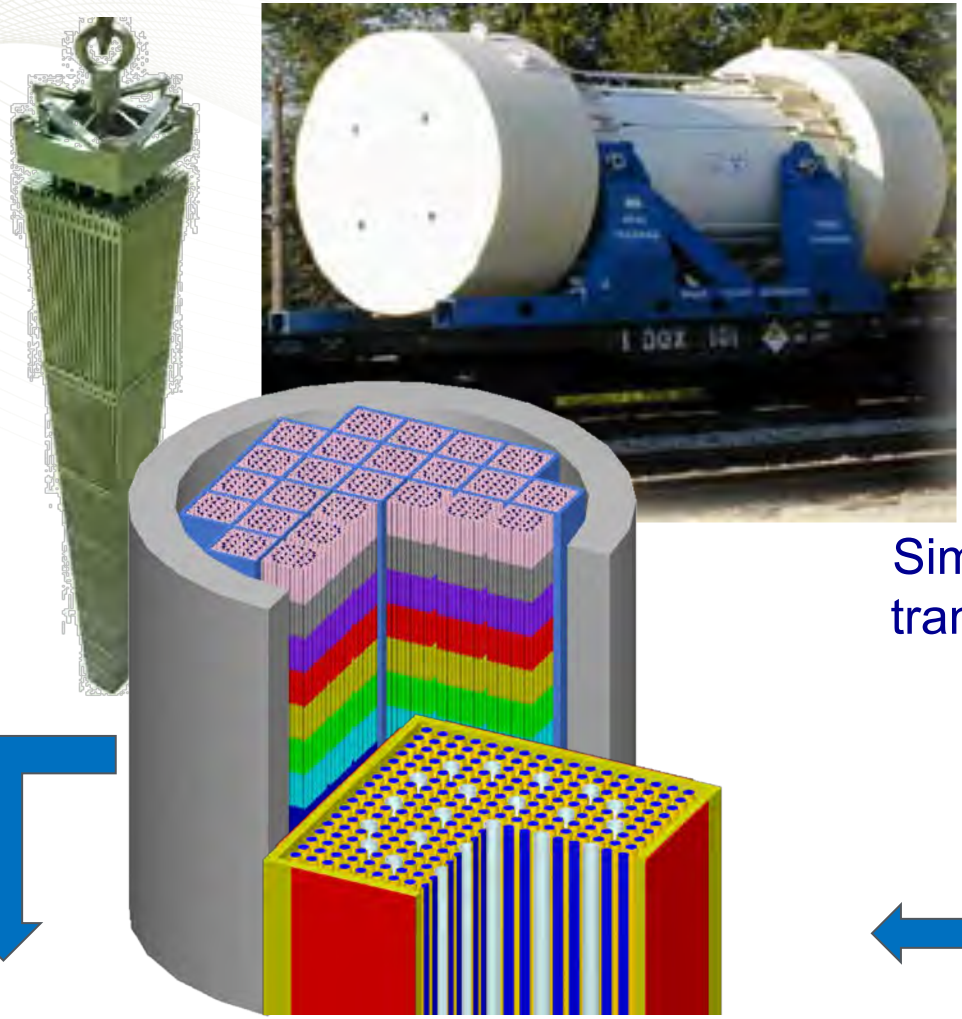
Sensitivity of k_{eff} to nuclear data quantifies how important each cross section is for application of interest



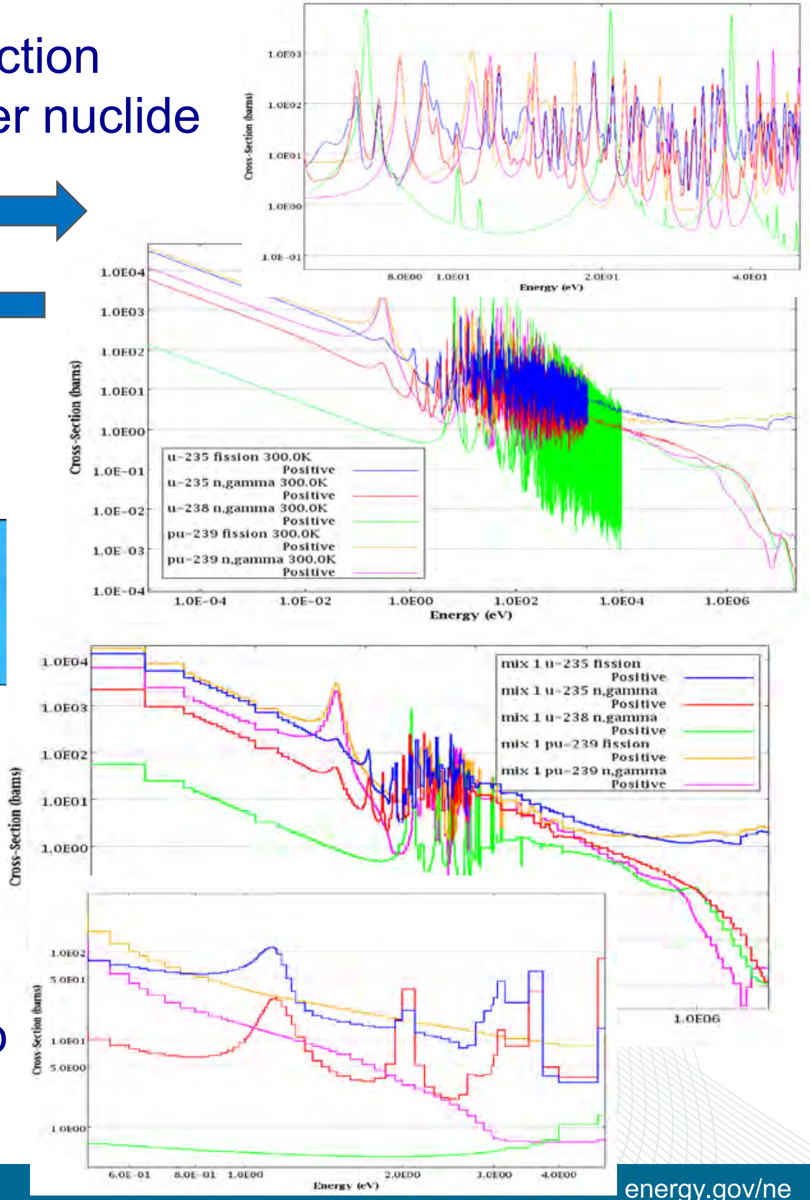
Role of Sensitivity and Uncertainty Analysis in Validation

- Clearly identifies processes that are important to validate
  - Materials, Nuclides, Reactions, Energy
- Assists with challenging areas of applicability where few or no similar experiments are available
- Premise of S/U-based validation
  - Computational biases are primarily caused by errors in the cross-section data
  - Errors are bounded by cross-section uncertainties represented in covariance data

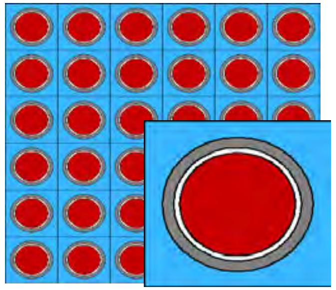
Example application of S/U methods: Safety assessment for transportation of burned nuclear fuel



Point-wise neutron cross-section data: ~60,000 data points per nuclide



Simplified neutron transport model of fuel pin

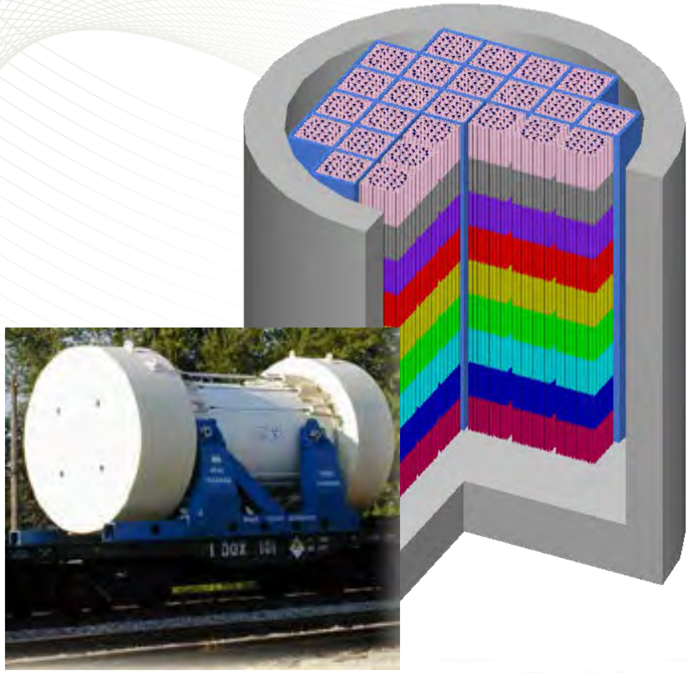


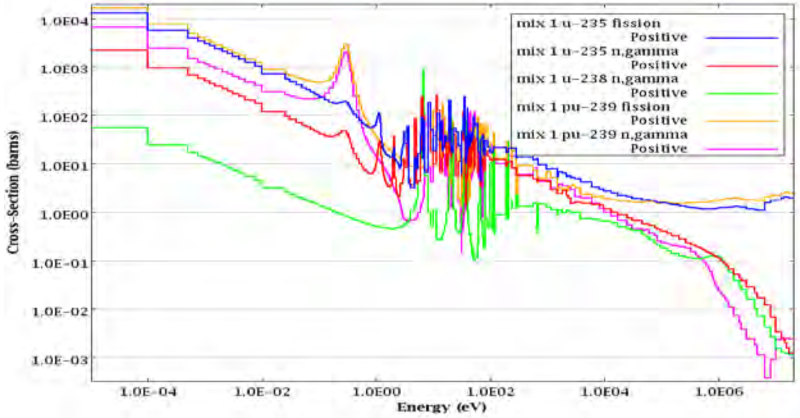
Problem-specific multi-group neutron cross-section data: 238 data points per nuclide

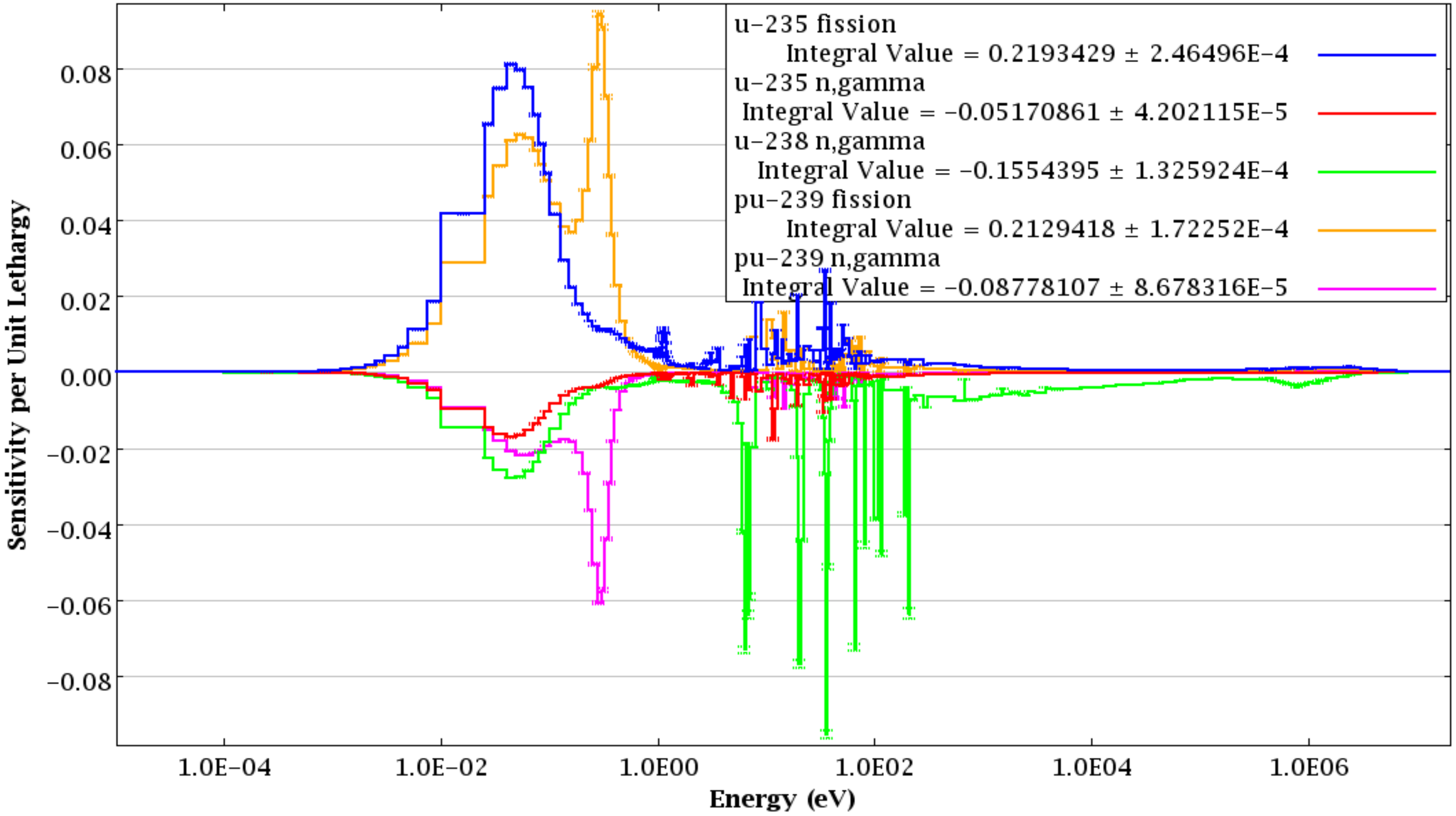
Explicit 3D neutron transport model of shipping cask

k_{eff}

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

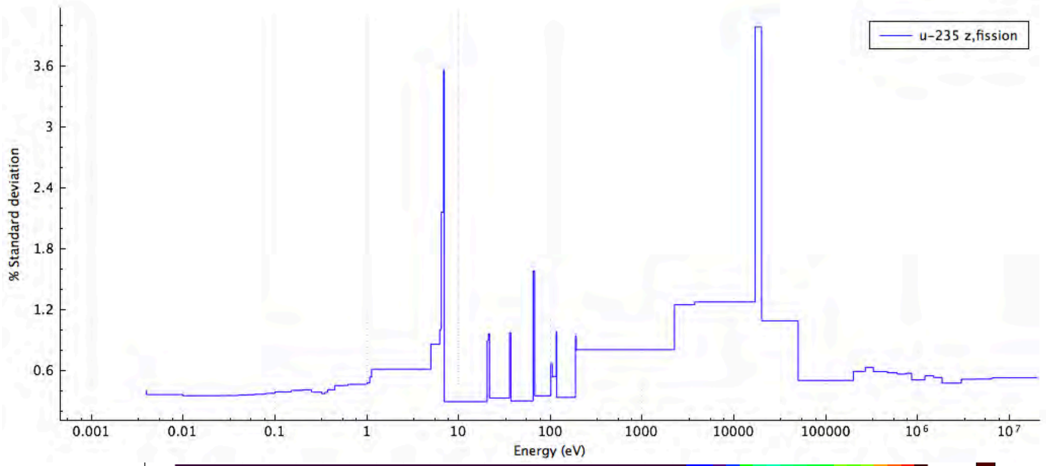


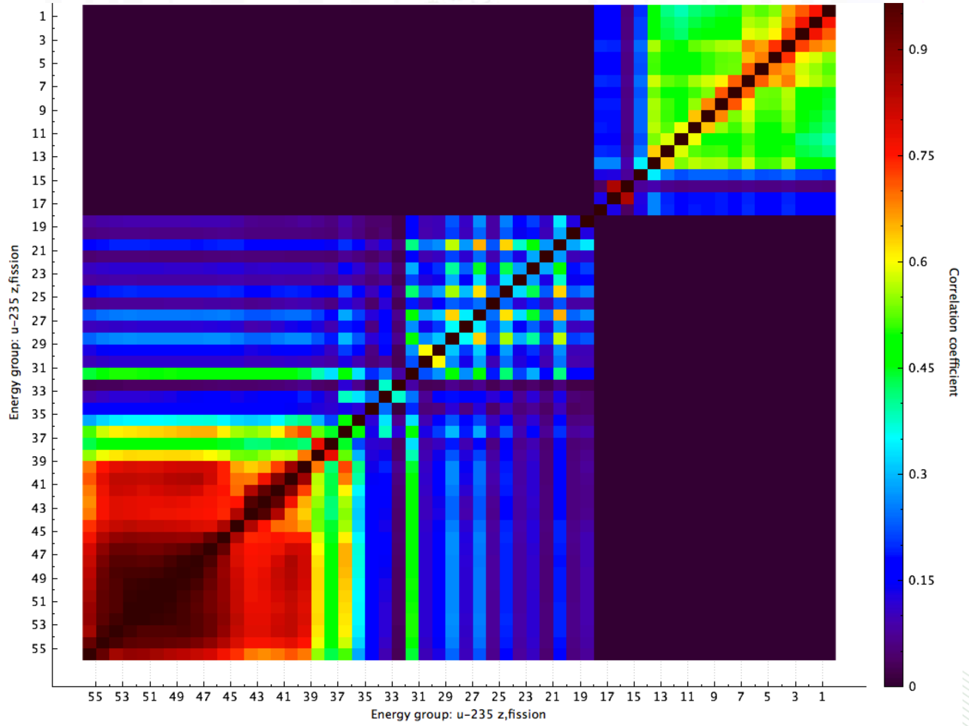




SCALE 6.2 covariance library

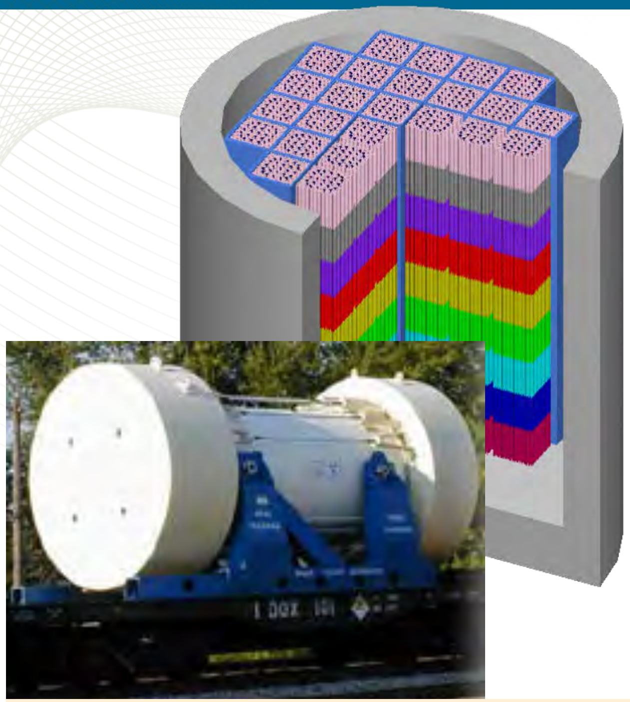
- ENDF/B-VII.1 contains data for 187 isotopes.
- SCALE 6.1 data retained for ~215 missing nuclides.
- Modified ENDF/B-VII.1 ^{239}Pu nubar, ^{235}U nubar, H capture, and several fission product uncertainties, with data contributed back to ENDF/A repository.
- Fission spectrum (chi) uncertainties processed from ENDF/B-VII.1 and from JENDL 4.0 (minor actinides).
- No uncertainties available for scattering secondary particle energy/angular distributions

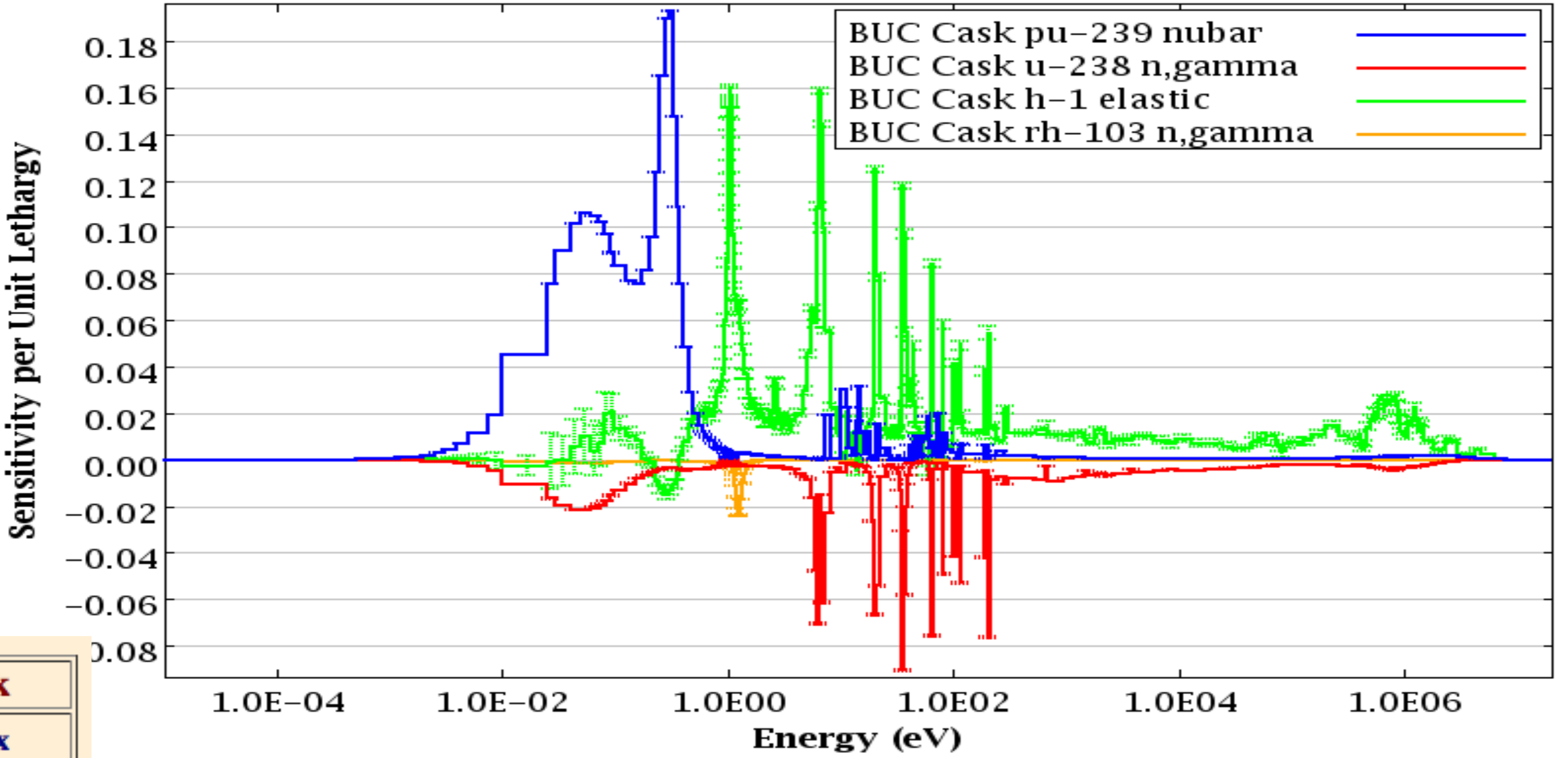




S/U analysis to identify important processes

Application specific

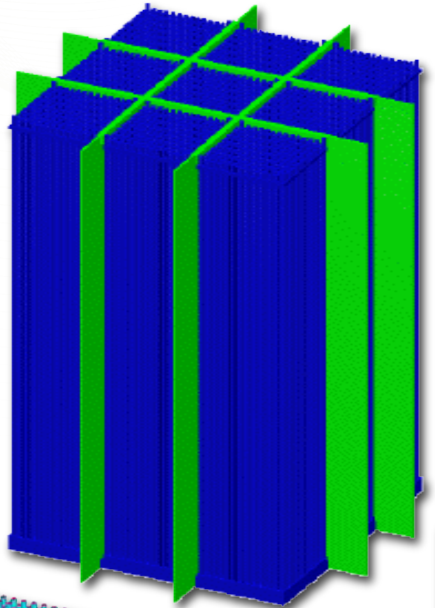


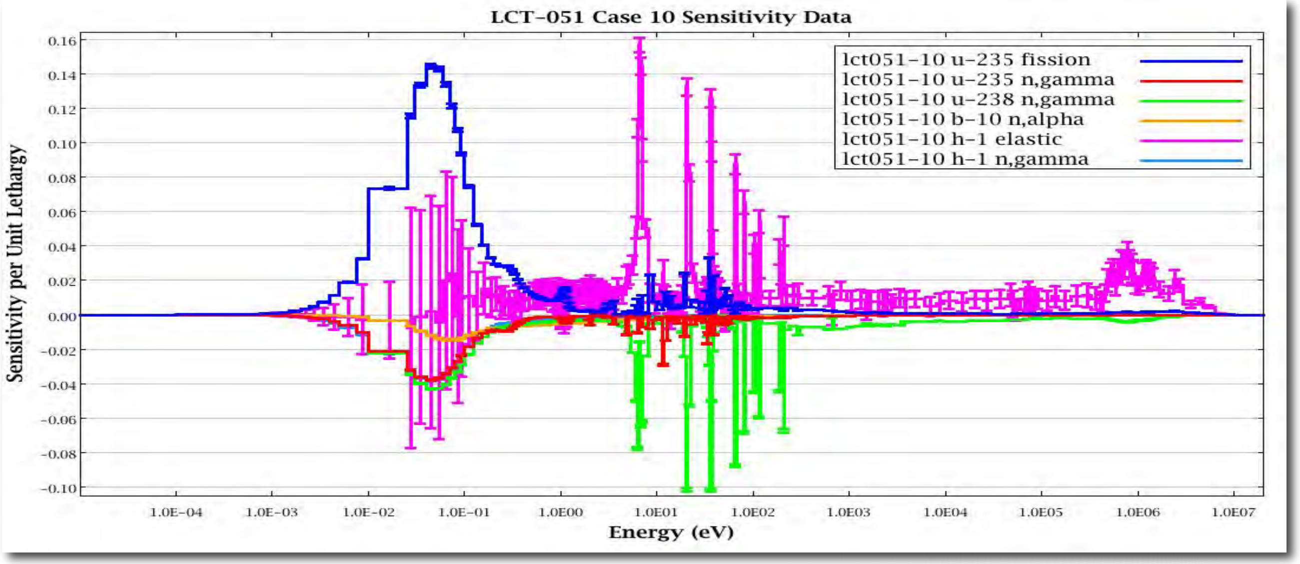


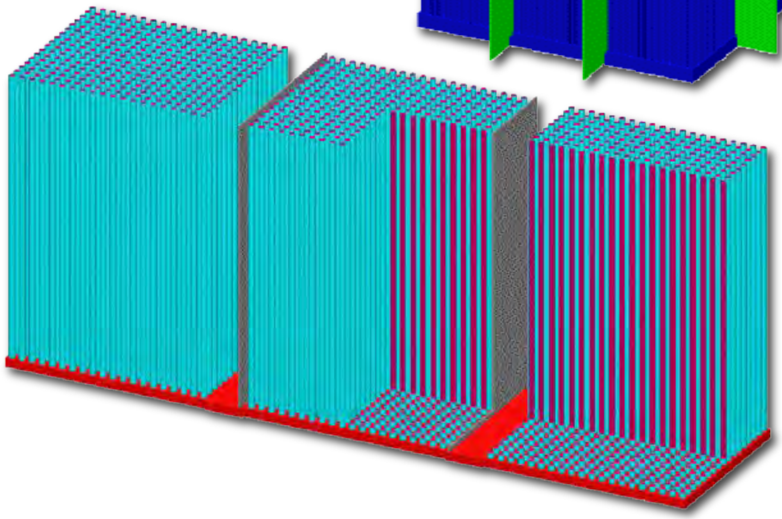
Covariance Matrix		Unc. in % dk/k
Nuclide-Reaction	Nuclide-Reaction	Due to this Matrix
^{239}Pu nubar	^{239}Pu nubar	4.0032E-01 ± 2.5625E-06
^{238}U n,gamma	^{238}U n,gamma	1.9457E-01 ± 1.2387E-05
^{239}Pu fission	^{239}Pu fission	1.5501E-01 ± 1.0838E-05
^{235}U nubar	^{235}U nubar	1.3981E-01 ± 5.0038E-07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

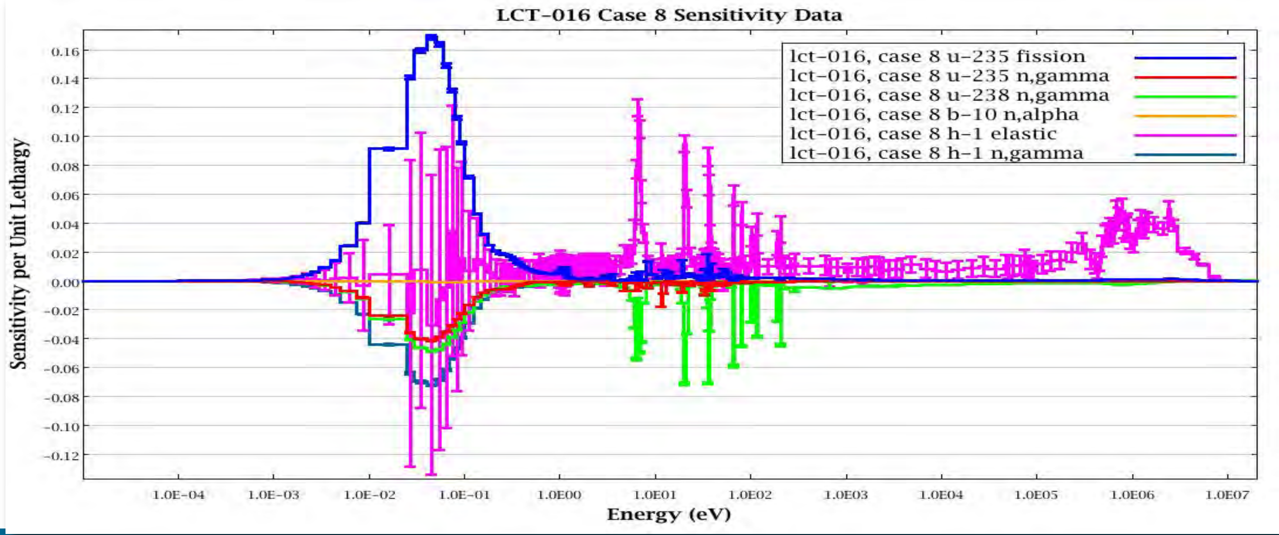
- Overall uncertainty: 0.52% Δk/k

Identify and analyze benchmark experiments to quantify bias in application









Correlation coefficient (c_k)

(a.k.a. representativity factor)

- Quantifies overall similarity potential sources of bias in k_{eff} between design application and benchmark experiment.

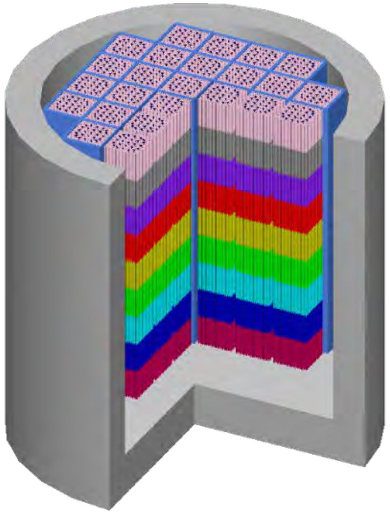
$$c_k = \frac{\sigma_{ae}^2}{\sigma_a \sigma_e}$$

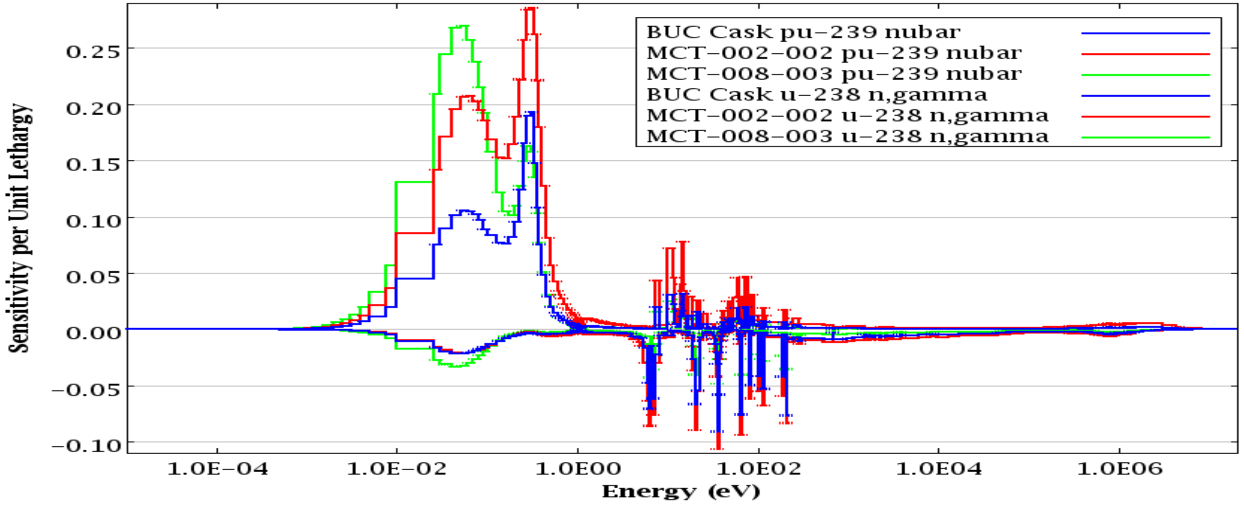
Covariance between Experiment (e) and Application (a) due to all nuclides and reactions

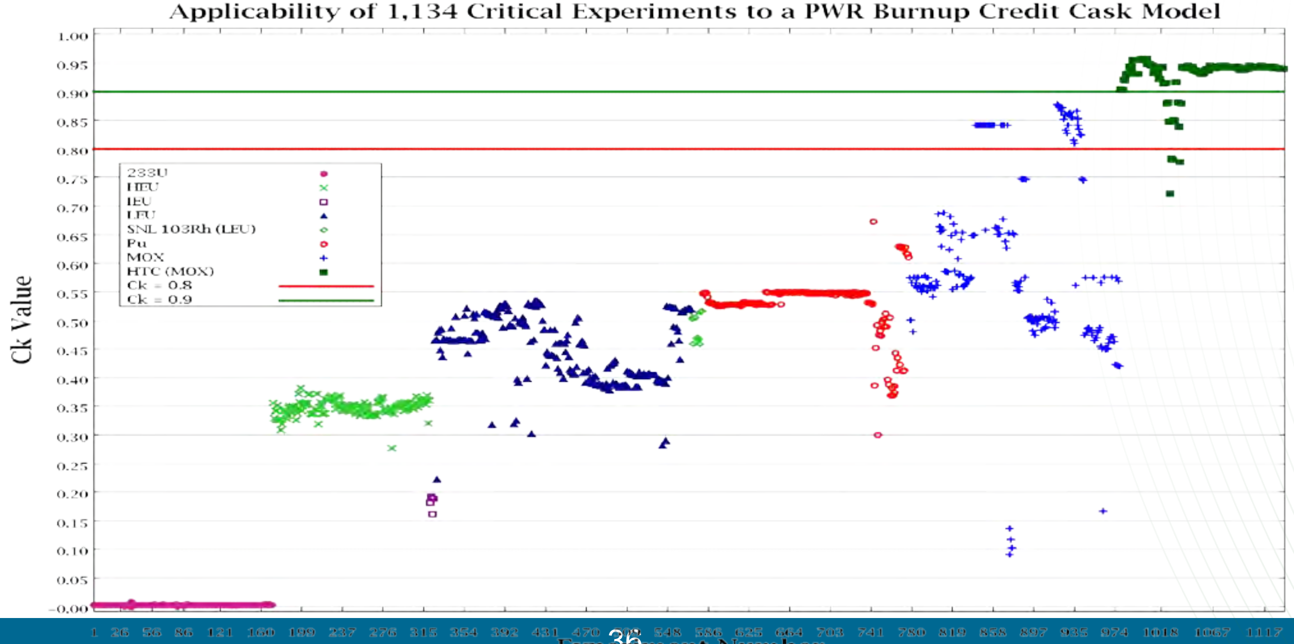
Standard deviations for Application (a) and Experiment (e) due to all nuclides and reactions

Code Validation: Identification of laboratory experiments that are similar to the targeted application

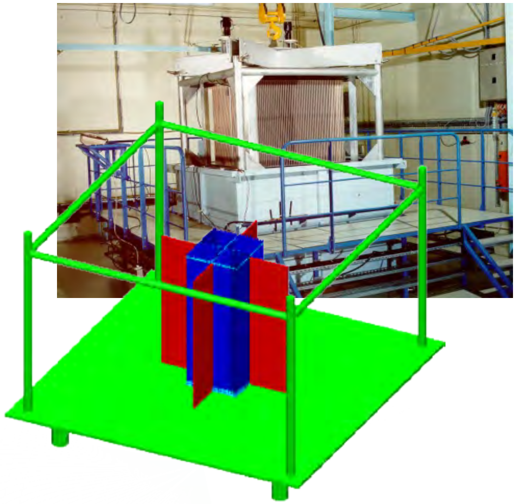
APPLICATION

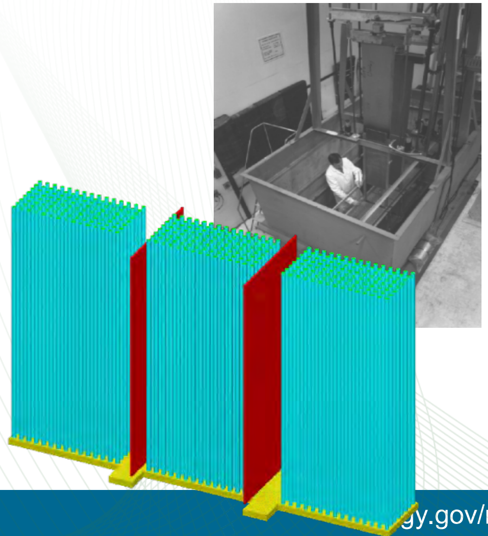




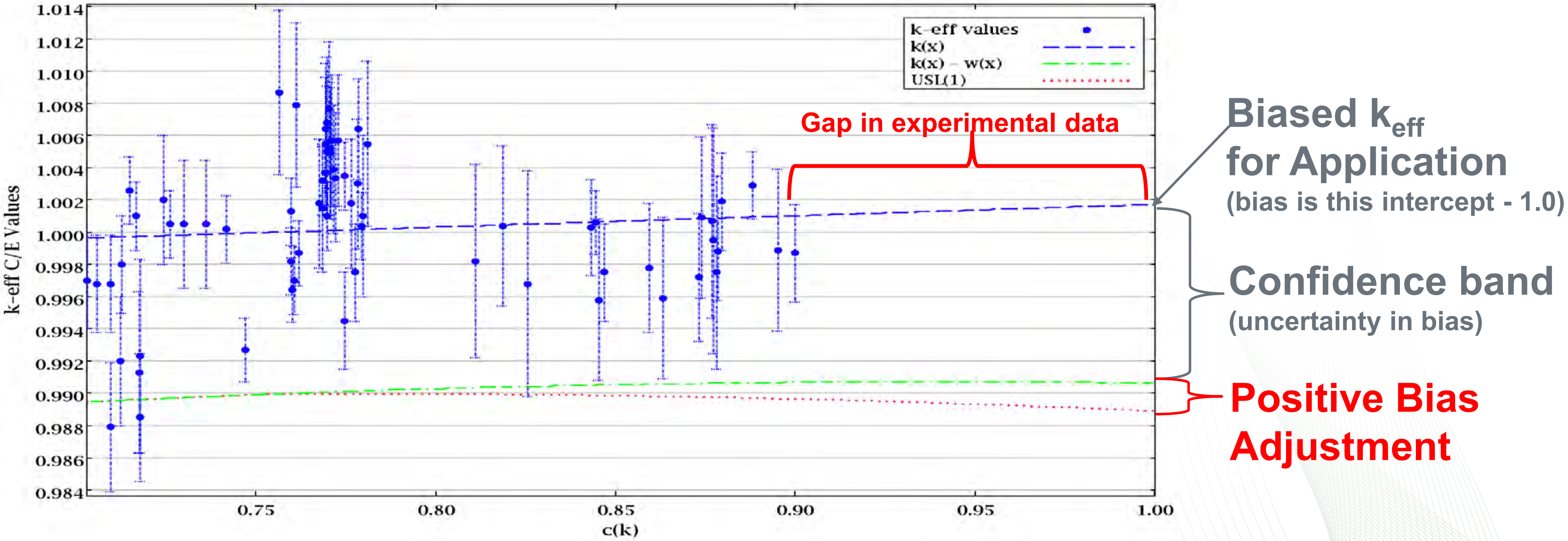


NUCLEAR CRITICALITY EXPERIMENTS

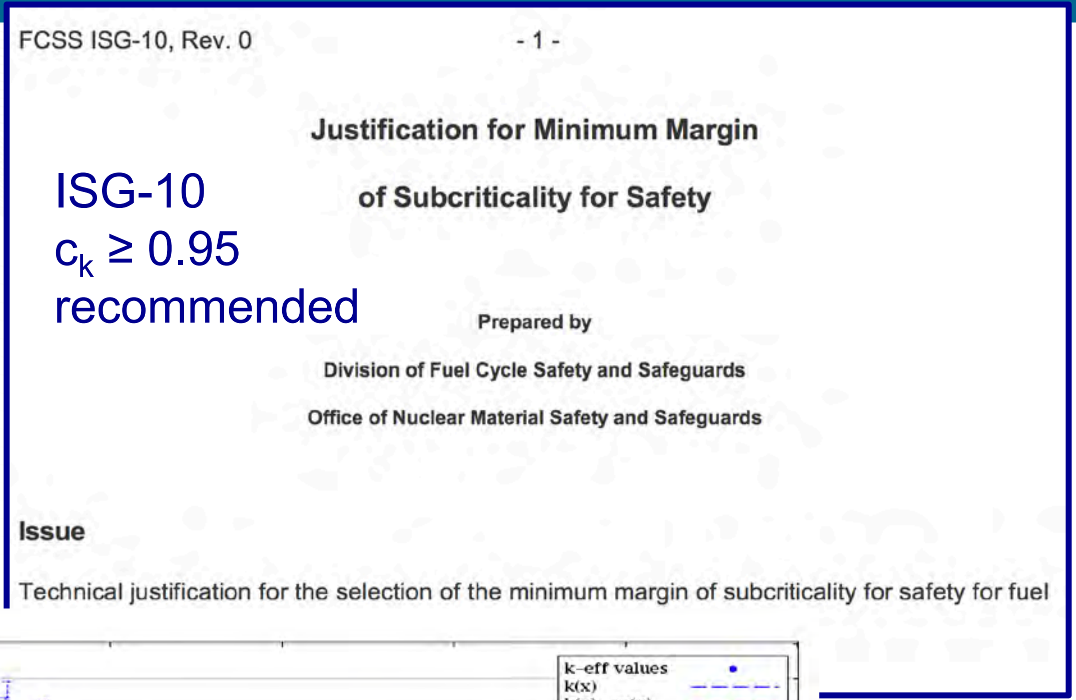




Similarity as independent parameter for trending analysis



Regulatory basis for validation applicability


FCSS ISG-10, Rev. 0 - 1 -

Justification for Minimum Margin of Subcriticality for Safety

Prepared by

Division of Fuel Cycle Safety and Safeguards

Office of Nuclear Material Safety and Safeguards

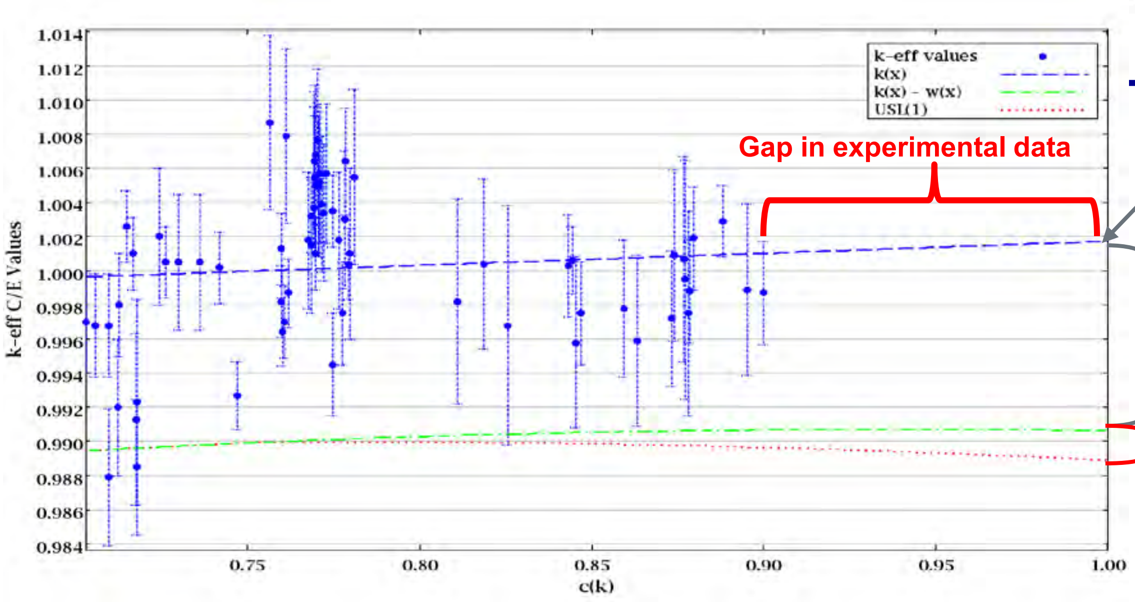
Issue

Technical justification for the selection of the minimum margin of subcriticality for safety for fuel

ISG-10

$c_k \geq 0.95$

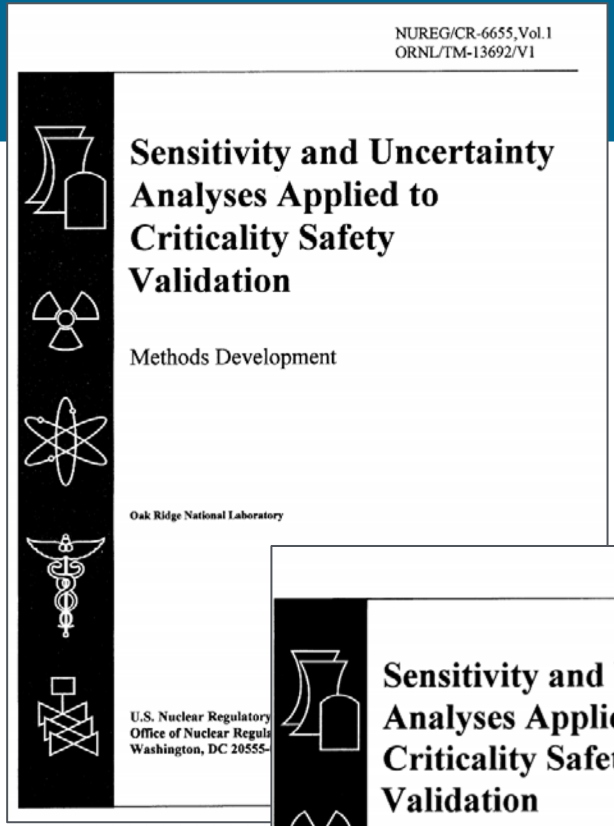
recommended



Biased k_{eff} for Application (bias is this intercept - 1.0)

Confidence band (uncertainty in bias)

Positive Bias Adjustment

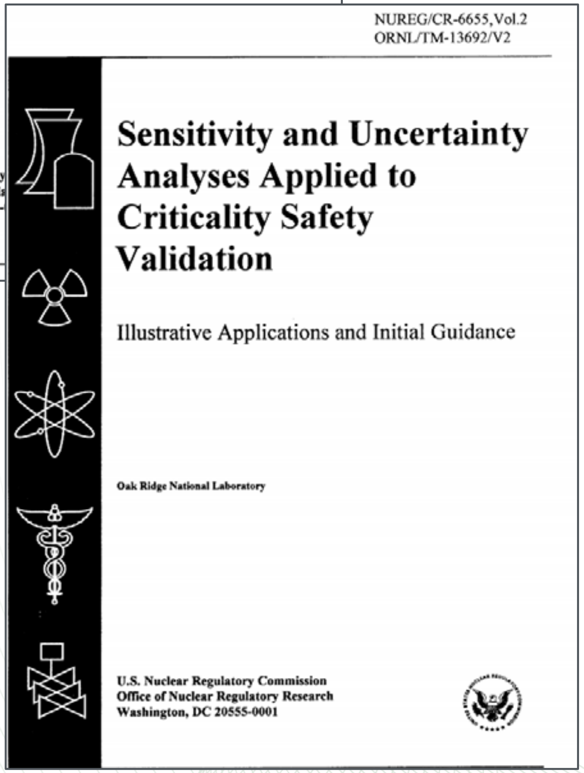

NUREG/CR-6655, Vol.1
ORNL/TM-13692/V1

Sensitivity and Uncertainty Analyses Applied to Criticality Safety Validation

Methods Development

Oak Ridge National Laboratory

U.S. Nuclear Regulatory
Office of Nuclear Regula
Washington, DC 20555-


NUREG/CR-6655, Vol.2
ORNL/TM-13692/V2

Sensitivity and Uncertainty Analyses Applied to Criticality Safety Validation

Illustrative Applications and Initial Guidance

Oak Ridge National Laboratory

U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Washington, DC 20555-0001

Regulatory basis for fission product burnup credit

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

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

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

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

NUREG/CR-7109
ORNL/TM-2011/5

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

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

September 2012

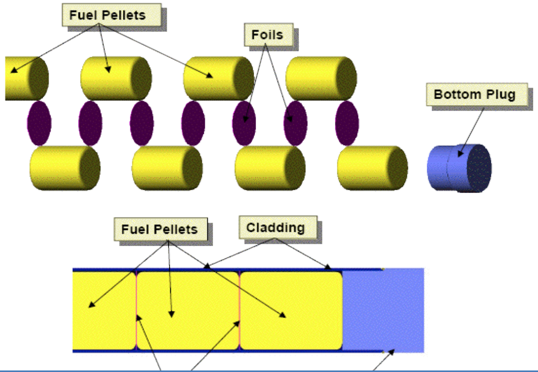
Issue: Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transportation and Storage Casks

Introduction:

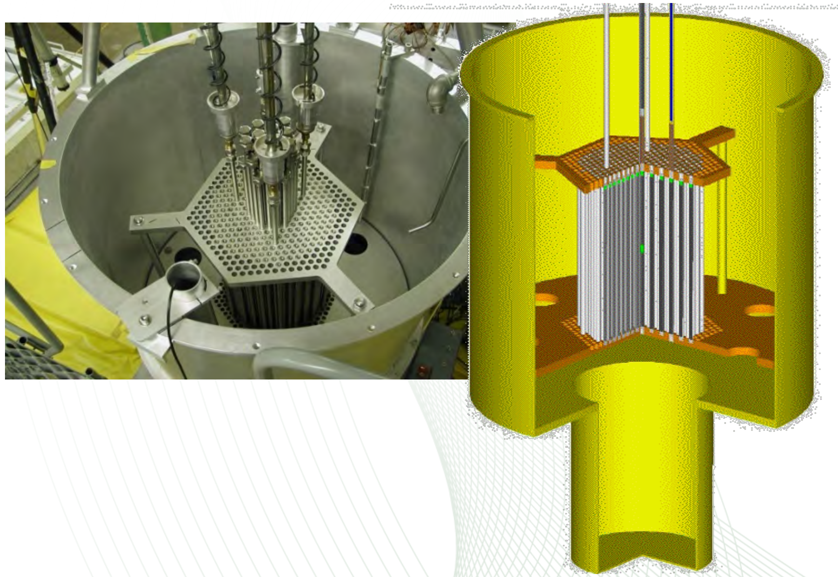
Title 10 of the Code of Federal Regulations (10 CFR) Part 71, *Packaging and Transportation of Radioactive Material*,[1] and 10 CFR Part 72, *Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste*,[2] require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

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

Sensitivity and Uncertainty Analysis of Commercial Reactor Criticals for Burnup Credit



Rh-103 Critical Experiment Design for Burnup Credit



Nuclear Data and Benchmarking Program
Initial Activities

- Nuclear data and validation studies:
  - Gap analysis for nonLWR (ORNL – Sobes/Bostelmann)
  - Investigation of HA-LEU transportation validation basis (ORNL – Rearden/Scaglione/Marshall/Clarity/Holcomb)
- Nuclear data generation:
  - Investigation and generation of application driven covariance data (ORNL – Sobes)
  - Improvements of nuclear data for depletion, activation, and decay (ORNL – Wieselquist)
  - New measurement of ^{238}U (n,n') with associated uncertainties (LBNL – Bernstein)
- International benchmarking activities:
  - Multi-Physics Experimental Data, Benchmark, and Validation (ORNL - Valentine)
  - International Physics Benchmark Programs: ICSBEP and IRPhEP (INL - Bess)
- University projects:
  - Generation of thermal scattering data for graphite (N.C. State, X-energy, ORNL)
  - Generation of thermal scattering sensitivity/uncertainty capabilities (U. Michigan, ORNL)

Clean. **Reliable.** **Nuclear.**

INL-NEI Technical Workshop on Transportation of High Assay Low Enriched Uranium

August 30-31, 2018

Melissa Mann
President, URENCO USA, Inc.

HALEU and the HALEU Community

- High Assay Low Enriched Uranium (HALEU) refers to enrichments above 5.0% U235 and below 20.0% U235.
- A broad community of users may benefit from HALEU:
  - Research and test reactors: including reactors fueled by DOE in the US/overseas and including those currently relying on HEU that may convert to HALEU
  - Advanced reactors, including non-LWRs
  - Advanced fuel designs
  - Producers of targets for medical isotope production
  - Operators of existing LWRs seeking improvements in fuel reliability and economics through higher burnup* and extended operating cycles
- As the enrichment levels needed by these users will vary, **fuel solutions are needed across the full span of HALEU enrichments** (although some "clumping" may develop in the ranges of 6.0%-8.0% U235 and 13.0-16.0% U235 and at 19.75% U235).

*Higher burnup is deemed to exceed an average burnup of roughly 45 gigawatt-days per metric ton of uranium (Gwd/MTU).

HALEU Fuel Cycle

- A complete and sustainable HALEU fuel cycle includes three fundamental capabilities:
  - A uranium enrichment facility to produce HALEU enrichments*: the material will be in the form of uranium hexafluoride (UF6)
  - A conversion facility to (de)convert HALEU UF6 into metal, oxide and/or salts
  - One or more fabrication facilities that can manufacture the specific fuel types required by the various reactor and fuel designs

*It is assumed that the “feed” material for HALEU enrichment is standard low enriched uranium as UF6 at roughly 4.95% U235.

- Packaging and transportation solutions are needed between each of these processing steps and to the ultimate user (for the purposes of today’s discussion, spent fuel packaging is not addressed).

Initial Observations

- Fuel cycle facilities producing and utilizing higher enrichments can be licensed in the US: two NRC-licensed facilities currently fabricate HEU fuel (Category I sites)

Nuclear Fuel Services (Erwin, TN)

BWXT Nuclear Operations Group (Lynchburg, VA)

- It is imperative that the enrichment, conversion and fabrication facilities - and the concordant packaging solutions - be developed on concurrent schedules.
- The licensing framework needs to support development of a HALEU fuel cycle and regulator resources are needed.
- Companies making investments in HALEU facilities need to be sufficiently assured of an economic return.

Potential HALEU Enrichment at URENCO USA (UUSA)


URENCO USA Uranium Enrichment Facility (Eunice, NM)

- 1st facility licensed, constructed and operated under a COL
- Application submitted 12/12/2003 and issued on 6/23/2006 (2 years, 6 months)
- Operations started in 2010
- Licensed for 10,000 million SWU/a; currently producing ~4.9 million SWU/a at up to 5.0% U235 as UF6
- Utilizes advanced gas centrifuge technology

UUSA advanced gas centrifuges are currently capable of producing at the full span of HALEU enrichments without further development or testing.

We estimate that if detailed design, site permitting, and contractor selection were undertaken during the NRC licensing process, we could construct, commission and start-up a HALEU module within 24 months of NRC licensing.

There are no treaty considerations associated with HALEU production at UUSA

Licensing HALEU Enrichment

- UUSA is currently licensed as a Category III facility. The licensing approach for adding a HALEU module may differ by assay bands:
  - For enrichments between 5.0% and 6.0% U235 – analytical approach?
  - For other assays below 10.0% U235 - amended license as a Category III site
  - For assays above 10.0% U235 and below 20.0% U235, Category II license

  Decision point: Initiate a separate license for a Category II HALEU module or license entire site as a Category II facility?

- NRC has clear guidance on MC&A/Fundamental Nuclear Material Control Plans for Category III (NUREG-1065) and Category I (NUREG-1280) facilities, but not for Category II sites.
- Physical protection appropriate to a Category II site and materials is required (as well as for transport).
- Additional criticality benchmark data will be required to support new licensing but questions also exist about how the NRC will approach criticality safety analyses. We would like to see a consistent and coordinated approach to criticality safety for all HALEU fuel cycle facilities.

Packagings for Fissile UF6

§ 71.55 General requirements for fissile material packages (excerpted)

b) Except as provided in paragraph (c) or (g) of this section, a package used for the shipment of fissile material must be so designed and constructed and its contents so limited that it would be subcritical if water were to leak into the containment system, or liquid contents were to leak out of the containment system so that, under the following conditions, maximum reactivity of the fissile material would be attained:

(1) The most reactive credible configuration consistent with the chemical and physical form of the material;

(2) Moderation by water to the most reactive credible extent; and

(3) Close full reflection of the containment system by water on all sides, or such greater reflection of the containment system as may additionally be provided by the surrounding material of the packaging.

(g) Packages containing uranium hexafluoride only are excepted from the requirements of paragraph (b) of this section provided that:

(1) Following the tests specified in § 71.73 ("Hypothetical accident conditions"), there is no physical contact between the valve body and any other component of the packaging, other than at its original point of attachment, and the valve remains leak tight;

(2) There is an adequate quality control in the manufacture, maintenance, and repair of packagings;

(3) Each package is tested to demonstrate closure before each shipment; and

(4) The uranium is enriched to not more than 5 weight percent uranium-235.

- **Approved packagings are needed for HALEU UF6, HALEU metal/oxide and for HALEU fabricated components**
- **Due to moderator exclusion requirements, packaging HALEU UF6 will likely be more complicated than packaging HALEU metals and oxides**
- **A rule change to 10 CFR Part 71.55 would be a lengthy process and likely unsuccessful**
- **Bespoke designs are likely required for different fabrication needs**

UF6 Packaging Considerations

- Are HALEU UF6 shipments limited to use of a small packaging?

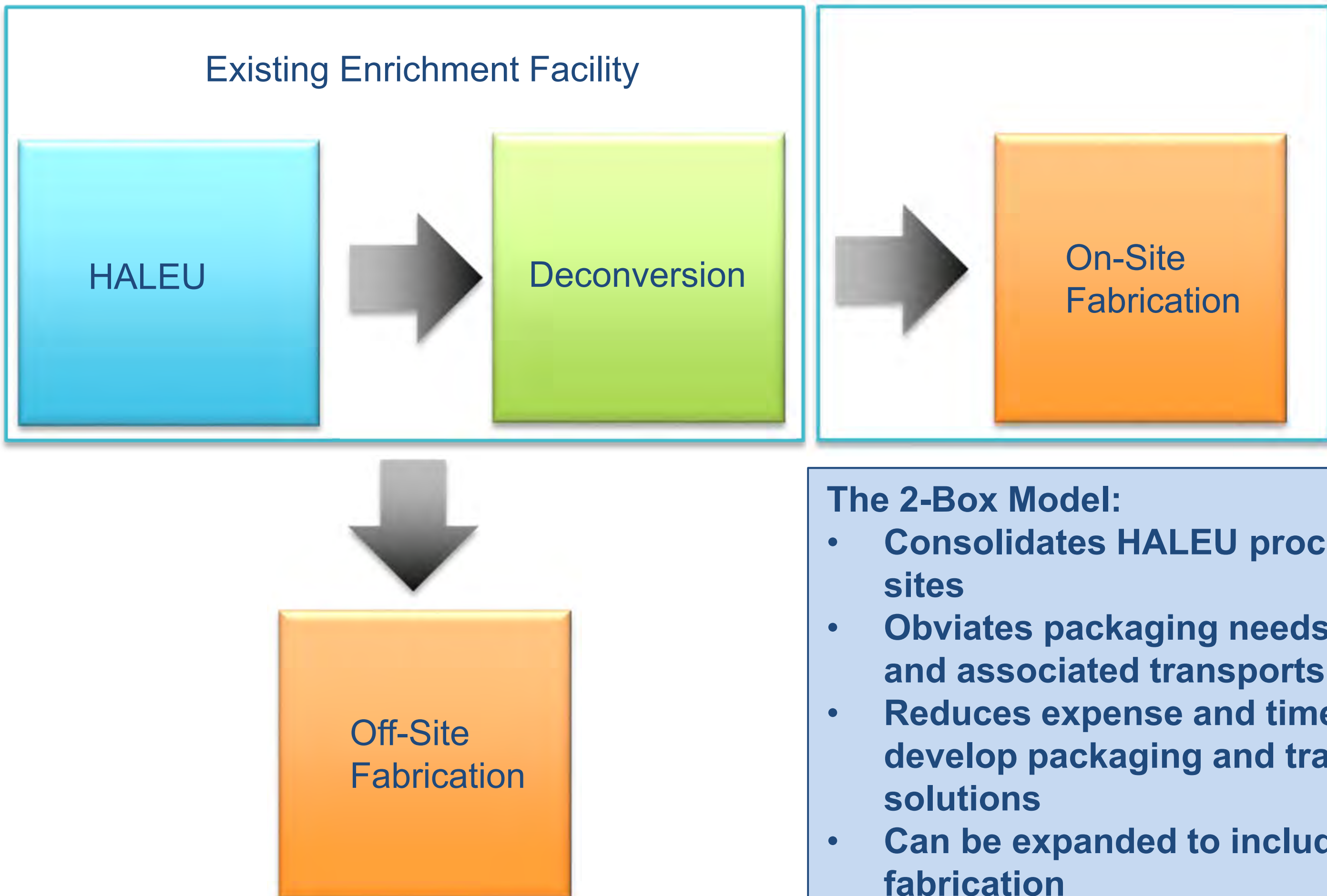
Existing UF6 Cylinders for Higher Assays

Cylinder Model	Diameter in inches	Maximum Enrichment	Maximum lb UF6*
1S	1.5	100.00%	1.0
2S	3.5	100.00%	4.9
5A	5.0	100.00%	54.9
5B	5.0	100.00%	54.9
8A	8.0	12.5%	255

*Ullage, purity and temperature limits apply.

- Are moderator exclusion requirements met through the cylinder or through an overpack?
- Criticality benchmarking data is needed for HALEU assays.

The 2-Box Model



The 2-Box Model:

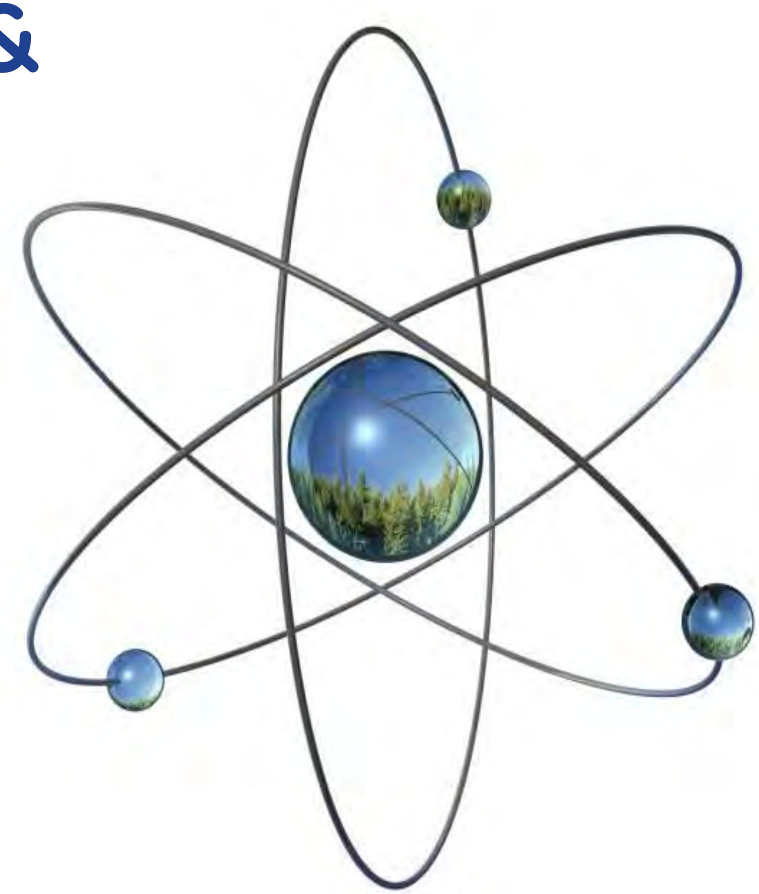
- **Consolidates HALEU processing at fewer sites**
- **Obviates packaging needs for HALEU UF6 and associated transports**
- **Reduces expense and time required to develop packaging and transport solutions**
- **Can be expanded to include some fabrication**
- **Leverages existing site characterization data, site infrastructure, and regulator familiarity**

Recommendations

- HALEU users and fuel cycle participants should coordinate on packaging design and development of criticality benchmark data. This drives consistency, reduces duplication of effort, and provides for a consolidated voice with the regulators.
- DOE/National Laboratory involvement in development of new criticality data for HALEU facilities and packagings – and possibly packagings themselves - would support development of new technologies/designs and provide endorsement of underlying benchmarks.
- Industry should engage with DOT/NRC in the near-term on resource requirements, regulatory approach, anticipated time frames, testing requirements, etc.
- Industry should similarly engage with ANSI (and ISO) to ensure that incorporated standards are updated on a concurrent schedule.
- Get started soon - package development, testing and approval takes time!

Global Nuclear Fuel

HALEU Fuel Fabrication & Transport: An Overview



Lon Paulson
Senior Nuclear Engineer
GNFA

INL-NEI Technical Workshop on High Assay Low-Enriched Uranium
August 30-31, 2018
Nuclear Energy Institute
Washington, DC

Global Nuclear Fuel

HALUE Fuel Fabrication

- Why HALEU?
- Fuel Form(s)
- Enrichment Facility*
- Feed Transport - 30B UF6 Cylinder / UX-30*
- Monte Carlo Methods
- Nuclear Criticality Safety Evaluations
- SNM-1097 License Amendment & ISAS
- Factory Implementation: Nodal Basis
- NDA and RP Instrumentation
- Misc. Programs
- Product Transport – Model RAJ-II, NPC
- Example NRC Review Timeline
- Cost Elements
- Summary

* Non-GNFA

Global Nuclear Fuel

Why Higher Assay LEU?

- Existing LWR fuel cycle is currently limited to 5.0 wt% U235 enrichment.

- The value to reactor utility is very high, as overall fuel cycle lengths *could* be increased in the existing fleet of Boiling Water Reactors (BWRs) or Pressurized Water Reactors (PWRs).

- Higher assay needed to reach +24-month cycle.

- Nearterm: BWRs/PWRs/SMRs utilizing UO_2 Accident Tolerant Fuel (ATF) designs such as IronClad (FeCrAl cladding) or other cladding types would require higher enrichment to permit higher exposure (up to ~80k MWD/MTU). Anticipated peak assay is < 6.5 wt. % U235.

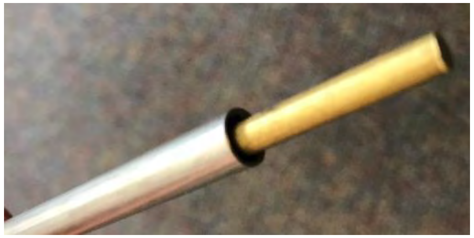
- Longterm: Advanced reactors and novel SMR designs utilizing SFR technology with metallic uranium fuel may require assays up to 19.9 wt. % U235.

Fuel Form(s)

HALEU fuel forms being considered by GNFA in support of SMRs include:

UO2 fuel matrix ➔ BWRX-300

U Metal fuel matrix ➔ PRISM, ARC, OKLO



U metal feed *could* be derived from national labs or converted from oxide....

Global Nuclear Fuel

ENR Facility

- The enrichment facility must successfully produce, and deliver to the fabricator, >5% assay in an approved UF6 cylinder.
- Technically feasible to *configure* cascade output to >5% enr.
- Licensing, analysis, ISA Summary perturbations required.
- NOTE: GLE licensing bases *evaluated* 30-inch and 48-inch UF6 cylinders up to 10.0 wt.% U235.

Global Nuclear Fuel

ENR Facility

3. NRC FORM 374

U.S. NUCLEAR REGULATORY COMMISSION

Page 1 of 8

MATERIALS LICENSE

Pursuant to the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974 (Public Law 93-438), and Title 10, Code of Federal Regulations, Chapter I, Parts 30, 31, 32, 33, 34, 35, 36, 39, 40, and 70, and in reliance on statements and representations heretofore made by the licensee, a license is hereby issued authorizing the licensee to receive, acquire, possess, and transfer byproduct, source, and special nuclear material designated below, to use such material for the purpose(s) and at the place(s) designated below, to deliver or transfer such material to persons authorized to receive it in accordance with the regulations of the applicable Part(s). This license shall be deemed to contain the conditions specified in Section 183 of the Atomic Energy Act of 1954, as amended, and is subject to all applicable rules, regulations, and orders of the Nuclear Regulatory Commission now or hereafter in effect and to any conditions specified below.

Licensee	
1. General Electric-Hitachi Global Laser Enrichment LLC	3. License Number: SNM-2019
2. 3901 Castle Hayne Road	4. Expiration Date: September 25, 2052
P.O. Box 780	5. Docket No. 70-7016
Wilmington, North Carolina 28402	

6. Byproduct, Source, and/or Special Nuclear Material	7. Chemical and/or Physical Form	8. Maximum Amount that Licensee May Possess at Any One Time
A. Uranium (natural and depleted) and daughter products	Physical: Solid, Liquid, and Gas Chemical: UF_6, UF_4, UO_2F_2, oxides and other compounds	140,000,000 kg (308,000,000 lbs)
B. Uranium enriched in isotope 235U up to 8 percent by weight and uranium daughter products	Physical: Solid, Liquid, and Gas Chemical: UF_6, UF_4, UO_2F_2, oxides and other compounds	2,600,000 kg (5,720,000 lbs)
C. Tc-99, transuranic isotopes and other contamination	Any	Amount that exists as contamination as a consequence of the historical feed of recycled uranium at other facilities

Under License SNM-2019, GLE CF *authorized* to produce UF6 with material enrichments up to 8.0 wt.% U235

Global Nuclear Fuel

Feed Transport: UF6 Cylinder / UX-30

Model 30B UF6 Cylinder

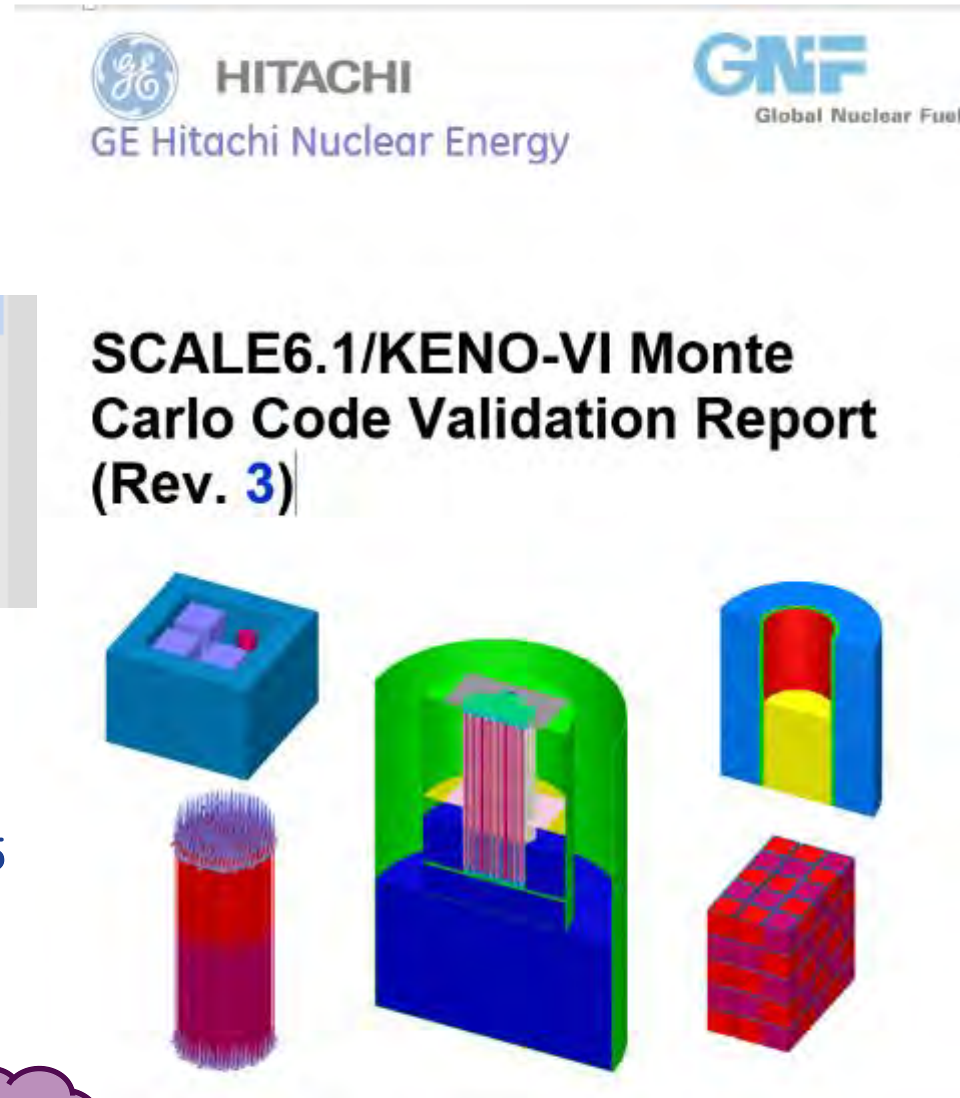
Model UX-30 Overpacks on Flatrack

Global Nuclear Fuel

Methods

GNFA Monte Carlo validation report(s) now support a variety of AOAs:

AOA-1: LEU Homogeneous Systems
AOA-2: HEU Solution Systems
AOA-3: LEU Heterogeneous Compound Systems without Absorbers
AOA-4: LEU Compound Systems with Cadmium
AOA-5: LEU Heterogeneous Compound Systems with Boron
AOA-6: Uranium Metal Systems
AOA-7: LEU Heterogeneous Compound System with Gadolinium


HITACHI
GE Hitachi Nuclear Energy
Global Nuclear Fuel

SCALE6.1/KENO-VI Monte Carlo Code Validation Report (Rev. 3)

LEU systems: ≤ 10.0 wt. % U235

HEU soln systems: 89 – 93.2 wt.% U235

U Metal systems: 9 – 97.6 wt. % U235

revisit

Global Nuclear Fuel

Criticality Safety Analyses

- The fabricator must re-evaluate the nuclear criticality safety bases associated with conversion of UF6 to UO2 and related ceramic processes to build LWR fuel bundles with a higher assay.

- Documented nuclear criticality safety evaluations (referred to herein as criticality safety analyses or CSAs) would be required to be *re-baselined* for the fuel manufacturing and support facilities on a node-by-node basis.

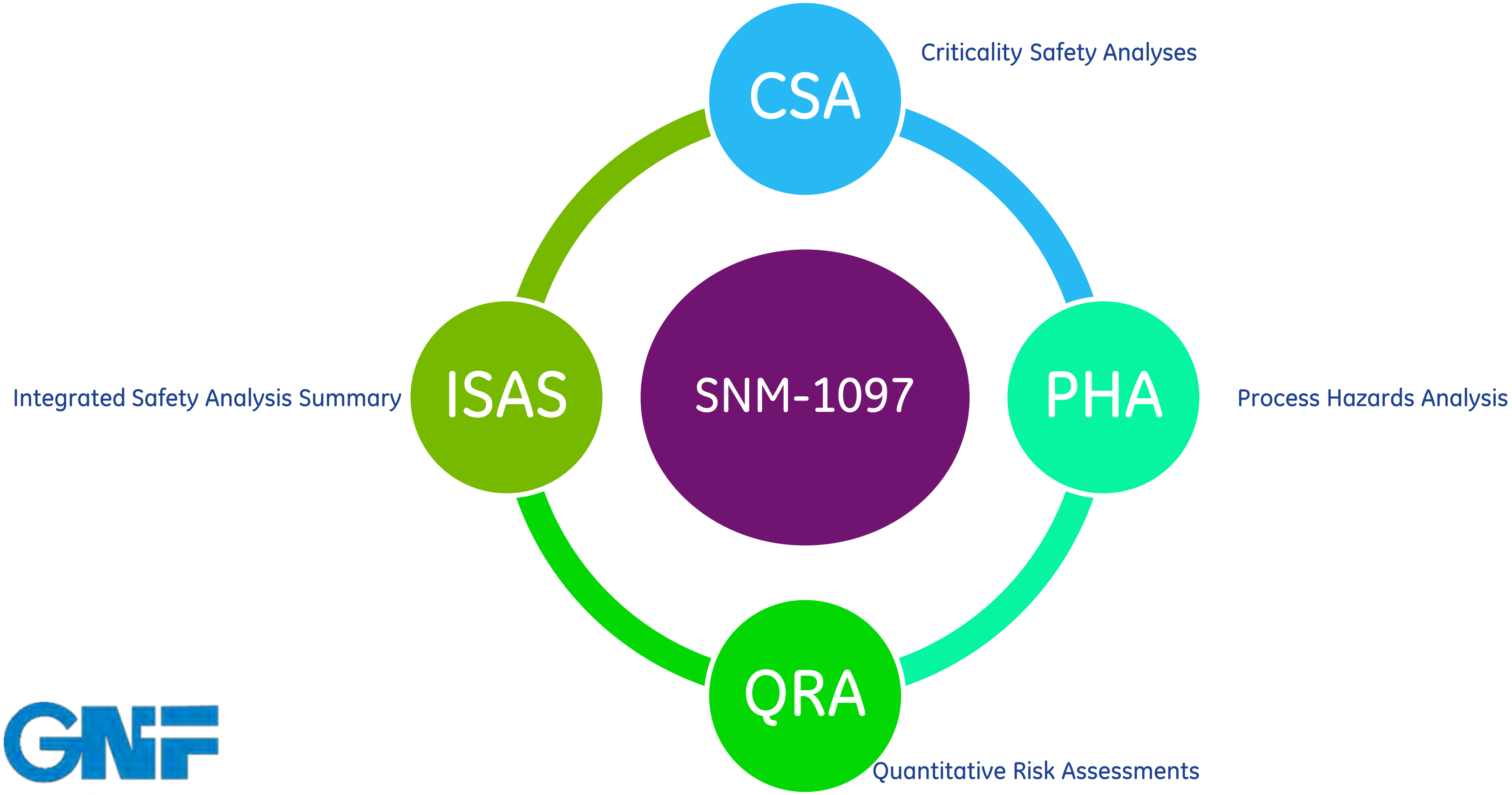
- Assay increase would result in low to intermediate impact on dry powder system(s), as well as on ceramics palletization, rod storage and bundle assembly, since the existing safety margins associated with non-uniform moderation safety limits / safe mass limits / safe rod quantity / etc. would be reduced with minor equipment modification required.

- Balance of plant systems are currently physically sized for favorable geometry classification (e.g., pipe tanks, annular vessels, containers, etc.) at 5%; and would in most cases not qualify as such at a higher assay; thus, equipment modifications would be expected for liquid waste systems.

Global Nuclear Fuel

Licensing & ISA

- Commensurate with the SNM-1097 license [amendment] application, an integrated safety analysis summary (ISAS) for the GNFA Wilmington fuel fabrication facility must also be revised to demonstrate that high consequence accident sequences remain highly unlikely pursuant 10CFR70.



Global Nuclear Fuel

Factory Implementation – Nodal Basis

- **Node 101 Cylinder Handling**
  - Receipt, Handling, Storage, and UF6 Heel Cylinder Shipment
- **Node 201 Vaporization**
  - Vaporization and Cold Trap Processes
- **Node 202 Conversion**
  - DCP Conversion Reactor-Kiln
- **Node 203 HF Recovery**
  - DCP HF Acid Recovery and HF Neutralization
- **Node 204 Homogenization**
  - DCP Powder Outlet
- **Node 205 Homogenization**
  - DCP Homogenization
- **Node 206 BPG**
  - DCP Blend, Precompact, Granulate, Tumble
- **Node 207 Powder Pack / Receipt**
  - DCP Powder Pack / Powder Receipt
- **Node 208 Miscellaneous**
  - DCP MRA Facility and Powder Containers

- **Node 301 DSR**
  - Dry Scrap Recylcle Furnace Feed, Blender, XFR Station, Furnace, Screener
- **Node 401-503 Fabrication Press**
  - Fabrication Rotary Press, Beaker Storage
- **Node 405-504 Fabrication Sinter**
  - Sintering Furnaces, Sinter Test, Pellet Boats, Boat Conveyors, Gamma Densitometer
- **Node 405a-504a Fabrication Sinter Hydrogen Gas Release**
  - N/A
- **Node 406-505 Fabrication Grind**
  - Pellet tray, Grind, Rod Load, Ministacker, Rod Storage
- **Node 407-506 Fabrication Rod Processing**
  - Fabrication Rod Processing
- **Node 501-502 Fabrication GAD Vibromill**
  - GAD MRA DM-10 Vibromill
- **Node 507 GAD DSR**
  - GAD Dry Scrap Recycle
  - GDSR DM-10 Vibromill
- **Node 601 Fabrication Bundle Assembly**
  - Fab Bundle Forest, Bundle Accumulation, Inner/Outer Pack
- **Node 701 Decon**
  - Decon General
- **Node 702-703 Radwaste/WTF**
  - Radwaste Processes, CAAS Needs Evaluation CWS DAM20
- **Node 705 FMO SPF**
  - FMO Scrap Pack Facility

- **Node 706-704 Incinerator Building**
  - Incinerator Building, Outside Storage Pads, CAAS Needs Evaluation - Wastebox
- **Node 801-802 Auxiliary Operations**
  - General Can Storage, Scrap Hood and Utility Hoods
- **Node 803-804 HVAC**
  - Primary and Secondary HEPA Filter Systems
- **Node 805 Laboratory**
  - Support Labs
- **Node 805a Hydrogen Gas Leak – Chemet Lab**
  - N/A
- **Node 807 Utilities**
  - N/A
- **Node 808 Adjacent Fire**
  - N/A
- **Node 900 *Generic Subcritical Limits***
  - CSA-900.00.200, Safe Uranium Mass Limits
  - CSA-900.00.200, Safe Mass Limits Appendix
  - CSA-900.01.100, Moderation_Limits
  - CSA-900.03.100, Sintered Pellets
  - CSA-900.04.100, Minimum Gd2O3 for Maintaining UO2/H2O Systems Subcritical

Global Nuclear Fuel

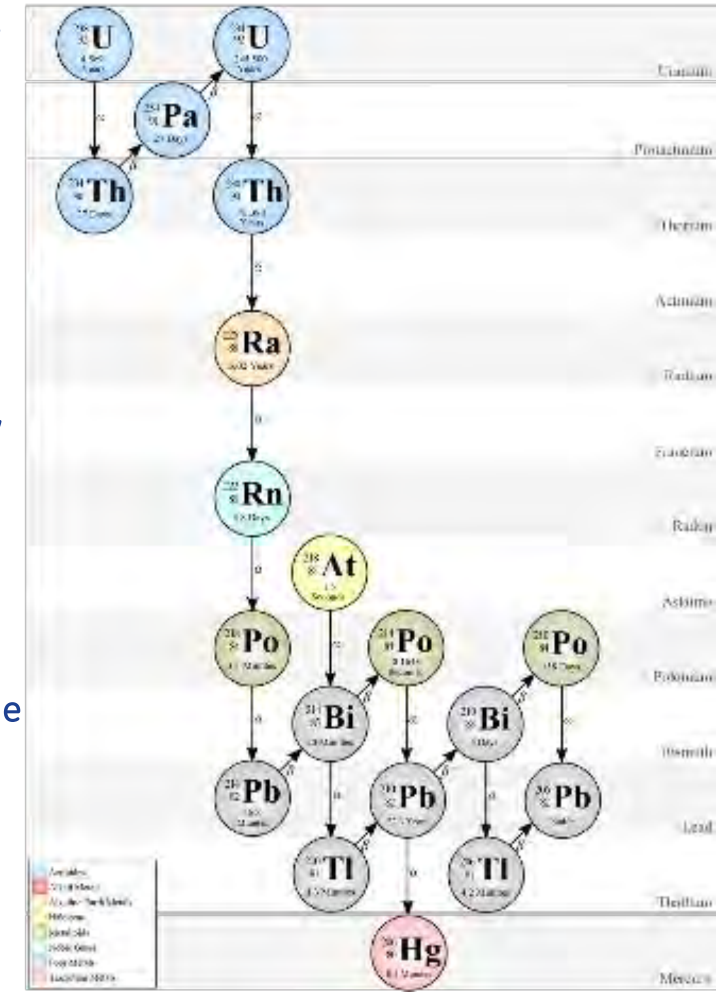
NDA Systems and RP Instrumentation

- To permit LWR fuel manufacture >5% enrichment, select non-destructive assay (NDA) nuclear measurement systems must be requalified.

- NDA systems based on interrogation of U238 (or total U) should not be directly affected by ENR changes
    - DECON waste cart monitors, DECON box monitors, Elephant-Gun, Gad powder XRF, Gad pellet XRF, Gad/UO2 Pellet Densitometers, hand-held Scout-II gamma monitor for detecting uranium buildup

- NDA systems based on interrogation of U235 will be directly impacted by an ENR change
    - Fat Albert, MAPS, DCP HF and FMO Radwaste, NaI scintillation detectors (a.k.a., “pipe detectors”), and the UF6 cylinder enrichment verification system

- Radiation Protection (RP) instrumentation is a special case, the alpha/beta counters (tennelec counters, airborne sample filter counters, and personnel exit survey personnel contamination monitors or PCMs) will require review and assessment.
    - When enrichment is changed, the expected uranium isotopic signature ratios (e.g., U234/U238) also change; and impact interpretation of the uranium content in a sample or deposit.



Global Nuclear Fuel

Miscellaneous Program(s)

- To permit LWR fuel manufacture >5% enrichment the corresponding SNM-1097 license must evaluate potential impacts on the following:

  - ❑ Decommissioning Funding Plan (DFP)
  - ❑ Radiological Contingency and Emergency Plan (RC&EP).
  - ❑ Physical Security Plan (10CFR73)

Note:

- ➢ Category III, Special nuclear material of low strategic significance - GNFA fuel fabrication facility may continue under Category III classification for physical security programs.

- ➢ Category II, Special nuclear material of moderate strategic significance - GNFA fuel fabrication facility may / may not require Category 2 designation to support U metal fuel fab; **depends on scale** of pilot.

Category	Qty. Permitted : Assay Range
II	≥ 10 kg U235: ≥10% but < 20%
III	≥ 1 kg but < 10 kg U235: ≥10% but < 20% ≥ 10 kg U235: ≤10%

Global Nuclear Fuel

Product Transport: RAJ-II

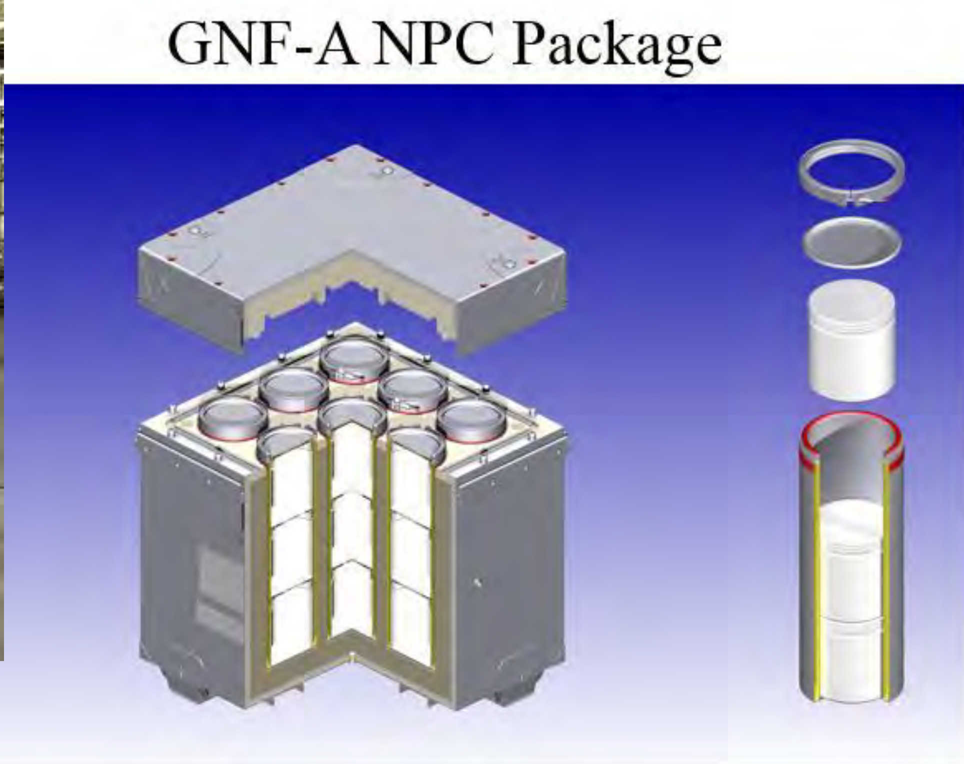
Model RAJ-II Type B Fissile Package
[USA/9309/B(U)F-96]
8x8, 9x9, 10x10 fuel assemblies
UO2 rods, UC rods, PWR rods

ECGU or RU per ASTM C996 material forms enriched to no more than 5.0 wt% U235

HALEU requires SAR update, corresponding CSI change

Global Nuclear Fuel

Product Transport: NPC



Model NPC Type A Fissile Package [USA/9294/AF-96]

UO2 powder, U3O8, UOx, UNH, U-bearing ash, calcium containing sludges, etc.
Heterogeneous UO2 pellets (BWR/PWR),
Heterogeneous UO2, U3O8, UOx

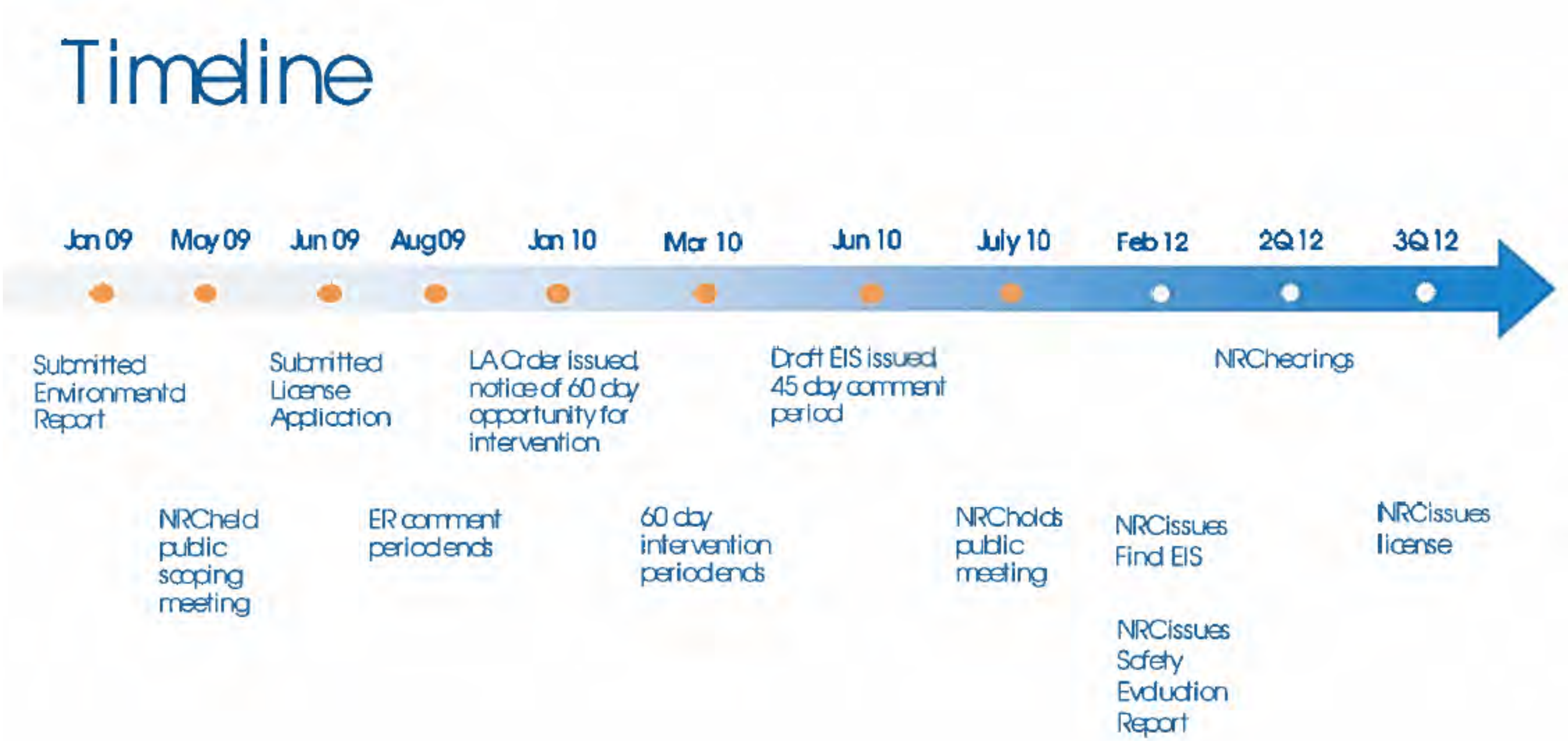
Material forms enriched to no more than 5.0 wt% U235

HALEU requires SAR update, corresponding CSI change

Global Nuclear Fuel

NRC Review Timeline – An Example

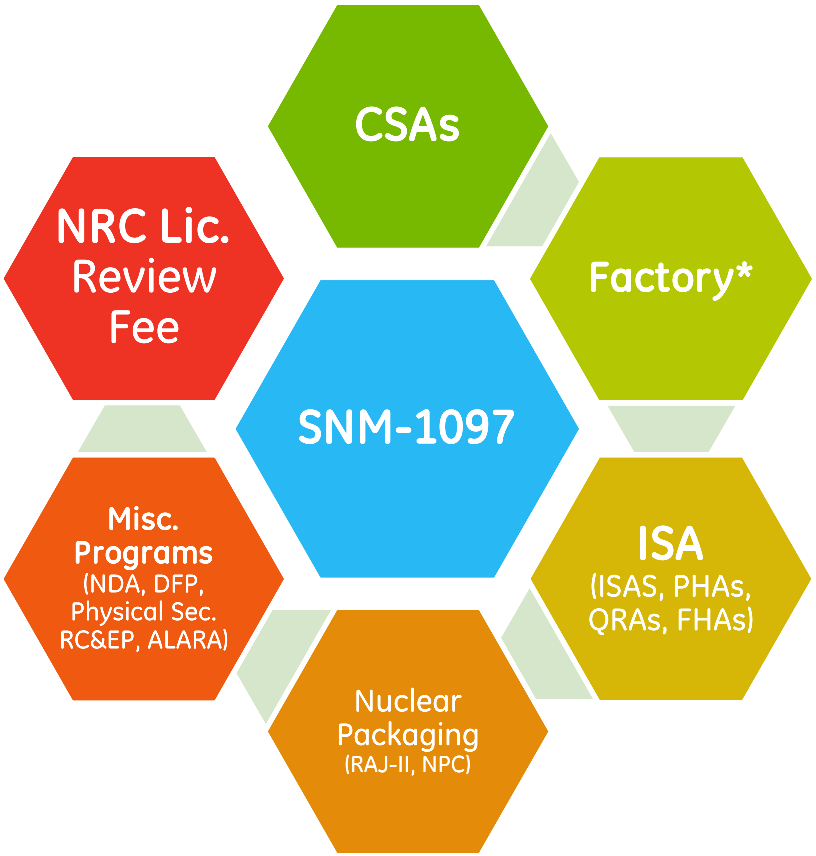
Below SNM-2019 License Application provides a recent real-world **example timeline** for full scope HALEU licensing.



Above time scale can be compressed in SNM-1097 License Amendment pursued.

Global Nuclear Fuel

GNF-A HALEU Cost Elements



SNM-1097 License Amendment (LA) approach

* Extent of factory changes depends on fuel form, enrichment limit

Global Nuclear Fuel

Summary

- Enrichment cascades can be licensed to produce higher assay UF6 feed; labs can support *feed* to u-metal pilot

- UF6 **feed transport** in 30B greater than 5% is technically feasible, but will create industry challenge to align regulations, standards, and certificate; u-metal transport can be authorized.

- **Product transport** will require SAR revisions/CSI change

- GNFA has **proven experience** in higher assay LEU licensing; fuel form selected will ultimately dictate cost.

- For u-metal, demonstration facility can capitalize on (i) existing environmental permits, (ii) NRC licensed facility, (iii) site security and infrastructure, (iv) established NRC/DOT nuclear packaging program, and (v) existing DFP.

Packages for Shipping 20% Enriched Materials

Author:
Andy Langston

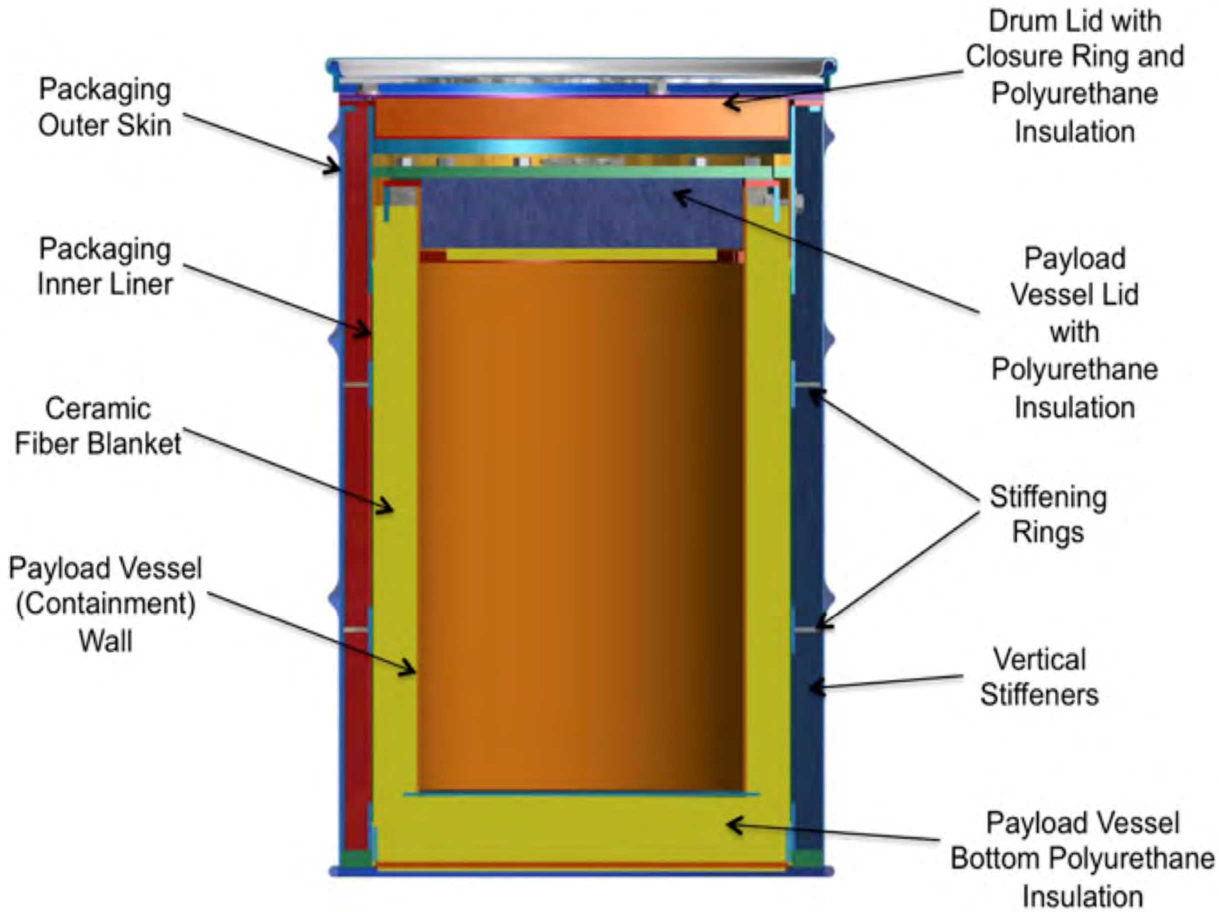
INL-NEI Technical Workshop on Transportation of High Assay Low-Enriched Uranium, August 30-31, 2018

Daher-TLI
8161 Maple Lawn Blvd., Suite 480
Fulton, MD 20759 U.S.A
(301) 421-4324

Currently Available Packages

- Majority of 20% shipped in DOE complex using drum type packages
- Example the Versa-Pac
- Currently under NRC Amendment Application for 1S/2S cylinder transport

General Design



Current Contents

- Contents limits based on uranium metal to bound likely contents
    - uranium oxides, uranium metal, uranyl nitrate crystals and other uranium compounds, e.g., uranium carbides, uranyl fluorides and uranyl carbonates, and thorium 232 as TRISO fuel
- 11.4 kw content limit

Weight percent U-235	U-235 Mass Limit (g)	
	General Limit	5-inch Pipe
≤ 100	350	695
≤ 20	410	1,215
≤ 10	470	1,605
≤ 5	580	1,065
≤ 1.25	2,000	--

Model No.	Packaging OD (in.)	Packaging Height (in.)	Payload Containment Cavity ID (in.)	Payload Containment Cavity Height (in.)	Pacakging Weight (lbs.)	Maximum loaded weight (lbs.)
VP-55	23-1/16	34 ¾	15	25-7/8	390	750
VP-110	30-7/16	42 ¾	21	29-3/4	702	965

Content Addition – UF_6 1S and 2S Cylinders

- ANSI N14.1-compliant 1S and 2S cylinders
- Criticality safety HAC evaluation assumes cylinders do not survive, NCT evaluations credit cylinder geometry
- Criticality results limit the quantity and type of UF_6 cylinders
  - 100 wt.% limited by fit of pipes in the cavity
- Thermal evaluation requires a 2 inch thick foam liner in the cavity

TABLE. Summary Table of 1S/2S Cylinder Modeling Configuration

Content	Configuration			
	20 wt.%		100 wt.%	
	1S	2S	1S	2S
Quantity of cylinders	7 cylinders	2 cylinders	1 cylinders * in 5-inch pipe	1 cylinder * in 5-inch pipe

*operation limit

Content Addition – Air Transport (criticality)

- Air transport – 1 package - packaging assumed to not survive

Allowable Payloads by Enrichment, Versa-Pac Configuration

wt.% ^{235}U	Configuration			
	VP-55 / VP-110		VP-55 (5-inch Pipe)	
	Mass ^{235}U (g)		Mass ^{235}U (g)	
	General	Air transport	General	Air Transport
≤ 100	350	350	695	395
≤ 20	410	410	1,215	495
≤ 10	470	470	1,605	590
≤ 5	580	580	1,065 *	790
≤ 1.25	2000	--	--	--

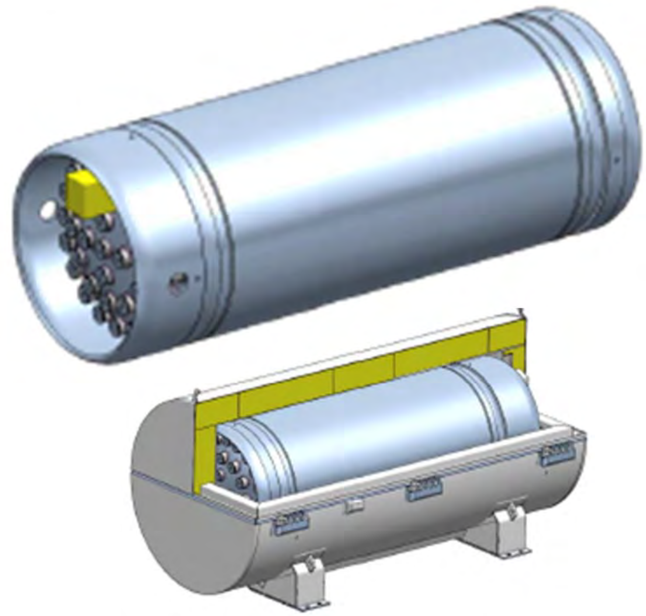
* This value is volume limited

Under Development – 30B cylinder designs up to 20% enrichment

The 30B-20 cylinder is designed based on the 30B cylinder as per the ISO 7195/ANSI N14.1 standards. Thanks to its state-of-the-art confinement system for criticality control, it can safely accommodate up to 20% enriched 1600 kg of UF6 which represents a tremendous increase in transport capacities.

The 30B-20 cylinder can be operated and handled the same way as a 30B cylinder and does not require any special retrofitting for already existing plants. This efficiently reduces costs for plant-related adjustments and additional staff training.

The cylinder will have to be licensed as part of a package system.



30B-20 cylinder Technical Data	
Nominal Diameter (mm)	762
Nominal Length (mm)	2060
Max. Net UF_6 Weight (kg)	1600
Max. Gross Weight (kg)	2710
Enrichtment (wt. % ^{235}U)	max. 20

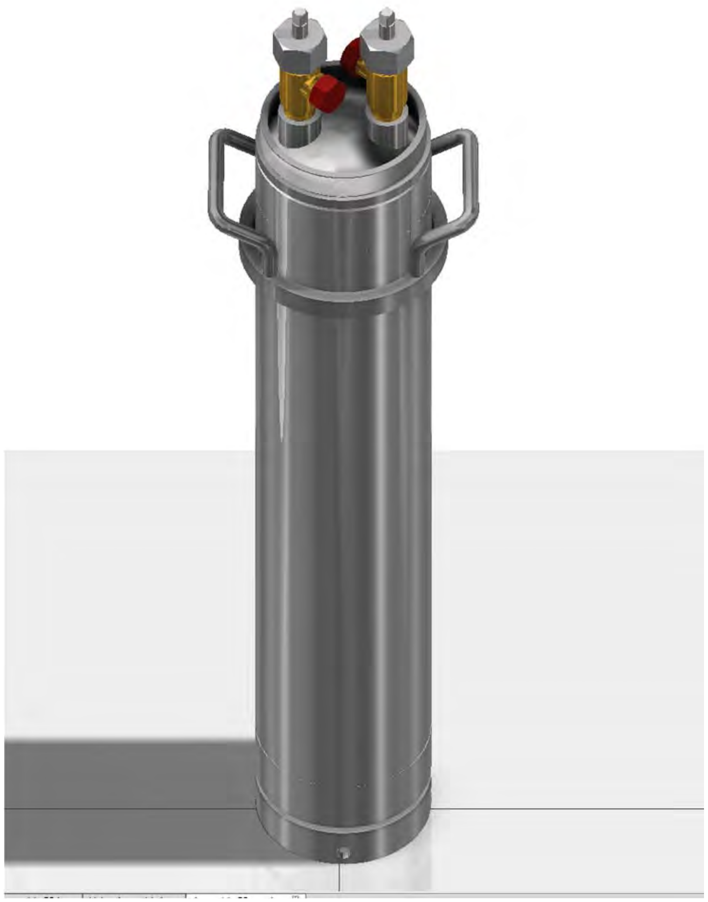
Under Development – Package for 5B/A Cylinders and TRIGA Fuel

- VP-55XL is an enhanced version of the TLI's NRC approved VP-55 (55-gal Type A package).
- In addition to the increased height the VP-55XL design includes an added thermal insulation.

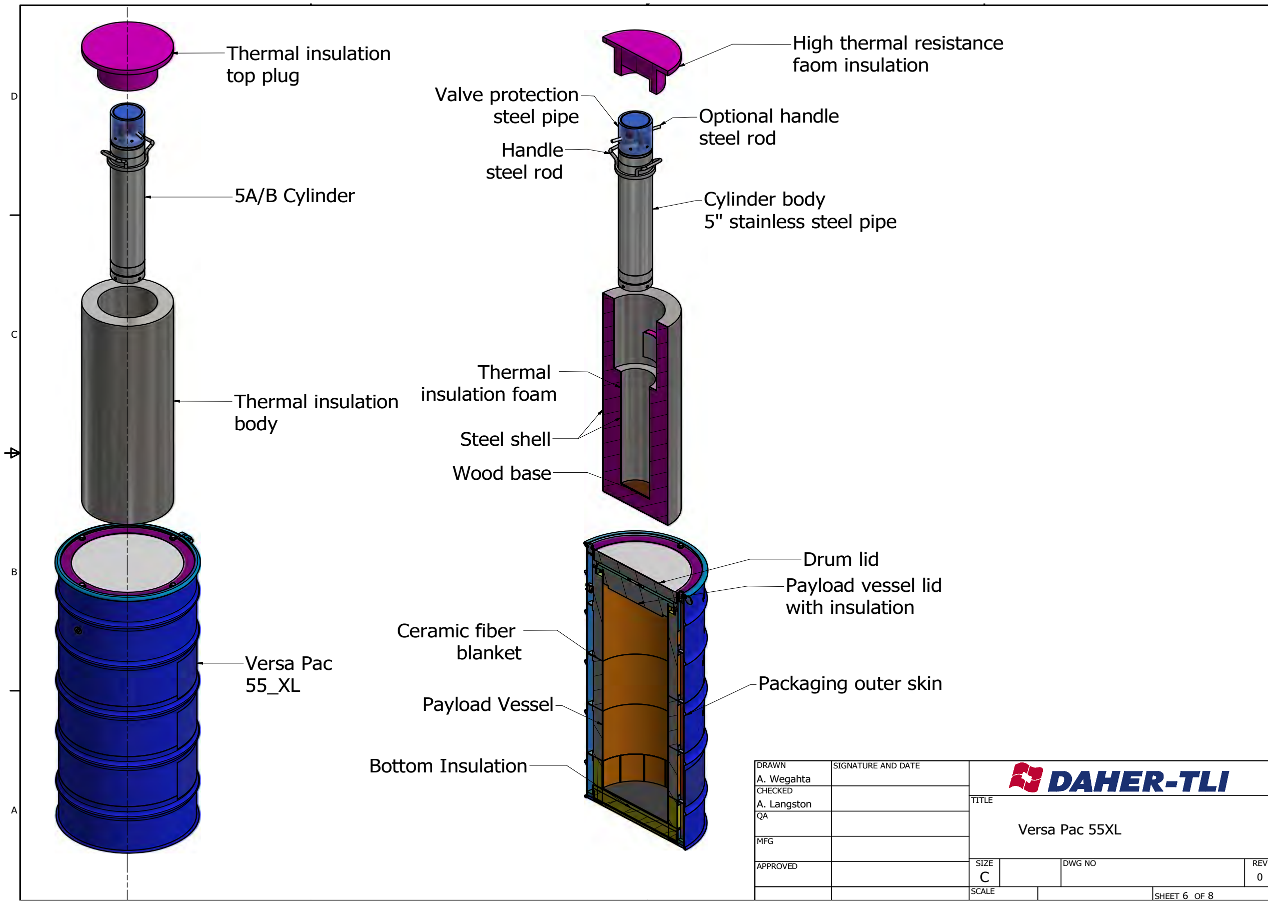
Dimensions		
	VP-55	VP-55XL
Overall height	34.8 in	55.92 in
Outer diameter	22.5 in	22.5 in
Cavity height	26 in	36.6 in
Cavity diameter	15 in	15 in
Tare weight	390 lb	780 lb
Gross weight	640 lb	1170 lb

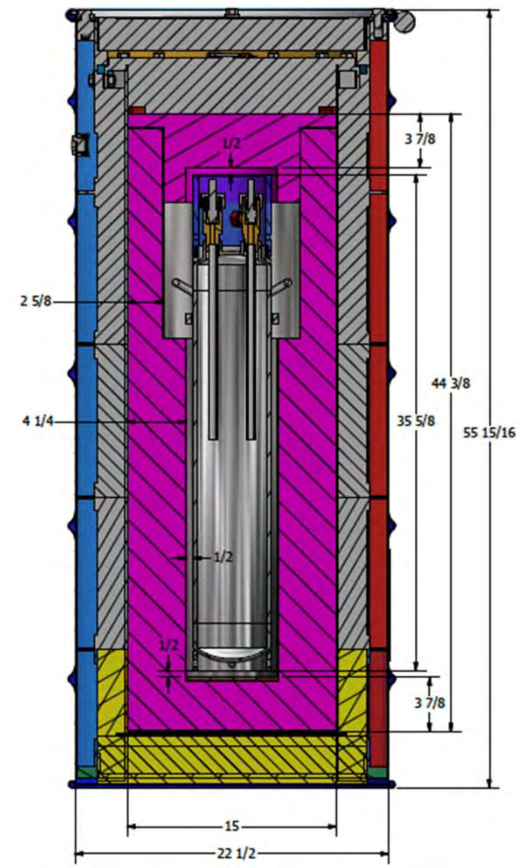
VP-55XL Contents

- Contents of the package include, but not limited to, 5B/A cylinders and TRIGA fuel bundles to transport fissile radioactive material.
- 5B/A dimensions are:
  - Gross weight = 110 lb
  - Overall height = 35.625 in
  - Outer diameter = 5.563 in
- Temperature range is -40°F to 250°F.



VP-55XL Components

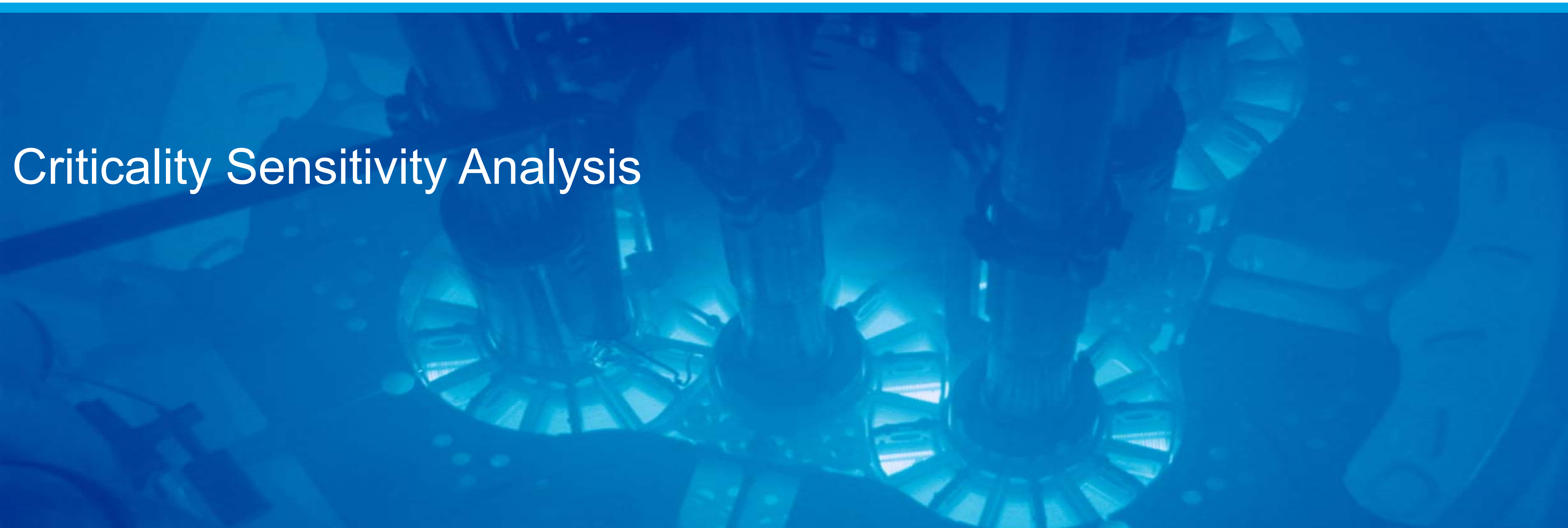




QUESTIONS?

U.S. DEPARTMENT OF ENERGY | Office of NUCLEAR ENERGY

Clean. Reliable. Nuclear.



Criticality Sensitivity Analysis

Presented by:
Bradley T. Rearden, Ph.D.
National Technical Director
Nuclear Data and Benchmarking Program

Presented to:
INL-NEI Technical Workshop on Transportation of High Assay Low-Enriched Uranium
August 30-31, 2018
Nuclear Energy Institute
August 30, 2018

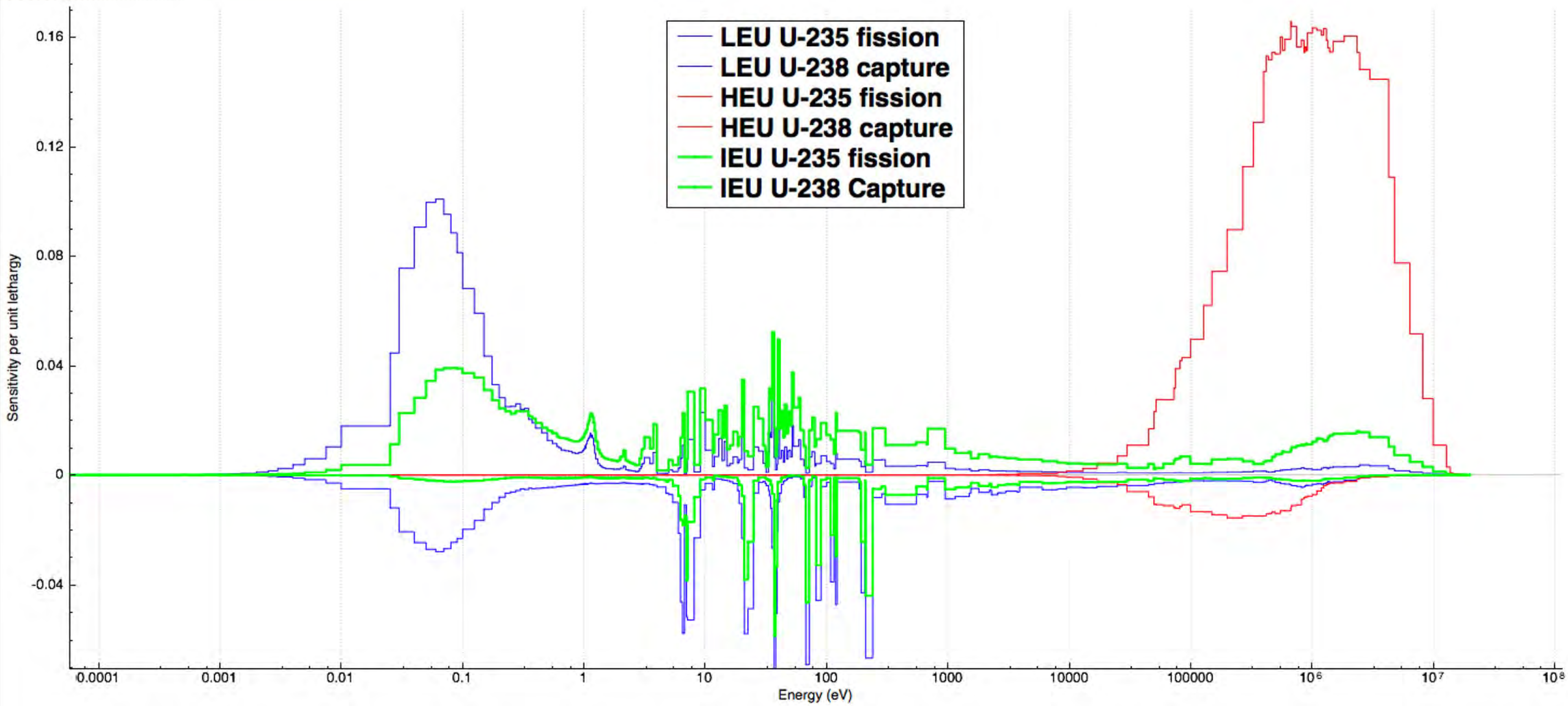
Knowledge Management

“There are known knowns; there are things we know that we know. There are known unknowns; that is to say, there are things that we now know we don't know. But there are also unknown unknowns – there are things we do not know we don't know.”
-United States Secretary of Defense, Donald Rumsfeld, 2002

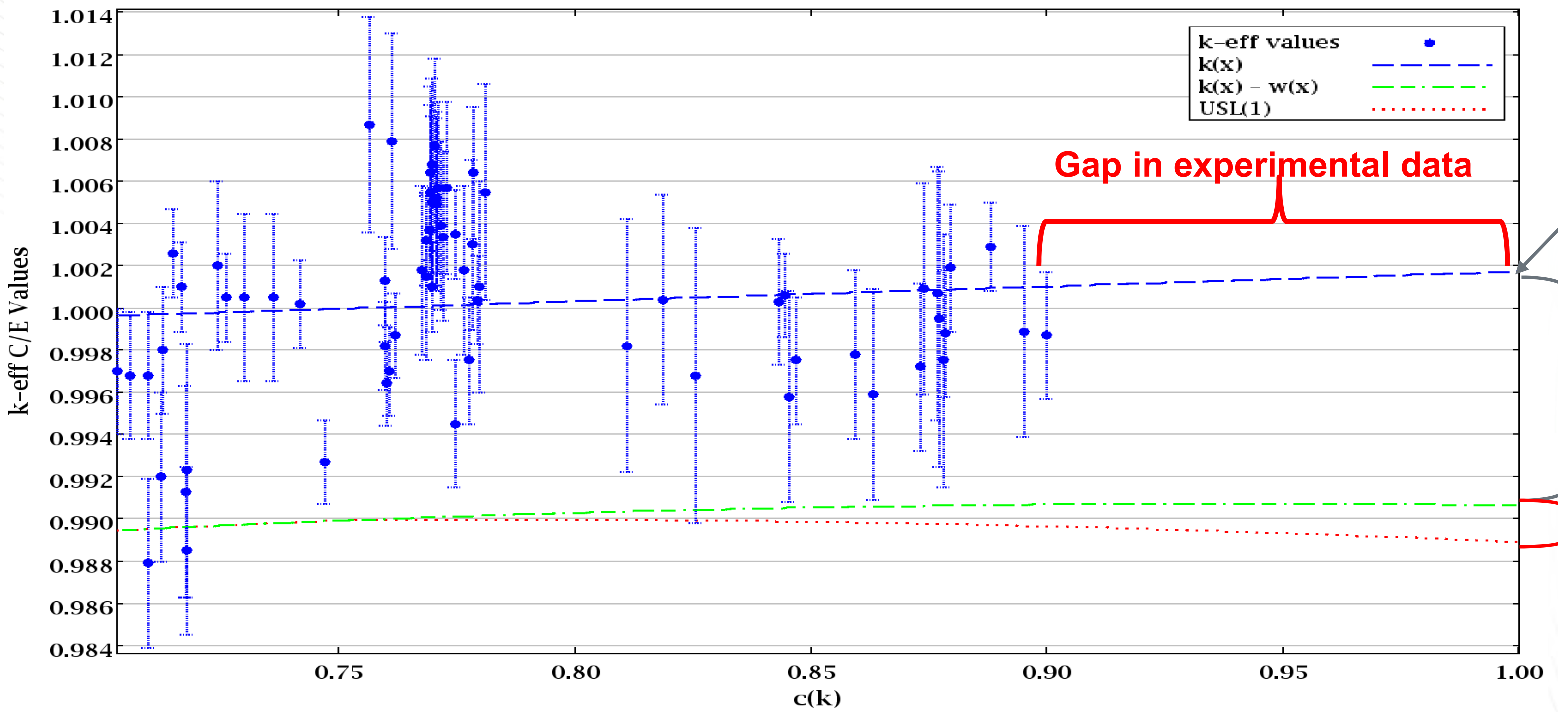
KNOWN KNOWNS Measurements/Observations	**KNOWN UNKNOWNS** Uncertainty Quantification
UNKNOWN KNOWNS Communication	**UNKNOWN UNKNOWNS** Safety Margins

“All models are wrong, some are useful.”
-George E. P. Box – Statistician, Professor, Univ. of Wisconsin

Reminder:
Sensitivity of k_{eff} to nuclear data for LEU, HEU and IEU benchmarks



Reminder: Cross section similarity as independent parameter for trending analysis



Biased k_{eff} for Application

(bias is this intercept - 1.0)

Confidence band

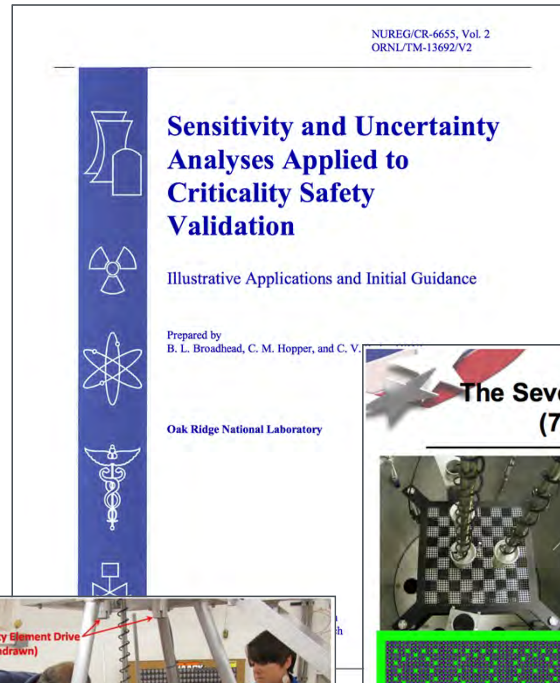
(uncertainty in bias)

Positive Bias Adjustment

Previous activities on fuel cycle analysis for high-burnup fuel

- S/U methods applied for investigation and design of experimental benchmarks and for safety margin assessment
- Need to move beyond 5% regulatory limit

The 30B cylinder: . . . *can contain 2270 kilograms of low-enriched uranium in the form of uranium hexafluoride. IAEA regulations include requirements for packages to meet the following test requirements: withstand a pressure test of at least 1.4 MPa; withstand a free drop test; withstand a thermal test at a temperature of 800 °C for 30 minutes* (World Nuclear News).



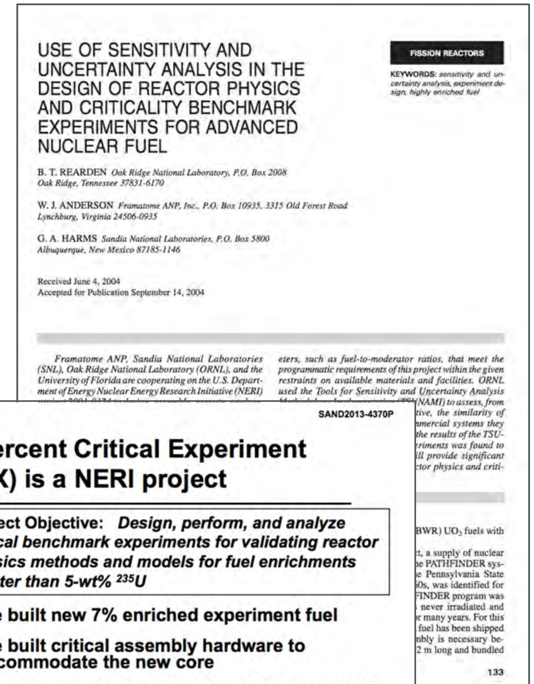
NUREG/CR-6655, Vol. 2
ORNL/TM-13692/V2

Sensitivity and Uncertainty Analyses Applied to Criticality Safety Validation

Illustrative Applications and Initial Guidance

Prepared by
B. L. Broadhead, C. M. Hopper, and C. V.

Oak Ridge National Laboratory



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

FISSION REACTORS

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

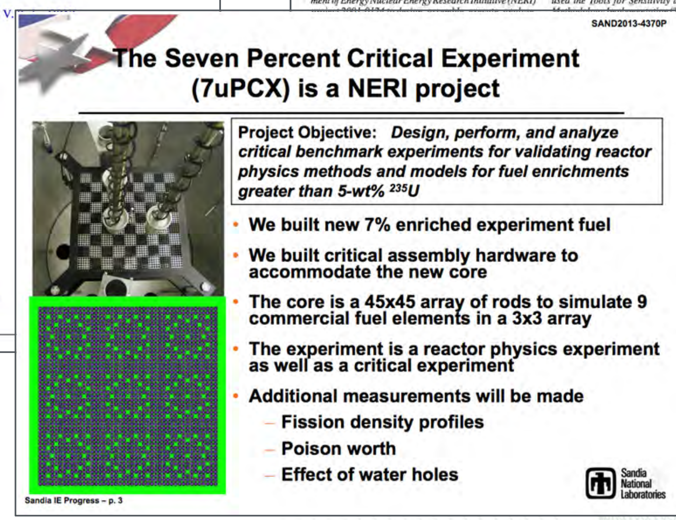
B. T. REARDEN Oak Ridge National Laboratory, P.O. Box 2008 Oak Ridge, Tennessee 37831-6170

W. J. ANDERSON Framatome ANP, Inc., P.O. Box 10935, 3315 Old Forest Road Lynchburg, Virginia 24506-0935

G. A. HARMS Sandia National Laboratories, P.O. Box 5800 Albuquerque, New Mexico 87185-1146

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

133



SAND2013-4370P

The Seven Percent Critical Experiment (7uPCX) is a NERI project

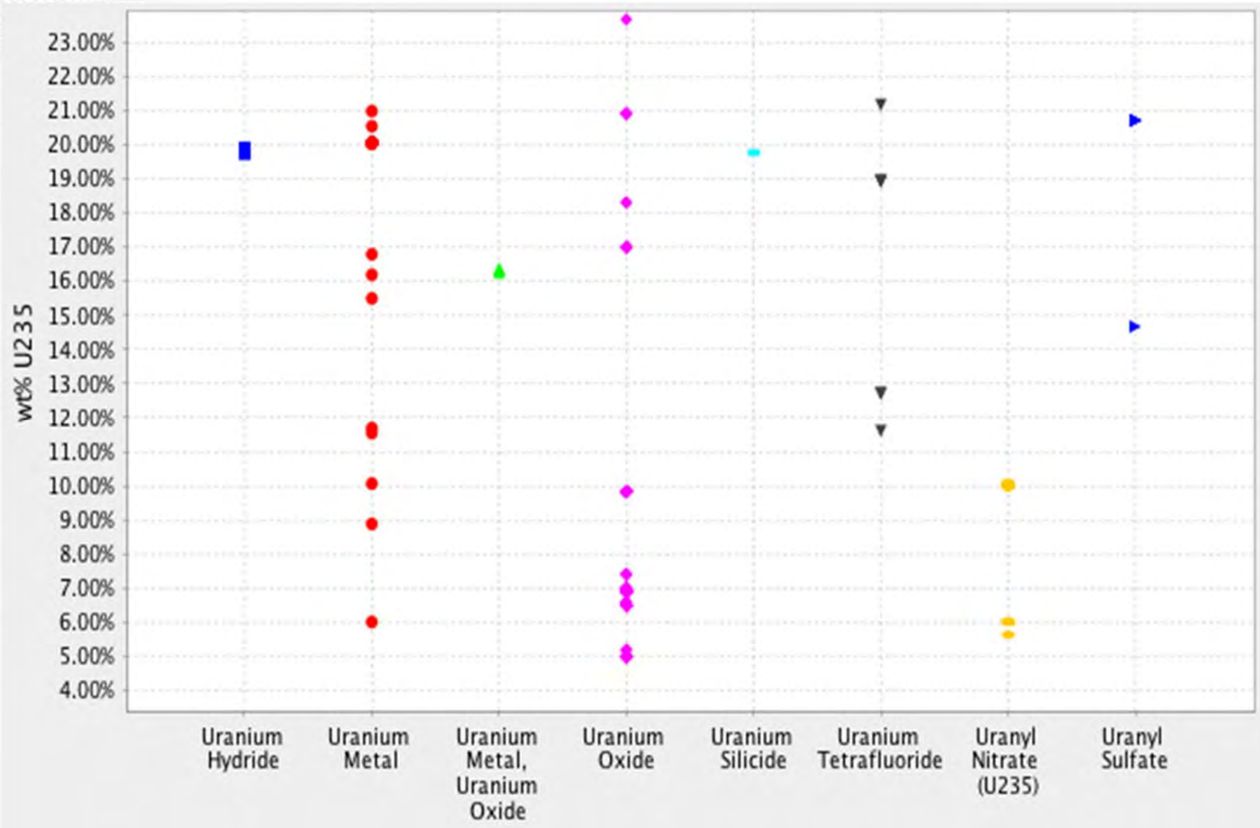
Project Objective: *Design, perform, and analyze critical benchmark experiments for validating reactor physics methods and models for fuel enrichments greater than 5-wt% ^{235}U*

- We built new 7% enriched experiment fuel
- We built critical assembly hardware to accommodate the new core
- The core is a 45x45 array of rods to simulate 9 commercial fuel elements in a 3x3 array
- The experiment is a reactor physics experiment as well as a critical experiment
- Additional measurements will be made
  - Fission density profiles
  - Poison worth
  - Effect of water holes

Sandia National Laboratories

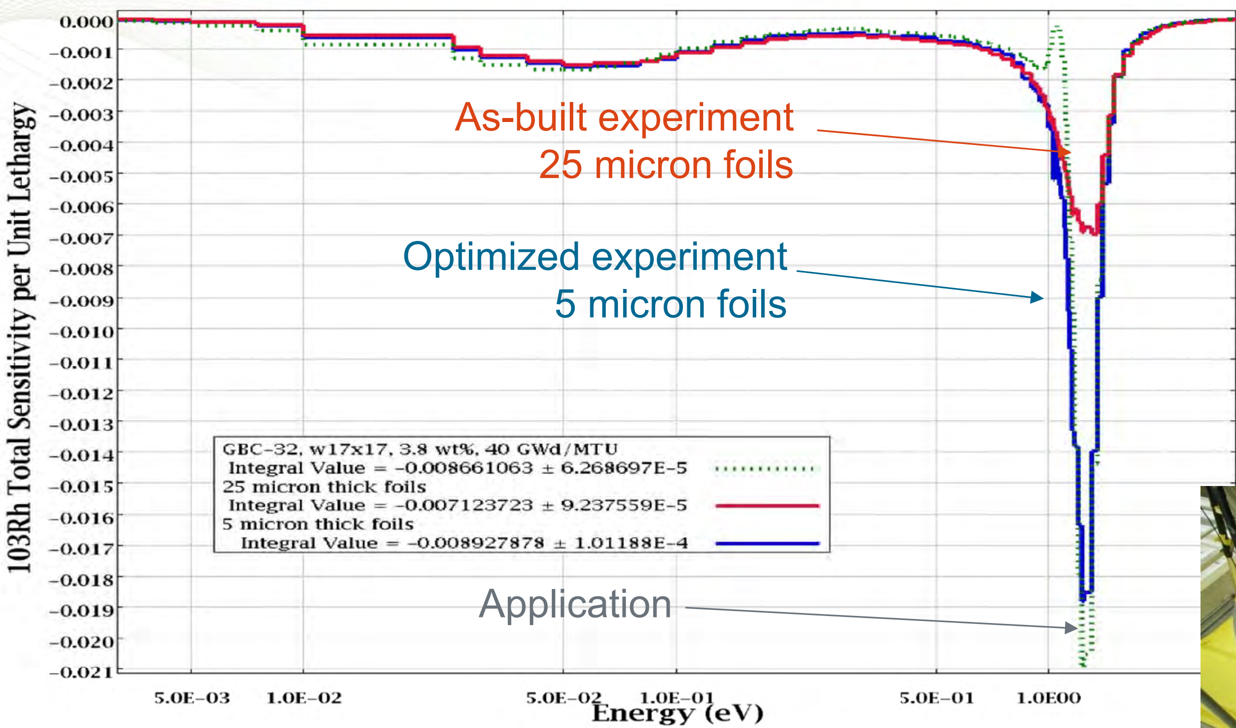
Sandia IE Progress – p. 3

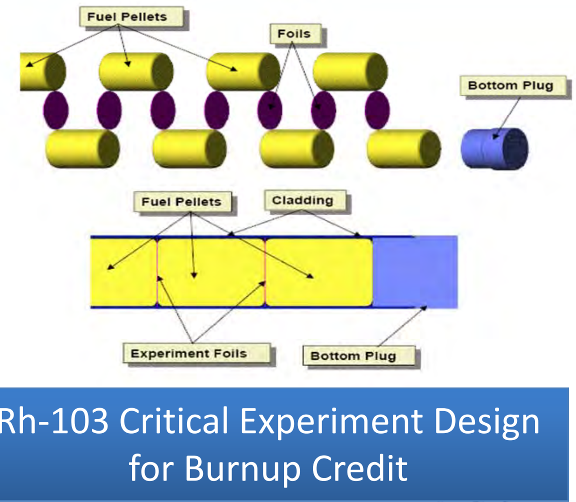
376 ICSBEP experiments with 5% < ^{235}U wt% < 25%

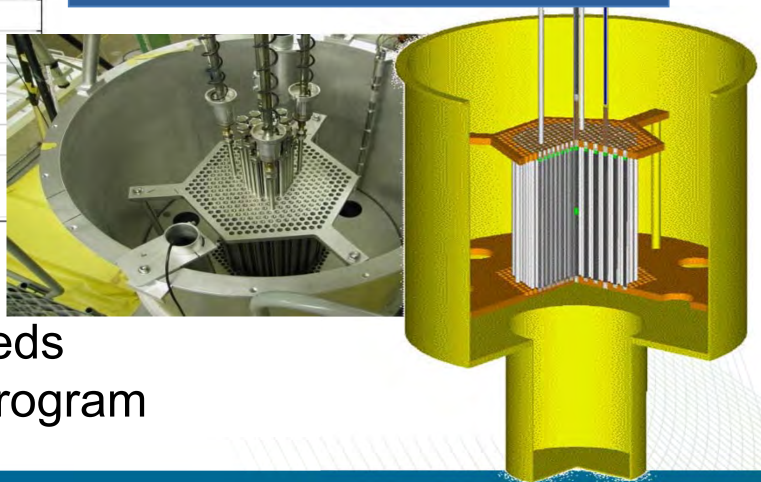


- Many legacy experiments for metallic cores
- IRPhEP has a few experiments for HTGR (HTR-10, HTTR)
- No experiments for molten salt (limited new measurements in Czech republic for non-fueled FLiBe)
- No data for FHR

Design of optimized experiments

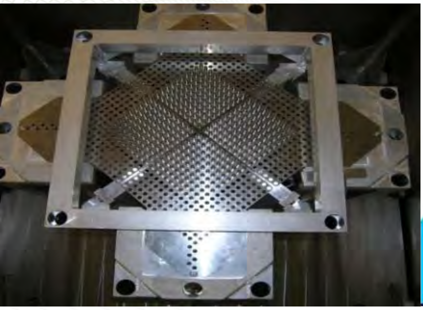
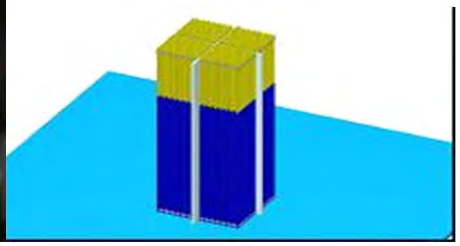
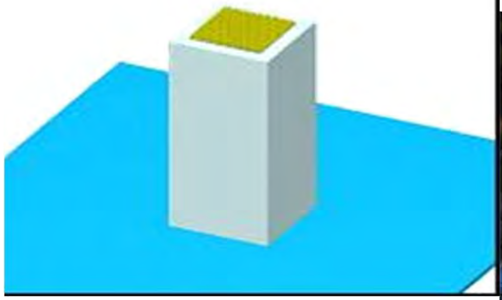




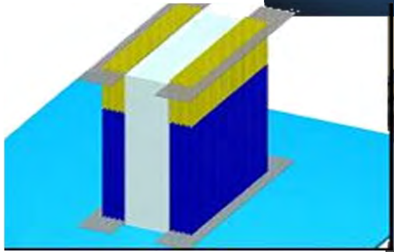
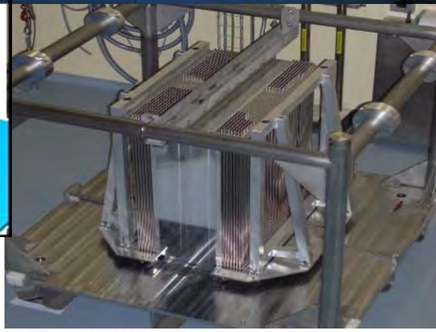


- Experiment designs optimized to meet application needs
- Required analysis in DOE Nuclear Criticality Safety Program

Design of MIRTE reference experiments


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

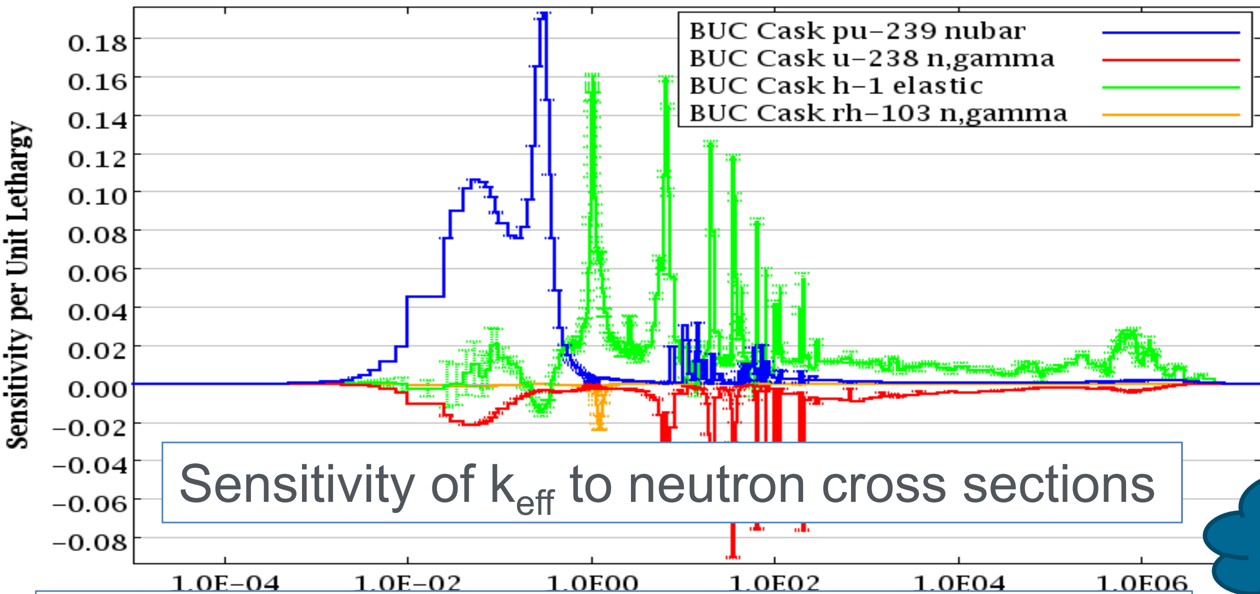
Comments on Use of S/U in Validation

- Provides advanced methods for challenging validation scenarios.
- Allows for combining information from many diverse experiments.
- Extracts and projects bias information from replacement experiments.
- Surrogate for validation to fill gaps where experiments are not available.
- Design of new experiments targeted to meet application needs.
- Data and tools readily available for production use.

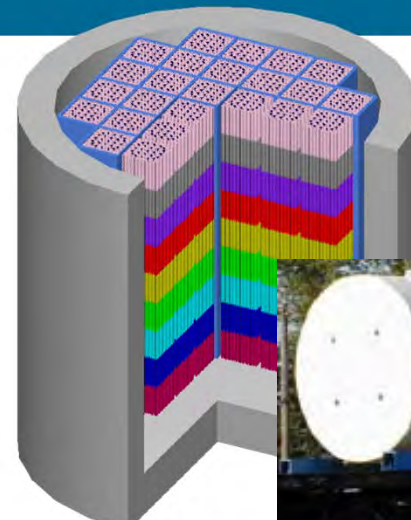
Reminder: Knowledge Management

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

Identifying important processes and uncertainties

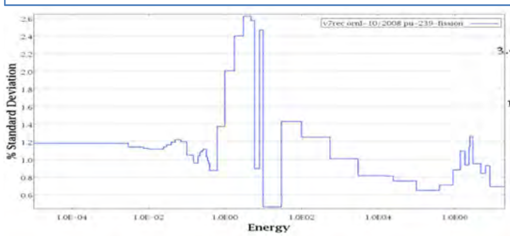


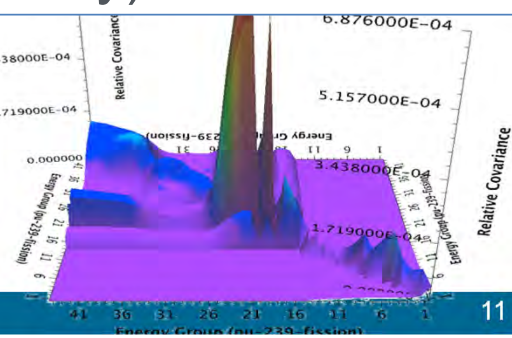
Sensitivity of k_{eff} to neutron cross sections



Known Unknown

Covariance (uncertainty) for cross sections

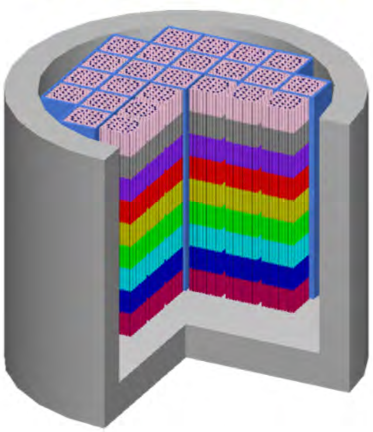


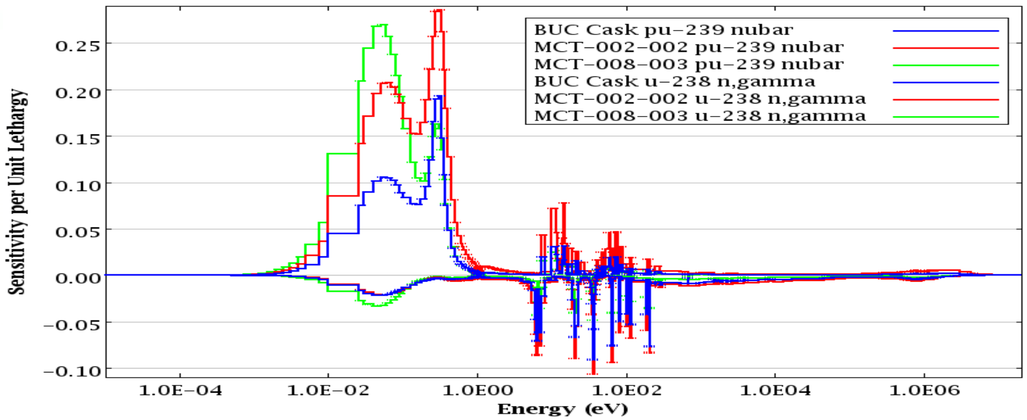


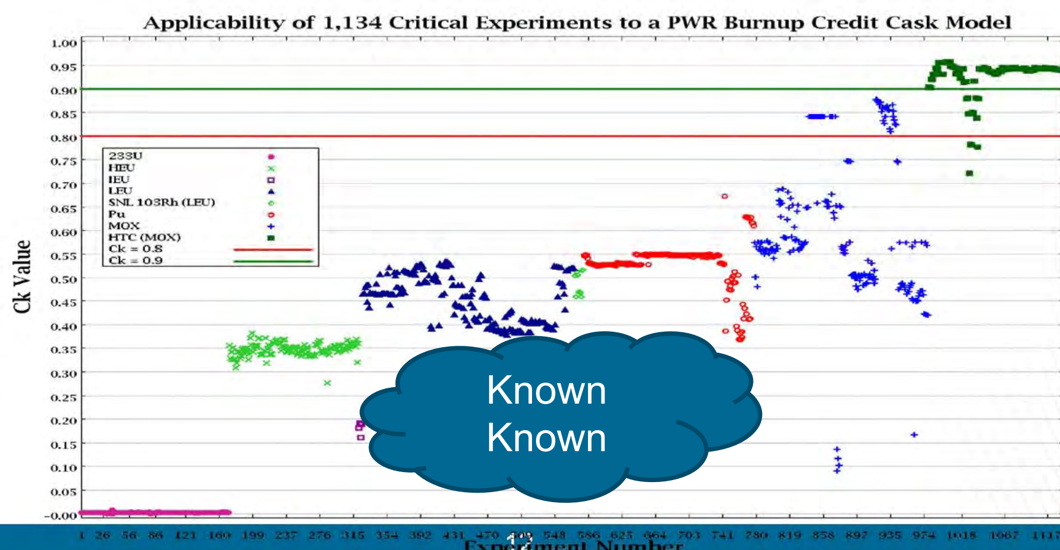
Covariance Matrix		Unc. in % dk/k
Nuclide-Reaction	Nuclide-Reaction	Due to this Matrix
^{239}Pu nubar	^{239}Pu nubar	4.0032E-01 ± 2.5625E-06
^{238}U n,gamma	^{238}U n,gamma	1.9457E-01 ± 1.2387E-05
^{239}Pu fission	^{239}Pu fission	1.5501E-01 ± 1.0838E-05
^{235}U nubar	^{235}U nubar	1.3981E-01 ± 5.0038E-07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

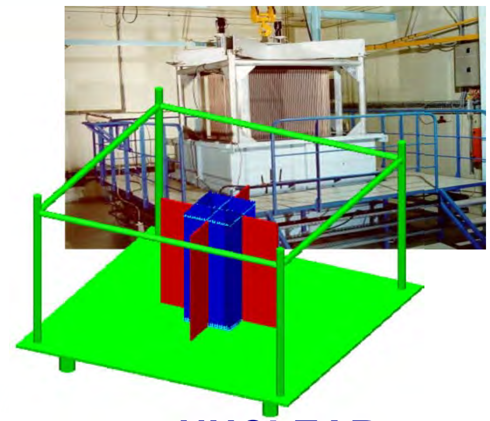
Code Validation: Identification of Laboratory Experiments that are Similar to the Targeted Application

APPLICATION

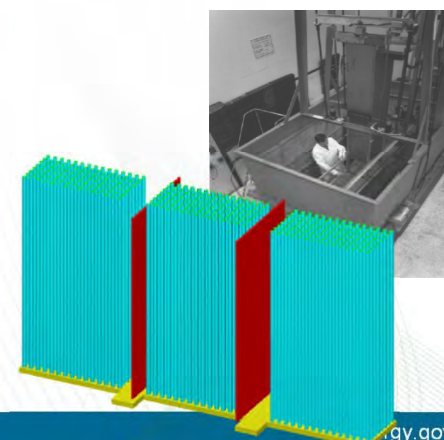




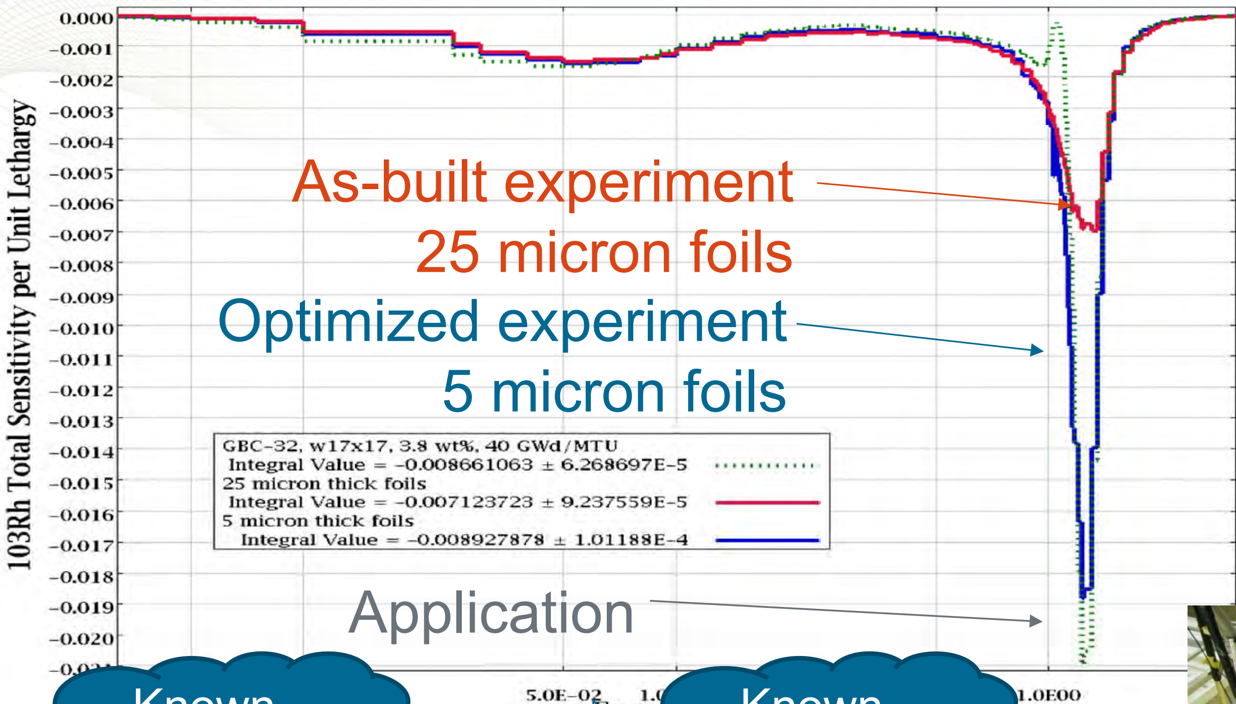


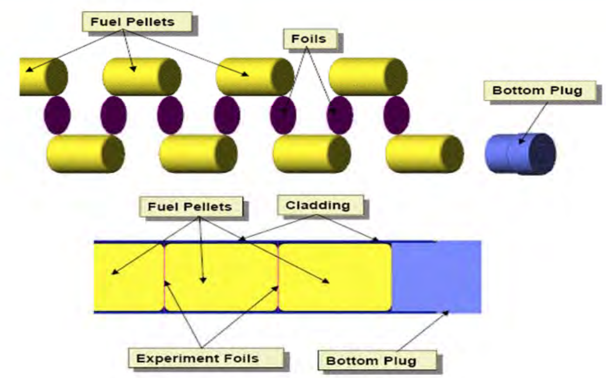


NUCLEAR CRITICALITY EXPERIMENTS



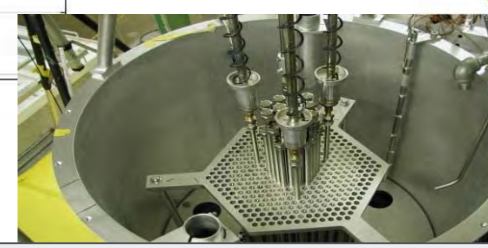
Design of optimized experiments in US and abroad

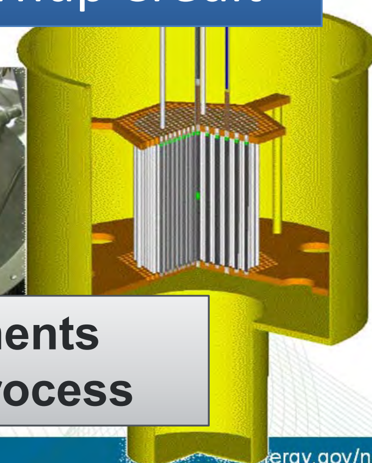




Rh-103 Critical Experiment Design for Burnup Credit

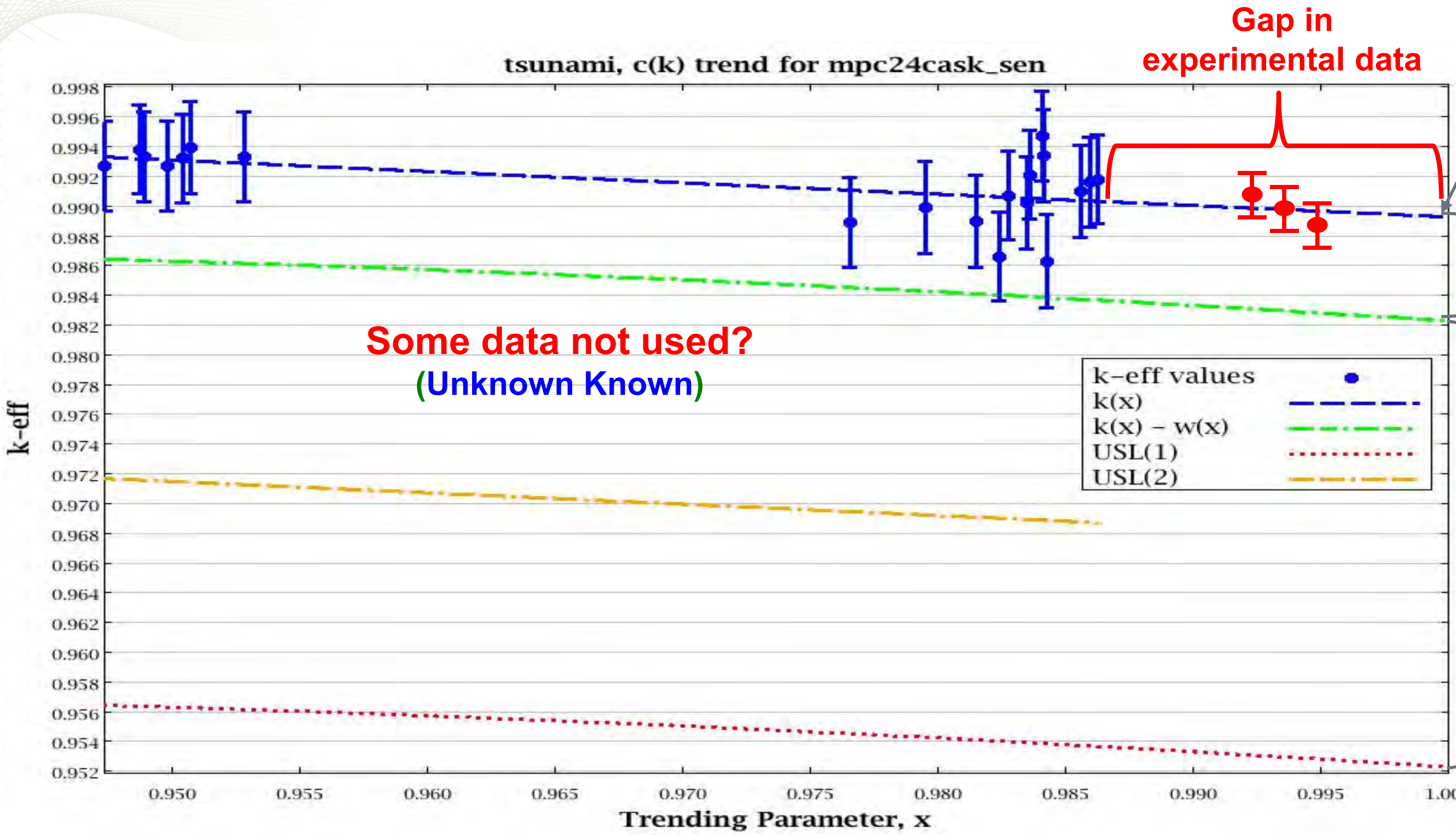
Known Unknown → Known Known





Experiment designs optimized to fill gaps not met by other experiments
Required analysis in DOE Nuclear Criticality Safety Program C_EdT Process

Setting safety limits



Experiments projected to application
(**Known Known**)

Confidence band
(**Known Unkown**)

Safety margin
(**Unknown Unkown**)

Sensitivity/uncertainty analysis methods in practice

- U.S. Nuclear Regulatory Commission
  - Nuclear Materials Safety and Safeguards, Nuclear Reactor Regulation, Office of New Reactors
- U.S. DOE / Areva / Duke Energy
  - Mixed Oxide Fuel Fabrication Facility
- Candu Energy
  - ACR-1000 Design Validation
- NRC / Atomic Energy of Canada, Ltd.
  - ACR-700 NRC Review/PIRT
- U.S. DOE
  - Yucca Mountain post-closure criticality safety
- Global Nuclear Fuels
  - Transportation package licensing
- Svensk Kärnbränslehantering AB (SKB)
  - Swedish used fuel repository
- Organization for Economic Cooperation and Development, Nuclear Energy Agency / International Atomic Energy Agency
  - International Expert Groups


U.S.NRC
Protecting People and the Environment
NUREG/CR-7109
ORNL/TM-2011/514

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

Nuclear Science
NEA/NSC/WPNCS/DOC(2013)7
October 2013

Overview of Approaches Used to Determine Calculational Bias in Criticality Safety Assessment

OECD
NEA

Fission nuclear data programs and prioritization

OECD Nuclear Energy Agency high priority request list

NEA Nuclear Data High Priority Request List

HPRL Main	High Priority Requests (HPR)	General Requests (GR)	Special Purpose Quantities (SPQ)	New Request	SGC/HPRL Documents

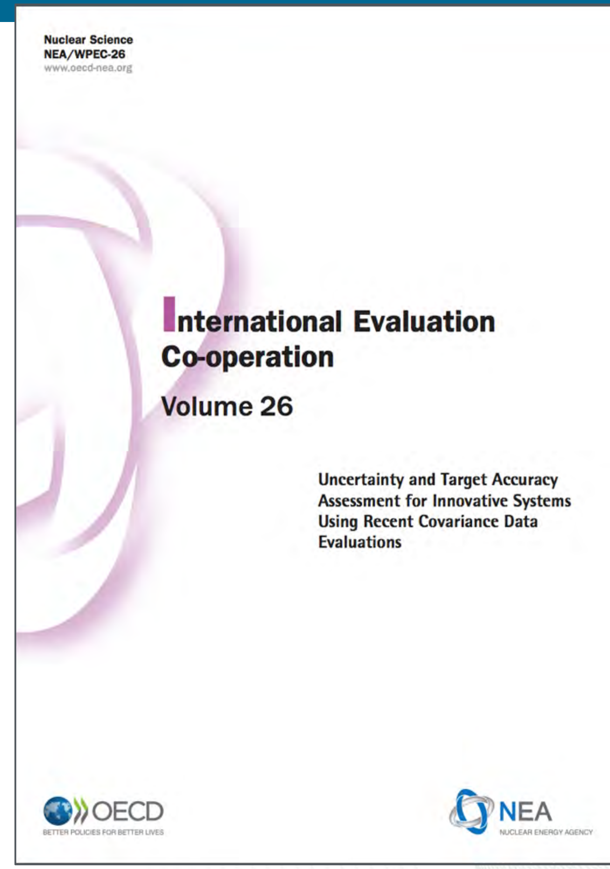
Results of your search in the request list

Requests are shown from the following list(s):
High Priority (H)

Explanations of each column can be found in the table heads. To view the details of a request, please click on the **link symbol** after the request ID.
To send a comment on a particular entry, please view the request, and click on the **'letter'** symbol there.

ID	View	Target	Reaction	Quantity	Energy range	Sec.E/Angle	Accuracy	Cov	Field	Date
2H		8-O-16	(n,a),(n,abs)	SIG	2 MeV-20 MeV		See details	Y	Fission	12-SEP-08
3H		94-PU-239	(n,f)	prompt g	Thermal-Fast	Eg=0-10MeV	7.5	Y	Fission	12-MAY-06
4H		92-U-235	(n,f)	prompt g	Thermal-Fast	Eg=0-10MeV	7.5	Y	Fission	12-MAY-06
5H		72-HF-0	(n,g)	SIG	0.5-5.0 keV		4	Y	Fission	16-APR-07
8H		1-H-2	(n,el)	DA/DE	0.1 MeV-1 MeV	0-180 Deg	5	Y	Fission	16-APR-07
12H		92-U-235	(n,g)	SIG,RP	100 eV-1 MeV		3	Y	Fission	06-NOV-07
15H		95-AM-241	(n,g),(n,tot)	SIG	Thermal		See details		Fission	10-SEP-08
18H		92-U-238	(n,inl)	SIG	65 keV-20 MeV	Emis spec.	See details	Y	Fission	11-SEP-08
19H		94-PU-238	(n,f)	SIG	9 keV-6 MeV		See details	Y	Fission	11-SEP-08
21H		95-AM-241	(n,f)	SIG	180 keV-20 MeV		See details	Y	Fission	11-SEP-08
22H		95-AM-242	(n,f)	SIG	0.5 keV-6 MeV		See details	Y	Fission	11-SEP-08
25H		96-CM-244	(n,f)	SIG	65 keV-6 MeV		See details	Y	Fission	12-SEP-08
27H		96-CM-245	(n,f)	SIG	0.5 keV-6 MeV		See details	Y	Fission	12-SEP-08
29H		11-NA-23	(n,inl)	SIG	0.5 MeV-1.3 MeV	Emis spec.	See details	Y	Fission	12-SEP-08
32H		94-PU-239	(n,g)	SIG	0.1 eV-1.35 MeV		See details	Y	Fission	12-SEP-08
33H		94-PU-241	(n,g)	SIG	0.1 eV-1.35 MeV		See details	Y	Fission	12-SEP-08
34H		26-FE-56	(n,inl)	SIG	0.5 MeV-20 MeV	Emis spec.	See details	Y	Fission	12-SEP-08
35H		94-PU-241	(n,f)	SIG	0.5 eV-1.35 MeV		See details	Y	Fission	12-SEP-08
36H		92-U-238	(n,g)	SIG	20 eV-25 keV		See details	Y	Fission	15-SEP-08
37H		94-PU-240	(n,f)	SIG	0.5 keV-5 MeV		See details	Y	Fission	15-SEP-08
38H		94-PU-240	(n,f)	nubar	200 keV-2 MeV		See details	Y	Fission	15-SEP-08
39H		94-PU-242	(n,f)	SIG	200 keV-20 MeV		See details	Y	Fission	15-SEP-08
40H		14-SI-28	(n,inl)	SIG	1.4 MeV-6 MeV		See details	Y	Fission	15-SEP-08
41H		82-PB-206	(n,inl)	SIG	0.5 MeV-6 MeV		See details	Y	Fission	15-SEP-08
42H		82-PB-207	(n,inl)	SIG	0.5 MeV-6 MeV		See details	Y	Fission	15-SEP-08
43H		1-H-1	(n,el)	SIG,DA	10 MeV-20 MeV	4 pi	1-2	Y	Standard	13-MAY-11
44H		93-NP-237	(n,f)	SIG,DE	200 keV-20 MeV			Y	Fission	18-MAY-15
45H		19-K-39	(n,p),(n,np)	SIG	10 MeV-20 MeV		10	Y	Fusion	11-JUL-17

Number of requests found: 28 (out of a total of 89 requests).
Download consolidated output report


Nuclear Science
NEA/WPEC-26
www.oecd-nea.org

International Evaluation Co-operation
Volume 26

Uncertainty and Target Accuracy Assessment for Innovative Systems Using Recent Covariance Data Evaluations

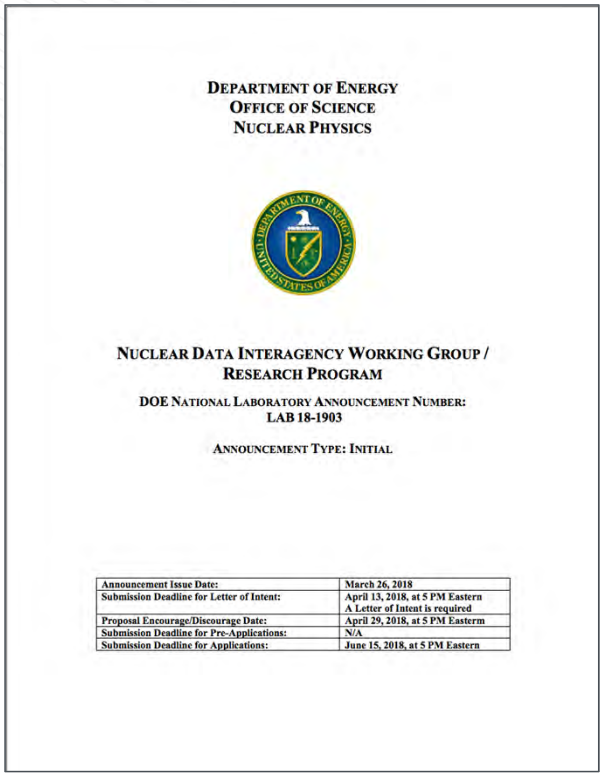
OECD
BETTER POLICIES FOR BETTER LIVES

NEA
NUCLEAR ENERGY AGENCY

https://www.oecd-nea.org/dbdata/hprl/

Nuclear Data Interagency Working Group

March 2018 FOA –
NP, ASCR, NE, NA-22



DEPARTMENT OF ENERGY
OFFICE OF SCIENCE
NUCLEAR PHYSICS

NUCLEAR DATA INTERAGENCY WORKING GROUP / RESEARCH PROGRAM

DOE NATIONAL LABORATORY ANNOUNCEMENT NUMBER: LAB 18-1903

ANNOUNCEMENT TYPE: INITIAL

Announcement Issue Date:	March 26, 2018
Submission Deadline for Letter of Intent:	April 13, 2018, at 5 PM Eastern A Letter of Intent is required
Proposal Encourage/Discourage Date:	April 29, 2018, at 5 PM Eastern
Submission Deadline for Pre-Applications:	N/A
Submission Deadline for Applications:	June 15, 2018, at 5 PM Eastern

Partners	Program Managers	Program Area	NDWG Member	Organization
NNSA/DNN R&D	Donny Hornback	Proliferation Detection	Catherine Romano (Chair) Candido Pereira	ORNL ANL
DOE/SC/Nuclear Physics	Tim Hallman Ted Barnes	Nuclear Physics/Nuclear Data	Lee Bernstein Dave Brown	LLNL BNL
NNSA/DNN R&D	Donna Wilt	Forensics / Post Detonation	Todd Bredeweg Jason Burke	LANL LLNL
DNDO/ Transformational & Applied Research	Namdoo Moon	Nuclear Detection		LANL
NNSA/NCSP	Angela Chambers	Criticality Safety	Mike Zerkle	NNL
NNSA/Defense Prog.	Ralph Schneider Staci Brown	Research and Development	Teresa Bailey	LLNL
NNSA/Defense Prog.	Douglas Wade Adam Boyd	Physics and Engineering Models	Bob Little	LANL
DOE/Nuclear Energy	Dan Funk Dave Henderson	Nuclear Energy	Brad Rearden	ORNL
DNDO /Forensics	William Ulicny Jeff Morrison	Forensics	Richard Essex	NIST
NNSA /Forensics	Tom Black Steve Goldberg	Nuclear Technical Forensics	Bob Rundberg	LANL
DOE/SC/ Isotope Office	Jehanne Gillo Dennis Phillips	Isotope Production	Meiring Nortier	LANL
NNSA/Nuclear Safeguards and Security	Arden Dougan	Safeguards Technology	Sean Stave	ORNL
NNSA/DNN R&D	Chris Ramos	Safeguards	Chris Pickett	ORNL
		Additional Expert Contributors	Mark Chadwick Patrick Talou Alejandro Sonzogni	LANL LANL BNL

Next steps

Now

- Identify candidate materials and transportation packages
- Perform nuclear data / benchmarking needs assessment and gap analysis
- <u>Proceed</u> with defensible safety margin (possibly at cost of efficiency)

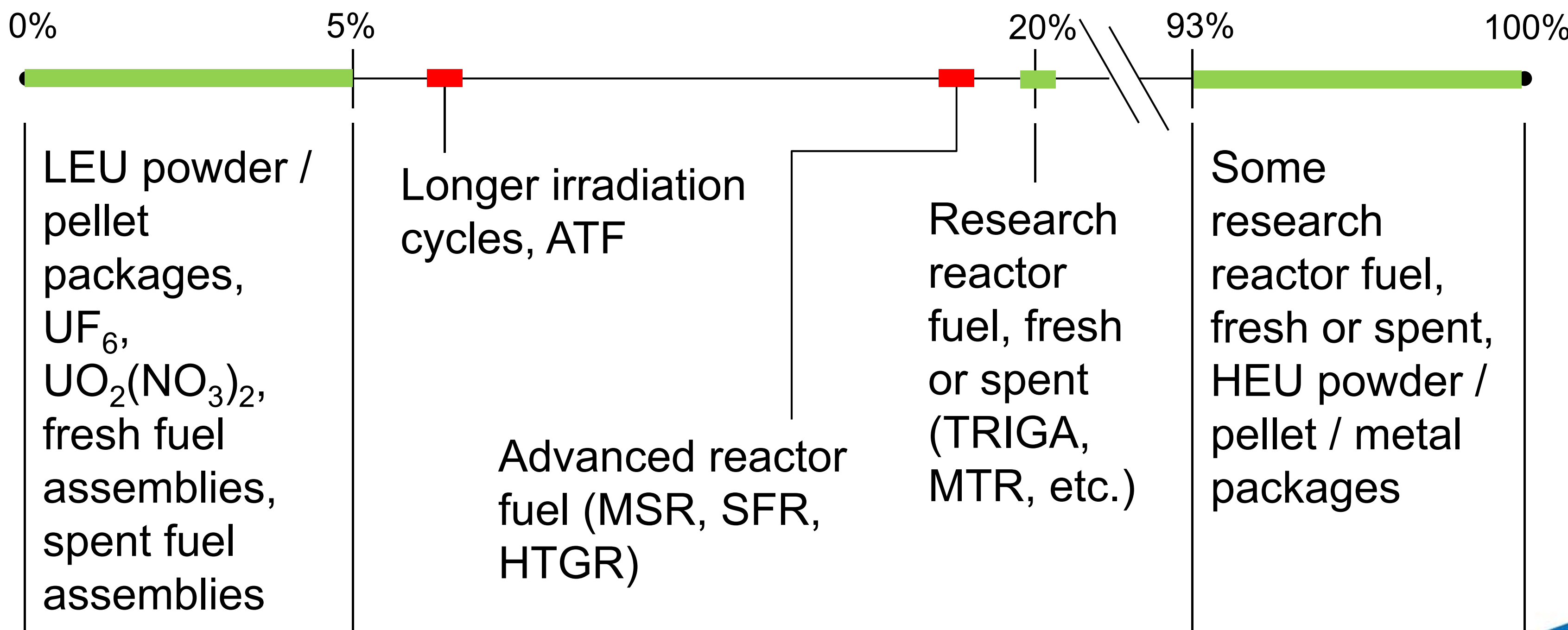
Ongoing R&D

- Extend use of existing data with advanced validation methods and training
- Nuclear data gaps
  - Generate improved evaluations
    - $1-2M,3-5 years per nuclide
- Benchmark experiment gaps
  - Mine existing experiments for information and document as archival benchmarks
    - $500k, 1-2 years per benchmark series
  - Build new critical experiments
    - Generate optimized experiment designs
    - Survey available facilities and materials and supplement as needed
    - $2-5M+++, 2-5 years for measurement
    - $500k,1-2 years to generate archival benchmark

Clean. **Reliable. Nuclear.**

>5.0 Weight Percent

Code Validation:



UF_6

- 10 CFR 71.55(g)(4) - *The uranium is enriched to not more than 5 weight percent uranium-235*.
- 49 CFR 173.417(a)(2) – “Heel” requirements: less than 5 weight percent in a 30-inch cylinder
- IAEA SSR-6 p. 680(a) – relief from water in-leakage requirement for UF_6 packages if enrichment is less than 5 weight percent
- ANSI N14.1 –
  - 30B/C, 12A/B enrichment limit: 5 weight percent
  - 8A: 12.5 weight percent
  - 5A/B: 100 weight percent
- ISO 7195 – similar to ANSI N14.1

Integral Experiments in the United States – Cost and Process

Douglas G. Bowen, Ph.D.
Nuclear Data and Criticality Safety Group Leader
Reactor and Nuclear Systems Division
Oak Ridge National Laboratory

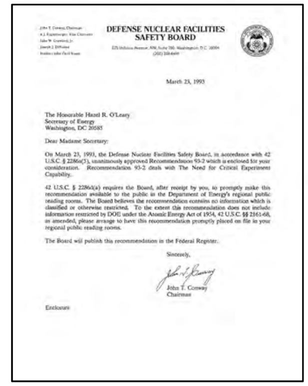
Nuclear Criticality Safety Program Execution Manager

INL-NEI Technical Workshop on Transportation of HALEU
August 31, 2018

Background / History

- Defense Nuclear Facilities Safety Board (DNFSB) Recommendations 93-2 and 97-2:
  - 93-2 (3/23/1993): Need for a general-purpose critical experiment capability that will ensure safety in handling and storage of fissionable material.
  - 97-2 (5/19/1997): Need for improved criticality safety practices and programs to alleviate potential adverse impacts on safety and productivity of DOE operations.
- 97-2 encompassed ongoing DOE activities of 93-2 while broadening scope to address important cross-cutting safety activities needed to ensure NCS throughout the Complex.
- DOE Implementation Plan for Board Recommendation 93-2 and 97-2 resulted in establishment of the US Nuclear Criticality Safety Program (NCSP)

Defense Nuclear Facilities Safety Board



DEFENSE NUCLEAR FACILITIES
SAFETY BOARD

March 23, 1993

The Honorable Hazel R. O'Leary
Secretary of Energy
Washington, DC 20585

Dear Madame Secretary:

On March 23, 1993, the Defense Nuclear Facilities Safety Board, in accordance with 42 U.S.C. § 2286a(5), unanimously approved Recommendation 93-2 which is enclosed for your consideration. Recommendation 93-2 deals with The Need for Critical Experiment Capability.

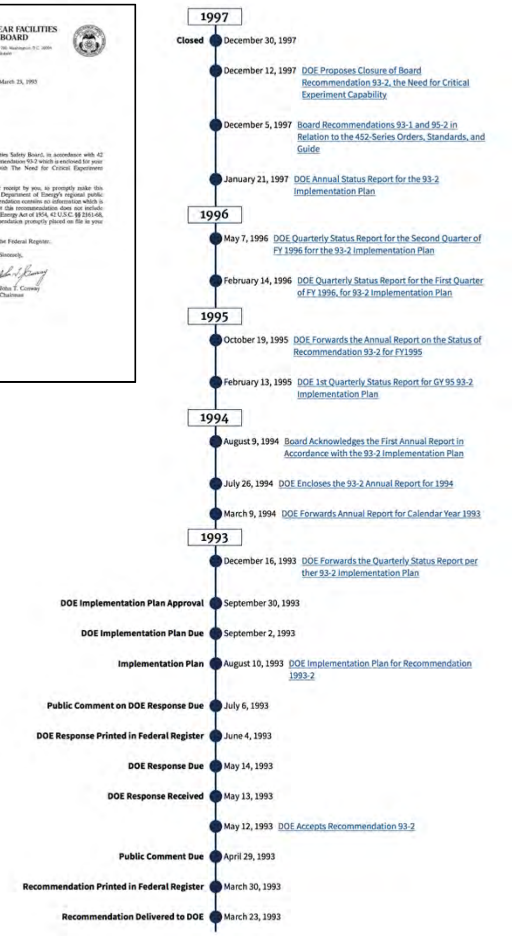
42 U.S.C. § 2286d(a) requires the Board, after receipt by you, to promptly make this recommendation available to the public in the Department of Energy's regional public reading rooms. The Board believes the recommendation contains no information which is classified or otherwise restricted. To the extent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

The Board will publish this recommendation in the Federal Register.

Sincerely,

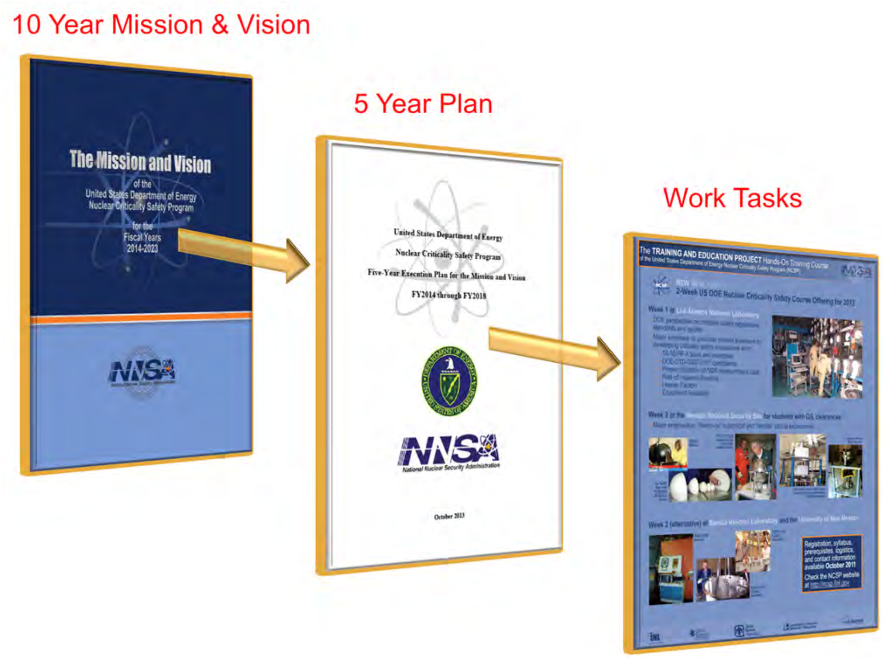
John T. Conway
Chairman

Enclosure



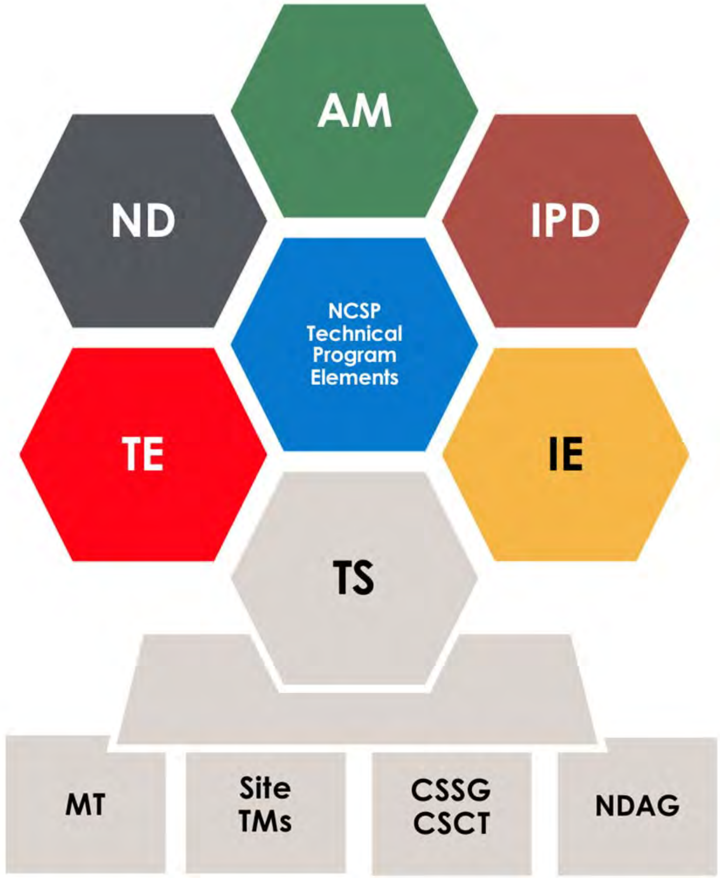
NCSP Organization and Overview

- Mission
  - Provide sustainable expert leadership, direction and the technical infrastructure necessary to develop, maintain and disseminate the essential technical tools, training and data required to support safe, efficient fissionable material operations within the Department of Energy.
- Vision
  - Continually improving, adaptable and transparent program that communicates and collaborates globally to incorporate technology, practices and programs to be responsive to the essential technical needs of those responsible for developing, implementing and maintaining nuclear criticality safety.



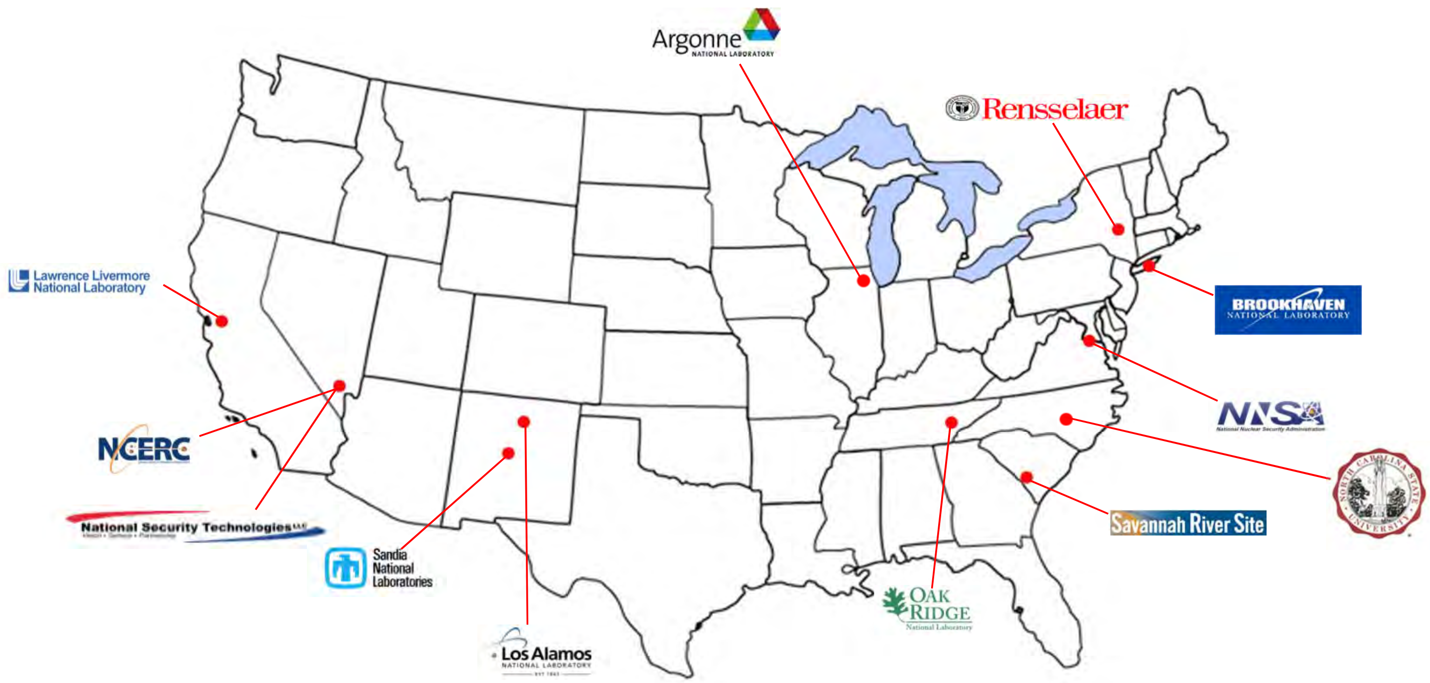
NCSP Technical Program Elements

- **Analytical Methods (AM) – 15% of budget**
  - Maintain and improve the Production Codes and Methods for Criticality Safety Engineers (MCNP/SCALE, NJOY/AMPX)
- **Nuclear Data (ND) – 13% of budget**
  - Perform Measurements of Basic Nuclear (Neutron) Physics Cross-Sections and Generate New Evaluated Cross-Section Libraries and Covariance Data for Use in Production Criticality Safety Codes
- **Information Preservation and Dissemination (IPD) – 4% of budget**
  - Protects Valuable Analyses and Information Related to Criticality Safety (includes ICSBEP)
- **Integral Experiments (IE) – 52% of budget**
  - Critical and Subcritical Experiments at the Critical Experiments Facility (CEF) at the Device Assembly Facility (DAF) in Nevada and Sandia National Laboratory Pulse Reactor Facility– provides integral tests of codes and data
- **Training and Education (TE) – 6% of budget**
  - Web-based training modules and 1- & 2-week Hands-On Criticality Safety courses for Criticality Safety Engineers, Line Management, and Oversight Personnel
- **Technical Support (TS) – 10% of budget**
  - Managerial and technical support



TS – Technical Support
MT – Management team
TMs – Task managers
CSSG – Criticality Safety Support Group
CSCT – Criticality Safety Coordinating Team
NDAG – Nuclear Data Advisory Group

Current NCSP Work Sites



FY2019 NCSP Budget: $26.8 million

US DOE NCSP Contributors

US Contributors

- National Laboratories
  - Argonne (ANL)
  - Brookhaven (BNL)
  - Lawrence Livermore (LLNL)
  - Los Alamos (LANL)
  - Oak Ridge (ORNL)
  - Pacific Northwest (PNNL)
  - Sandia (SNL)
- Sites
  - Nevada National Security Site (NNSS)
  - Savannah River (SRNL)
  - Y-12
- Universities
  - Rensselaer Polytechnic Institute (RPI)
  - Georgia Institute of Technology (Ga Tech)
  - North Carolina State University (NCSU)
  - Massachusetts Institute of Technology (MIT)
  - University of Florida (Gainesville) (UF)
  - University of Tennessee (Knoxville) (UTK)

International Partners

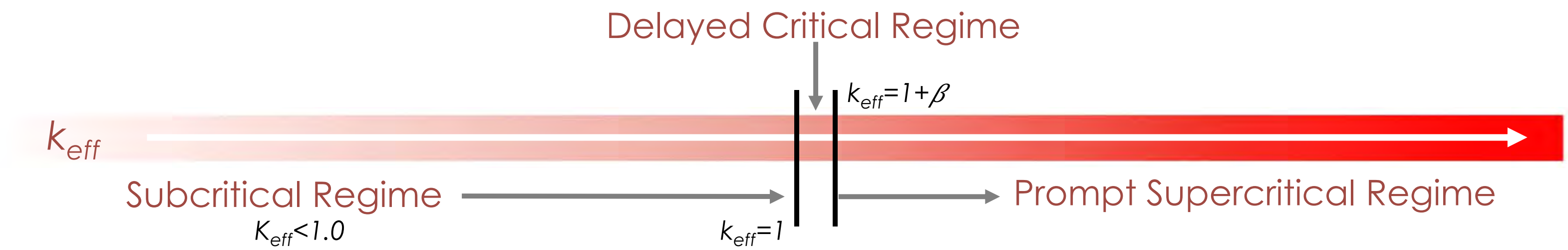
- U.K.: AWE (JOWOG-30)
- France:
  - IRSN (Formal MOU with NCSP)
  - CEA (Nuclear Data)
- Belgium: Institute for Reference Materials and Measurements (IRMM) differential nuclear data measurements
- OECD/NEA
  - ICSBEP
  - WPEC
  - WPNCS

NCSP Integral Experiments

- NCSP integral measurements are performed at
  - Sandia National Laboratories (SNL) and
  - National Criticality Experiments Research Center (NCERC), currently operated by Los Alamos National Laboratory
    - NCERC is located at the Nevada National Security Site (NNSS) inside the Device Assembly Facility (DAF)
- Types of experiments that can be performed
  - Subcritical
    - Rocky Flats shells, BeRP ball, Np-237 sphere, TACS shells, etc.
  - Critical/Delayed Supercritical
    - NCERC: Planet, Comet, Godiva IV, Flattop
    - Sandia: Sandia Pulse Reactor critical assembly (2 fuel types, currently)
  - Prompt Supercritical
    - NCERC: Godiva IV (< 300 deg. C pulse)

DAF/NCERC

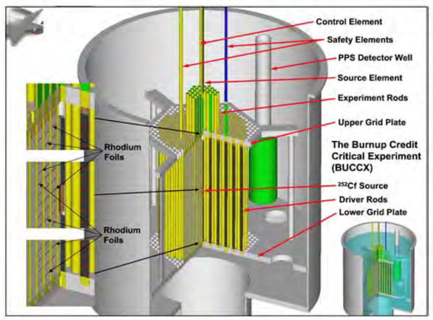
SNL/TA-V/SPR Facility



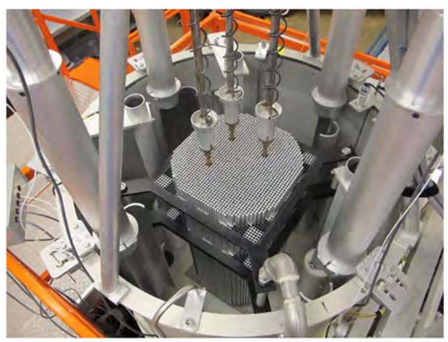
NCSP Critical Assemblies

Sandia National Laboratory

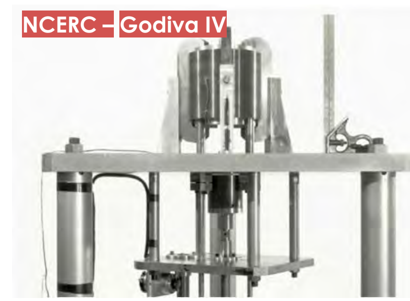
SNL – BUCCX – U(4.31)/Fission Product Experiments



SNL – 7uPCX – U(6.9) UO_2 rods

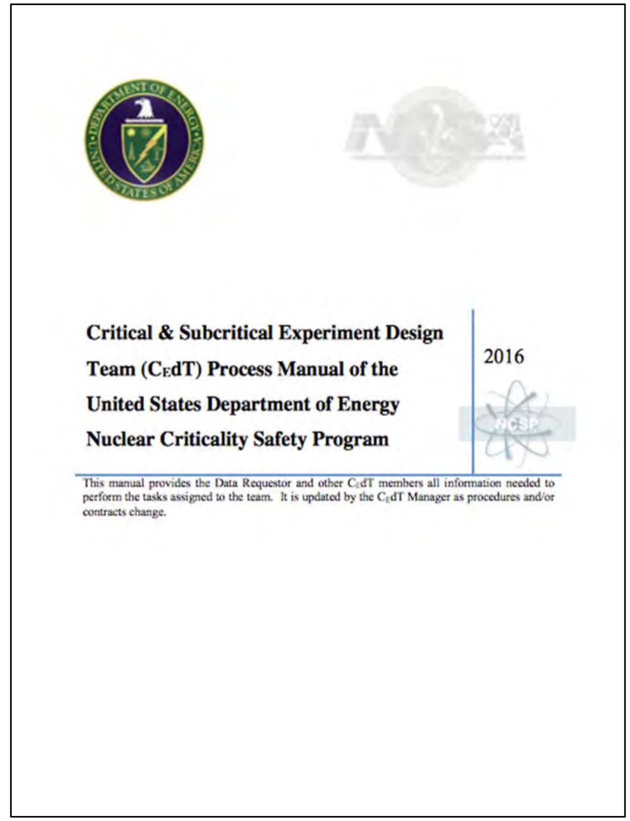


NCERC/DAF



Overview of the NCSP C_EdT Process

- Experimental phases
  - CED-0 – experiment proposal is submitted
  - CED-1 – preliminary design of the experiment
  - CED-2 – final design of the experiment
  - CED-3
    - CED-3a – Schedule/cost/procurement/installations/etc.
    - CED-3b – experiment execution
  - CED-4
    - CED-4a – summary of experimental data collected during the experiment to ensure it met requirements
    - CED-4b – publish final laboratory report or formal critical experiment benchmark report
- Each experiment is assigned a team of experts to provide support
- The experiments take years to complete and are dependent upon the regulatory environment, critical experiment assembly availability, availability of trained operators, etc.


Critical & Subcritical Experiment Design Team (C_EdT) Process Manual of the United States Department of Energy Nuclear Criticality Safety Program

2016

This manual provides the Data Requestor and other C_EdT members all information needed to perform the tasks assigned to the team. It is updated by the C_EdT Manager as procedures and/or contracts change.

CEdT Manual
- Roles & Responsibilities
- Guidance for
  - Completing the experimental phases
  - Obtaining approval from the NCSP Manager
  - Requesting schedule/scope baseline changes
  - Technical conflict resolutions
  - Using the NCSP experiment database
  - Requesting a new experiment

Costs to Design and Perform Critical Experiments

C_EdT Phase Gate		Description	Cost (k$) (low)	Cost (k$) (high)	Duration	Comments
CED-1		Preliminary Design	$ 75	$ 150	3-12 months	Depends significantly on the complexity of the experiment
CED-2		Final Design	$ 100	$ 250	6-12 months	
CED-3	CED-3a	Costs estimated for procurements and procedure development; resource loaded schedule developed; component fabrication	$ 50	$ 300	3-6 months	Material procurements, reactor safety committee approvals, safety basis changes, and procedure reviews can be expensive
	CED-3b	Experiment execution	$ 100	$ 1,000	3-6 months	Approximate costs per site: SNL – $45k/week; NCERC – $75k/week
CED-4	CED-4a	Process experimental data; Begin to document final report	$ 50	$ 250	3-6 months	
	CED-4b	Publish final report	$ 50	$ 150	6-12 months	Sponsor report or an evaluation for the International Handbook of Evaluated Criticality Safety Benchmark Experiments
		Total Estimated Cost	**$ 425**	**$ 2,100**	**24-54 months**	

Experimental Cost Discussion

- Sandia Example (6.9% Fuel Benchmark)
  - Experiments for ICSBEP handbook
    - Series of 19 configurations
  - Experimental duration and costs

Phase	Date/Duration	Cost (x$1,000)
CED-0	Late 2012	–
CED-1	3/2013	80
CED-2	9/2013	75
CED-3a	1/2014	200
CED-3b	9/2014	195
CED-4a	9/2015	243
CED-4b		
Total Cost	**Duration ~3 yr.**	**793**

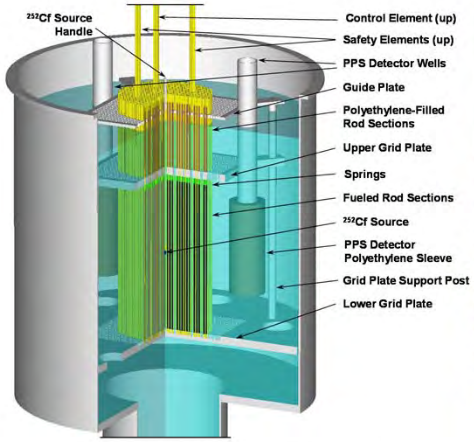
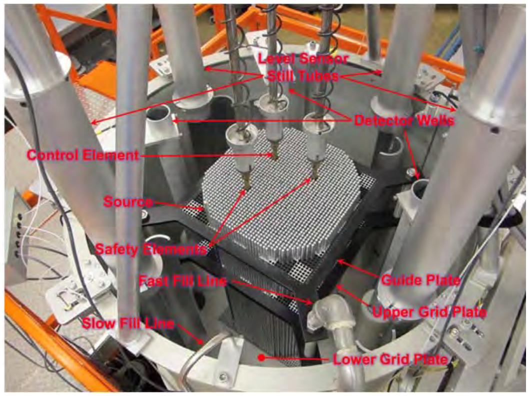
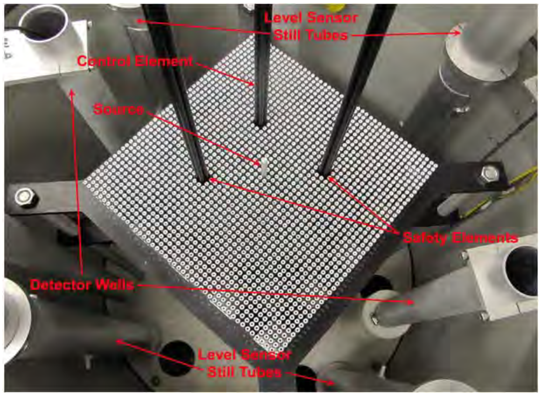


Figure 2. Critical Assembly Concept of the 7uPCX.



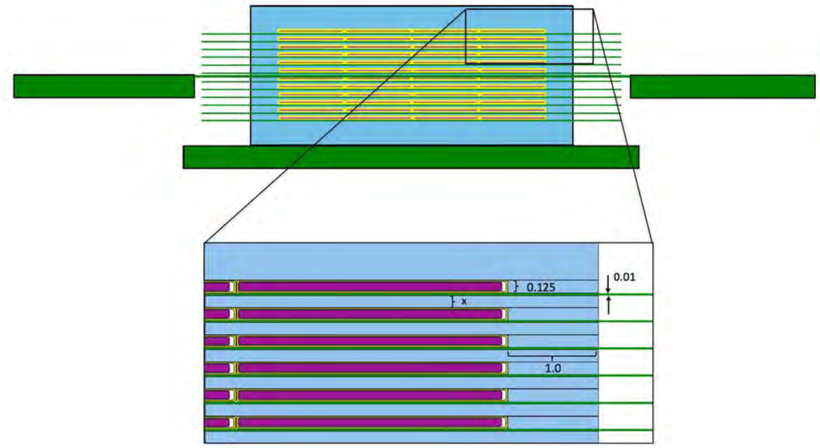


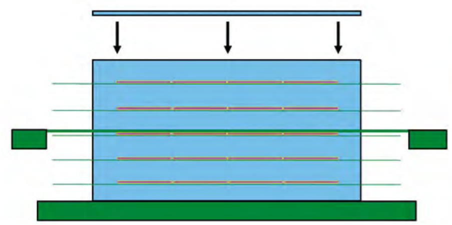
OAK RIDGE National Laboratory

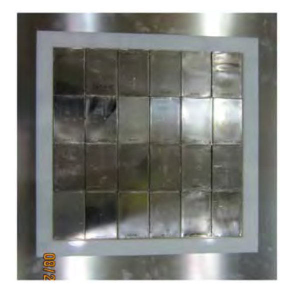
Experimental Cost Discussion

- NCERC Example (LLNL Pu TEX Experiments)
  - Experiments for ICSBEP handbook
    - Series of 10 experiments
      - Five baseline thermal, intermediate, and fast experiments
      - Five with a tantalum layer to test cross sections
  - Experimental duration and costs

Phase	Date/Duration	Cost (x$1,000)
CED-0	5/2011	–
CED-1	9/2012	100
CED-2	11/2014	150
CED-3a	10/2017	200 – 65 (Component Fabrication) – 125 (Procedure Dev.)
CED-3b	In progress (2018)	600 (est.)
CED-4a	TBD	250 (est.)
CED-4b		
Total Cost	**7+ years so far**	**1,300**



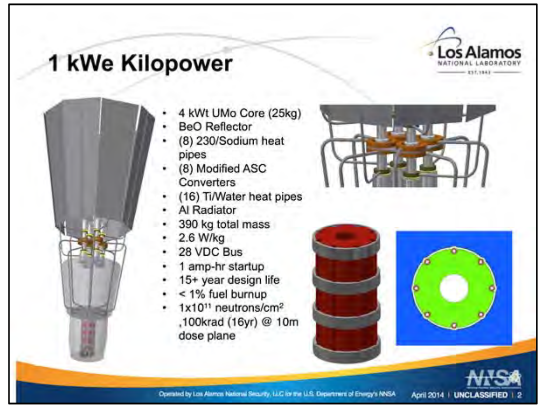




Experimental Cost Discussion

- NCERC Example (Extreme)
  - KRUSTY Critical Experiment
    - NNSA/NASA collaboration
    - C_EdT Team consisted of LANL personnel and the NNSS M&O operator
  - Phase Durations and Costs

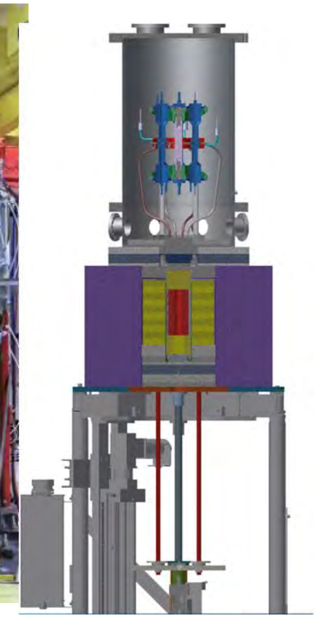
Phase	Duration
CED-1	1 yr.
CED-2	1.5 yr.
CED-3a	7 mo.
CED-3b	3 mo.
CED-4a	1.5 yr. expected
CED-4b	
Total Cost	**Duration ~3 yr.**

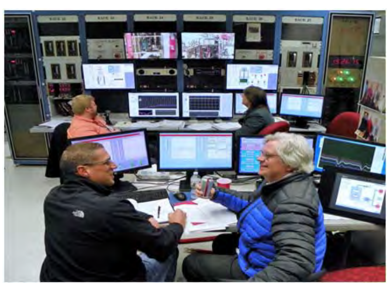


- KRUSTY Funding

Year	NASA	NNSA
FY15	$3.6M	$0
FY16	$3.9M	$0.5M
FY17	$4.0M	$2.9M
FY18	$0.8M*	$2.5M
FY19	$0M	$0.2M
Total	**$12.3M**	**$6.1M**

*estimate provide by NASA based upon currently available data





Questions

Overview of Criticality Analysis Validation

Presented by:

John M. Scaglione

Reactor and Nuclear Systems Division

Oak Ridge National Laboratory

Methods using sensitivity and uncertainty (S/U) analysis to assess similarity of models are available in existing computer codes

- The *International Handbook of Evaluated Criticality Safety Benchmark Experiments* (IHECSBE) contains ~5,000 laboratory critical experiments performed at various critical facilities around the world
- Computational tools are available to survey the critical experiments and use a mathematics-based approach to select benchmarks that are applicable to the application model of interest (e.g., transportation package model)
- Techniques are available to fill in gaps using cross section data uncertainty (NUREG/CR-7109)

Performance of criticality calculations requires detailed knowledge of the application system (package and contents) and modes for reconfiguration

- Parameters important for nuclear criticality safety control include materials, mass, geometry, density, enrichment, reflection, moderation, concentration, interaction, neutron absorption, and volume
- Fuel forms to focus on
  - Powder
  - Pellets
  - Rods
  - Fuel assemblies
- Configuration development considers both normal conditions of transport and hypothetical accident conditions
  - Demonstrate under all credible transport conditions that the system is subcritical

Traditional	Advanced reactors
UO_3	Triso
UO_2	Metal
UF_6	Oxide
	Molten salt

Criticality safety analyses are performed to show that a proposed fuel transport configuration meets applicable requirements

– 10 CFR 71.55 general requirements for fissile material packages:

… a package used for the shipment of fissile material must be so designed and constructed and its contents so limited that it would be subcritical if water were to leak into the containment system, or liquid contents were to leak out of the containment system so that, under the following conditions, maximum reactivity of the fissile material would be attained:

1) *The most reactive credible configuration consistent with the chemical and physical form of the material;*
2) *Moderation by water to the most reactive credible extent; and*
3) *Close full reflection of the containment system by water on all sides, or such greater reflection of the containment system as may additionally be provided by the surrounding material of the packaging.*

Calculated results frequently do not exhibit exact agreement with expectations

- The *computational method* is the combination of the computer code, the data used by the computer code, and the calculational options selected by the user
- Criticality safety evaluations require **validation** of the calculational method with critical experiments that are as similar as possible to the safety analysis models and for which the k_{eff} values are known
- The goal of this validation is to establish a predictable relationship between calculated results and reality
  - A quantitative understanding of the difference or "bias" between calculated and expected results
  - Uncertainty in this difference (bias uncertainty)

The traditional approach to criticality validation is to compute bias and bias uncertainty values through comparisons with critical experiments

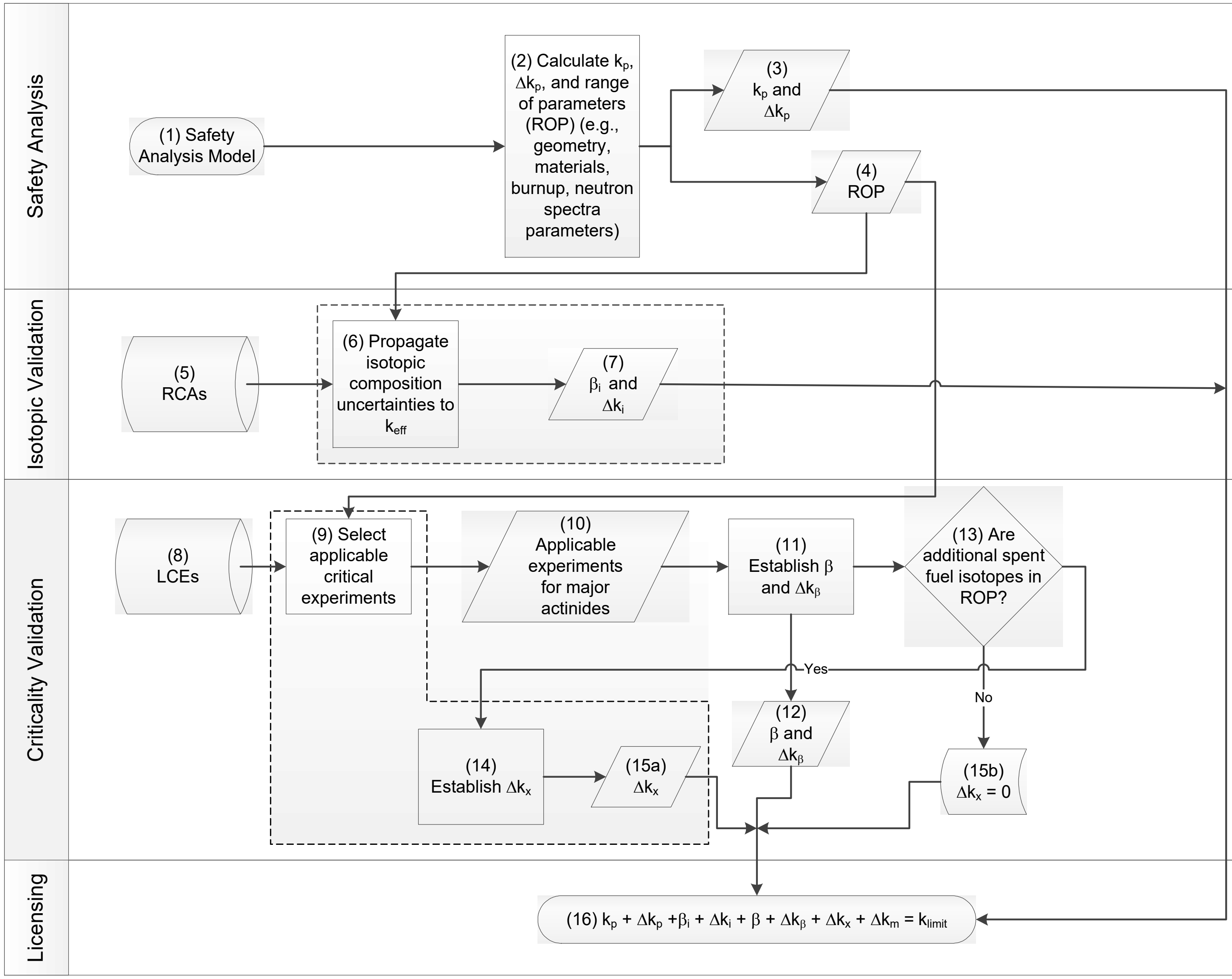
- Trending analyses are typically used in these comparisons
- The difference between the expected and calculated values of the effective neutron multiplication factor, k_{eff}, of a critical experiment is considered the computational bias for that experiment
- The uncertainty in the bias is established through a statistical analysis of the trend

Criticality analysis process

- Develop application model and identify metrics that define it
- Select appropriate benchmark experiments
- Calculate bias and uncertainty
- Process is agnostic to application model

RCA = Radiochemical assay
LCE = laboratory critical experiment

OAK RIDGE National Laboratory



Methodology illustrated from NUREG/CR-7109

Acceptance criterion

$$k_p + \Delta k_p + \beta_i + \Delta k_i + \beta + \Delta k_\beta + \Delta k_x + \Delta k_m \leq k_{limit}$$

$\mathbf{k_p}$ is the calculated multiplication factor of the model for the system being evaluated

$\mathbf{\Delta k_p}$ is an allowance for statistical or convergence uncertainties, or both, in the determination of k_p, material and fabrication tolerances, uncertainties due to geometric or material representation limitations of the models used in the determination of k_p

$\boldsymbol{\beta}$ is the bias that results from the calculation of the benchmark criticality experiments using a particular calculation method and nuclear cross section data

$\mathbf{\Delta k_\beta}$ is bias uncertainty that includes

- statistical or convergence uncertainties, or both, in the computation of β,
- uncertainties in the benchmark criticality experiments,
- uncertainty in the bias resulting from application of the linear least-squares fitting technique to the critical experiment results, and
- a tolerance interval multiplier to yield a single-sided 95% probability and 95% confidence level

$\mathbf{\Delta k_x}$ is a supplement to β and Δk_β that may be included to provide an allowance for the bias and uncertainty from nuclide cross section data that might not be adequately accounted for in the benchmark criticality experiments used for calculating β

$\mathbf{\Delta k_m}$ is a margin for unknown uncertainties and is deemed adequate to ensure subcriticality of the physical system being modeled. This term is typically referred to as an *administrative margin*

$\mathbf{k_{limit}}$ is the upper limit on the k_{eff} value for which the system is considered acceptable

Selection of critical experiments

- The critical experiments and the safety basis model need to use the nuclear data in a similar energy-dependent manner; otherwise, an incorrect bias could be generated
- Historically, similarity has been left largely to professional judgment using qualitative and integral quantitative comparisons to select critical experiments
  - Qualitative parameters considered might include
    - fissionable, moderating, and neutron-absorbing materials present;
    - type of geometry (e.g., fuel pin lattices);
    - type of neutron reflection (i.e., bare, water reflected, steel reflected, etc.);
    - qualitative characterization of the energy dependence of the neutron flux as thermal, intermediate, or fast
  - Quantitative parameters include
    - Energy of average lethargy of a neutron causing fission (EALF)
    - ratio of moderating nuclei to fissile nuclei (e.g., H/X)
    - fuel enrichment
    - lattice fuel pitch

Sensitivity/uncertainty (S/U) tools can be used to assess application and critical experiment model similarity with a quantifiable metric

- Uncertainty analysis is performed for the safety analysis (application) model and for each candidate critical experiment model
  - Uncertainty analysis results rely heavily on the cross-section uncertainty data in the covariance data file
  - Sensitivity is the fractional change in k_{eff} due to a fractional change in a nuclear data value or $S \equiv (\Delta k/k)/(\Delta\sigma/\sigma)$
- Energy-dependent k_{eff} uncertainties for each application model and each critical experiment are compared, producing a correlation coefficient (ck) for each application/experiment model pair
  - A high ck value of near 1 for an application/critical experiment pair indicates that both models have similar sensitivities to the same nuclear data and consequently should have similar biases
  - Low ck values indicate that the two systems differ significantly and may have significantly different biases

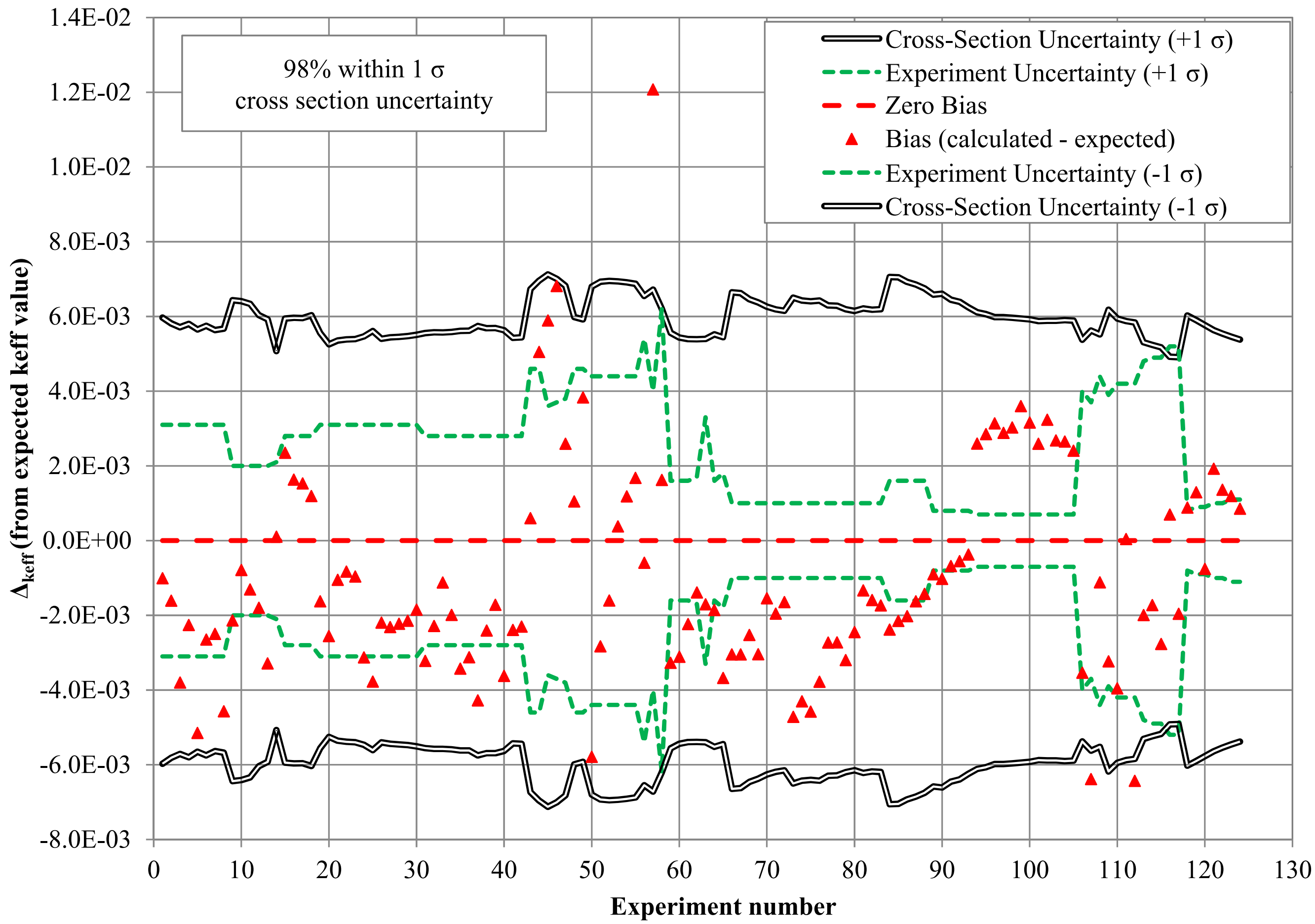
In many instances there are nuclides in the application model for which there are few or no appropriate critical experiments available

- Historically, when a particular material could not be evaluated in a safety analysis model, the material was either removed or a Δk penalty was used based on engineering judgment
- NUREG/CR-7109 provides a validation approach for nuclides that lack experimental data (e.g., minor actinides and structural materials) for criticality safety evaluations
  - The approach is based on the uncertainty in k_{eff} due to nuclear data uncertainties
  - Model-specific sensitivity data, which are in units of $(\Delta k/k)/(\Delta\sigma/\sigma)$, can be used to translate nuclear data uncertainties, which are in units of $\Delta\sigma/\sigma$, into uncertainty in the model k_{eff} value

Plots of computational and experimental uncertainty

- The plot suggests that the nuclear data uncertainties are overestimated
- It also demonstrates the relative merits of analytical techniques that can be used to address validation gaps using nuclear data uncertainties

Comparison of calculated biases and experiment-specific nuclear data uncertainty in k_{eff} for LEU experiments



Source: J. M. Scaglione, D. E. Mueller & J. C. Wagner (2014) An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses: Criticality (keff) Predictions, Nuclear Technology, 188:3, 266-279, DOI: 10.13182/NT13-151

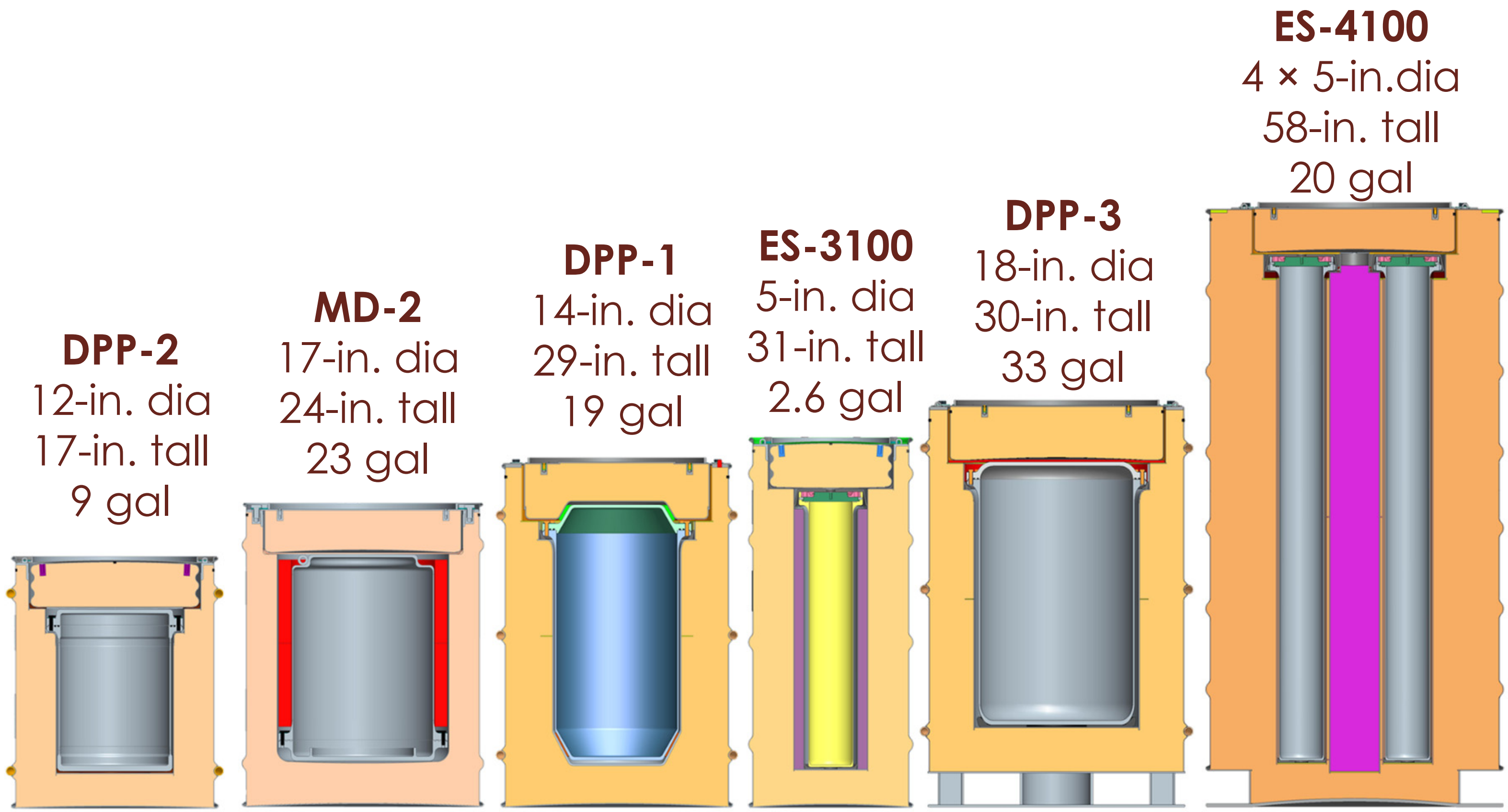
Example application of process

Standard UF_6 cylinder data

Model #	Nominal diameter (in.)	Maximum enrichment (wt% ^{235}U)	Fill limit (lb. UF_6)	Model #	Nominal diameter (in.)	Maximum enrichment (wt% ^{235}U)	Fill limit (lb. UF_6)
1S	1.5	100.0	1.0	48F	48	4.5	27,030
2S	3.5	100.0	4.9	48Y	48	4.5	27,560
5A	5.0	100.0	54.9	48T	48	1.0	20,700
5B	5.0	100.0	54.9	48O	48	1.0	26,070
8A	8.0	12.5	255.0	48OM Allied	48	1.0	27,030
12A	12.0	5.0	460.0	48OM	48	1.0	26,070
12B	12.0	5.0	460.0	48H, 48HX	48	1.0	27,030
30B, 30C	30.0	5.0	5,020.0	48G	48	1.0	26,840
48A, 48X	48.0	4.5	21,030.0				

Source: ANSI N14.1-2012

Kaolite-insulated packages



Courtesy of Jeff Arbital

Y-12 National Security Complex

Example criticality validation process using the ES-4100 package

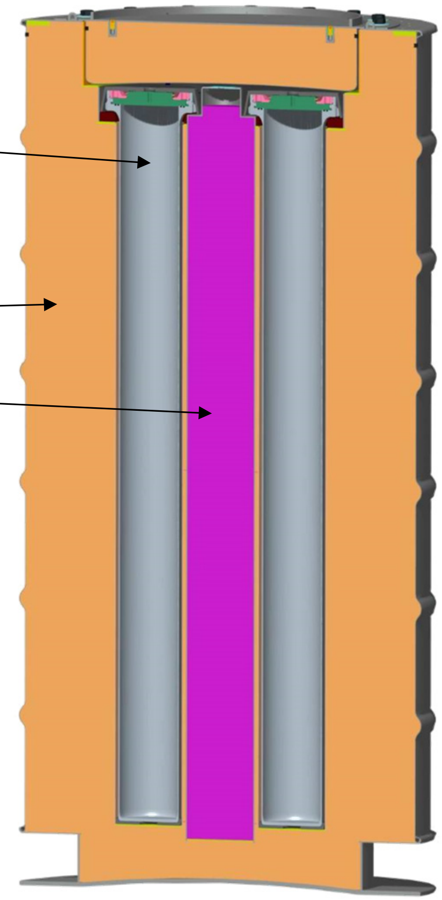
Containment vessel

Photos Courtesy of Jeff Arbital, Y-12 National Security Complex

ES-4100 design features

- Multi-pack: 4 containment vessels (CVs) per drum
- CV inner dimensions: 5.0-in. dia × 58 in. tall
- Outer drum size: 34.0-in. dia × 71 in. tall
- Insulation: Kaolite 1600
- Neutron absorber: 277-4 cast ceramic w/B_4C
- Gross weight: approximately 2,000 lb
  - Less than gross weight of four 6M-110s
- Content weight allowance: 4 × 88 lb
  - Over 350 lb of content weight

ES-4100

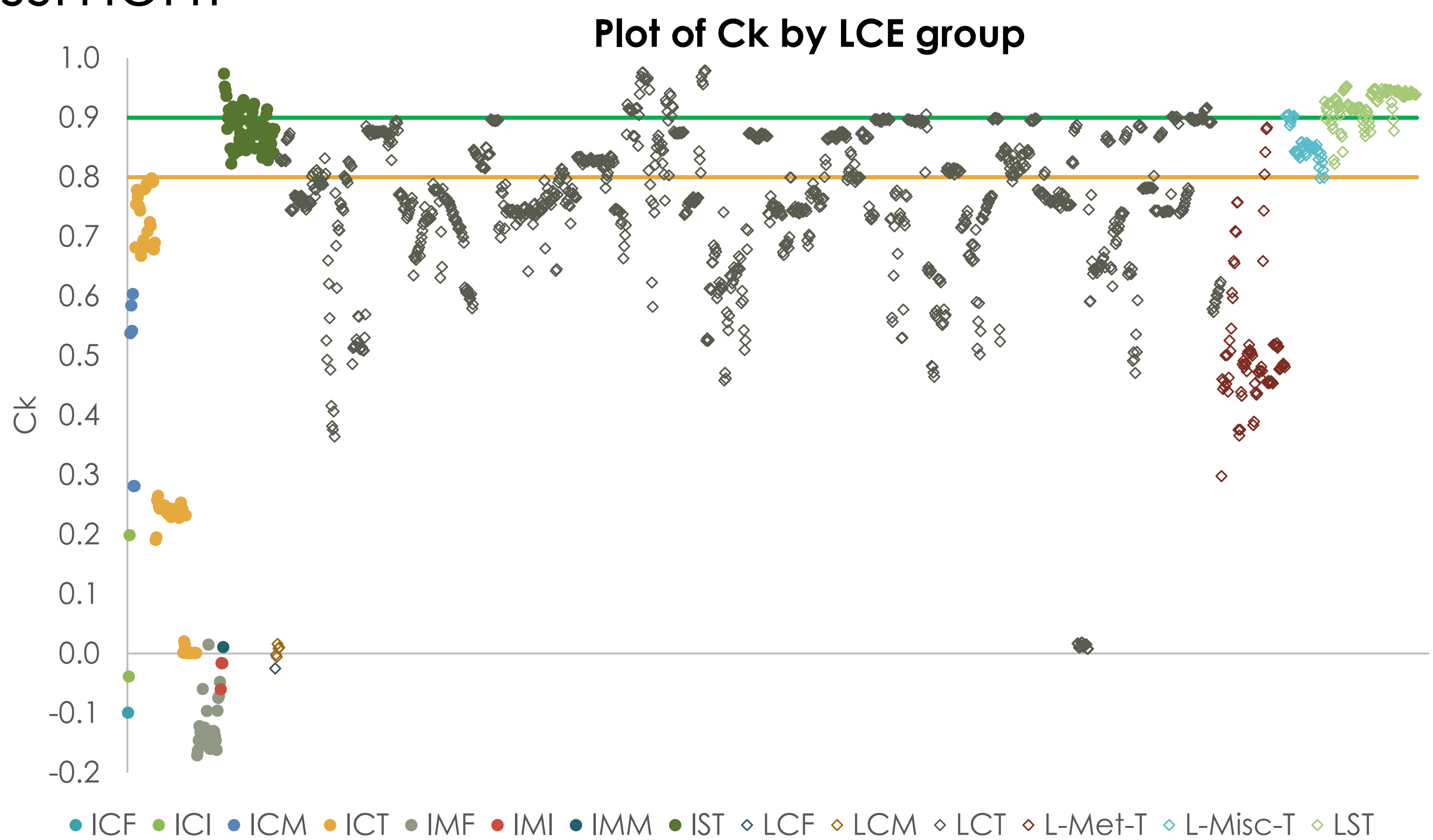


Allowable contents

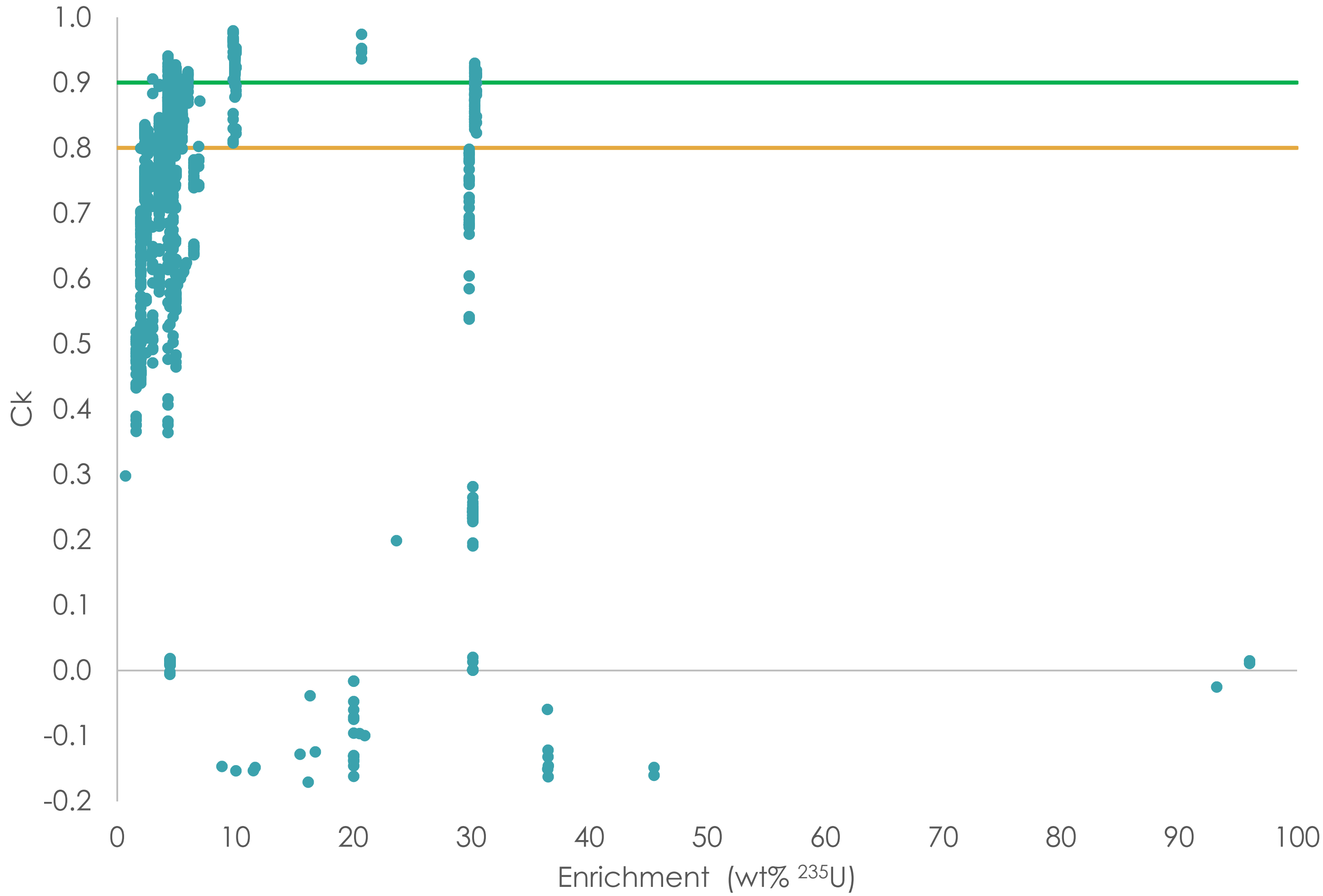
- University of Missouri Research Reactor (MURR) fuel
- Massachusetts Institute of Technology (MIT) reactor fuel
- Loose Advanced Test Reactor (ATR) fuel rods
- Materials Test Reactor (MTR)-type fuel elements and components
- Foreign Research Reactor (FRR) fuels
- Other fuels
- 1,000 g ^{235}U per CV limit
  - Typical US pressurized water reactor (PWR) fuel assembly has ~23,000 g ^{235}U
  - Typical US boiling water reactor (BWR) fuel assembly has ~8,700 g ^{235}U

Selection of applicable critical experiments using similarity assessment

Ck is a correlation coefficient indicating how similar an experiment is to an application model



Ck trended with enrichment

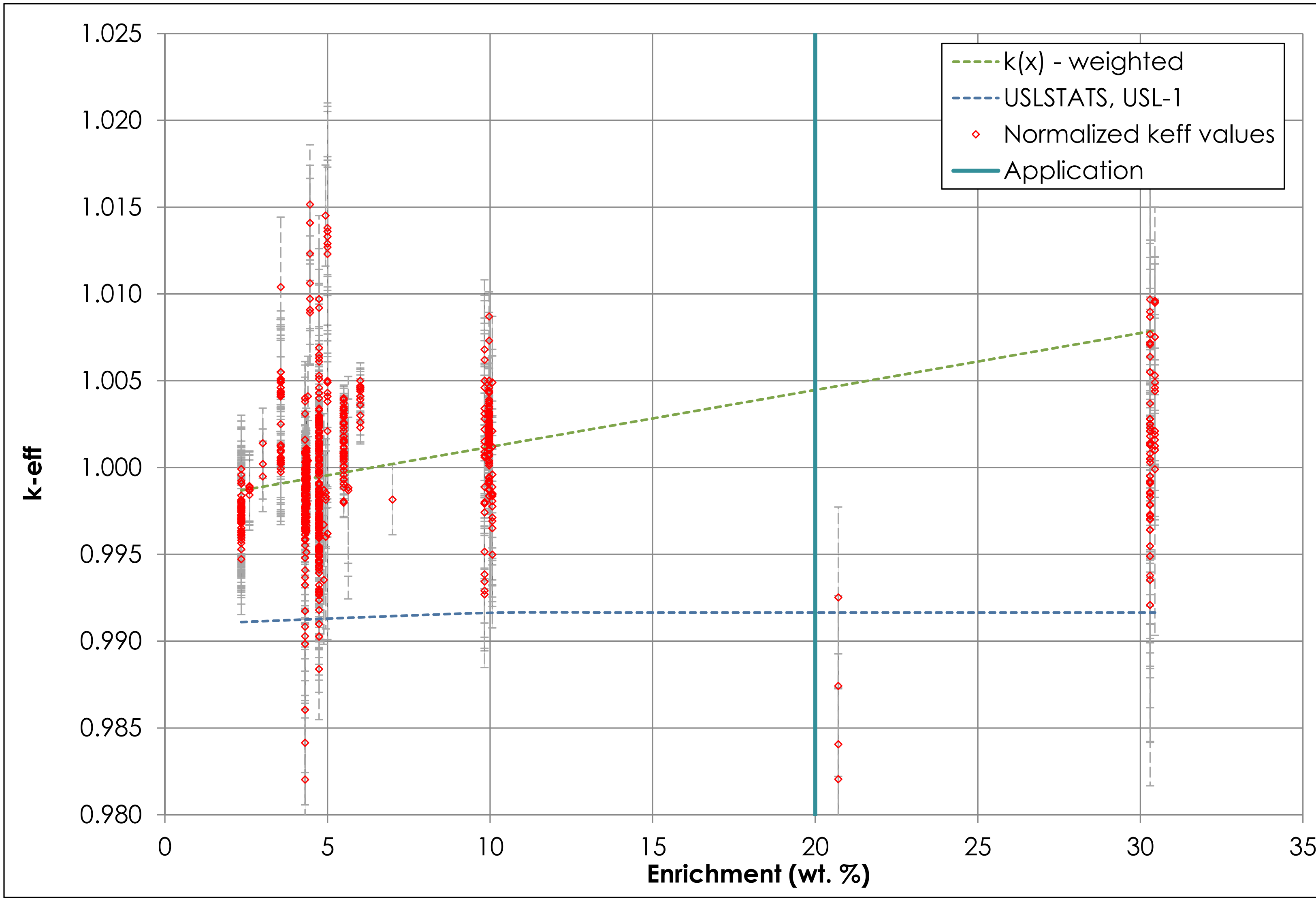


Summary of applicable critical benchmarks

LCEs by group

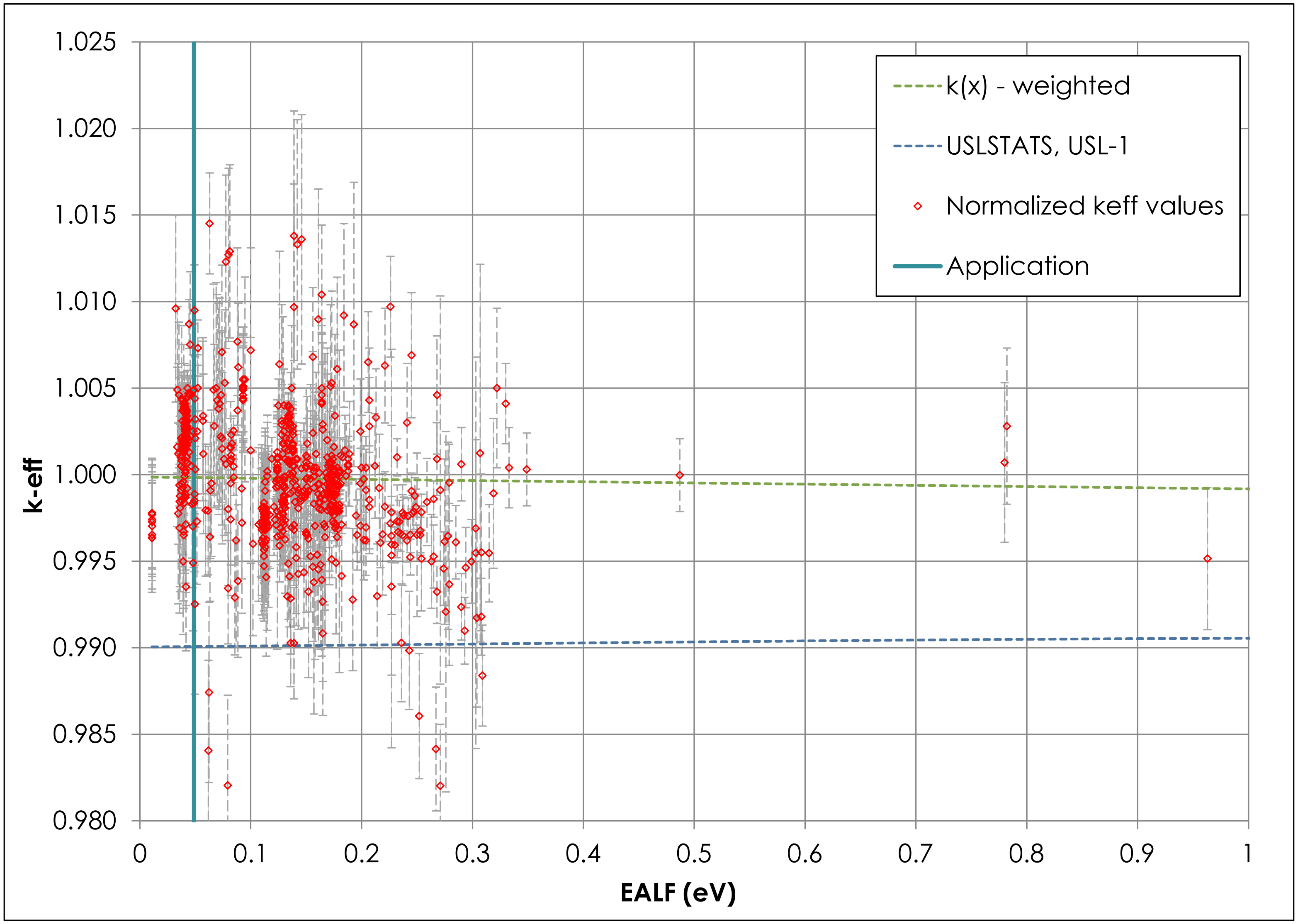
Application system		Number of applicable critical experiments														
Package	Enrichment / BU	ICF	ICI	ICM	ICT	IMF	IMI	IMM	IST	LCF	LCM	LCT	L-Met-T	L-Misc-T	LST	Total
ES4100	Evaluated	1	2	6	76	29	3	1	63	1	5	1,157	79	48	113	1,584
	Ck > 0.9	0	0	0	0	0	0	0	19	0	0	52	0	7	95	173
	Ck > 0.8	0	0	0	0	0	0	0	63	0	0	472	4	46	113	698

Trend analysis using initial enrichment



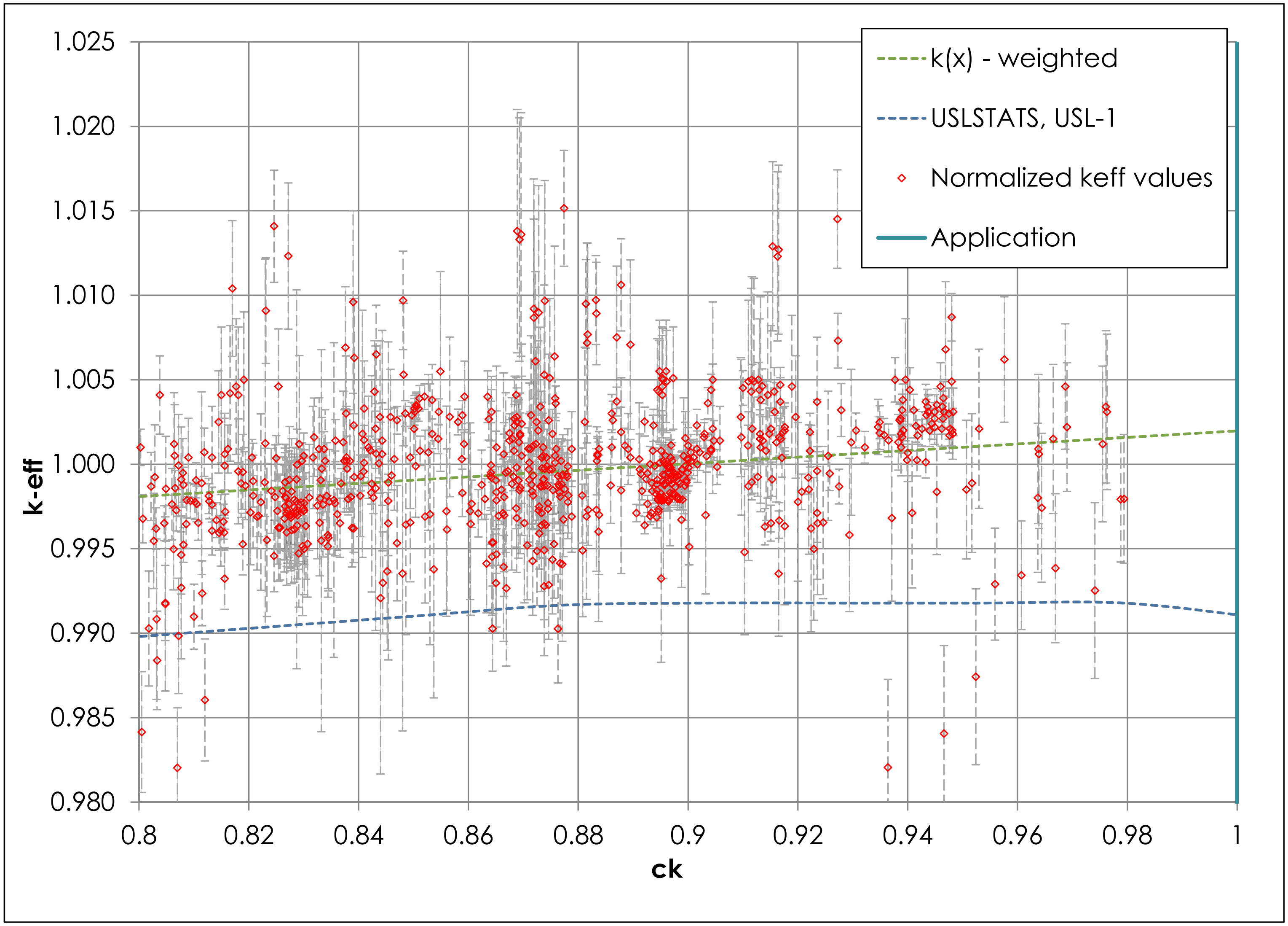
OAK RIDGE National Laboratory

Trend analysis using EALF



OAK RIDGE
National Laboratory

Trend analysis using ck similarity coefficient



Criticality (k_{eff}) validation summary

- Validate criticality calculational method using available critical experiment data and appropriate statistical analysis techniques
- Uncertainty in k_{eff} due to nuclear data uncertainties can be used to cover validation gaps
- If new critical experiments are needed, a process exists to ensure that the critical experiment is designed to fill the gaps using existing computational tools
- The fuel form and the package's internal design are important for development of appropriate design basis configurations and selection of applicable benchmarks
- Note that it is also required to demonstrate that the fuel can be stored safely after use in the reactor (10 CFR 50)
  - The same criticality experiments may or may not be applicable
  - Any new experiment design should also consider storage conditions to maximize range of applicability

BACKUP

All nuclear data used in criticality calculations have some error

- Sources of error include
  - the type of data
  - the experimental apparatus and procedure used to measure the data
  - the quality and amount of measured data
  - nuclear models used to fill in data gaps
  - the evaluation technique used to combine measured and modeled data and resolve conflicting data
  - conversion of the data into formats suitable for use in the computational method