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

Josh Jarrell

September 27, 2018
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A Proposed Path Forward for Transportation of High-Assay Low-Enriched Uranium

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September 2018

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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.
# ACRONYMS

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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 UF₆ currently can be transported in Type 30B packages at up to 5% enrichment (as illustrated in Figure 1), while only 24.9 kg of UF₆ can be transported in Type 5A/B packages at up to 100% enrichment [ANSI 2012].
Another example is the DOE-certified ES-3100 package, which has been design to hold 24 kg of UO\textsubscript{2} as illustrated in Figure 2.

For fresh fuel packaging, GNF has a package, the GNF-A NPC, currently designed to move 5% enriched UO\textsubscript{2}, U\textsubscript{3}O\textsubscript{8}, UO\textsubscript{x}, and other uranium materials as shown in Figure 3.
In addition, Daher-TLI is in the process of developing a package based on the 30B package. It could accommodate 20% enriched UF₆ and is called the 30B-20. It is being developed with a goal to transport up to 1,600 kg of UF₆ enriched to 20% as illustrated in Figure 4.

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 B4C poison element in the central location as illustrated in Figure 5.

![Containment vessel](image)

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

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

![Plot of ck](image)

Figure 6. Plot of ck when comparing criticality experiments with the ES-4100 package with 20% enriched UF6 [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$_3$O$_8$, 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  

NEI 2018a  

NEI 2018b  
NEI February 22, 2018 Statement on HALEU  
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 specifying 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
ii. 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 criticality/benchmarks
   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 C_k over 0.8, when considering HALEU UF_6 in the ES-4100 package. Therefore, optimism that experiments exist to defend future packaging 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 availability 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

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

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

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Part III
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

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QUESTIONS TO CONSIDER

- Will the fuel cycle process be similar to current fleet?
- 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
<table>
<thead>
<tr>
<th>Time</th>
<th>Agenda Item</th>
</tr>
</thead>
<tbody>
<tr>
<td>12:00 pm</td>
<td>Lunch / Introductions / Goals of Meeting – Josh Jarrell, INL and Everett Redmond, NEI</td>
</tr>
<tr>
<td>1:00 pm</td>
<td>Validation and role of critical experiments and nuclear data – Brad Rearden, ORNL</td>
</tr>
<tr>
<td>1:45 pm</td>
<td>Potential for material recovery and form/dose/isotope concerns – Monica Regalbuto, INL</td>
</tr>
<tr>
<td>2:15 pm</td>
<td>Potential for enrichment up to 20% - Melissa Mann, URENCO</td>
</tr>
<tr>
<td>2:45 pm</td>
<td>Break</td>
</tr>
<tr>
<td>3:00 pm</td>
<td>Evaluation of HALEU fabrication issues – Lon Paulson, GEH/GNF</td>
</tr>
<tr>
<td>3:30 pm</td>
<td>HALEU UF6 transportation issues – Andy Langston, DAHER-TLI</td>
</tr>
<tr>
<td>4:00 pm</td>
<td>Cask supplier perspective – Rick Migliore, TN Americas</td>
</tr>
<tr>
<td>4:15 pm</td>
<td>Criticality sensitivity analysis – Brad Rearden, ORNL</td>
</tr>
<tr>
<td>4:45 pm</td>
<td>Discussion and preparation for NRC Visit – Nima Ashkeboussi, NEI</td>
</tr>
<tr>
<td>5:15 pm</td>
<td>Adjourn</td>
</tr>
<tr>
<td>8:30 am</td>
<td>NRC perspective – Drew Barto, NRC</td>
</tr>
<tr>
<td>9:30 am</td>
<td>Break</td>
</tr>
<tr>
<td>9:40 am</td>
<td>Criticality facilities and cost and process for new criticality experiments – Doug Bowen (ORNL)</td>
</tr>
<tr>
<td>10:10 am</td>
<td>Validation discussion: How important are material forms to establishing applicability of criticality experiments? Is there &quot;common ground&quot;? If not, what forms should be the focus? – John Scaglione, ORNL</td>
</tr>
<tr>
<td>10:30 am</td>
<td>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</td>
</tr>
<tr>
<td>11:00 am</td>
<td>DOE perspective – John Herczeg, DOE</td>
</tr>
<tr>
<td>11:10 am</td>
<td>Industry/NEI Needs and Recommendations for DOE – Nima Ashkeboussi, NEI</td>
</tr>
<tr>
<td>11:40 am</td>
<td>Wrap up /Action Items – Josh Jarrell, INL</td>
</tr>
<tr>
<td>12:00 pm</td>
<td>Adjourn</td>
</tr>
</tbody>
</table>
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 UF₆ package (5A or 5B) hold ~24.9 kg of UF₆ or ~16.8 kg of 20% enriched uranium
- Current UF₆ packages for 5% enriched (30B) hold ~2277 kg of UF₆
- DOE certified package (ES-3100) holds ~24 kg of UO₂ or ~21 kg of 20% enriched uranium

20 MTUs of UF₆ HALEU would require:
- ~1191 5A package shipments
- ~13 30B packages shipments (assuming 20% enriched was allowable)

20 MTUs of UO₂ HALEU would require:
- ~953 ES-3100 package shipments
Consider these questions:

1. What form(s) of HALEU should be considered in this scope (e.g., UF₆, 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 UF₆ transported to a fuel fabrication facility, converted to metal fuel, which is subsequently transported to reactor site.

   - 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?
Validation and Role of Critical Experiments and Nuclear Data
Nuclear Data and Benchmarking Program

• 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
## Abbreviated advanced reactor technology matrix (1/2)

<table>
<thead>
<tr>
<th>Reactor Type</th>
<th>Companies</th>
<th>Licensing action expected</th>
<th>Fuel / Enrichment</th>
<th>Thermal spectrum</th>
<th>Fast Spectrum</th>
<th>Coolant</th>
<th>Radial core expansion</th>
<th>Flowing Fuel</th>
<th>Fuel Form</th>
<th>Control elements</th>
</tr>
</thead>
<tbody>
<tr>
<td>HPR</td>
<td>Oklo</td>
<td>2019</td>
<td>~20%</td>
<td>✔</td>
<td>✔</td>
<td>Sodium heat pipes</td>
<td>✔</td>
<td>Metallic Castings</td>
<td>External drums</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Westinghouse (eVinci)</td>
<td>2019</td>
<td>19.75%</td>
<td>Thermal/Epithermal</td>
<td>Sodium heat pipes (dual condenser)</td>
<td>Sodium</td>
<td>✔</td>
<td>Oxide</td>
<td>External drums</td>
<td></td>
</tr>
<tr>
<td>SFR</td>
<td>TerraPower (TWR)</td>
<td>~20%</td>
<td>✔</td>
<td></td>
<td></td>
<td>Sodium</td>
<td>✔</td>
<td>Metallic Rods</td>
<td>Internal rods</td>
<td></td>
</tr>
<tr>
<td></td>
<td>GE PRISM</td>
<td>~20%</td>
<td>✔</td>
<td></td>
<td></td>
<td>Sodium</td>
<td>✔</td>
<td>Metallic Rods</td>
<td>Internal rods</td>
<td></td>
</tr>
<tr>
<td>LFR</td>
<td>Westinghouse</td>
<td>15-20%</td>
<td>✔</td>
<td></td>
<td>Lead</td>
<td>✔</td>
<td>Oxide/Nitride</td>
<td>Internal rods</td>
<td></td>
<td></td>
</tr>
<tr>
<td>HTGR</td>
<td>X-energy (Xe-100)</td>
<td>2020s</td>
<td>15.5%</td>
<td>✔</td>
<td></td>
<td>Helium</td>
<td></td>
<td>Pebbles</td>
<td>TRISO</td>
<td>External rods</td>
</tr>
<tr>
<td></td>
<td>Areva (SC-HTGR)</td>
<td>~20%</td>
<td>✔</td>
<td></td>
<td></td>
<td>Helium</td>
<td></td>
<td>TRISO</td>
<td>Internal rods</td>
<td></td>
</tr>
<tr>
<td>FHR</td>
<td>Kairos</td>
<td>2020s</td>
<td>~17%</td>
<td>✔</td>
<td></td>
<td>FLiBe</td>
<td></td>
<td>Pebbles</td>
<td>TRISO</td>
<td>External rods</td>
</tr>
</tbody>
</table>
Abbreviated advanced reactor technology matrix (2/2)

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

Send updates to Brad - reardenb@ornl.gov
Neutronics calculations rely on nuclear data for criticality, reactivity, power distributions, depletion, decay heat, and more.

Nuclear data is of fundamental importance in nuclear science and engineering.
Different reactor designs have different nuclear data needs
Nuclear data lifecycle

- **Nuclear Data Needs**
- **Validation and Applications (SCALE)**
- **Nuclear Data Processing (AMPX)**
- **Evaluated Nuclear Data Files (ENDF)**
- **Data Evaluation (SAMMY)**

1. **Differential Data Measurements**
2. **Nuclear Data**
3. **Data Evaluation (SAMMY)**
4. **Evaluated Nuclear Data Files (ENDF)**
5. **Nuclear Data Processing (AMPX)**
6. **Validation and Applications (SCALE)**
7. **Nuclear Data Needs**
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 $k_{\text{eff}}$

Eric Bauge* reported on an analysis where components of the Brûyères-le-Châtel (BRC) $^{239}$Pu evaluation were replaced with those from ENDF/B-VII.1. At each step in the replacement process, $k_{\text{eff}}$ of the Jezebel critical assembly was computed. While both the BRC and ENDF/B-VII.1 give the same $k_{\text{eff}}$ 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.


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, ($\alpha$,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

<table>
<thead>
<tr>
<th></th>
<th>MET1000</th>
<th>MOX3600</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Eigenvalue</strong></td>
<td>nominal</td>
<td>uncertainty</td>
</tr>
<tr>
<td></td>
<td>1.0841(1)</td>
<td>1.49(1)%</td>
</tr>
<tr>
<td><strong>CR worth</strong></td>
<td>12081(11) pcm</td>
<td>2.81(1)%</td>
</tr>
</tbody>
</table>

### CE TSUNAMI: Top 3 contributors

<table>
<thead>
<tr>
<th></th>
<th>MET1000</th>
<th>MOX3600</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Eigenvalue</strong></td>
<td>CR worth</td>
<td></td>
</tr>
</tbody>
</table>

fuel assemblies
primary control assembly
48.7413 cm

fuel assemblies
primary control assembly
84.882 cm
Recent nuclear data developments of interest to the advanced reactor community
# Changes in graphite data

**ENDF/B-VII.0 (2006)**

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

<table>
<thead>
<tr>
<th>HTTR loading</th>
<th>ENDF-VII.0 C/E</th>
<th>ENDF-VII.1 C/E</th>
</tr>
</thead>
<tbody>
<tr>
<td>Initial criticality</td>
<td>1.0165</td>
<td>1.0011</td>
</tr>
<tr>
<td>Full core</td>
<td>1.0097</td>
<td>1.0015</td>
</tr>
</tbody>
</table>

**ENDF/B-VII.1 (2011)**

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

**ENDF/B-VIII.0 (2018)**

<table>
<thead>
<tr>
<th>Incident neutron data / / C*Nat / MT = 102 : (z,y) / Cross section</th>
</tr>
</thead>
<tbody>
<tr>
<td>Incident energy (MeV)</td>
</tr>
<tr>
<td>Cross-section (mb)</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Configuration</th>
<th>ENDF-VII.1 C/E</th>
<th>ENDF-VIII.0 C/E</th>
</tr>
</thead>
<tbody>
<tr>
<td>First core</td>
<td>1.00267</td>
<td>1.00582</td>
</tr>
</tbody>
</table>

HTR-10 Benchmark

A. Hawari
NC State

- Ideal Graphite Density = 2.25 g/cm³
- Nuclear Graphite Density = 1.5 – 1.8 g/cm³
HTR-10 pebble: KENO-VI eigenvalue comparison

<table>
<thead>
<tr>
<th>Library</th>
<th>Code</th>
<th>XS lib</th>
<th>$k_\infty$</th>
<th>$\Delta k$ (pcm)</th>
</tr>
</thead>
<tbody>
<tr>
<td>ENDF/B-VII.1</td>
<td>KENO</td>
<td>CE</td>
<td>1.6770(4)</td>
<td>(ref)</td>
</tr>
<tr>
<td>ENDF/B-VIII.0</td>
<td>KENO</td>
<td>CE</td>
<td>1.6722(4)</td>
<td>−438(57)</td>
</tr>
</tbody>
</table>

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

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

<table>
<thead>
<tr>
<th>Basis: ENDF 7.1</th>
<th>$\Delta k$ to all ENDF 7.1 (pcm)</th>
</tr>
</thead>
<tbody>
<tr>
<td>But: graphite from ENDF 8.0</td>
<td>−7</td>
</tr>
<tr>
<td>But: $^{235}$U from ENDF 8.0</td>
<td>−702</td>
</tr>
<tr>
<td>But: $^{238}$U from ENDF 8.0</td>
<td>239</td>
</tr>
<tr>
<td>All ENDF 8.0</td>
<td>−438</td>
</tr>
</tbody>
</table>

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

<table>
<thead>
<tr>
<th>Data Library</th>
<th>$k_{\text{eff}}$</th>
</tr>
</thead>
<tbody>
<tr>
<td>ENDF/B-VII.0</td>
<td>$1.02993 \pm 0.00002$</td>
</tr>
<tr>
<td>ENDF/B-VII.1</td>
<td>$1.04924 \pm 0.00002$</td>
</tr>
</tbody>
</table>

Simplified Molten Chloride Fast Reactor

<table>
<thead>
<tr>
<th>Reaction</th>
<th>Sensitivity</th>
</tr>
</thead>
<tbody>
<tr>
<td>$\text{Cl-35 (n,p) Capture Reaction}$</td>
<td>-0.958</td>
</tr>
<tr>
<td>Pu-239 Nu-bar</td>
<td>0.603</td>
</tr>
<tr>
<td>U-238 Nu-bar</td>
<td>0.281</td>
</tr>
<tr>
<td>Na-23 Elastic Scatter Reaction</td>
<td>0.114</td>
</tr>
</tbody>
</table>

No data for FLiBe / FLiNaK thermal scattering
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

- **HEU-MET-FAST**
- **HEU-SOL-TERM**
- **MIX-COMP-THERM**
- **LEU-COMP-THERM**

**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/

• 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

Code Validation:

- LEU powder / pellet packages, UF₆, UO₂(NO₃)₂, fresh fuel assemblies, spent fuel assemblies
- Longer irradiation cycles, ATF
- Advanced reactor fuel (MSR, SFR, HTGR)
- Research reactor fuel, fresh or spent (TRGA, MTR, etc.)
- Some research reactor fuel, fresh or spent, HEU powder / pellet / metal packages

Part 71/72 Interface

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.
Example criticality validation process using the ES-4100 package

Photos Courtesy of Jeff Arbital
Y-12 National Security Complex

Containment vessel
ES-4100 w/ 20 w/o UF₆ study: Counteracting errors in ENDF/B-VII.1 – ENDF/B-VIII.0

ENDF-7.1: $k_{\text{eff}} = 0.86464 (8)$

ENDF-7.1 $^{238}\text{U}$ → ENDF-8.0 $^1\text{H}$

$\begin{align*}
+42 & \text{ pcm} \\
-132 & \text{ pcm}
\end{align*}$

ENDF-7.1 $^{235}\text{U}$ → ENDF-8.0 $^1\text{H}$

$\begin{align*}
+216 & \text{ pcm} \\
-238 & \text{ pcm}
\end{align*}$

ENDF-7.1 $^{16}\text{O}$ → ENDF-8.0 $^{235}\text{U}$

$\begin{align*}
+95 & \text{ pcm} \\
-83 & \text{ pcm}
\end{align*}$

ENDF-7.1 $^1\text{H}$ → ENDF-8.0 $^{238}\text{U}$

$\begin{align*}
-65 & \text{ pcm}
\end{align*}$

$\sim 450 \text{ pcm}$

$^{235}\text{U} + ^{238}\text{U}$ evaluations

ENDF-7.1 from ENDF-8.0

$\text{ENDF-7.1 from ENDF-8.0}$

$\text{ENDF-8.0^{*}}$
Cross section changes ENDF/B-VII.1 – ENDF/B-VIII.0
OECD/NEA SG-46
Sensitivity of $k_{\text{eff}}$ 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

Explicit 3D neutron transport model of shipping cask

Problem-specific multi-group neutron cross-section data: 238 data points per nuclide
Sensitivities of $k_{eff}$ of a shipping cask to cross section data

### Integral Values

**u-235 fission**
- Integral Value = $0.2193429 \pm 2.46496E-4$

**u-235 n, gamma**
- Integral Value = $0.05170861 \pm 4.202115E-5$

**u-238 n, gamma**
- Integral Value = $-0.1554395 \pm 1.325924E-4$

**pu-239 fission**
- Integral Value = $0.2129418 \pm 1.72252E-4$

**pu-239 n, gamma**
- Integral Value = $-0.08778107 \pm 8.678316E-5$

---

**Graphical Representation**

The graph shows the sensitivity of $k_{eff}$ with respect to cross section data across different energies. The x-axis represents energy in eV, and the y-axis represents sensitivity per unit lethargy.
Uncertainties in nuclear 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

- Overall uncertainty: 0.52% $\Delta k/k$
Identify and analyze benchmark experiments to quantify bias in application.
• Quantifies overall similarity potential sources of bias in $k_{\text{eff}}$ 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
Similarity as independent parameter for trending analysis

- **Biased** $k_{\text{eff}}$ for Application
  (bias is this intercept - 1.0)

- **Confidence band**
  (uncertainty in bias)

- **Positive Bias Adjustment**

- **Gap in experimental data**
Regulatory basis for validation applicability

ISG-10

$c_k \geq 0.95$

recommended

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

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

Confidence band
(uncertainty in bias)

Positive Bias Adjustment

Gap in experimental data

k-eff values

- Red
- Blue
- Red - White
- USN13
Regulatory basis for fission product burnup credit

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

August 30-31, 2018

Melissa Mann
President, URENCO USA, Inc.
• 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)

- 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?

<table>
<thead>
<tr>
<th>Cylinder Model</th>
<th>Diameter in inches</th>
<th>Maximum Enrichment</th>
<th>Maximum lb UF6*</th>
</tr>
</thead>
<tbody>
<tr>
<td>1S</td>
<td>1.5</td>
<td>100.00%</td>
<td>1.0</td>
</tr>
<tr>
<td>2S</td>
<td>3.5</td>
<td>100.00%</td>
<td>4.9</td>
</tr>
<tr>
<td>5A</td>
<td>5.0</td>
<td>100.00%</td>
<td>54.9</td>
</tr>
<tr>
<td>5B</td>
<td>5.0</td>
<td>100.00%</td>
<td>54.9</td>
</tr>
<tr>
<td>8A</td>
<td>8.0</td>
<td>12.5%</td>
<td>255</td>
</tr>
</tbody>
</table>

*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:

- 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
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
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 UO₂ 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....
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.
**ENR Facility**

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

---

**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, 37, 38, 39, 40, 41, 42, 43, 44, 45, 46, 47, 48, 49, 50, 51, 52, 53, 54, 55, 56, 57, 58, 59, 60, 61, 62, 63, 64, 65, 66, 67, 68, 69, 70, and 71, and in reliance on statements and representations heretofore made by the licensees, 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 materials 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.

<table>
<thead>
<tr>
<th>Licensee</th>
<th>License Number/SNM-2019</th>
</tr>
</thead>
<tbody>
<tr>
<td>1. General Electric Hitachi Global Laser Enrichment LLC</td>
<td></td>
</tr>
</tbody>
</table>

**Licensee**

1. General Electric Hitachi Global Laser Enrichment LLC
2. 3901 Castle Hayne Road
3. Exp. Date: September 25, 2052
4. P.O. Box 780
5. Docket No.: 70-7018
6. Wilmington, North Carolina 28402

<table>
<thead>
<tr>
<th>Byproduct, Source, and/or Special Nuclear Material</th>
<th>Chemical and/or Physical Form</th>
<th>Maximum Amount that Licensee May Possess at Any One Time</th>
</tr>
</thead>
<tbody>
<tr>
<td>A. Uranium (natural and depleted) and daughter products</td>
<td>Physical: Solid, Liquid, and Gas Chemical: UF₆, UF₆, UF₆O₂, oxides and other compounds</td>
<td>140,000,000 kg (308,000,000 lbs)</td>
</tr>
<tr>
<td>B. Uranium enriched in isotope 235U up to 8 percent by weight and uranium daughter products</td>
<td>Physical: Solid, Liquid, and Gas Chemical: UF₆, UF₆, UF₆O₂, oxides and other compounds</td>
<td>2,600,000 kg (5,720,000 lbs)</td>
</tr>
<tr>
<td>C. Tc-99, transuranic isotopes and other contamination</td>
<td>Any</td>
<td>Amount that exists as contamination as a consequence of the historical feed of recycled uranium at other facilities</td>
</tr>
</tbody>
</table>

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*GNF Global Nuclear Fuel*
Feed Transport: UF6 Cylinder / UX-30

Model 30B UF6 Cylinder

Model UX-30 Overpacks on Flatrack
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

**LEU systems**: ≤ 10.0 wt. % U235

**HEU soln systems**: 89 – 93.2 wt.% U235

**U Metal systems**: 9 – 97.6 wt. % U235
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.
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.
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 Recycle 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

- **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 - Wastebax

- **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 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
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.
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)
  - 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.

<table>
<thead>
<tr>
<th>Category</th>
<th>Qty. Permitted : Assay Range</th>
</tr>
</thead>
<tbody>
<tr>
<td>II</td>
<td>≥ 10 kg U235: ≥10% but &lt; 20%</td>
</tr>
</tbody>
</table>
| III      | ≥ 1 kg but < 10 kg U235: ≥10% but < 20%  
  ≥ 10 kg U235: ≤10% |
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
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
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.
GNF-A HALEU Cost Elements

- CSAs
- Factory*
- NRC Lic. Review Fee
- Misc. Programs (NDA, DFP, Physical Sec., RC&EP, ALARA)
- Nuclear Packaging (RAJ-II, NPC)
- ISA (ISAS, PHAs, QRAs, FHAs)

SNM-1097 License Amendment (LA) approach

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

- Drum Lid with Closure Ring and Polyurethane Insulation
- Payload Vessel Lid with Polyurethane Insulation
- Stiffening Rings
- Vertical Stiffeners
- Payload Vessel Bottom Polyurethane Insulation
- Packaging Outer Skin
- Packaging Inner Liner
- Ceramic Fiber Blanket
- Payload Vessel (Containment) Wall
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

<table>
<thead>
<tr>
<th>Weight percent U-235</th>
<th>U-235 Mass Limit (g)</th>
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<tbody>
<tr>
<td></td>
<td>General Limit</td>
</tr>
<tr>
<td>≤ 100</td>
<td>350</td>
</tr>
<tr>
<td>≤ 20</td>
<td>410</td>
</tr>
<tr>
<td>≤ 10</td>
<td>470</td>
</tr>
<tr>
<td>≤ 5</td>
<td>580</td>
</tr>
<tr>
<td>≤ 1.25</td>
<td>2,000</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Model No.</th>
<th>Packaging OD (in.)</th>
<th>Packaging Height (in.)</th>
<th>Payload Containment Cavity ID (in.)</th>
<th>Payload Containment Cavity Height (in.)</th>
<th>Packaging Weight (lbs.)</th>
<th>Maximum loaded weight (lbs.)</th>
</tr>
</thead>
<tbody>
<tr>
<td>VP-55</td>
<td>23-1/16</td>
<td>34 ¾</td>
<td>15</td>
<td>25-7/8</td>
<td>390</td>
<td>750</td>
</tr>
<tr>
<td>VP-110</td>
<td>30-7/16</td>
<td>42 ¾</td>
<td>21</td>
<td>29-3/4</td>
<td>702</td>
<td>965</td>
</tr>
</tbody>
</table>
Content Addition – UF₆ 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 UF₆ 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**

<table>
<thead>
<tr>
<th>Content</th>
<th>Configuration</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>20 wt.%</td>
</tr>
<tr>
<td>Quantity of</td>
<td>2S</td>
</tr>
<tr>
<td>cylinders</td>
<td></td>
</tr>
<tr>
<td>7 cylinders</td>
<td>2 cylinders</td>
</tr>
</tbody>
</table>

*operation limit
Content Addition –
Air Transport (criticality)

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

<table>
<thead>
<tr>
<th>wt.% $^{235}\text{U}$</th>
<th>Configuration</th>
<th>VP-55 / VP-110</th>
<th>VP-55 (5-inch Pipe)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Mass $^{235}\text{U}$ (g)</td>
<td>Mass $^{235}\text{U}$ (g)</td>
<td></td>
</tr>
<tr>
<td>General</td>
<td>Air transport</td>
<td>General</td>
<td>Air Transport</td>
</tr>
<tr>
<td>≤ 100</td>
<td>350</td>
<td>350</td>
<td>695</td>
</tr>
<tr>
<td>≤ 20</td>
<td>410</td>
<td>410</td>
<td>1,215</td>
</tr>
<tr>
<td>≤ 10</td>
<td>470</td>
<td>470</td>
<td>1,605</td>
</tr>
<tr>
<td>≤ 5</td>
<td>580</td>
<td>580</td>
<td>1,065 *</td>
</tr>
<tr>
<td>≤ 1.25</td>
<td>2000</td>
<td>--</td>
<td>--</td>
</tr>
</tbody>
</table>

* 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.

<table>
<thead>
<tr>
<th>30B-20 cylinder Technical Data</th>
</tr>
</thead>
<tbody>
<tr>
<td>Nominal Diameter (mm)</td>
</tr>
<tr>
<td>Nominal Length (mm)</td>
</tr>
<tr>
<td>Max. Net UF₆ Weight (kg)</td>
</tr>
<tr>
<td>Max. Gross Weight (kg)</td>
</tr>
<tr>
<td>Enrichment (wt. % ^235U)</td>
</tr>
</tbody>
</table>
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.

<table>
<thead>
<tr>
<th>Dimensions</th>
<th>VP-55</th>
<th>VP-55XL</th>
</tr>
</thead>
<tbody>
<tr>
<td>Overall height</td>
<td>34.8 in</td>
<td>55.92 in</td>
</tr>
<tr>
<td>Outer diameter</td>
<td>22.5 in</td>
<td>22.5 in</td>
</tr>
<tr>
<td>Cavity height</td>
<td>26 in</td>
<td>36.6 in</td>
</tr>
<tr>
<td>Cavity diameter</td>
<td>15 in</td>
<td>15 in</td>
</tr>
<tr>
<td>Tare weight</td>
<td>390 lb</td>
<td>780 lb</td>
</tr>
<tr>
<td>Gross weight</td>
<td>640 lb</td>
<td>1170 lb</td>
</tr>
</tbody>
</table>
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

- SA/B Cylinder
- Thermal insulation body
- Versa Pac 55_XL
- Thermal insulation foam
- Steel shell
- Wood base
- Cylinder body
  - 5" stainless steel pipe
- Valve protection
  - steel pipe
- Handle
  - steel rod
- Optional handle
  - steel rod
- High thermal resistance
  - foam insulation
- Valve protection
  - steel pipe
- Handle
  - steel rod
- Cylinder body
  - 5" stainless steel pipe
- Thermal insulation foam
- Steel shell
- Wood base
- Drum lid
- Payload vessel lid
  - with insulation
- Ceramic fiber
  - blanket
- Payload Vessel
- Bottom Insulation
- Packaging outer skin
- VP-55XL Components
QUESTIONS?
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

<table>
<thead>
<tr>
<th>KNOWN KNOWNS</th>
<th>KNOWN UNKNOWNS</th>
</tr>
</thead>
<tbody>
<tr>
<td>Measurements/Observations</td>
<td>Uncertainty Quantification</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>UNKNOWN KNOWNS</th>
<th>UNKNOWN UNKNOWNS</th>
</tr>
</thead>
<tbody>
<tr>
<td>Communication</td>
<td>Safety Margins</td>
</tr>
</tbody>
</table>

“All models are wrong, some are useful.”
-George E. P. Box – Statistician, Professor, Univ. of Wisconsin
Reminder:
Sensitivity of $k_{\text{eff}}$ 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

Gap in experimental data
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).
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**

<table>
<thead>
<tr>
<th>KNOWN KNOWNS</th>
<th>KNOWN UNKNOWNS</th>
</tr>
</thead>
<tbody>
<tr>
<td>Measurements/Observations</td>
<td>Uncertainty Quantification</td>
</tr>
<tr>
<td>Communication</td>
<td></td>
</tr>
<tr>
<td></td>
<td>UNKNOWNS UNKNOWNS</td>
</tr>
<tr>
<td></td>
<td>Safety Margins</td>
</tr>
</tbody>
</table>
Identifying important processes and uncertainties

Sensitivity of $k_{\text{eff}}$ to neutron cross sections

Covariance (uncertainty) for cross sections

<table>
<thead>
<tr>
<th>Nuclide-Reaction</th>
<th>Unc. in % $dk/k$ Due to this Matrix</th>
</tr>
</thead>
<tbody>
<tr>
<td>$^{239}\text{Pu} \text{ nubar}$</td>
<td>$4.0032E-01 \pm 2.5625E-06$</td>
</tr>
<tr>
<td>$^{238}\text{U} \text{ n,\gamma}$</td>
<td>$1.9457E-01 \pm 1.2387E-05$</td>
</tr>
<tr>
<td>$^{239}\text{Pu} \text{ fission}$</td>
<td>$1.5501E-01 \pm 1.0838E-05$</td>
</tr>
<tr>
<td>$^{235}\text{U} \text{ nubar}$</td>
<td>$1.3981E-01 \pm 5.0038E-07$</td>
</tr>
<tr>
<td>$^{239}\text{Pu} \text{ fission}$</td>
<td>$1.2261E-01 \pm 4.3564E-06$</td>
</tr>
</tbody>
</table>
Code Validation: Identification of Laboratory Experiments that are Similar to the Targeted Application
Design of optimized experiments in US and abroad

As-built experiment
25 micron foils

Optimized experiment
5 micron foils

Rh-103 Critical Experiment
Design for Burnup Credit

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 Unknown)

Safety margin (Unknown Unknown)

Gap in experimental data

Some data not used? (Unknown Known)
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
Fission nuclear data programs and prioritization
OECD Nuclear Energy Agency high priority request list

### NEA Nuclear Data High Priority Request List

<table>
<thead>
<tr>
<th>HPRL Main</th>
<th>High Priority Requests (HPR)</th>
<th>General Requests (GR)</th>
<th>Special Purpose Quantities (SPQ)</th>
<th>New Request</th>
<th>SGC/HPRL Documents</th>
</tr>
</thead>
</table>

#### 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.

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<th>Target</th>
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<th>Sec./Angle</th>
<th>Accuracy</th>
<th>Corr Field</th>
<th>Date</th>
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<td>2H</td>
<td></td>
<td>n,e,n</td>
<td>(n,x,n,3n)</td>
<td>SIG</td>
<td>3 MeV-23 MeV</td>
<td>Y Fission</td>
<td>Y Fission</td>
<td>Y Fission</td>
<td>13-SEP-08</td>
</tr>
<tr>
<td>3H</td>
<td></td>
<td>n,e,n</td>
<td>(n,f)</td>
<td>prompt g</td>
<td>Thermal-Fast</td>
<td>7.5 Y Fission</td>
<td>Y Fission</td>
<td>Y Fission</td>
<td>12-MAY-06</td>
</tr>
<tr>
<td>4H</td>
<td></td>
<td>n,e,n</td>
<td>(n,f)</td>
<td>prompt g</td>
<td>Thermal-Fast</td>
<td>7.5 Y Fission</td>
<td>Y Fission</td>
<td>Y Fission</td>
<td>12-MAY-06</td>
</tr>
<tr>
<td>5H</td>
<td></td>
<td>n,e,n</td>
<td>(n,g)</td>
<td>SIG</td>
<td>0.3-3.2 MeV</td>
<td>4 Y Fission</td>
<td>Y Fission</td>
<td>Y Fission</td>
<td>16-APR-07</td>
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<tr>
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<td>1-8-2</td>
<td>n,e,n</td>
<td>(n,e)</td>
<td>SIG</td>
<td>0.1 MeV-1 MeV</td>
<td>0-180 Deg</td>
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<td>Y Fission</td>
<td>16-APR-07</td>
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<tr>
<td>12H</td>
<td></td>
<td>n,e,n</td>
<td>(n,g)</td>
<td>SIG,RP</td>
<td>100 keV-1 MeV</td>
<td>3 Y Fission</td>
<td>Y Fission</td>
<td>Y Fission</td>
<td>06-SEP-07</td>
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<td>n,e,n</td>
<td>(n,g,Σ)</td>
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<td>Emissspec</td>
<td>Y Fission</td>
<td>Y Fission</td>
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<td>n,e,n</td>
<td>(n,In)</td>
<td>SIG</td>
<td>65 keV-25 MeV</td>
<td>Emissspec</td>
<td>Y Fission</td>
<td>Y Fission</td>
<td>10-SEP-08</td>
</tr>
<tr>
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<td>(n,In)</td>
<td>SIG</td>
<td>9 keV-3 MeV</td>
<td>Y Fission</td>
<td>Y Fission</td>
<td>Y Fission</td>
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<td>(n,In)</td>
<td>SIG</td>
<td>100 keV-25 MeV</td>
<td>Y Fission</td>
<td>Y Fission</td>
<td>Y Fission</td>
<td>11-SEP-08</td>
</tr>
<tr>
<td>22H</td>
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<td>n,e,n</td>
<td>(n,In)</td>
<td>SIG</td>
<td>0.5 keV-4 MeV</td>
<td>Y Fission</td>
<td>Y Fission</td>
<td>Y Fission</td>
<td>11-SEP-08</td>
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<td>(n,In)</td>
<td>SIG</td>
<td>0.5 keV-4 MeV</td>
<td>Y Fission</td>
<td>Y Fission</td>
<td>Y Fission</td>
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<td>(n,In)</td>
<td>SIG</td>
<td>0.5 keV-4 MeV</td>
<td>Y Fission</td>
<td>Y Fission</td>
<td>Y Fission</td>
<td>12-SEP-08</td>
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<td>(n,In)</td>
<td>SIG</td>
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<td>Y Fission</td>
<td>Y Fission</td>
<td>12-SEP-08</td>
</tr>
<tr>
<td>32H</td>
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<td>(n,In)</td>
<td>SIG</td>
<td>0.1 eV-1.35 MeV</td>
<td>Y Fission</td>
<td>Y Fission</td>
<td>Y Fission</td>
<td>12-SEP-08</td>
</tr>
<tr>
<td>33H</td>
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<td>(n,In)</td>
<td>SIG</td>
<td>0.1 eV-1.35 MeV</td>
<td>Y Fission</td>
<td>Y Fission</td>
<td>Y Fission</td>
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<tr>
<td>34H</td>
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<td>(n,In)</td>
<td>SIG</td>
<td>0.5 keV-25 MeV</td>
<td>Emissspec</td>
<td>Y Fission</td>
<td>Y Fission</td>
<td>12-SEP-08</td>
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<tr>
<td>35H</td>
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<td>(n,In)</td>
<td>SIG</td>
<td>0.5 eV-1.35 MeV</td>
<td>Y Fission</td>
<td>Y Fission</td>
<td>Y Fission</td>
<td>12-SEP-08</td>
</tr>
<tr>
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<td>(n,In)</td>
<td>SIG</td>
<td>0.5 keV-25 MeV</td>
<td>Y Fission</td>
<td>Y Fission</td>
<td>Y Fission</td>
<td>12-SEP-08</td>
</tr>
<tr>
<td>37H</td>
<td></td>
<td>n,e,n</td>
<td>(n,In)</td>
<td>SIG</td>
<td>0.5 keV-3 MeV</td>
<td>Y Fission</td>
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<tr>
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<td>(n,In)</td>
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<td>Y Fission</td>
<td>Y Fission</td>
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</tr>
<tr>
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<td>n,e,n</td>
<td>(n,In)</td>
<td>SIG</td>
<td>200 keV-2 MeV</td>
<td>Y Fission</td>
<td>Y Fission</td>
<td>Y Fission</td>
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<td>Y Fission</td>
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<td>(n,In)</td>
<td>SIG</td>
<td>0.5 MeV-4 MeV</td>
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<td>Y Fission</td>
<td>Y Fission</td>
<td>15-SEP-08</td>
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<td>42H</td>
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<td>n,e,n</td>
<td>(n,In)</td>
<td>SIG</td>
<td>0.5 MeV-4 MeV</td>
<td>Y Fission</td>
<td>Y Fission</td>
<td>Y Fission</td>
<td>15-SEP-08</td>
</tr>
<tr>
<td>43H</td>
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<td>n,e,n</td>
<td>(n,In)</td>
<td>SIG</td>
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<td>Y Fission</td>
<td>Y Fission</td>
<td>Y Fission</td>
<td>15-SEP-08</td>
</tr>
<tr>
<td>44H</td>
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<tr>
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<td>(n,In)</td>
<td>SIG</td>
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<td>10 Y Fission</td>
<td>Y Fission</td>
<td>Y Fission</td>
<td>11-JUL-17</td>
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</table>

Number of requests found: 28 (out of a total of 89 requests).

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https://www.oecd-nea.org/dbdata/hprl/
## Nuclear Data Interagency Working Group

### Partners

<table>
<thead>
<tr>
<th>Program Managers</th>
<th>Program Area</th>
<th>NDWG Member</th>
<th>Organization</th>
</tr>
</thead>
<tbody>
<tr>
<td>Donny Hornback</td>
<td>Proliferation Detection</td>
<td>Catherine Romano (Chair)</td>
<td>ORNL</td>
</tr>
<tr>
<td>Ted Barnes</td>
<td>Nuclear Physics/Nuclear Data</td>
<td>Candido Pereira</td>
<td>ANL</td>
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<td>Forensics / Post Detonation</td>
<td>Todd Bredeweg</td>
<td>LANL</td>
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<td>Mike Zerkle</td>
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<td>NNL</td>
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<td>Staci Brown</td>
<td>Research and Development</td>
<td>Tereasa Bailey</td>
<td>LLNL</td>
</tr>
<tr>
<td>Douglas Wade</td>
<td>Physics and Engineering Models</td>
<td>Bob Little</td>
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<tr>
<td>Adam Boyd</td>
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<tr>
<td>Dan Funk</td>
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<td>Brad Rearden</td>
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<td>William Ulicny</td>
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<td>Richard Essex</td>
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<td>Bob Rundberg</td>
<td>LANL</td>
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<tr>
<td>Steve Goldberg</td>
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<tr>
<td>Jehanne Gilio</td>
<td>Isotope Production</td>
<td>Meiring Nortier</td>
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<tr>
<td>Dennis Phillips</td>
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<td></td>
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<tr>
<td>Arden Dougan</td>
<td>Safeguards Technology</td>
<td>Sean Stave</td>
<td>ORNL</td>
</tr>
<tr>
<td>Chris Ramos</td>
<td>Safeguards</td>
<td>Chris Pickett</td>
<td>LANL</td>
</tr>
<tr>
<td>Additional Expert Contributors</td>
<td></td>
<td>Mark Chadwick, Patrick Talou, Alejandro Sonzogni</td>
<td>LANL</td>
</tr>
</tbody>
</table>
Next steps

Now
• Identify candidate materials and transportation packages
• Perform nuclear data / benchmarking needs assessment and gap analysis
• Proceed 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
>5.0 Weight Percent

Code Validation:

- LEU powder / pellet packages, UF₆, UO₂(NO₃)₂, fresh fuel assemblies, spent fuel assemblies
- Longer irradiation cycles, ATF
- Advanced reactor fuel (MSR, SFR, HTGR)
- Research reactor fuel, fresh or spent (TRIGA, MTR, etc.)
- Some research reactor fuel, fresh or spent, HEU powder / pellet / metal packages
• 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:
  - 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)
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)
NCSP Critical Assemblies

Sandia National Laboratory

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

SNL – 7uPCX – U(6.9) UO2 rods

NCERC/DAF

NCERC – Np-237 Sphere
NCERC – BeRP Ball
NCERC – TACS
NCERC – Godiva IV
NCERC – Planet
NCERC – Flattop
Overview of the NCSP CEdT 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.

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

<table>
<thead>
<tr>
<th>CeDt Phase Gate</th>
<th>Description</th>
<th>Cost (k$) (low)</th>
<th>Cost (k$) (high)</th>
<th>Duration</th>
<th>Comments</th>
</tr>
</thead>
<tbody>
<tr>
<td>CED-1</td>
<td>Preliminary Design</td>
<td>$75</td>
<td>$150</td>
<td>3-12 months</td>
<td>Depends significantly on the complexity of the experiment</td>
</tr>
<tr>
<td>CED-2</td>
<td>Final Design</td>
<td>$100</td>
<td>$250</td>
<td>6-12 months</td>
<td></td>
</tr>
<tr>
<td>CED-3a</td>
<td>Costs estimated for procurements and procedure development; resource loaded schedule developed; component fabrication</td>
<td>$50</td>
<td>$300</td>
<td>3-6 months</td>
<td></td>
</tr>
<tr>
<td>CED-3b</td>
<td>Experiment execution</td>
<td>$100</td>
<td>$1,000</td>
<td>3-6 months</td>
<td>Approximate costs per site: SNL – $45k/week; NCERC – $75k/week</td>
</tr>
<tr>
<td>CED-4a</td>
<td>Process experimental data; Begin to document final report</td>
<td>$50</td>
<td>$250</td>
<td>3-6 months</td>
<td></td>
</tr>
<tr>
<td>CED-4b</td>
<td>Publish final report</td>
<td>$50</td>
<td>$150</td>
<td>6-12 months</td>
<td>Sponsor report or an evaluation for the International Handbook of Evaluated Criticality Safety Benchmark Experiments</td>
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<tr>
<td><strong>Total Estimated Cost</strong></td>
<td></td>
<td>$425</td>
<td>$2,100</td>
<td>24-54 months</td>
<td></td>
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</table>
Experimental Cost Discussion

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

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

Figure 2: Critical Assembly Concept of the 7xPCX.

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

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

- NCERC Example (Extreme)
  - KRUSTY Critical Experiment
    - NNSA/NASA collaboration
    - CE$\text{d}$T Team consisted of LANL personnel and the NNSS M&O operator
  - Phase Durations and Costs

<table>
<thead>
<tr>
<th>Phase</th>
<th>Duration</th>
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<tbody>
<tr>
<td>CED-1</td>
<td>1 yr.</td>
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<tr>
<td>CED-2</td>
<td>1.5 yr.</td>
</tr>
<tr>
<td>CED-3a</td>
<td>7 mo.</td>
</tr>
<tr>
<td>CED-3b</td>
<td>3 mo.</td>
</tr>
<tr>
<td>CED-4a</td>
<td>1.5 yr. expected</td>
</tr>
<tr>
<td>CED-4b</td>
<td></td>
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<tr>
<td>Total Cost</td>
<td>Duration ~3 yr.</td>
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- KRUSTY Funding

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<th>Year</th>
<th>NASA</th>
<th>NNSA</th>
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<tr>
<td>FY15</td>
<td>$3.6M</td>
<td>$0</td>
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<td>FY16</td>
<td>$3.9M</td>
<td>$0.5M</td>
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<td>FY17</td>
<td>$4.0M</td>
<td>$2.9M</td>
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<tr>
<td>FY18</td>
<td>$0.8M*</td>
<td>$2.5M</td>
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<tr>
<td>FY19</td>
<td>$0M</td>
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<tr>
<td>Total</td>
<td>$12.3M</td>
<td>$6.1M</td>
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</table>

*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

ORNL is managed by UT-Battelle, LLC for the US Department of Energy
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

<table>
<thead>
<tr>
<th>Traditional</th>
<th>Advanced reactors</th>
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</thead>
<tbody>
<tr>
<td>UO$_3$</td>
<td>Triso</td>
</tr>
<tr>
<td>UO$_2$</td>
<td>Metal</td>
</tr>
<tr>
<td>UF$_6$</td>
<td>Oxide</td>
</tr>
<tr>
<td></td>
<td>Molten salt</td>
</tr>
</tbody>
</table>
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 $k_{\text{eff}}$ 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

Methodology illustrated from NUREG/CR-7109
Acceptance criterion

\[ k_p + \Delta k_p + \beta_i + \Delta k_i + \beta + \Delta k_\beta + \Delta k_x + \Delta k_m \leq k_{\text{limit}} \]

- \( k_p \) is the calculated multiplication factor of the model for the system being evaluated.
- \( \Delta k_p \) 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 \( k_p \).
- \( \beta \) is the bias that results from the calculation of the benchmark criticality experiments using a particular calculation method and nuclear cross section data.
- \( \Delta k_\beta \) is bias uncertainty that includes:
  - statistical or convergence uncertainties, or both, in the computation of \( \beta \),
  - 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.
- \( \Delta k_x \) is a supplement to \( \beta \) and \( \Delta k_\beta \) 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 \( \beta \).
- \( \Delta k_m \) 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.
- \( k_{\text{limit}} \) is the upper limit on the \( k_{\text{eff}} \) 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 $k_{\text{eff}}$ due to a fractional change in a nuclear data value or $S \equiv (\Delta k/k)/(\Delta \sigma/\sigma)$

- Energy-dependent $k_{\text{eff}}$ 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 $\Delta k$ 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 $k_{\text{eff}}$ 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 $k_{\text{eff}}$ value
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.

Example application of process
Standard UF₆ cylinder data

<table>
<thead>
<tr>
<th>Model #</th>
<th>Nominal diameter (in.)</th>
<th>Maximum enrichment (wt% ²³⁵U)</th>
<th>Fill limit (lb. UF₆)</th>
<th>Model #</th>
<th>Nominal diameter (in.)</th>
<th>Maximum enrichment (wt% ²³⁵U)</th>
<th>Fill limit (lb. UF₆)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1S</td>
<td>1.5</td>
<td>100.0</td>
<td>1.0</td>
<td>48F</td>
<td>48</td>
<td>4.5</td>
<td>27,030</td>
</tr>
<tr>
<td>2S</td>
<td>3.5</td>
<td>100.0</td>
<td>4.9</td>
<td>48Y</td>
<td>48</td>
<td>4.5</td>
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<td>5.0</td>
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<td>100.0</td>
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<td>5,020.0</td>
<td>48G</td>
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<td>21,030.0</td>
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</tbody>
</table>

Source: ANSI N14.1-2012
Kaolite-insulated packages

**DPP-2**
- 12-in. dia
- 17-in. tall
- 9 gal

**MD-2**
- 17-in. dia
- 24-in. tall
- 23 gal

**DPP-1**
- 14-in. dia
- 29-in. tall
- 19 gal

**ES-3100**
- 5-in. dia
- 31-in. tall
- 2.6 gal

**DPP-3**
- 18-in. dia
- 30-in. tall
- 33 gal

**ES-4100**
- 4 × 5-in. dia
- 58-in. tall
- 20 gal

*Courtesy of Jeff Arbital*

*Y-12 National Security Complex*
Example criticality validation process using the ES-4100 package

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$_4$C
- 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
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

<table>
<thead>
<tr>
<th>Application system</th>
<th>Number of applicable critical experiments</th>
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</thead>
<tbody>
<tr>
<td></td>
<td>ICF</td>
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<tr>
<td>Package</td>
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<td>ES4100</td>
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<td>Ck &gt; 0.9</td>
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<td>Ck &gt; 0.8</td>
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</tbody>
</table>
Trend analysis using initial enrichment

Trend analysis using initial enrichment

- \( k(x) \) - weighted
- USLSTATS, USL-1
- Normalized \( k_{eff} \) values
- Application
Trend analysis using EALF
Trend analysis using $c_k$ similarity coefficient

![Graph showing trend analysis using $c_k$ similarity coefficient. The graph plots $k$-eff against $c_k$ with different lines and markers representing various data sets.](image)

- $k(x)$ - weighted
- USLSTATS, USL-1
- Normalized $k_{eff}$ values
- Application
Criticality ($k_{\text{eff}}$) validation summary

- Validate criticality calculational method using available critical experiment data and appropriate statistical analysis techniques
- Uncertainty in $k_{\text{eff}}$ 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
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