DOE/NRC COLLABORATION FOR CRITICALITY SAFETY SUPPORT FOR COMMERCIAL-SCALE HALEU FUEL CYCLES AND TRANSPORTATION (DNCSH) – CALL #2 WORKSHOP

AUGUST 27, 2025

DNCSH

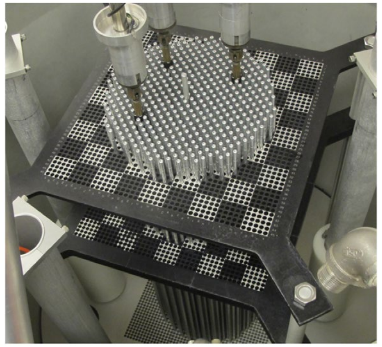
DOE/NRC Criticality Safety for Commercial-Scale HALEU for Fuel Cycle and Transportation

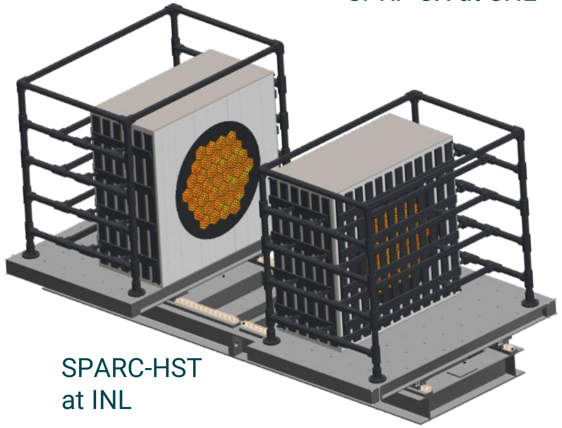
U.S. DEPARTMENT OF ENERGY

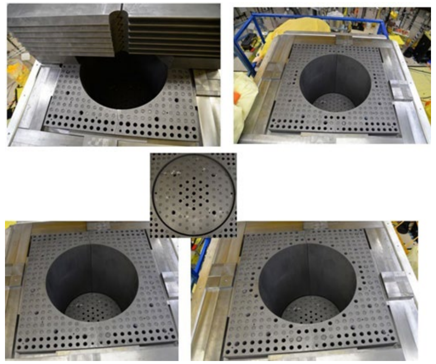
U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

DNCSH is well-positioned to help the U.S. nuclear industry meet the President's Executive Orders (EO14156, EO14299, EO14300, EO14301, EO14302, EO14303) with the project touching on each of the below points:

1. Speed Up Nuclear Reactor Licensing
2. Add 300 Gigawatts of New Capacity by 2050
3. Lay the Groundwork for Faster Reactor Testing
4. Deploy for Artificial Intelligence and Military
5. Explore Fuel Recycling and Reprocessing
6. Amp up Domestic Nuclear Fuel Production
7. Bolster the American Nuclear Workforce
8. Assess Spent Nuclear Fuel Management
9. Expand U.S. Nuclear Energy Exports


SPRF-CX at SNL


SPARC-HST at INL


Deimos at NCERC

OAK RIDGE National Laboratory

DNCSH

DNCSH Pre-Workshop 2 Survey - Analysis

Walid A. Metwally, Alex Shaw, and Veronica Karriem

Workshop #2 on the Collaboration Between the U.S. Department of Energy and the U.S. Nuclear Regulatory Commission for Development of Criticality Safety Benchmarking Data for HALEU Fuel Cycle and Transportation (DNCSH)

August 27, 2025

ORNL IS MANAGED BY UT-BATTELLE LLC
FOR THE US DEPARTMENT OF ENERGY

Outline

- Survey Objectives
- Responses
- Observations

Survey Objectives

Workshop #2:

- Validation gaps related to the manufacturing and fabrication of HALEU fuel at front-end facilities

Survey:

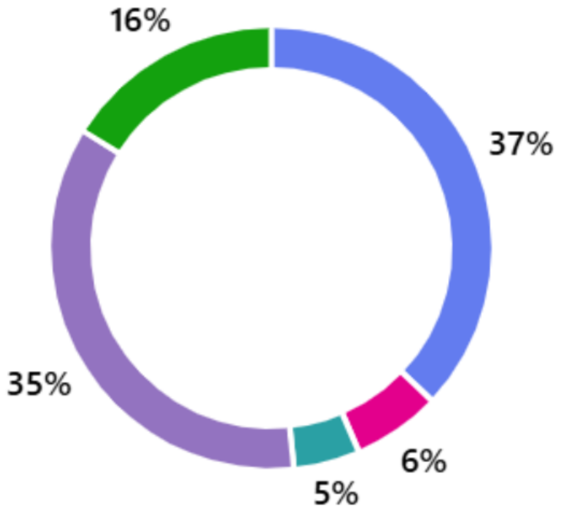
- Gather input from community members and industry stakeholders on current or anticipated criticality safety validation needs
- Provide feedback on Larry Wetzel's report: *Benchmark Gap Assessment for the Manufacturing of High-Assay Low-Enriched Uranium Fuels* - ORNL/TM-2025/3744 https://www.nrc.gov/docs/ML2506/ML25062A173.pdf
- Identify benchmark needs for HALEU fuel facilities

Survey Responses

51 responses

Which of the following best describes your current role?

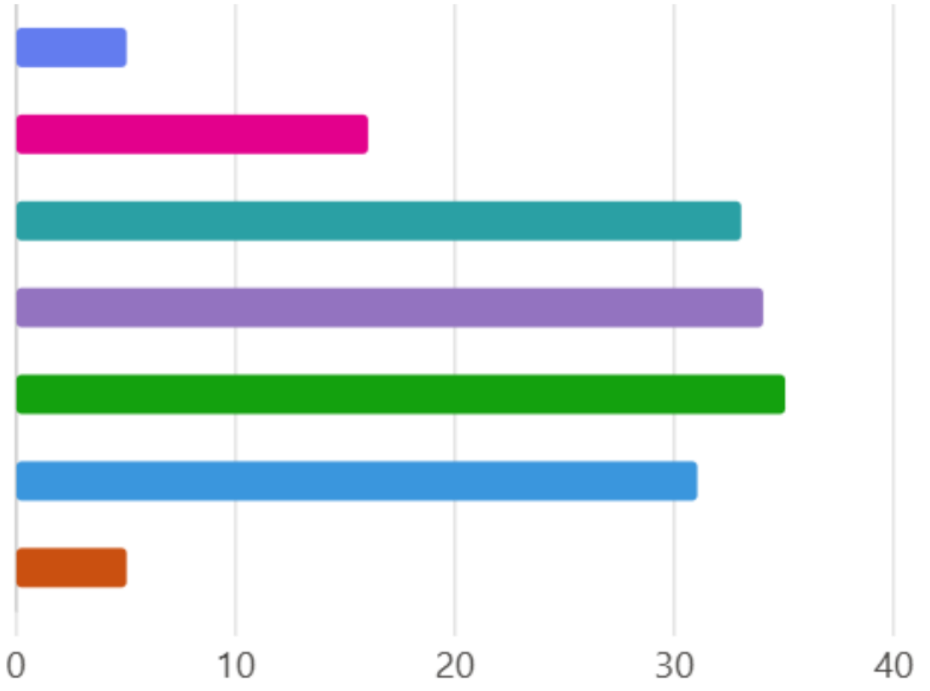
Criticality safety practitioner	23
Regulator	4
Operations	3
Research and Development	22
Other	10



Survey Responses

Which fuel cycle facilities are you most interested in?

- Mining and Milling 5
- Conversion 16
- Enrichment 33
- Fabrication 34
- Transportation 35
- Storage 31
- Other 5

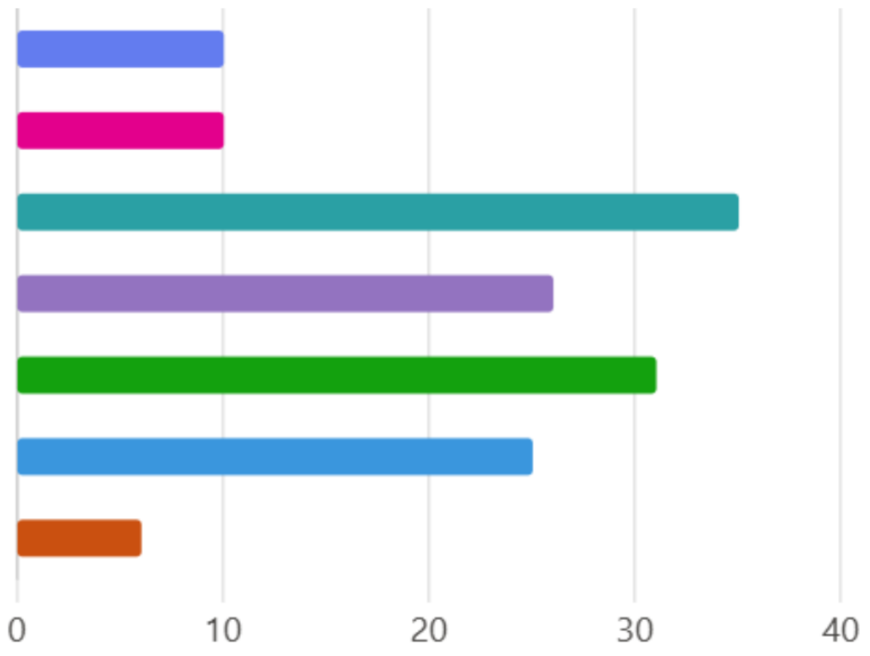


Oak Ridge National Laboratory

Survey Responses

Which HALEU fuel type are you most interested in?

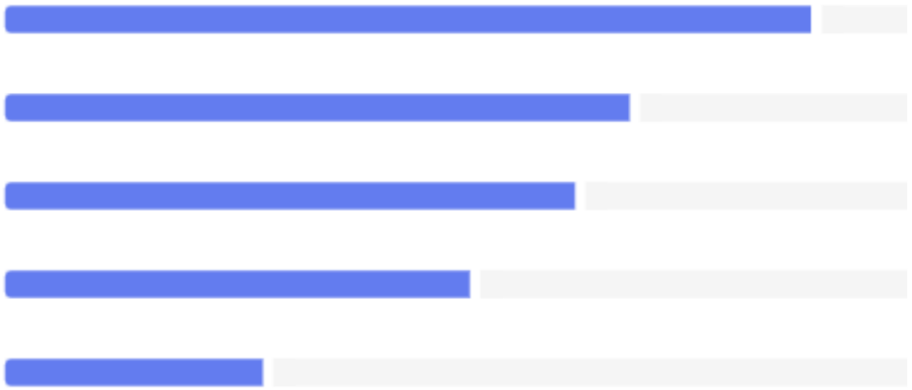
UF4- Based fuel	10
UCL3- Based fuels	10
TRISO-fuels	35
Oxide fuel	26
Metallic Fuels	31
U3O8/UF6	25
Other	6



Survey Responses

Rank the importance of the following areas to meet your facility need

1. Additional benchmark experiments
2. Validation methods in nuclear criticality safety
3. Nuclear data
4. Application models with HALEU fuel
5. Supplemental analysis tools



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Survey Responses

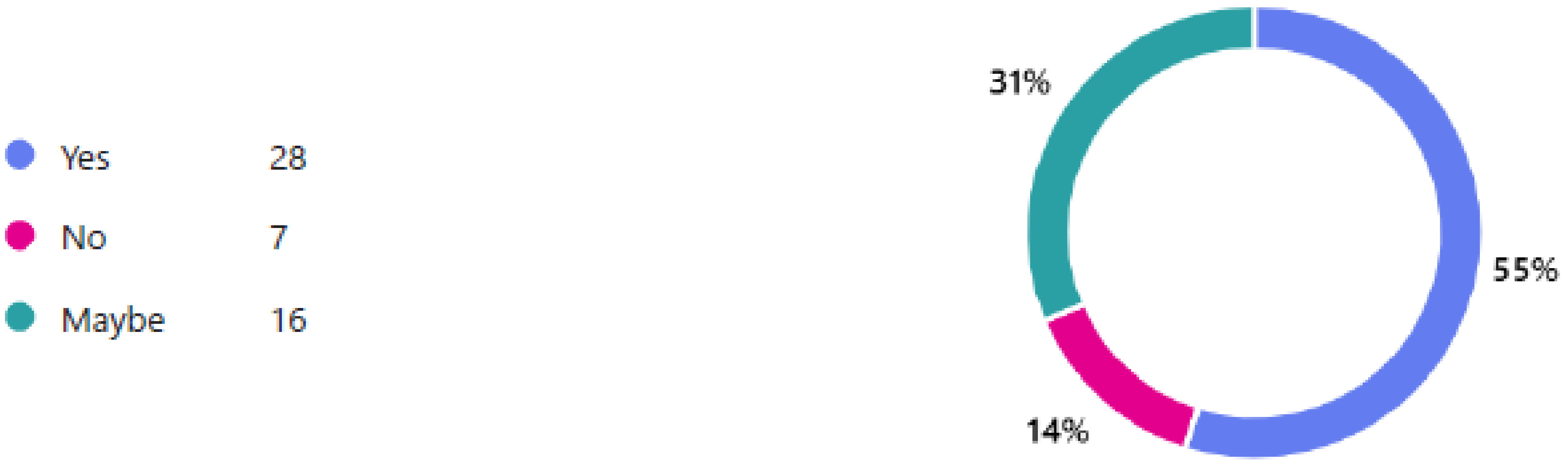
Please share your thoughts on the ORNL/TM-2025/3744 report by addressing the following:

Focus Areas: *Are there any topics that you feel were either missing or not thoroughly addressed in the report and could help complete the discussion during the workshop?*

- Improving the basis for burnup credit and support for minor actinides.
- Including casting operations with metal reflection and FLiBe salt systems.
- Studying the impact of absorber materials on similarity assessment
- Identifying the most relevant experiments for the areas with gaps
- Communicating the publicly available inputs and SDFs.

Survey Responses

Would you be interested in attending a training on validation methods in nuclear criticality safety related to commercial and licensing applications?



Observations

- *Of more interest:*
  - *Facilities: enrichment, fabrication, and transportation*
  - *Fuels: TRISO, metallic, and oxide*
- *Benchmarks, nuclear data, and methods remain a need*
- *A few topics suggested to complement ORNL/TM-2025/3744*
- *More than half expressed interest in validation methods training*

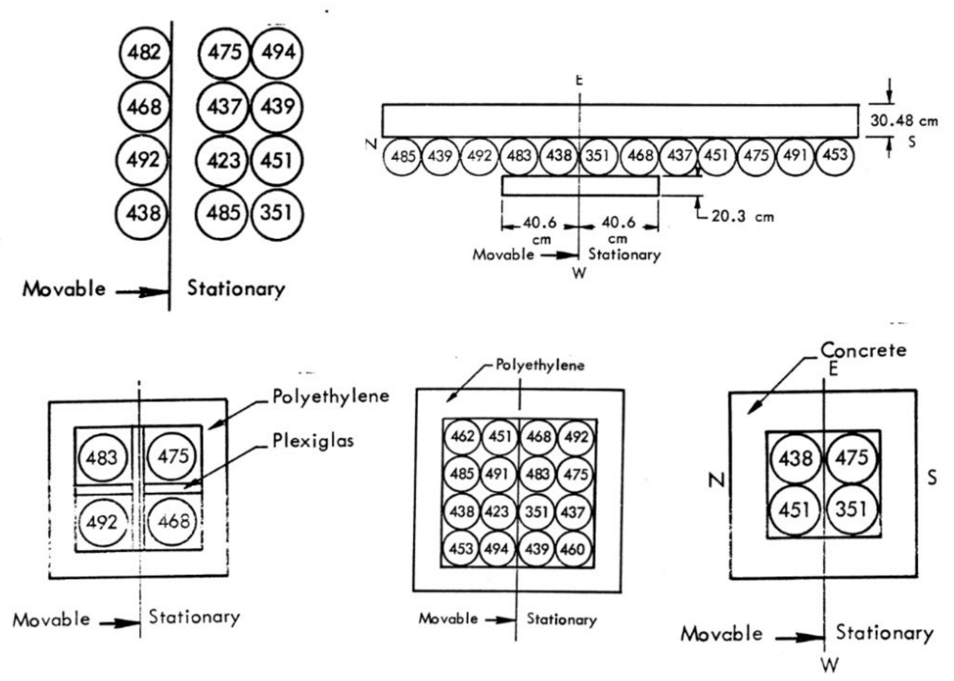
DNCSH Funded Criticality Safety Benchmark Projects

Catherine Percher
Control Account Manager for Critical Benchmark Execution
Lawrence Livermore National Laboratory

Call #1 Funded Activities

Title	Lead Lab	Partners	Facility
Benchmark Validation for Transportation of TRISO HALEU Fuel for Advanced Reactors	INL	BWXT, Kairos, X-Energy, Radiant, JFoster & Associates, University of Michigan	PROTEUS PSI, Switzerland
THETA: TRISO-form HALEU-fueled Experimant for Transport Applications	LANL	Kairos	NCERC, USA
Benchmark of Historical Y-12 Critical Experiments with UF6 Cylinder Model 8A Containers	LLNL	ORNL, CS Engineering, University of Tennessee	Y-12, USA
PETALE Benchmark	LLNL	EPFL, University of California-Berkeley	CROCUS at EPFL, Switzerland
Thermal/Epithermal eXperiments (TEX) Additional Chlorine Configurations to Provide Validation for TerraPower’s Molten Chloride Salt Fuel	LLNL	LANL, TerraPower	NCERC, USA
Critical Experiments for New 19.75 wt% 235U Enriched IPEN/MB-01 Core	ORNL	GE Vernova, IPEN	MB01 at IPEN, Brazil
SLOWPOKE-2 Refuel Measurements	ORNL	Canadian Nuclear Laboratories	Royal Military College, Canada
High-Temperature Graphite Double Differential Scattering Cross Sections: Measurement, Evaluation, and Validation	ORNL	Yarmouk University	Spallation Neutron Source, USA
RPI Reactor Criticals with Noteworthy Non-Fissile SS Element Sensitivity	ORNL	INL, Rensselaer Polytechnic Institute	RCF at RPI, USA
Characterization of ISU’s AGN-201 Reactor for Qualification as an ICSBEP Benchmark	ORNL	GE Vernova, Idaho State University	AGN-201 at ISU, USA
Evaluation of Critical Configurations of the Missouri S&T Reactor	ORNL	Missouri S&T	MSTR at MST, USA
ZED-2 Measurements with In-Core Absorbers	ORNL	Canadian Nuclear Laboratories	ZED-2 at Chalk River, Canada
Temperature Dependent Transmission and S(alpha, beta) Measurements of Advanced Nuclear Moderators	ORNL	X-Energy	Spallation Neutron Source, USA
eDeimos Experiments with Westinghouse for New HALEU Benchmarks	LANL	Westinghouse	NCERC, USA
HALEU Critical Experiments in Water Moderated UO2 Fuel Rod Lattices	SNL	Orano	SPRF/CX at SNL, USA
Critical Experiments Targeting Optimum Moderation Conditions	PNNL	ORNL, SNL	SPRX/CX at SNL, USA

Benchmark of Historical Y-12 Critical Experiments with UF_6 Cylinder Model 8A Containers

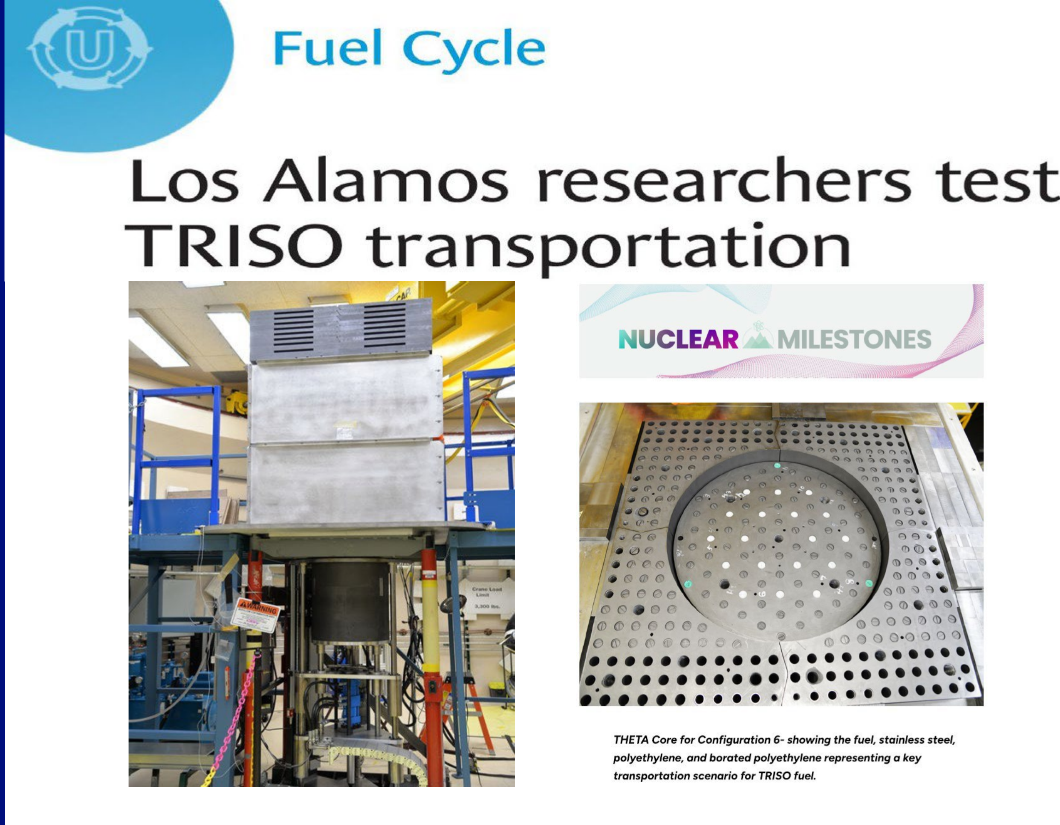


Examples of cylinder layouts (numbered circles) in unreflected and reflected arrays.

- Approximately 16 unique configurations
- Bare, polyethylene-reflected and concrete-reflected arrays specifically designed for NCS validation of transportation and storage
- UF_6 cylinder model 8A containers (still in use today for enrichments up to 12.5%)
- Fuel was highly enriched uranium (HEU) at 97.7% enrichment UF_6
- Benchmarks would fill an important gap as the ICSBEP does not currently have any UF_6 cylinder benchmarks
- Lead Lab: LLNL and ORNL
- Partner: University of Tennessee, Knoxville

THETA: TRISO-form HALEU-fuel Experiment for Transportation Applications

- Experiment in collaboration with Kairos Power
- Focused on transportation of HALEU TRISO fuel
  - Configurations with stainless steel, polyethylene, and borated polyethylene
- Builds upon Deimos experiment
  - Executed February 2025
  - 5 unique configurations
- Uses the recently characterized CNPS fuel from Deimos



Benchmark Validation for Transportation of TRISO HALEU Fuel for Advanced Reactors

Consolidation

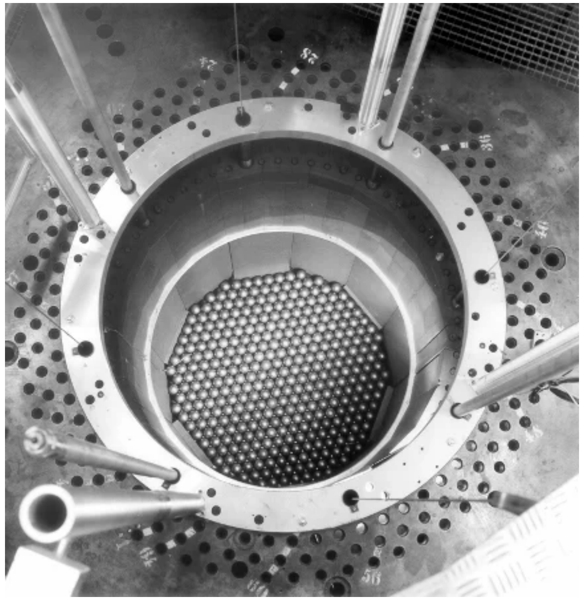
HTR-PROTEUS CRIT portions of
4 IRPhEPs into 1 ICSBEP report
IEU-COMP-THERM-TBD

Improvements

- quality of uncertainty evaluation
- better address uncertainties in mass, density, and composition w. modern M&S
- adjoint sensitivity analysis via ksen card
- reduced statistical uncertainties
- better address uncertainties in random TRISO and pebble placement

Investigation

- solutions from deterministic tools
- impact of newer nuclear data libraries and TSLs
- identify, and if possible, evaluate, <u>subcritical measurement data</u>



- Lead: Idaho National Laboratory
- Partners:
  - <u>active</u>: JFoster & Associates, University of Michigan
  - <u>passive</u>: BWXT, Kairos, X-Energy, Radiant

DNCSH
DOE/NRC Criticality Safety for Commercial-Scale HALEU for Fuel Cycle and Transportation

Pacific Northwest
NATIONAL LABORATORY

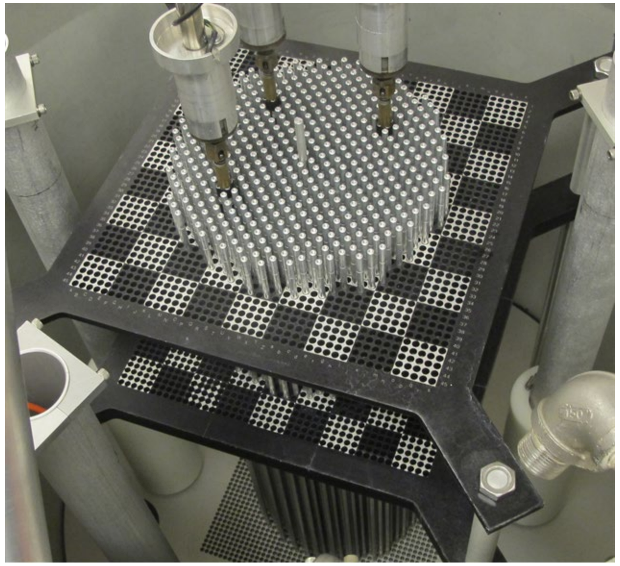
Critical Experiments Targeting Optimum Moderation

- Low moderator density (optimum moderation) conditions accentuate the importance of structural materials such as stainless steel.
- This project is developing experiments for SPRF/CX that are sensitive to these materials.

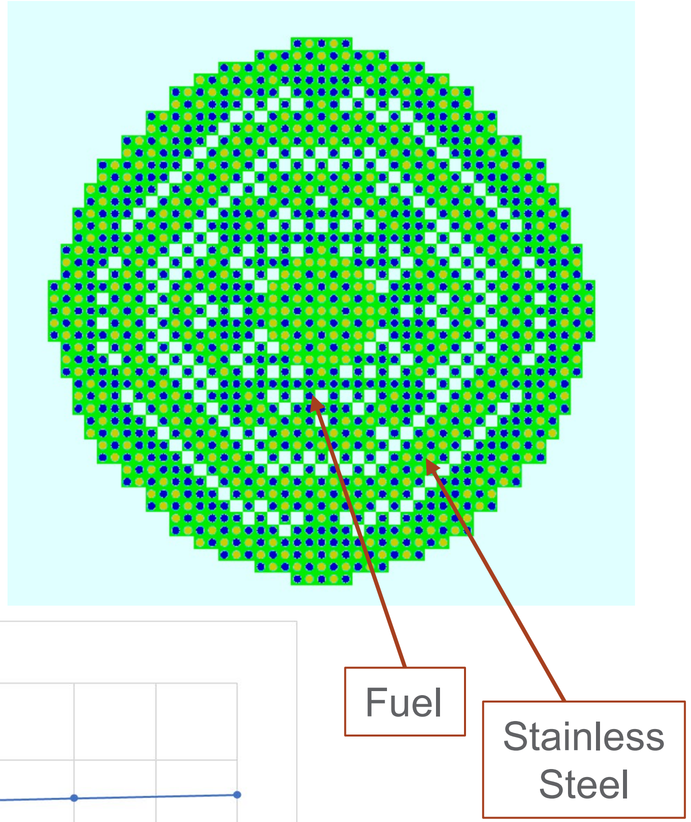
Lead Lab: PNNL

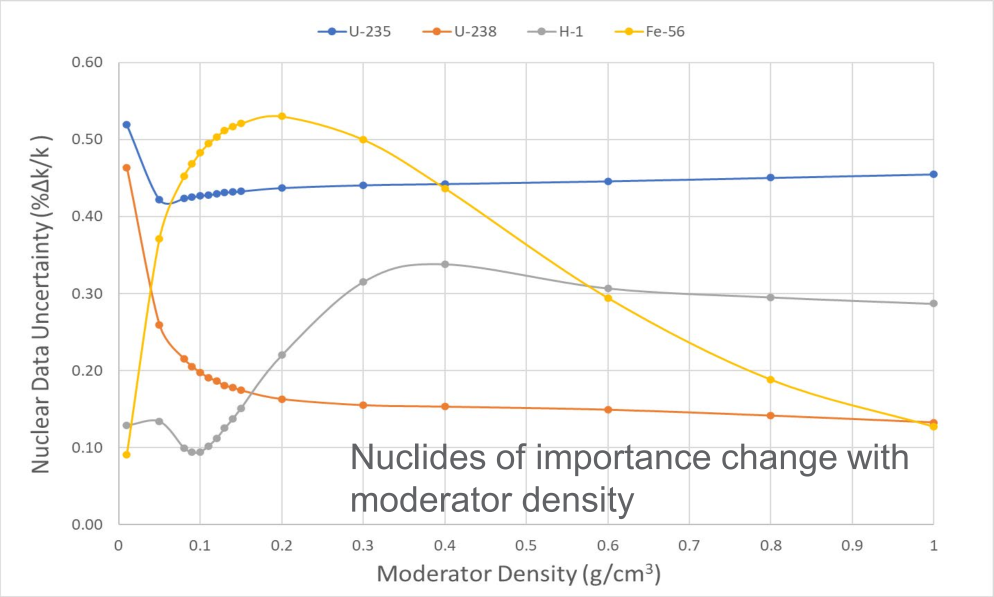
Partners: Sandia, ORNL

SPRF/CX Reactor



Possible Experimental Configuration





Evaluation of already performed critical experiments and design of new critical experiments with the new 19.75 wt% U-235 enriched IPEN/MB-01 core for ICSBEP publication

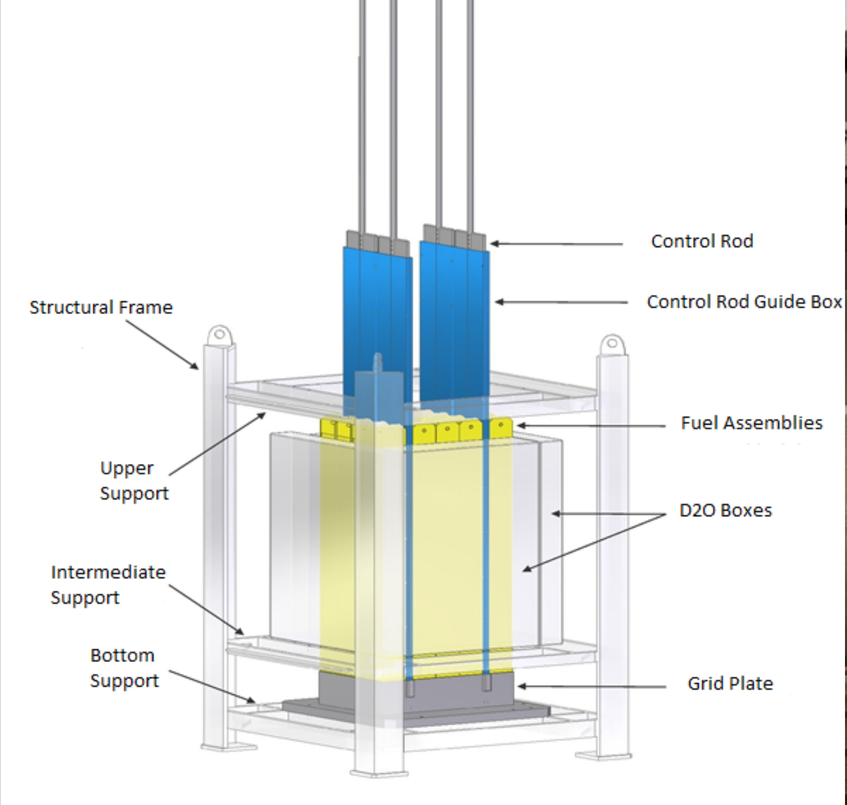
The IPEN/MB01 reactor in Brazil has recently been reconfigured with a new plate-type fuel core using U_3Si_2-Al fuel enriched at 19.75 wt% in ^{235}U.

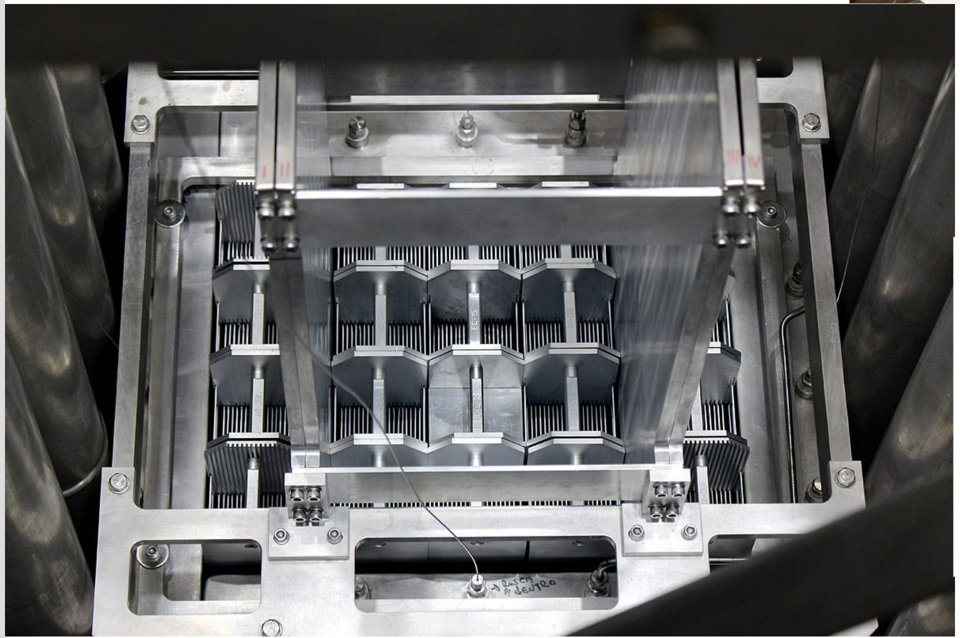
Two evaluations of recently performed and newly designed experiments will be submitted for inclusion in the ICSBEP handbook.

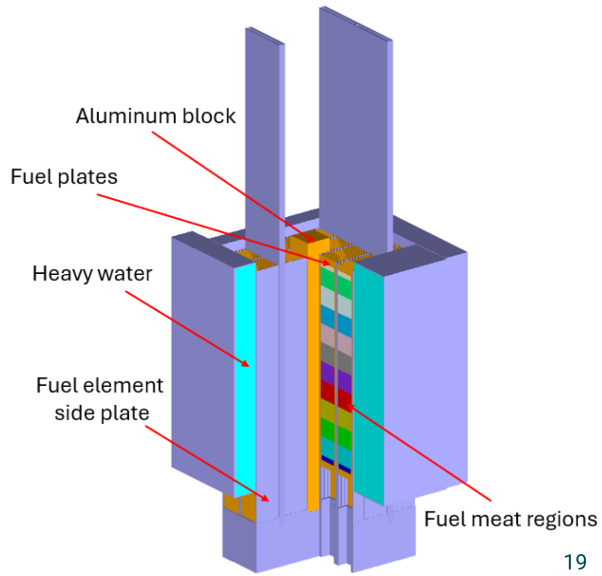
Lead Lab: ORNL

Partners: GE Vernova, Instituto de Pesquisas Energéticas e Nucleares (IPEN), Brazil

Oak Ridge National Laboratory





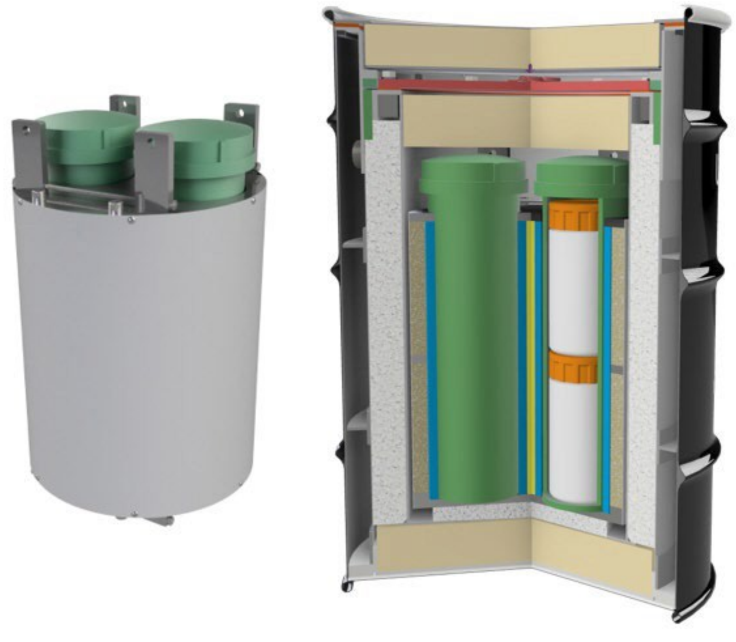


Theresa Cutler
Control Account Manager –
Facility Enhancements
Los Alamos National Laboratory

Facility Enhancements: eDeimos Experiments with Westinghouse for new HALEU Benchmarks

- Builds upon planned experiment with Westinghouse, known as eDeimos
- Fuel to be used for Westinghouse benchmark experiments and DNCSH experiments, and **remain in NCERC inventory**
  - Full ICSBEP level characterization planned
- Feedstock identified and fabrication in-process at BWXT
- Fuel on track to deliver to NCERC in FY2026

A replica of TRISO Compact, the fuel used in a microreactor, at the BWXT facility in Lynchburg Virginia. (Cal Cary/For The Washington post)



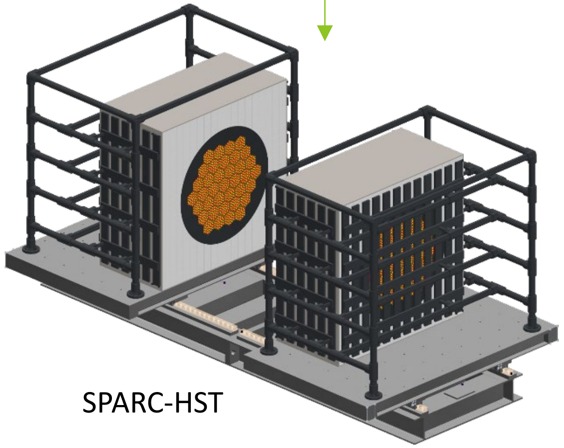
High-capacity Basket

Versa-Pac VP-55 cutaway showing the High-capacity Basket

System Physics Advanced Reactor Critical facility (SPARC)

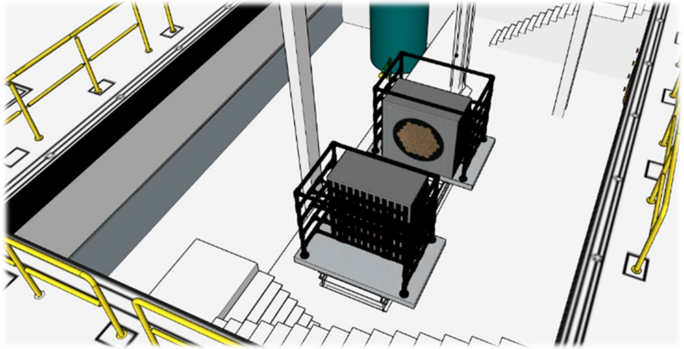
- New horizontal split table (HST) being developed for near term deployment in former SPERT-IV reactor building
  - Large “open basement” with low room return, ample electrical services for electrically heated tests
  - HST can support 2m cube-shaped core (when assembled) and 24,000 kg weight capacity
  - Versatile open table design with fixture points, not hardcoded with mechanical fixture system (e.g., drawers)
  - Enables HALEU-driven “large-and-leaky” tests sensitive to intermediate neutron energies
- Near term fuel system potentials to build inaugural experiments
  - ~600 fuel rods, 4.8% enr. UO_2, 36 in. fuel length, SST clad, from decommissioning Walthousen reactor critical facility (RCF)
  - ~360 U-10Mo plates, 19.75% enr. (9.5×3.5×0.20 in.), former HPRR material

Historic HSTs in the US (all now decommissioned)



SPARC-HST

Former SPERT-IV Building (aka PBF-613)



Rendering of SPARC-HST in basement

U-Mo Ingots

RCF Rods in Storage

HALEU CRITICAL EXPERIMENTS IN WATER MODERATED UO_2 FUEL ROD LATTICES

- New critical benchmark experiments expanding on existing experiments performed at SPRF/CX
- Experiments require UO_2 fuel rods that meet the **same dimensional requirements** as the 7uPCX fuel rods, but with **increased ^{235}U enrichment to nearly 20 wt.%** (HALEU)
- Targets the need for high quality integral critical benchmark experiments within the enrichment range of 10-20 wt.%
  - **Same fuel dimensions** allow critical benchmark experiments with mixtures of 7uPCX and HALEU
- **Lead Lab**: SNL
- **Partner**: Orano

SPRF/CX

Core Tank with 7uPCX fuel

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OAK RIDGE
National Laboratory

August 27, 2025

Nuclear Data: Needs and Current Activities

Iyad Al-Qasir

Control Area Manager: Nuclear Data Enhancement

U.S. DEPARTMENT of ENERGY

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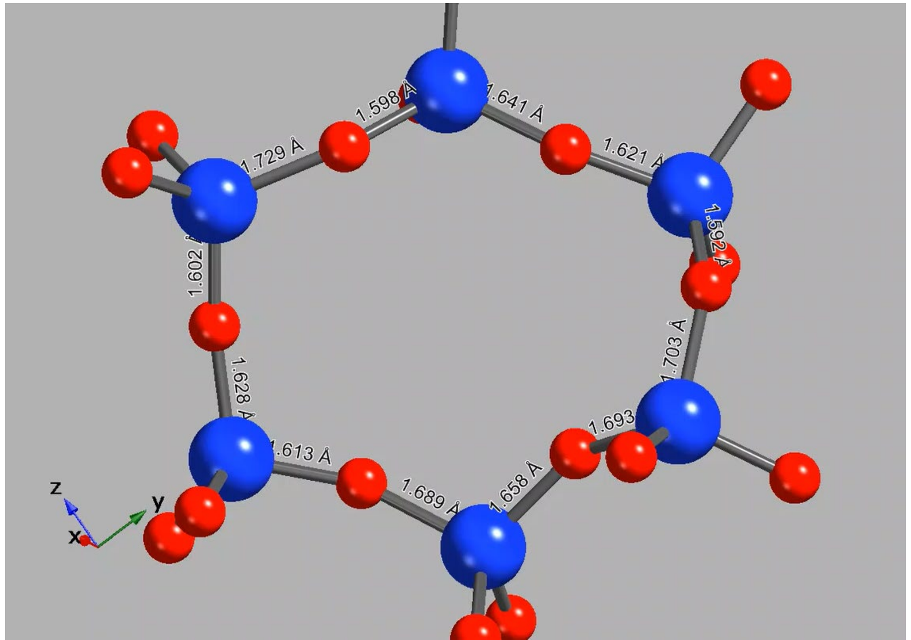
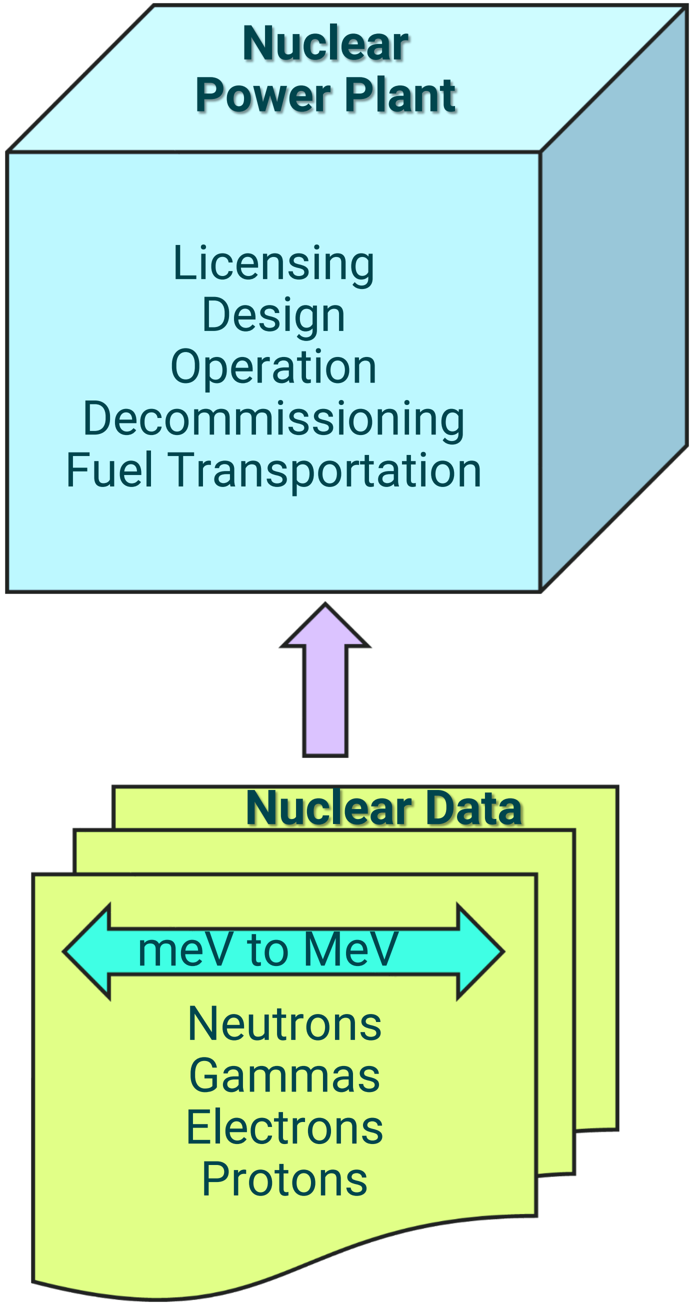
Introduction

- The use of high-quality nuclear data in calculations is essential for accuracy, safety, reliability, optimization fuel performance, research, and <u>regulatory compliance</u> in various nuclear-related fields and applications.
- **Thermal neutron scattering law (TSL)** describes the neutron scattering intensity as a function of energy and momentum transfer between the thermal neutron and the vibrating atoms of the scattering medium





Nuclear Data Needs

UF_6 Transportation
10% - 20% Enrichment Gap
Nonfissile Material Validation
Fissile Salts
✓ Graphite and Advanced Moderator Nuclear Data

Topic Area #5: Benchmarks targeting graphite and advanced moderator nuclear data

Material	Available TSL ENDF Files	Differential XS Measurements	Integral XS Measurements	*Benchmark Experiments
Graphite	Yes	Yes	Yes	Yes
$ZrH_{1.6}$ & ZrH_2	Yes	Yes	Yes	Yes
YH_2	Yes	Yes	Yes	No
Be metal	Yes	Yes	Yes	No
BeO	Yes	No	Yes	No
MgO	No	No	Yes	No
Be_2C	Yes	No	No	No
SiC	Yes	No	No	No
ZrC	No	No	No	No

* Benchmark experiments involve fuel compositions ranging from 5 to 19.75 wt% enrichment of ^{235}U exhibiting a neutron flux of <0.625 eV (*International Handbook of Evaluated Reactor Physics Benchmark Experiments* [IRPhE]).

Current Activities

UF_6 Transportation
10% - 20% Enrichment Gap
Nonfissile Material Validation
Fissile Salts
✓ Graphite and Advanced Moderator Nuclear Data

- Two proposals have been awarded by DNCSH Call #1

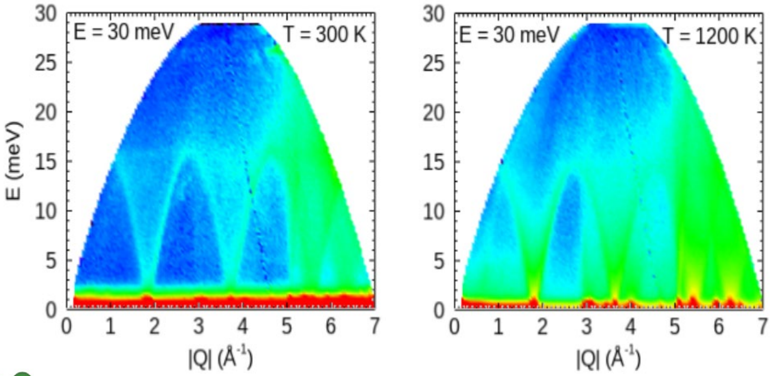
- High-Temperature Graphite Double Differential Scattering Cross Sections: Measurement, Evaluation, and Validation

- Temperature dependent transmission and S(alpha, beta) measurements of advanced nuclear moderators

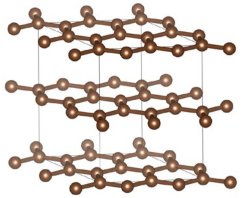
Neutron Moderator	Available TSL ENDF Files	Differential XS Measurements	Integral XS Measurements	*Benchmark Experiments
Graphite	Yes	Ongoing	Ongoing	Yes
$ZrH_{1.6}$ & ZrH_2	Yes	Yes	Yes	Yes
YH_2	Yes	Yes	Yes	No
Be metal	Yes	Yes	Yes	No
BeO	Yes	No	Yes	No
MgO	Yes	Ongoing	Ongoing	No
Be_2C	Yes	No	No	No
SiC	Yes	Ongoing	Ongoing	No
ZrC	Ongoing	Ongoing	Ongoing	No

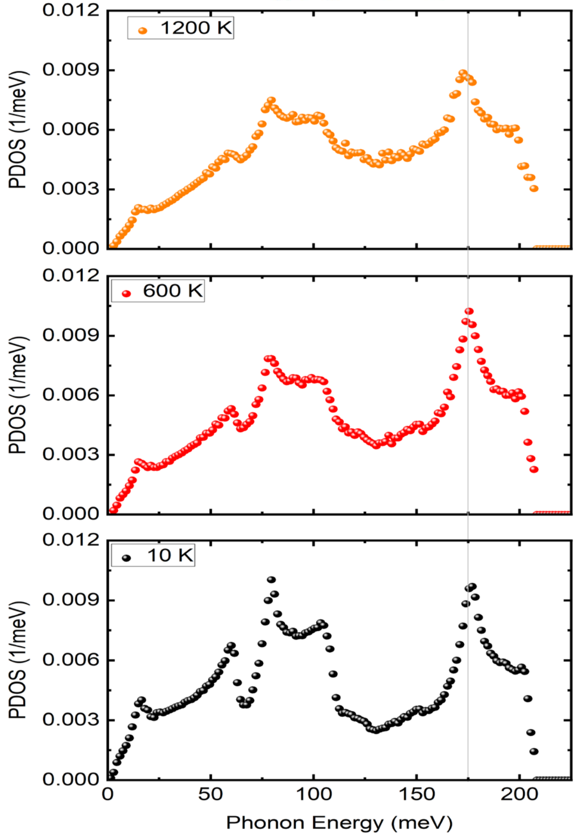
OAK RIDGE National Laboratory

Measurements

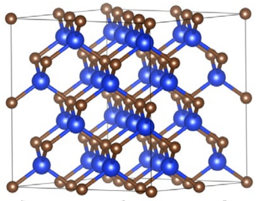


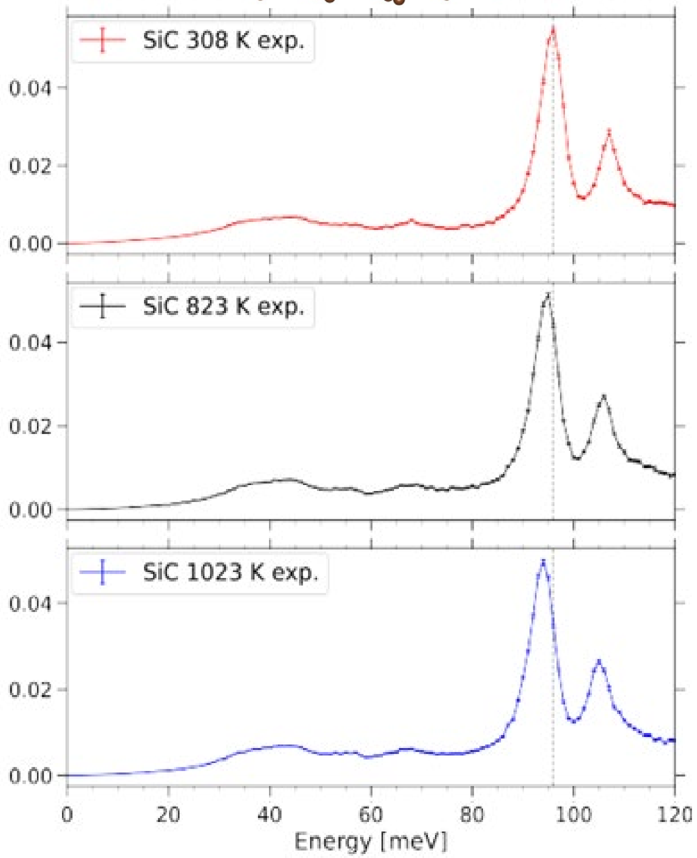
Graphite



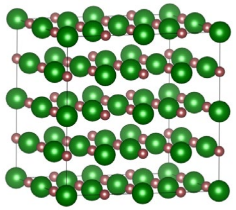


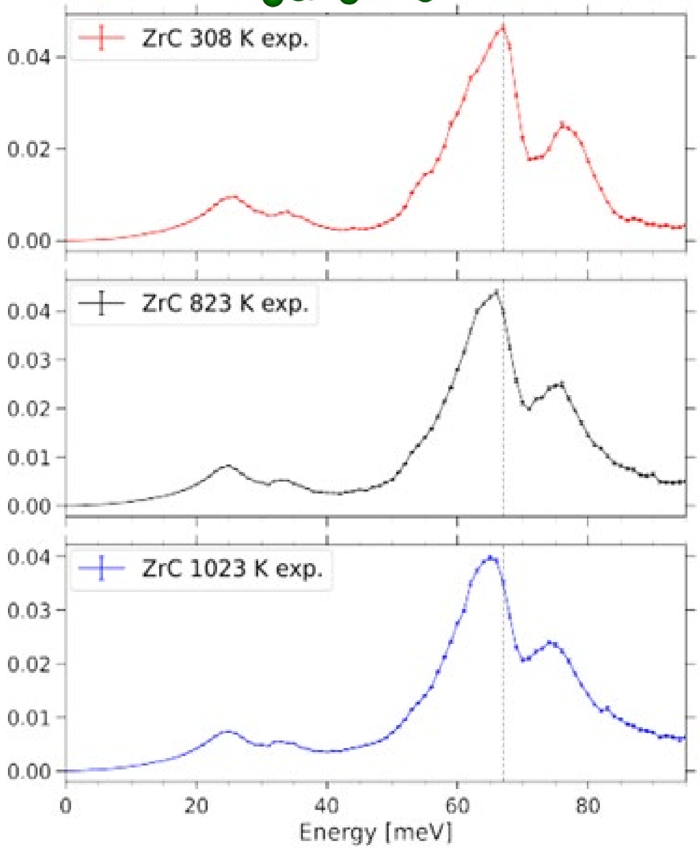
SiC

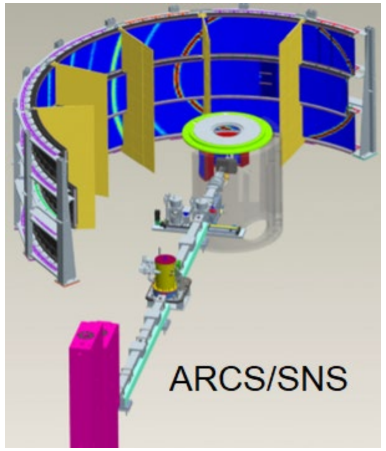




ZrC







Conclusions

- ✓ **Bridging Data Gaps**: Present TSL measurements are filling gaps in available moderator data
- ✓ **Validation of Models**: Present TSL experimental results are essential to validate temperature – dependent calculated phonon density of states
- ✓ **Understanding Moderator Behavior**: Present TSL measurements improve knowledge of atomic vibrations and neutron scattering functions at high temperatures
- ✓ **Impact on Applications**: Enhanced moderator TSL data directly support more reliable criticality calculations and reactor design optimizations
- ✓ **Future Outlook**: Continued integration of state-of-the art measurements with atomistic modeling will enable comprehensive, accurate nuclear data libraries for next-generation systems

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AUGUST 27, 2025

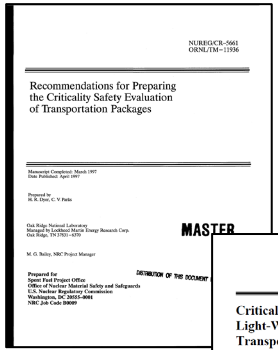
RECENT NUREG/CR REPORTS FOR CRITICALITY SAFETY VALIDATION

Andrew Barto
US Nuclear Regulatory Commission

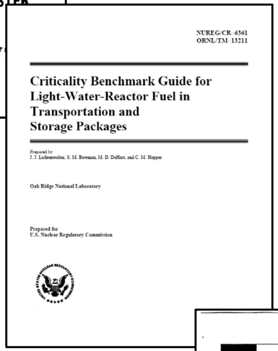
DOE/NRC Criticality Safety for Commercial-Scale HALEU for Fuel Cycle and Transportation

NUREG/CR-7311 - Determination of Bias and Bias Uncertainty for Criticality Safety Computational Methods

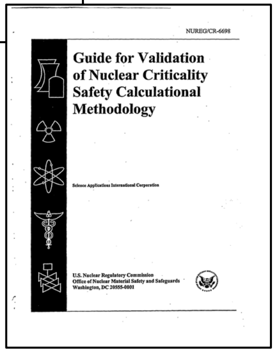
- Provides criticality safety computational method validation techniques for analyses involving all types of fissionable material operations
- Includes information from previous validation guidance (NUREG/CRs-5661, -6361, and -6698), and provides new recommendations where previous guidance was lacking



NUREG/CR-5661



NUREG/CR-6361



NUREG/CR-6698

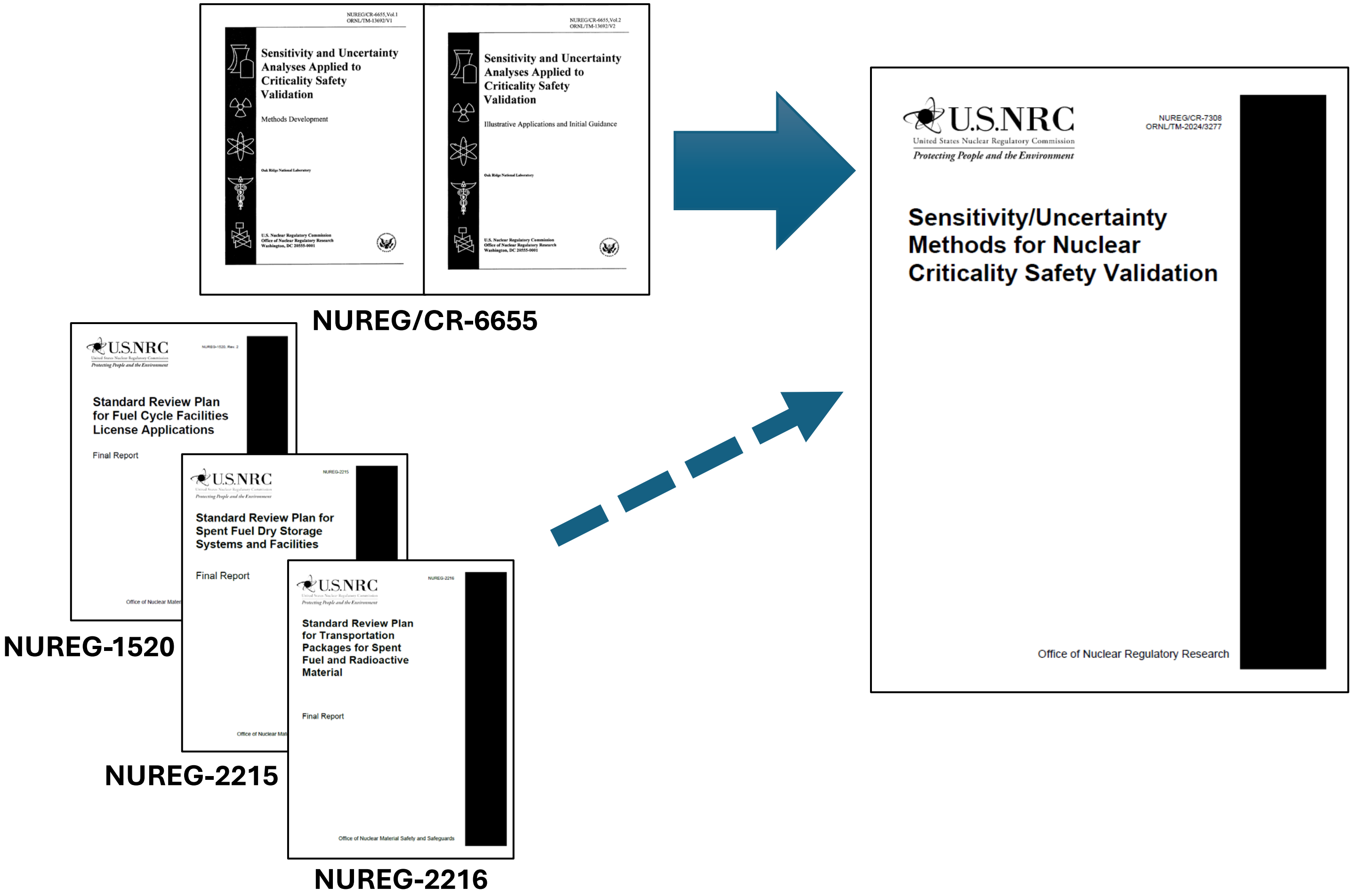
NUREG/CR-7311

NUREG/CR-7311 - Determination of Bias and Bias Uncertainty for Criticality Safety Computational Methods

- Purpose of Validation
- Definition of Computational Method
- Critical Experiment Selection and Area of Applicability Determination
- Statistical Background
  - Hypothesis Testing
  - Assessment of Normality
  - Goodness-of-Fit Testing
- Determination of Bias and Uncertainty
  - Non-Trending Methods
  - Non-Parametric Methods
  - Analysis of Trends
- Identifying and Addressing Validation Weaknesses and Gaps
- Documentation

NUREG/CR-7308 - Sensitivity/Uncertainty Methods for Nuclear Criticality Safety Validation

- Provides recommendations for using S/U methods to calculate sensitivity coefficients, confirm their accuracy, perform uncertainty analysis of validation gaps, and assess benchmark similarity for criticality validation
- Incorporates and expands on information from NUREG/CR-6655 (1999)
- Expands on recommendations from Standard Review Plans



NUREG/CR-7308 - Sensitivity/Uncertainty Methods for Nuclear Criticality Safety Validation

- Validation Overview
- Previous Guidance on S/U
  - NUREG/CR-6655
  - Nuclear Technology article
  - TSUNAMI Primer
- Theoretical Aspects of S/U Analysis Applied to NCS Validation
  - Sensitivity coefficients, perturbation theory, nuclear data
  - TSUNAMI implementation
  - Covariance data
  - Uncertainty analysis and similarity assessment
- Recommendations for S/U Methods in NCS Validation
  - Direct perturbation
  - Multigroup and continuous energy methods
  - Uncertainty analysis and similarity assessment
  - Sources of sensitivity data
- Case Studies
- Advanced Capabilities

Manufacturing Analysis, Call #2

Larry L. Wetzel P.E.

Aug. 27, 2025

Background

- My background
  - ~2 years of supporting DNCSH
  - 34 years of NCS experience at BWXT (fresh fuel manufacturing)
  - 2 years of NCS experience at Naval Reactors Facility (spent fuel analysis)
  - Chair of Nuclear Criticality Safety Consensus Committee
  - Chair of ANS 8.24 on validation
- Focus of this work
  - Identify gaps in benchmarks which are needed to support fuel manufacturing NCS analysis
  - These gaps also extend to transport of related fuel material as feed material and finished assemblies
- ORNL/TM-2025/3744 https://www.nrc.gov/docs/ML2506/ML25062A173.pdf

Object of this work

- Comparison of expected manufacturing configuration in a NCS analysis
  - Select fuel forms based on planned reactor designs
  - Establishing minimum critical mass (optimum H/X) bare and reflected
  - Evaluating infinite arrangements of some of the fuel forms in water or graphite
  - Evaluating selected cases for TRISO manufacturing.

- TSUNAMI used for generating SDF files and TSUNAMI-IP for comparison with benchmark SDF files
  - SDF files generated with SCALE 6.3.1 TSUNAMI
    - ce_v7.1_endf cross section library
  - Compared against the SDF files for ICSBEP downloaded from NEA

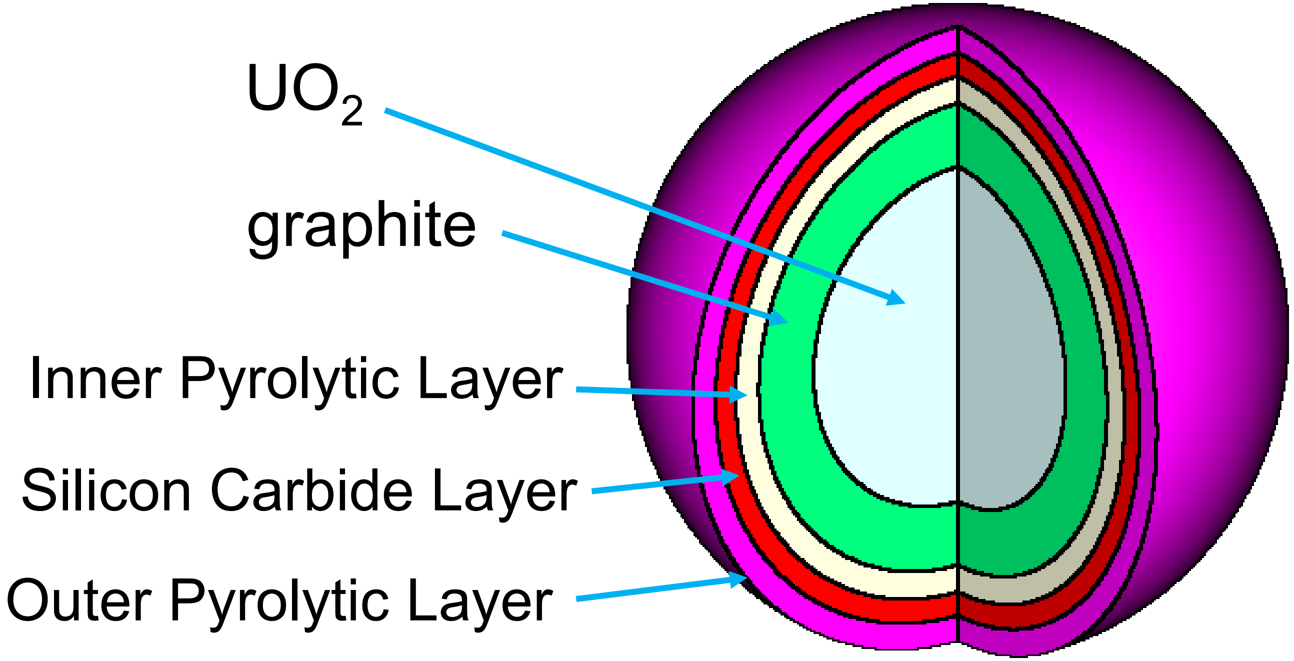
Fuel Types Evaluated

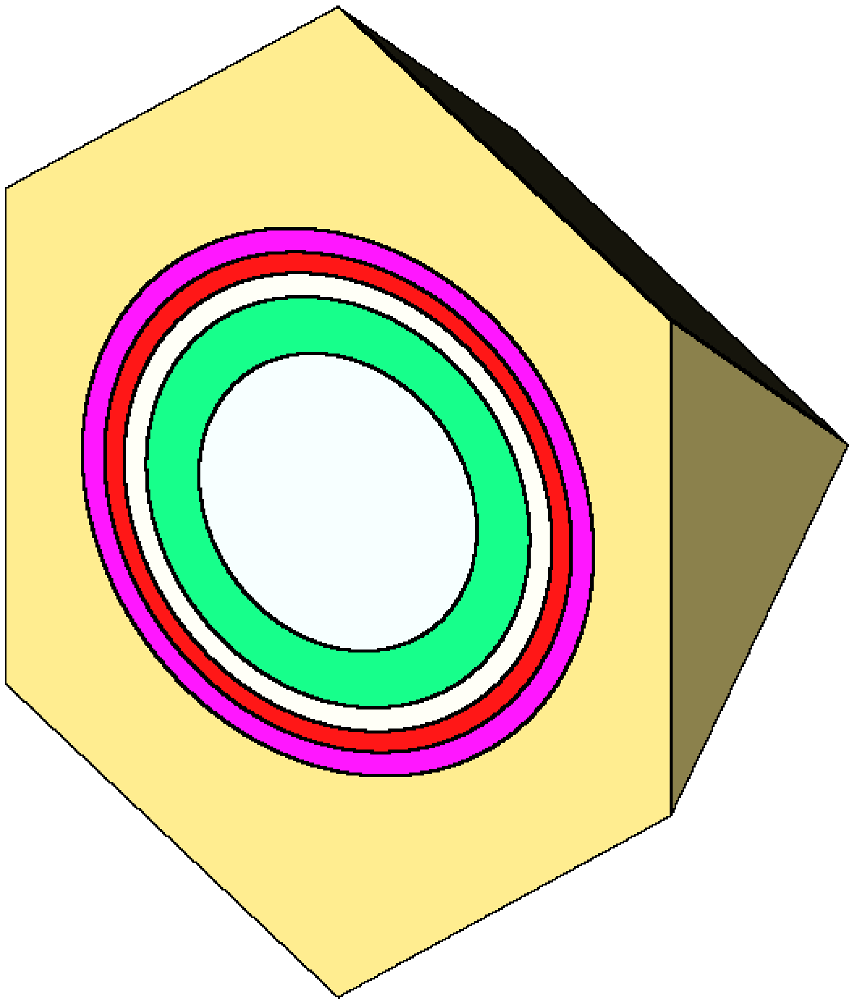
- Intermediate and finished fuel forms
  - TRISO (19.75 wt% and 9.9 wt%)
  - Uranium metal (19.75 wt%)
  - U-Mo(10) (19.75 wt%)
  - U-Zr(10) (19.75 wt%)
  - UO_2 (19.75 wt%)
  - UCl_3 (19.75 wt%)

Configurations Evaluated

- TRISO particles
  - Unreflected, water-reflected, or graphite-reflected sphere with interstitial void
  - Unreflected, water-reflected, or graphite-reflected sphere with interstitial water
  - Unreflected or silicon carbide sphere with interstitial silicon carbide
  - Infinite lattice with water, graphite, or silicon carbide moderator

Basic TRISO Model



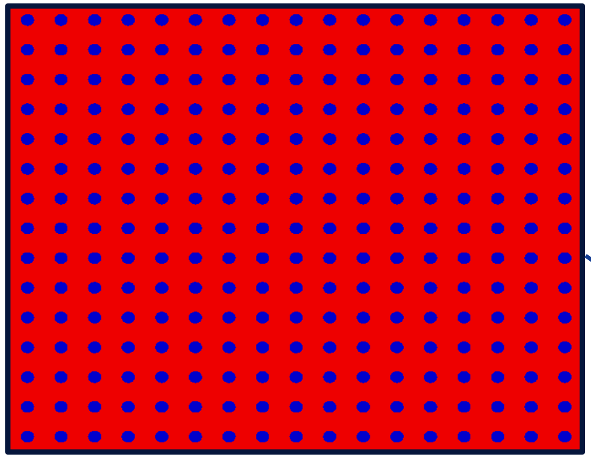


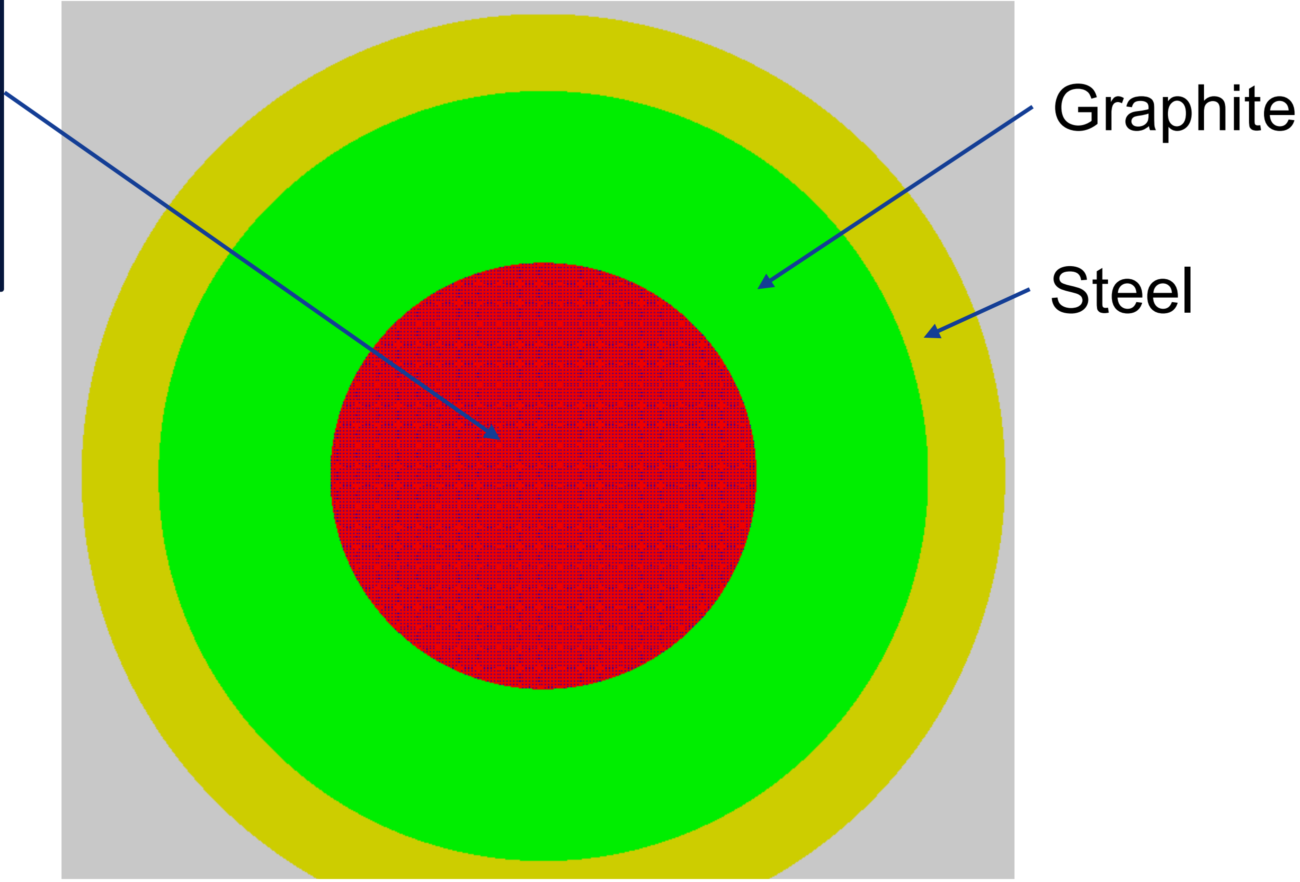
Particle in a dodecahedron lattice

TRISO Configurations Evaluated

- TRISO manufacturing
  - Sol-gel formation
  - Kernel conversion
  - Coating
  - Densification

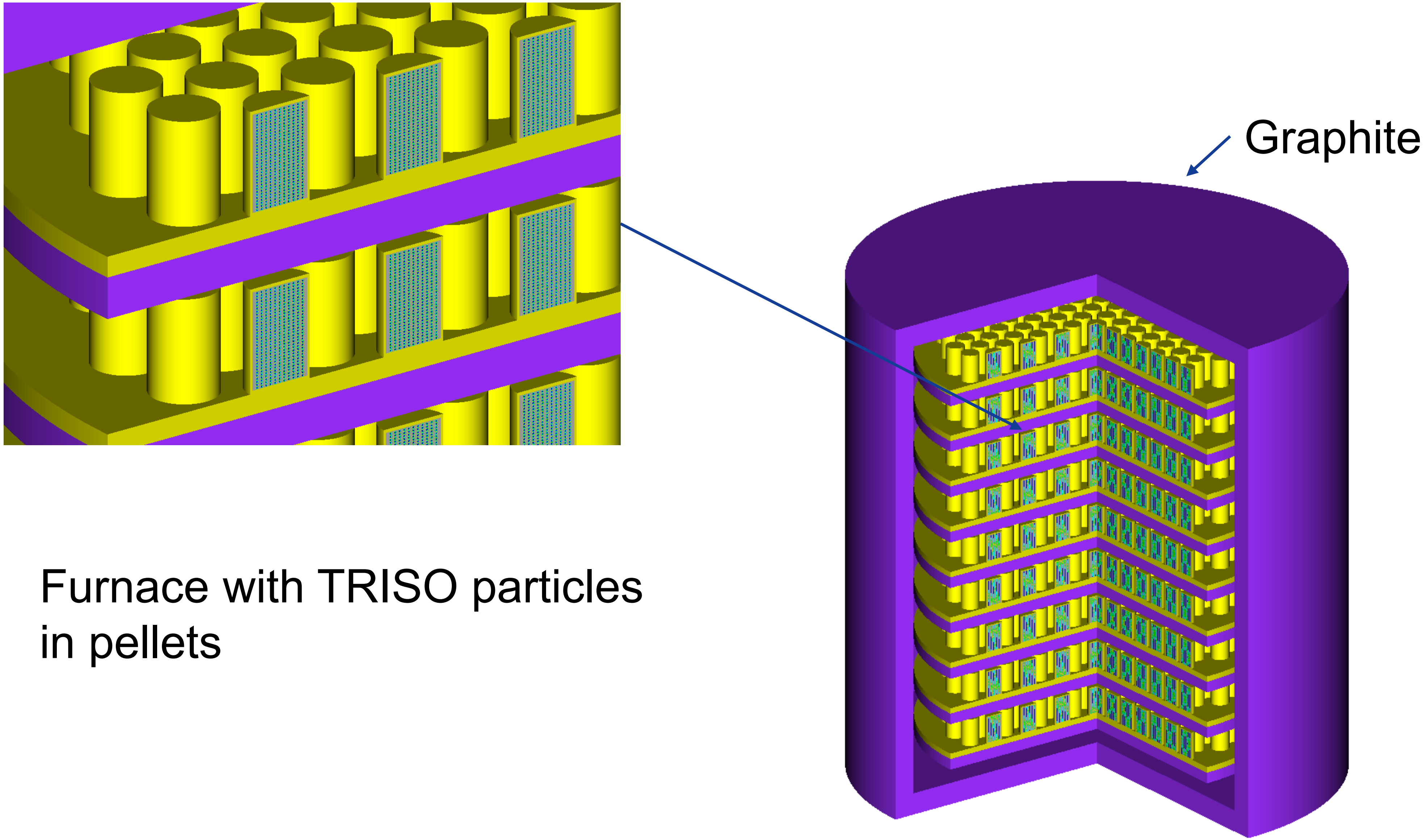
Detailed Models





Lattice of TRISO particles
in a furnace

Detailed Models

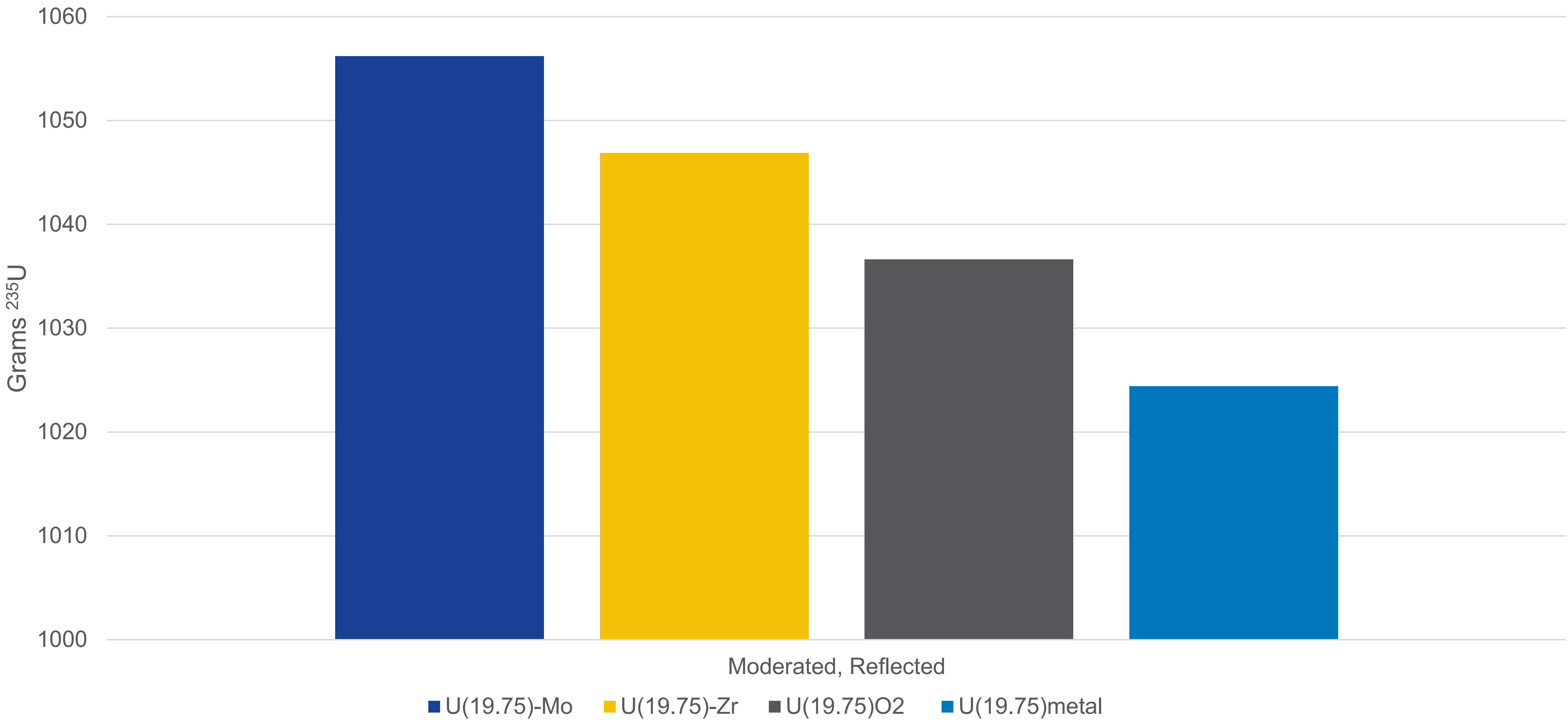


Furnace with TRISO particles in pellets

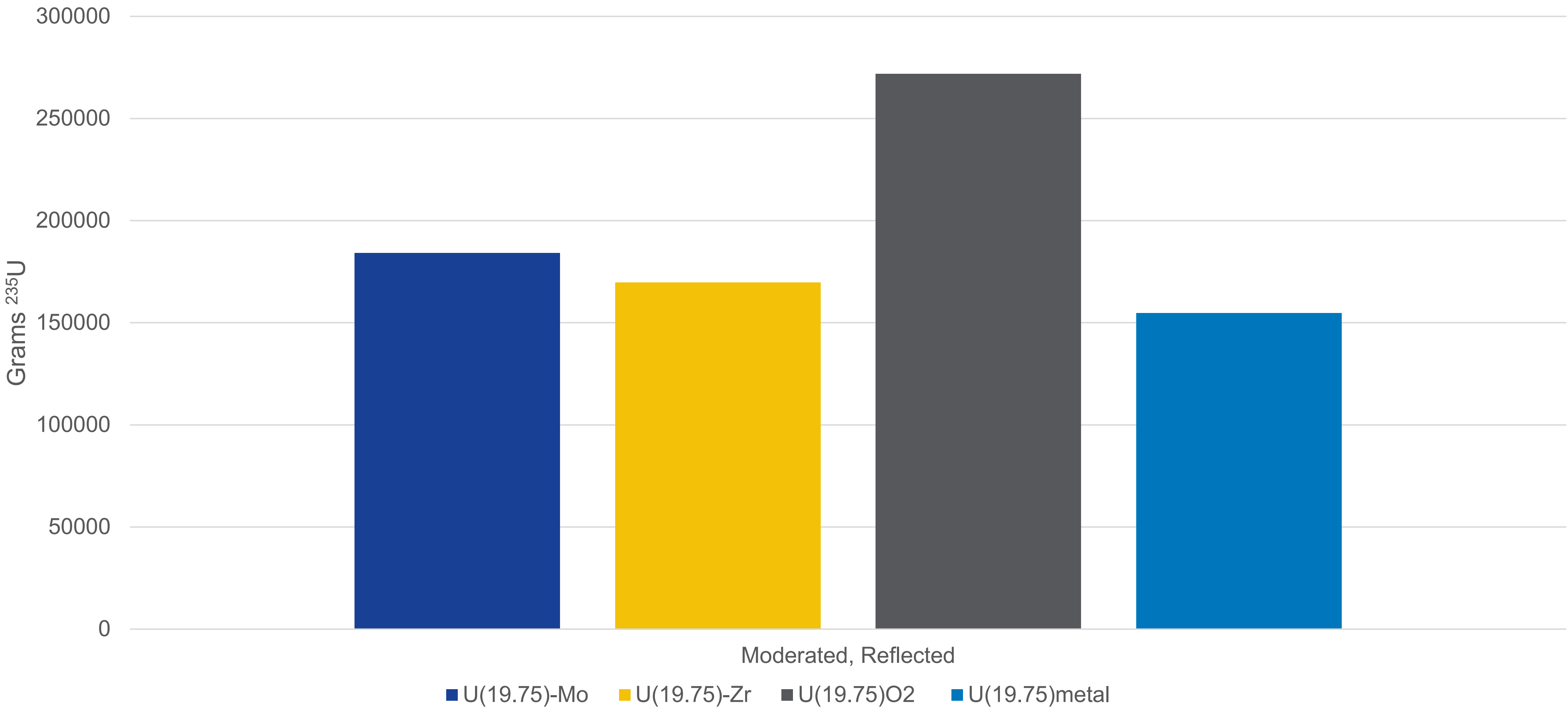
Other Configurations Evaluated

- Uranium metal, U-Mo(10), U-Zr(10), and UO_2
  - Bare Sphere
    - Optimum water moderation
    - No water moderation
  - Water Reflected Sphere
    - Optimum water moderation
    - No water moderation
- UCl_3 (19.75 wt%)
  - Infinite moderated system
  - Unmoderated sphere
    - Bare
    - Water-Reflected

Moderated and Water-Reflected Critical Masses



Unmoderated and Unreflected Critical Masses



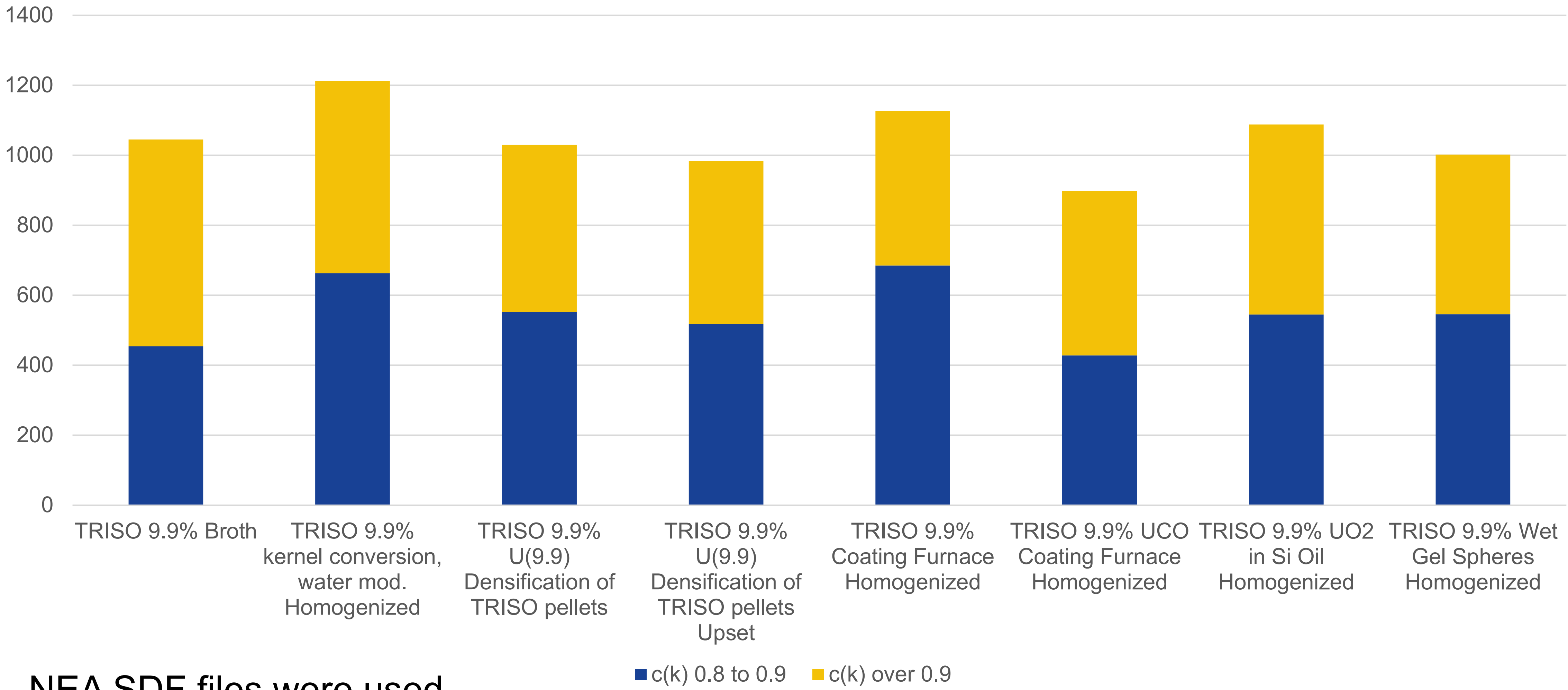
System ^{235}U and U Mass Values

Configuration	Mass (^{235}U g)	Mass (U g)
U(19.75)-Mo Moderated Reflected	1047	5301
U(19.75)-Mo Moderated Unreflected	1818	9207
U(19.75)-Mo Unmoderated Reflected	76680	388272
U(19.75)-Mo Unmoderated Unreflected	169800	859500
U(19.75)-Zr Moderated Reflected	1047	5301
U(19.75)-Zr Moderated Unreflected	1818	9207
U(19.75)-Zr Unmoderated Reflected	76680	388300
U(19.75)-Zr Unmoderated Unreflected	169800	859600

System ^{235}U and U Mass Values

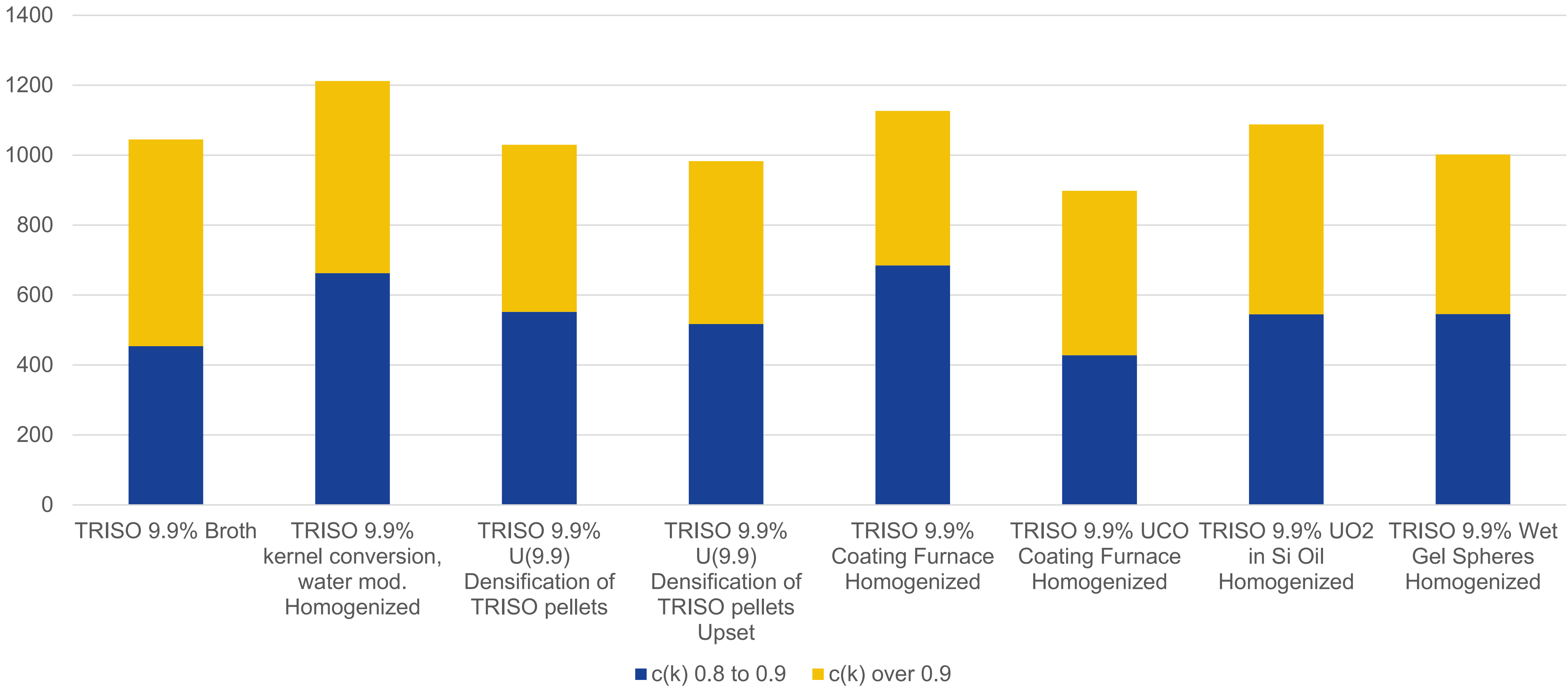
Configuration	Mass (^{235}U g)	Mass (U g)
U(19.75)Cl3 Unmoderated Reflected	2207000	11180000
U(19.75)Cl3 Unmoderated Unreflected	3194000	16170000
U(19.75)O2 Moderated Unreflected	1037	5249
U(19.75)O2 Moderated Reflected	1831	9270
U(19.75)O2 Unmoderated Reflected	123900	627500
U(19.75)O2 Unmoderated Unreflected	271800	1376000
U(19.75)metal Moderated Unreflected	1024	5187
U(19.75)metal Moderated Reflected	1815	9191
U(19.75)metal Unmoderated Reflected	66200	335200
U(19.75)metal Unmoderated Unreflected	154800	783800

TSUNAMI-IP Results for TRISO

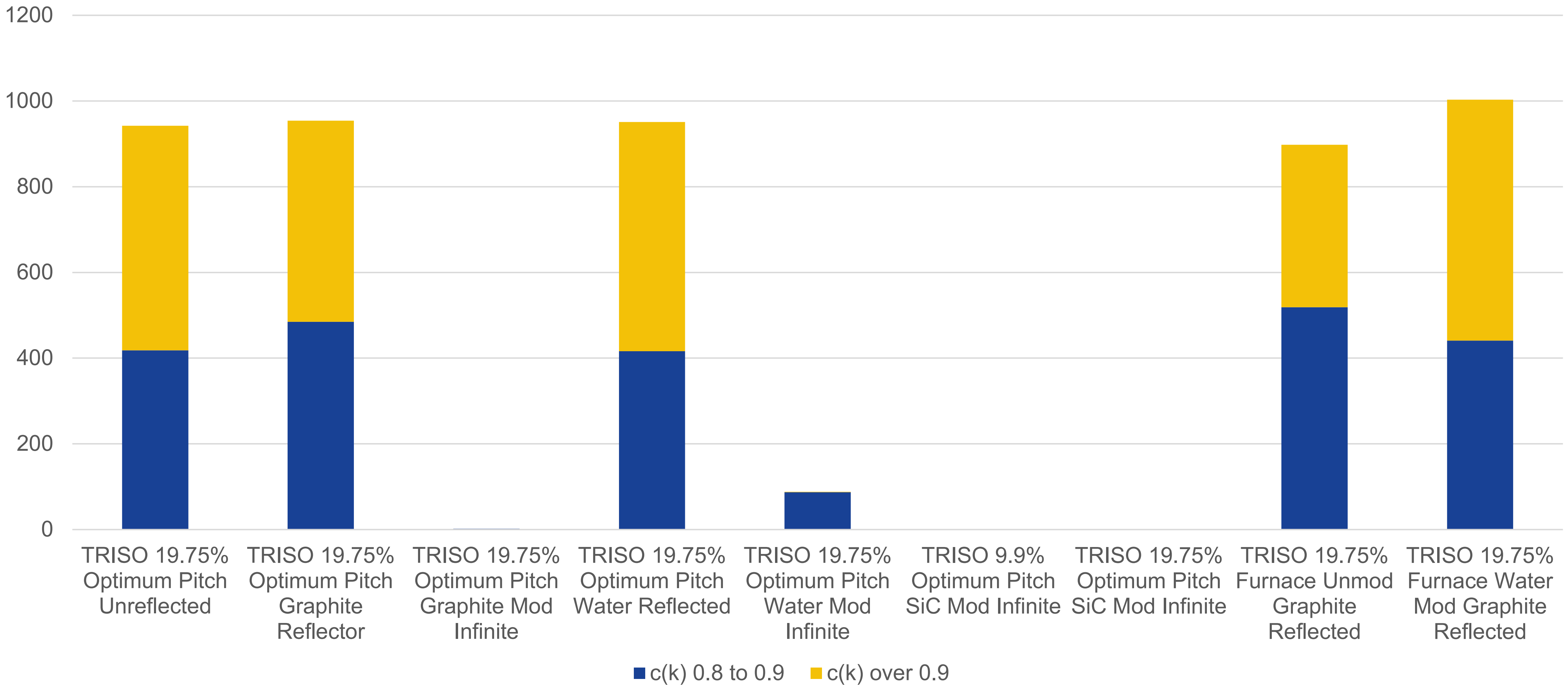


NEA SDF files were used

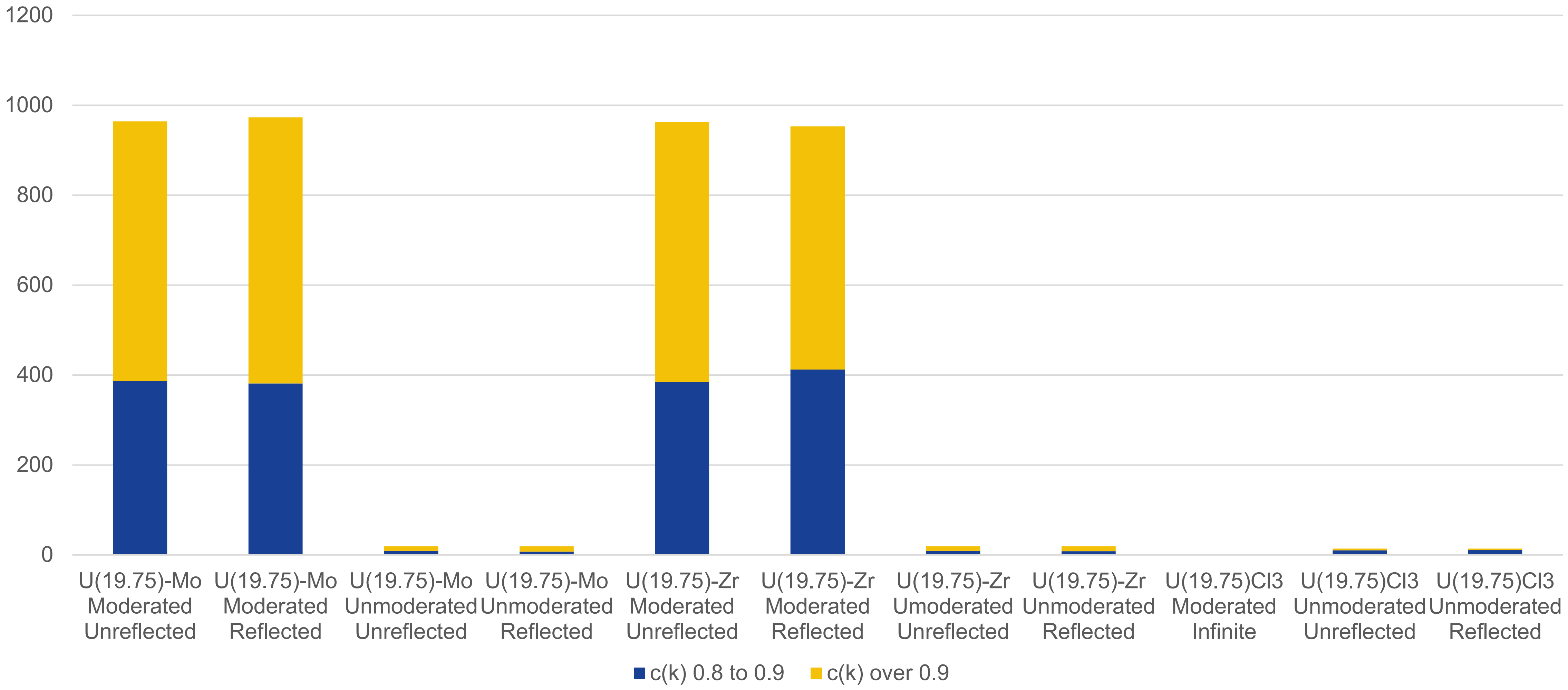
TSUNAMI-IP Results for TRISO



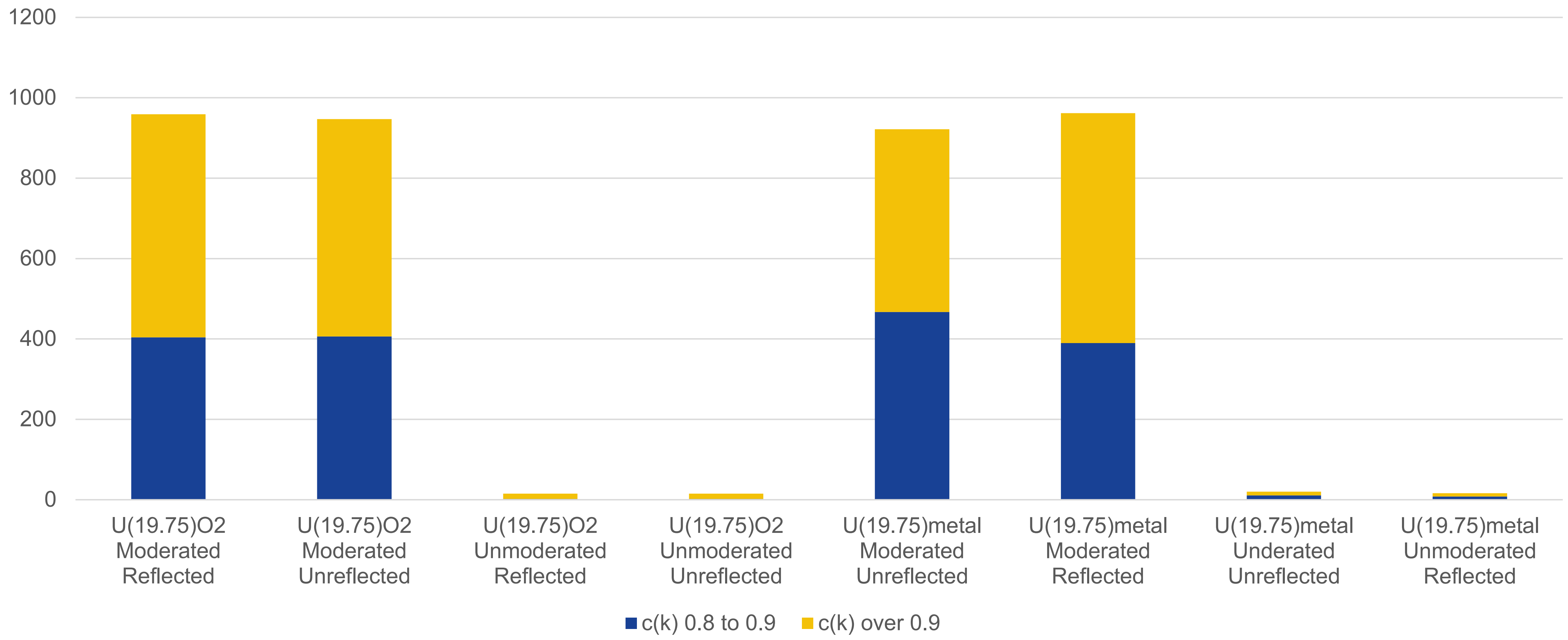
TSUNAMI-IP Results for TRISO



TSUNAMI-IP Results for Other Fuel Types



TSUNAMI-IP Results for Other Fuel Types



TSUNAMI-IP Results (NEA)

	Description	c(k)>0.8	c(k)>0.9
1	TRISO 9.9% Broth	1045	591
2	TRISO 9.9% kernel conversion, water mod. Homogenized	1212	549
3	TRISO 9.9% U(9.9) Densification of TRISO pellets	1030	478
4	TRISO 9.9% U(9.9) Densification of TRISO pellets Upset	983	466
5	TRISO 9.9% Coating Furnace Homogenized	1127	442
6	TRISO 9.9% UCO Coating Furnace Homogenized	898	470
7	TRISO 9.9% UO2 in Si Oil Homogenized	1088	543
8	TRISO 9.9% Wet Gel Spheres Homogenized	1002	456
9	TRISO 19.75% Broth	1044	632
10	TRISO 19.75% kernel conversion, water mod. Homogenized	1124	490
11	TRISO 19.75% U(19.75) Densification of TRISO pellets	1028	581
12	TRISO 19.75% U(19.75) Densification of TRISO pellets Upset	961	520
13	TRISO 19.75% Coating Furnace Homogenized	997	407
14	TRISO 19.75% UCO Coating Furnace Homogenized	637	1
15	TRISO 19.75% UO2 in Si Oil Homogenized	1094	685

TSUNAMI-IP Results (NEA)

	Description	c(k)>0.8	c(k)>0.9
16	TRISO 19.75% Wet Gel Spheres Homogenized	1070	642
17	TRISO 19.75% Optimum Pitch Unreflected	942	524
18	TRISO 19.75% Optimum Pitch Graphite Reflector	954	469
19	TRISO 19.75% Optimum Pitch Graphite Mod Infinite	2	0
20	TRISO 19.75% Optimum Pitch Water Reflected	951	535
21	TRISO 19.75% Optimum Pitch Water Mod Infinite	88	1
22	TRISO 9.9% Optimum Pitch SiC Mod Infinite	0	0
23	TRISO 19.75% Optimum Pitch SiC Mod Infinite	0	0
24	TRISO 19.75% Furnace Unmod Graphite Reflected	898	379
25	TRISO 19.75% Furnace Water Mod Graphite Reflected	1003	562
26	U(19.75)-Mo Moderated Unreflected	964	578
27	U(19.75)-Mo Moderated Reflected	973	592
28	U(19.75)-Mo Unmoderated Unreflected	19	10
29	U(19.75)-Mo Unmoderated Reflected	19	12
30	U(19.75)-Zr Moderated Unreflected	962	578

TSUNAMI-IP Results (NEA)

	Description	c(k)>0.8	c(k)>0.9
31	U(19.75)-Zr Moderated Reflected	953	541
32	U(19.75)-Zr Unmoderated Unreflected	19	10
33	U(19.75)-Zr Unmoderated Reflected	19	11
34	U(19.75)Cl3 Moderated Infinite	0	0
35	U(19.75)Cl3 Unmoderated Unreflected	14	4
36	U(19.75)Cl3 Unmoderated Reflected	14	3
37	U(19.75)O2 Moderated Reflected	959	555
38	U(19.75)O2 Moderated Unreflected	947	541
39	U(19.75)O2 Unmoderated Reflected	15	13
40	U(19.75)O2 Unmoderated Unreflected	15	13
41	U(19.75)metal Moderated Unreflected	922	455
42	U(19.75)metal Moderated Reflected	962	572
43	U(19.75)metal Unmoderated Unreflected	20	9
44	U(19.75)metal Unmoderated Reflected	16	8

Summary of Gaps

- Finite system gaps:
  - Highly moderated systems
    - Moderated TRISO particles
    - TRISO particles with SiC and moderated
    - Moderated UCl_3
  - Unmoderated or low moderation (H/X<3)
    - U(19.75) unreflected and water-reflected, any form (metal, oxide, UF6, U-Mo(10), etc.)
- Infinite system:
  - Highly moderated systems
    - TRISO particles in water or graphite
    - UCl_3 moderated

Recommended Focus

1. Moderated TRISO at 20 wt% fuel
   - Reflected with water or graphite
2. Unmoderated uranium at 20 wt% systems (U, UO_2, UF_6, etc.)
   - Bare and water-reflected
3. Low Moderation (high uranium density at 20 wt%, H/X<3)
   - Bare and water-reflected
4. Other Uranium compounds at 20 wt% (U-Mo, UCl_3, U-Zr, etc.)
   - Bare and water-reflected

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August 27,2025 | DNCSH

Application Models Repository

Cihangir Celik, Veronica Karriem, Alex Shaw, Ahmed Shama, Riley Cumberland, Larry Wetzel, Zhaopeng Zhong, Ahmed A. E. Abdelhameed, Nicolas Stauff

DNCSH Workshop #2

U.S. DEPARTMENT of ENERGY

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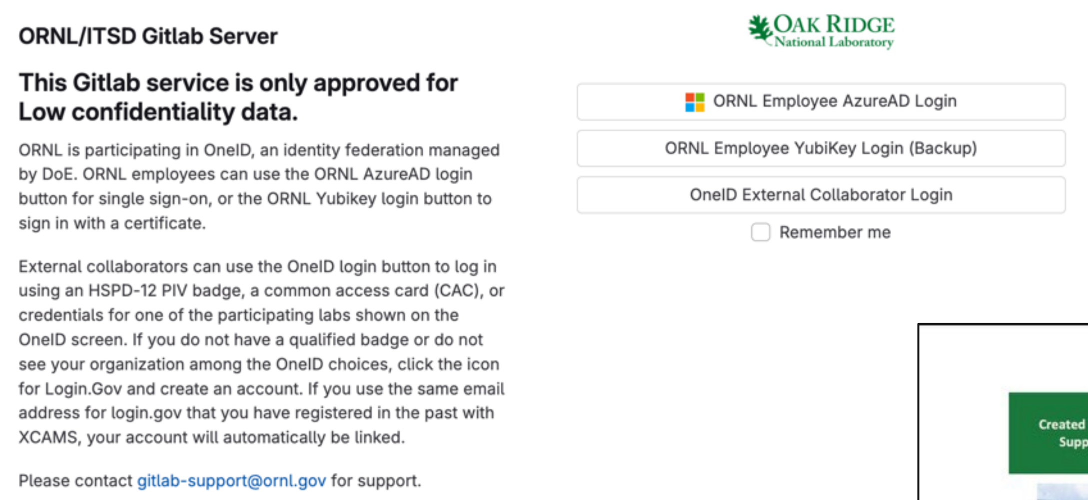
Outline

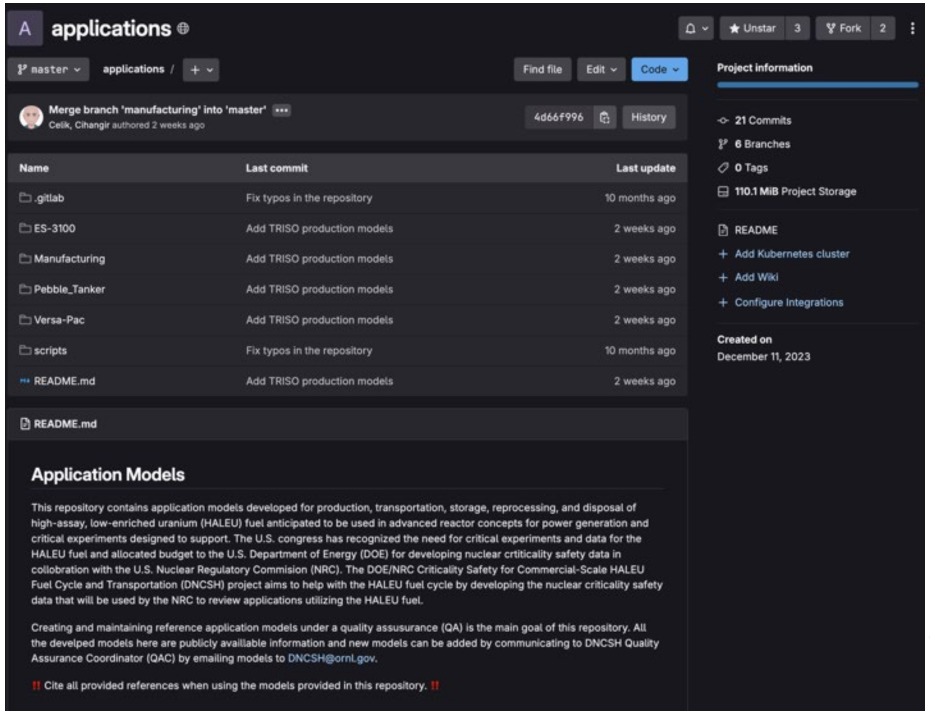
- Goal of the application models repository
- Available models
- Summary and future work

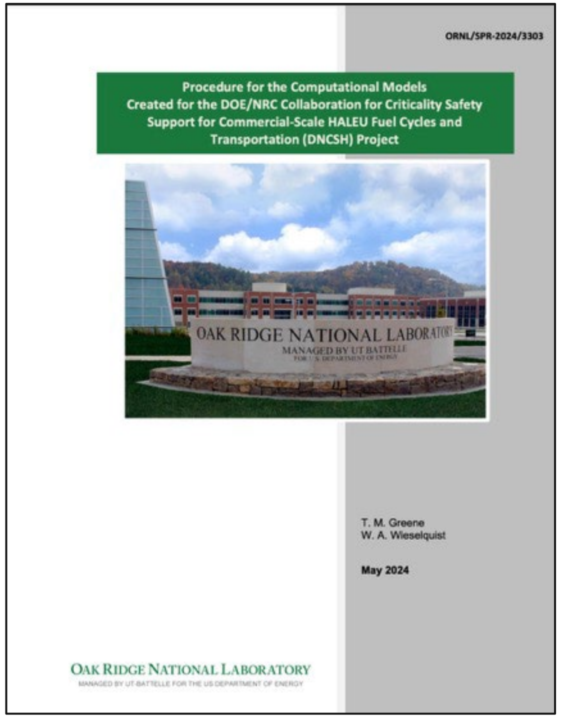
Goal of the application models repository

Create and maintain a ***public*** repository with application models relevant to ***HALEU fuel transportation and storage*** that can be used by ***regulators, industry, academia, or anyone*** to support ***verification and validation*** of their own applications.

- GitLab repository hosted at ORNL
- maintain and expand with new models under [QA]
- all models are documented and peer reviewed
- contains computational models (SCALE and MCNP codes) and derived sensitivity data files (SDFs)
- publicly accessible at [https://code.ornl.gov/dncsh/applications]







DNCSH
DOE/NRC Criticality Safety for Commercial-Scale HALEU for Fuel Cycle and Transportation

OAK RIDGE National Laboratory

Available models: ES-3100

Developed for transportation of fresh/used HALEU nuclear fuel in the form of fuel pins.

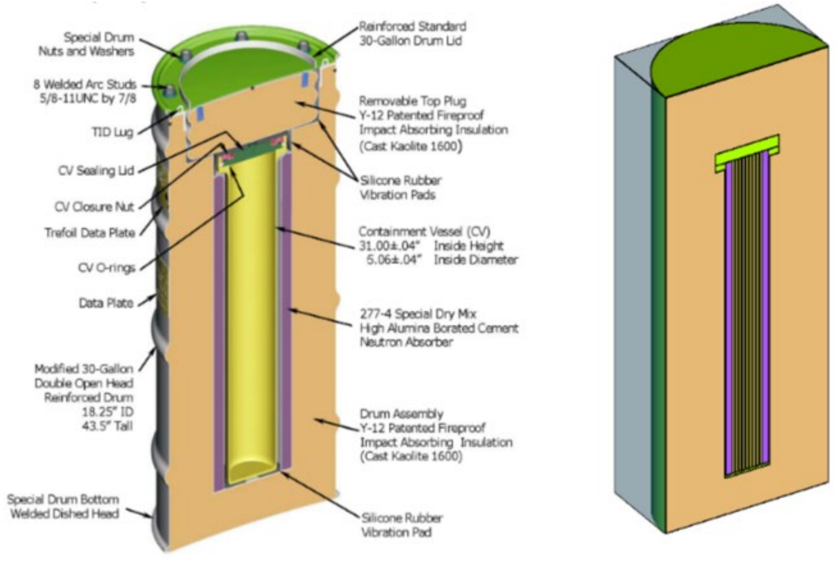
- 30-gal drum developed by Y-12 for HEU transportation

Sodium Fast Reactor (SFR) fuel (20 wt% ^{235}U in U-10Zr)

- Advanced Burner Test Reactor (ABTR) fuel

SCALE models are available

MCNP models are in review



ES-3100 Schematics **ES-3100 SCALE Model**

Figure 1: ES-3100 Schematic and its SCALE model

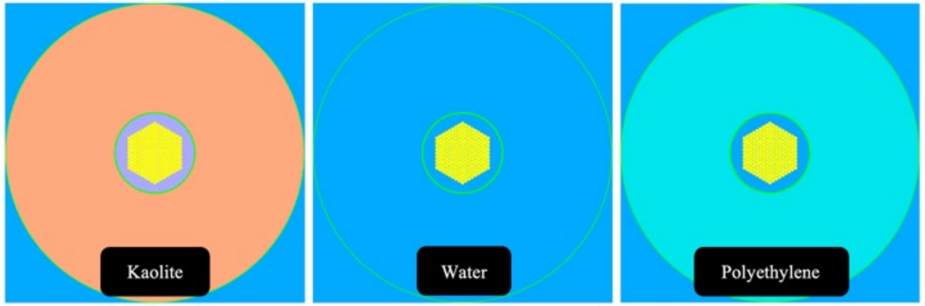
Calculated k_{eff} and c_k values[1]

	Dry	Wet	Wet (0.3814 cm)
k_{eff}	0.51690	0.58452	0.69454
$c_k \geq 0.8$	0	0	373
$c_k \geq 0.9$	0	0	0
Maximum c_k	0.7236	0.7862	0.8947

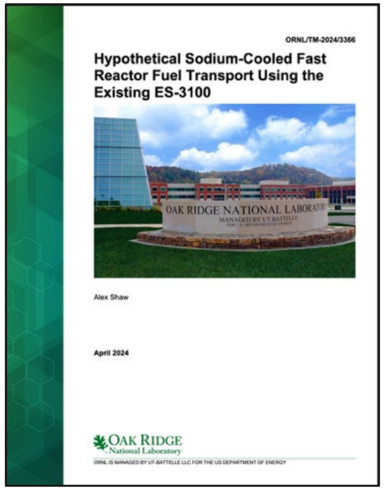
Comparison of k_{eff} values – water flooded cases

SCALE 6.3*	MCNP 6.3**	Δ (pcm)
0.58435	0.58501	66

*σ < 20 pcm ** σ < 30 pcm



ES-3100 models[1]



[1]Alex Shaw (2024), *Hypothetical Sodium-Cooled Fast Reactor Fuel Transport Using the Existing ES-3100*, *ORNL/TM-2024/3366, Oak Ridge National Laboratory, Oak Ridge, TN.*

Available models: Versa-Pac

Developed for transportation of fresh/used HALEU nuclear fuel in various forms.

- 55-gal configuration (VP-55) approved for transporting fuel pebbles by U.S. NRC

Pebble Bed Modular Reactor (PBMR) fuel pebbles with 20 wt% ^{235}U in TRISO particles

SCALE models are available

MCNP models are in review

TRISO filled VP-55 models are in review

Molten Salt Reactor (MSR) fuel salts with VP-55 models are in review

VP-55 model[1]

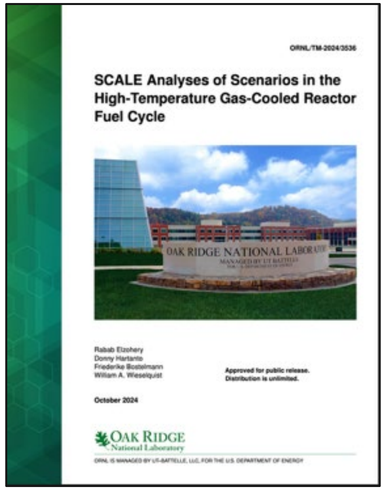
Comparison of k_{eff} values – water flooded cases

SCALE 6.3*	MCNP 6.3**	Δ (pcm)
0.96061	0.96260	199

*σ < 20 pcm ** σ < 30 pcm

TRISO-loaded VP-55 similarity analysis

TRISO Pitch (cm)	Container Pitch (cm)	Canister Fill Material	k_{eff}	ICSBEP $c_k > 0.8$
0.047	60	Water	1.1247	892
0.055	60	Water	1.1959	791
0.047	90	Air	1.0989	0
0.047	60	Air	1.0847	0
0.047	60	SiO2	1.0524	0



ORNL/TM-2024/3536

SCALE Analyses of Scenarios in the High-Temperature Gas-Cooled Reactor Fuel Cycle

Rabab Elzohery
Donny Hartanto
Friederike Bostelmann
William A. Wieselquist

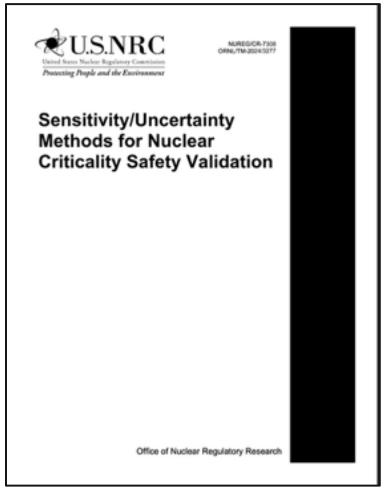
Approved for public release. Distribution is unlimited.

October 2024

Oak Ridge National Laboratory

Fuel salt-loaded VP-55 similarity analysis

Fuel salt-mix	k_{eff}	#EXP c_k>0.8
UF_4	1.172475	203
FLiBeZr	0.908385	52
FLiBe	0.791329	0
UCL3	1.084030	0
NaCl-UCl3	0.866155	0
KCl-UCl3	0.690583	0



U.S.NRC

NUREG/CR-7308
ORNL/TM-2024/3277

Sensitivity/Uncertainty Methods for Nuclear Criticality Safety Validation

Office of Nuclear Regulatory Research

[1]R. Elzohery, D. Hartanto, F. Bostelmann, W Wieselquist (2024), SCALE Analyses of Scenarios in the High-Temperature Gas-Cooled Reactor Fuel Cycle, ORNL/TM-2024/3536, Oak Ridge National Laboratory, Oak Ridge, TN.

[2]W. J. Marshall, T. Greene, A. M. Shaw, C. Celik, M. N. Dupont (2024), Sensitivity/Uncertainty Methods for Nuclear Criticality Safety Validation, NUREG/CR-7308;ORNL/TM-2024/3277, Oak Ridge National Laboratory, Oak Ridge, TN.

Oak Ridge National Laboratory

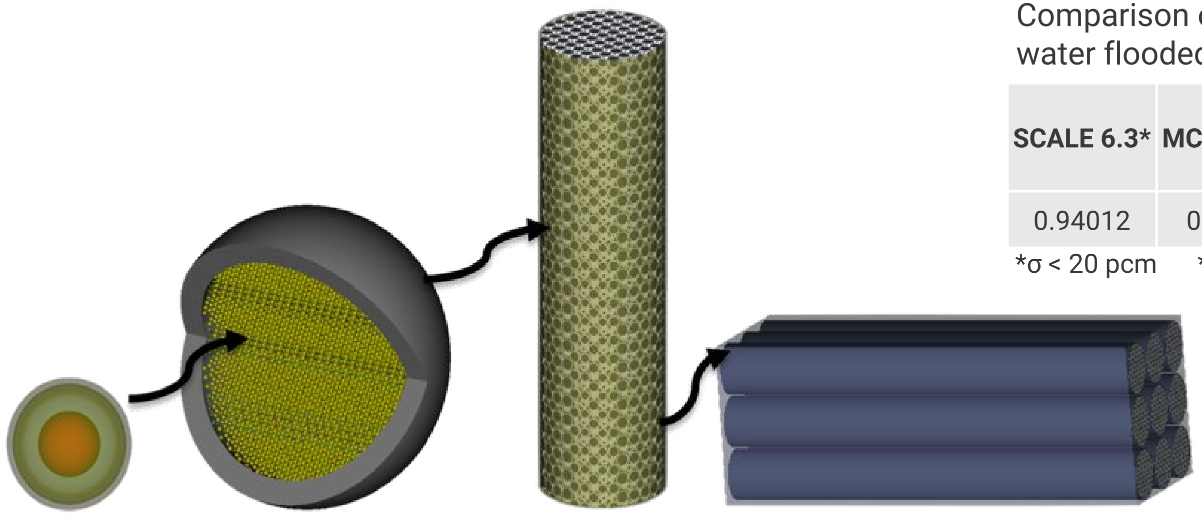
Available models: Pebble Tanker

A hypothetical model developed for transportation of fresh HALEU fuel pebbles.

PBMR and HERMES fuel pebbles with 19.75 wt% ^{235}U in TRISO particles

SCALE models are available

MCNP models are in review

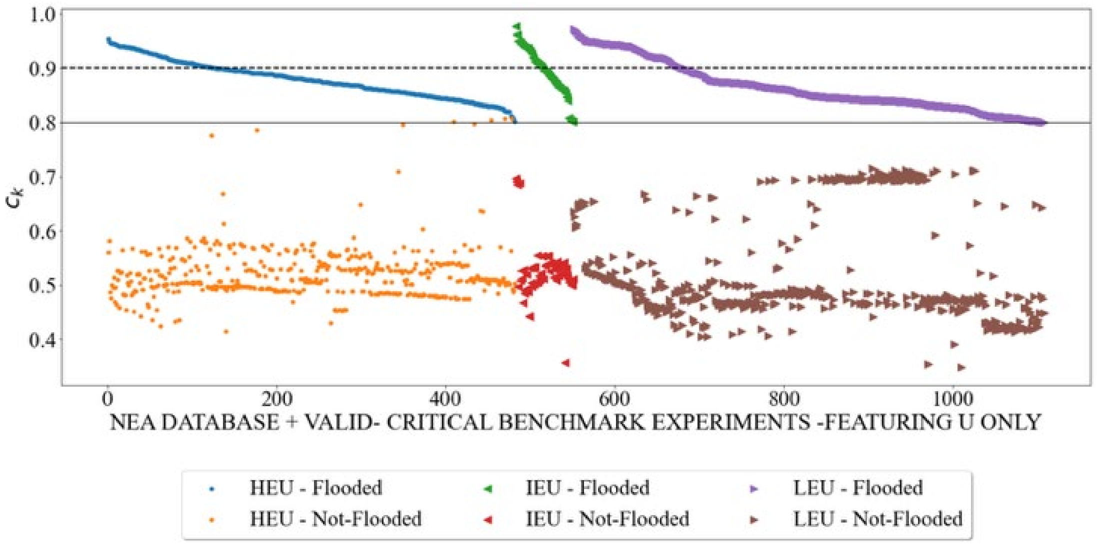


Pebble Tanker model with PBMR pebble[1]

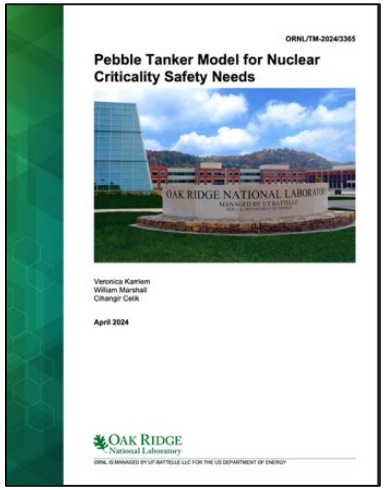
Comparison of k_{eff} values – water flooded cases

SCALE 6.3*	MCNP 6.3**	Δ (pcm)
0.94012	0.94168	156

*σ < 20 pcm ** σ < 30 pcm



Applicable critical benchmarks for Pebble Tanker model with PBMR pebble[1]



[1]*Veronica Karriem, William Marshall, and Cihangir Celik (2024), Pebble Tanker Model for Nuclear Criticality Safety Needs, ORNL/TM-2024/3365, Oak Ridge National Laboratory, Oak Ridge, TN.*

Available models: Fuel manufacturing

Generic models for manufacturing TRISO with HALUE-level enriched UO_2 and UCO

- Coating furnace
- Conversion furnace
- Densification furnace
- General Geometry

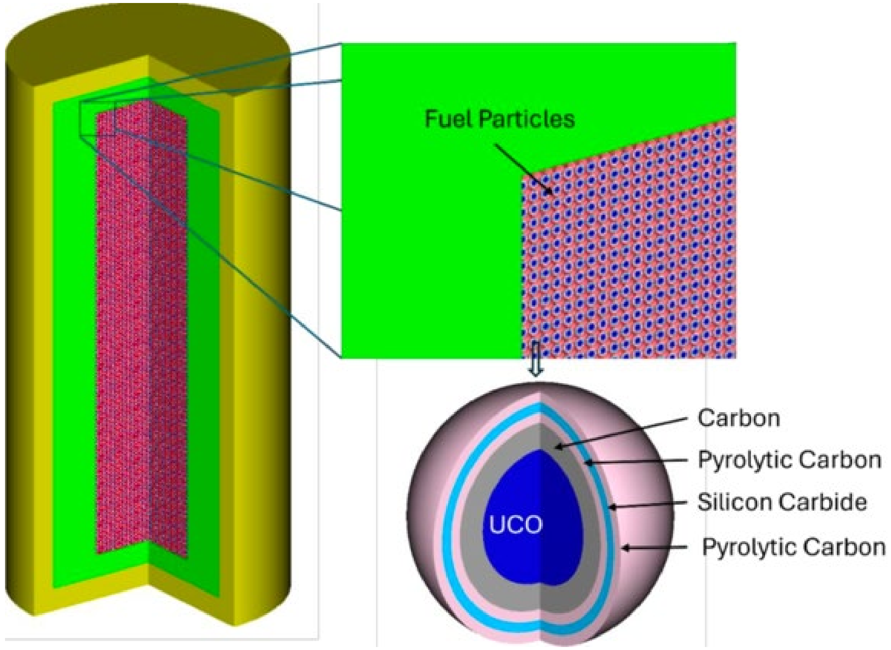
Reflected and moderated models for various metallic and hydride uranium fuel forms

- chloride salt, hydride, nitride, oxide
- metal, molybdenum alloy, zirconium alloy, zirconium hydride

Manufacturing process analyses

SCALE models are available

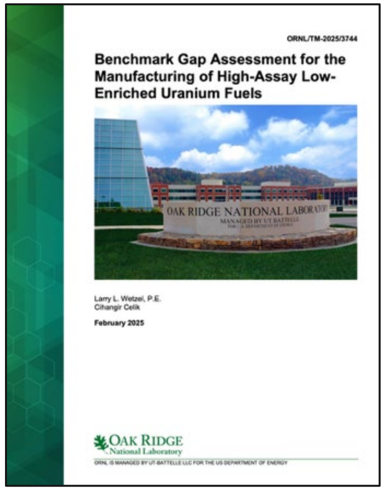
Main set of analyses for Workshop #2



A hypothetical furnace for UCO TRISO coating[1]

A subset of analyses with $c_k > 0.8$ for TRISO manufacturing[1]

Fuel Type	Description	ICSBEP $c_k > 0.8$
TRISO(19.75)	TRISO, Unreflected sphere, half pitch = 0.068 cm, R = 21.5 cm	942
TRISO(19.75)	TRISO, graphite-reflected sphere, half pitch = 0.065 cm, R = 11.8 cm	954
TRISO(19.75)	TRISO particles infinite graphite lattice	2
TRISO(19.75)	TRISO, water-reflected sphere, half pitch = 0.062 cm, R = 16.6 cm	951
TRISO(19.75)	TRISO particles in graphite, infinite lattice	88
TRISO(9.9)	TRISO particles in silicon carbide, infinite lattice	0
TRISO(19.75)	TRISO particles in silicon carbide, infinite lattice	0



[1] *Larry L. Wetzel and Cihangir Celik (2025), Benchmark Gap Assessment for the Manufacturing of High-Assay Low-Enriched Uranium Fuels, ORNL/TM-2025/3744, Oak Ridge National Laboratory, Oak Ridge, TN.*

Summary and future work

Available models

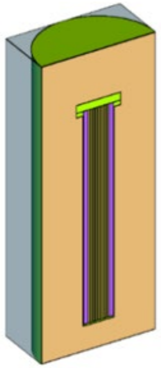
- Es-3100 with SFR fuel (SCALE, MCNP in review)
- Versa-Pac with fuel pebbles and TRISOs (SCALE, MCNP in review)
- Pebble Tanker with PBMR and HERMES fuel pebbles (SCALE, MCNP in review)
- Fuel manufacturing of TRISO, hydride/oxide/nitrite/metal fuels (SCALE)

Models under review

- GCMR with UCO TRISO (SCALE, analyses will be extended)
- MSR fuel salts within GBC-32 and Versa-Pac (SCALE)
- MAP-12 package with ABR fuel assemblies
- TN B1 package with ABTR fuel assemblies

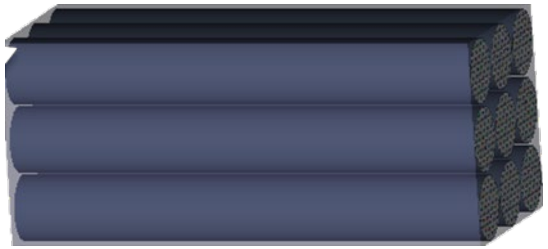
Server installation and setup in progress

- will be used for running models from the repository

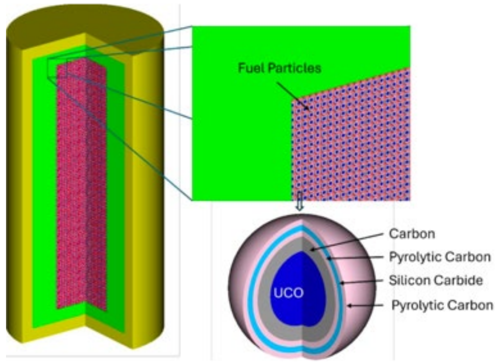


ES-3100

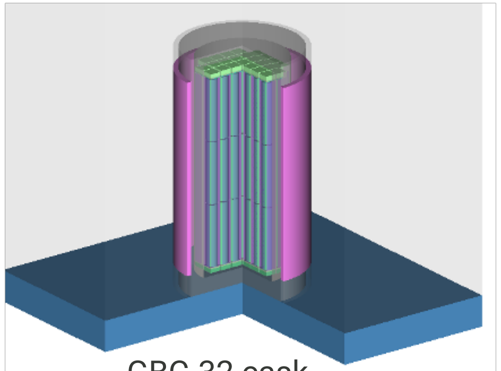
VP-55



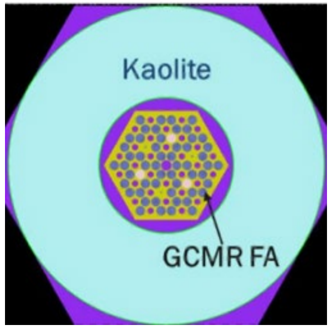
Pebble Tanker



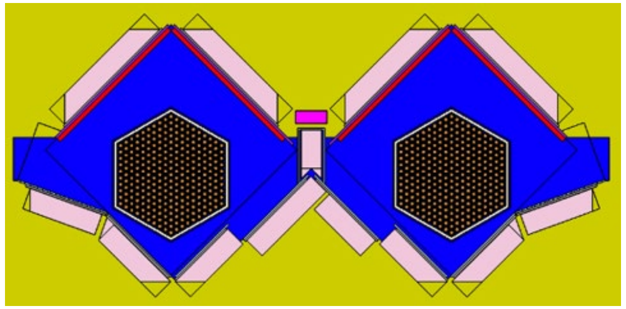
Furnace for UCO TRISO coating



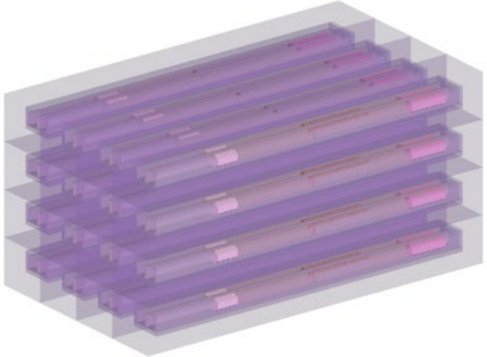
GBC-32 cask



GCMR assembly in ES-3100



MAP-12 package with ABR assembly



Array of TN B1 packages with ABTR assemblies

See Application Models Repository at

https://code.ornl.gov/dncsh/applications

Any specific applications of interest or interested in submitting models?
Reach out to us at dncsh@ornl.gov

OAK RIDGE
National Laboratory

August 27, 2025

DOE/NRC Collaboration for Criticality Benchmarks supporting HALEU-based Fuel Cycles

Workshop #2

William Wieselquist, ORNL

National Technical Director, DNCSH

Revisiting Call #1

Held a public workshop to collect information Feb. 29, 2024

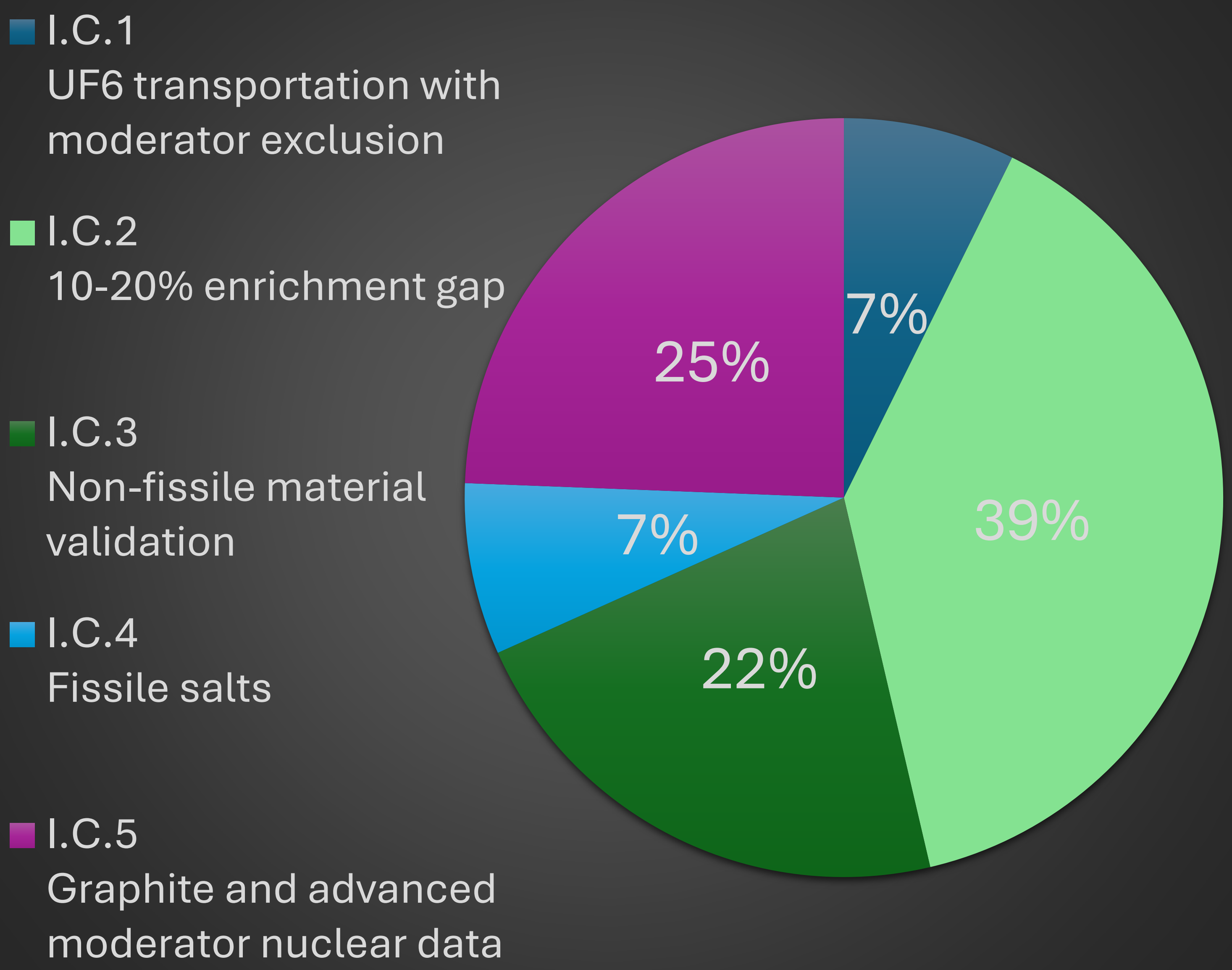
Resulted in a call for proposals

Received 30 proposals totaling $28M across 5 topic areas in June 2024

Funded **$17M** in proposals starting in FY25

- 9 to turn **existing experiments** into benchmarks
- 5 for **new experiments** with existing fuel
- 2 for new experiments **needing new HALEU fuel**

Submitted Proposals vs. Topic Area



Oak Ridge National Laboratory

Timeline and Focus Areas for Remaining Calls

- **Call #2: Facilities and Fabrication**
  - **Sept. 27, 2025** - Call document finalized
  - **Oct. 1** - Call document released
  - **Oct. 15** - Intent to propose
  - **Nov. 7** - Proposals due
  - **Dec. 12** - ~$6M in awards
- **Call #3: Microreactors**
  - **Early February 2026** –Workshop #3
  - **May 2026** – ~$6M in awards

- **Call #2 will include #1 topic areas**
  1. **Facility and fabrication gaps (new for Call #2)**
  2. UF6 transportation with moderator exclusion
  3. 10-20% enrichment gap
  4. Non-fissile material validation
  5. Fissile salts
  6. Graphite and advanced reactor moderator nuclear data
- **Call #3 will add microreactor areas**

Summary of Call #2

- Unfunded proposals **from Call #1 can re-submit**
- Proposals must focus on making **new experimental data for validation** **publicly** available
- **Unlikely to fund new fuel purchases** because of funding/time constraints
- Proposals using **recently acquired** fuels are **highly encouraged**
- **Coordinate with facility owner prior to submitting any proposals**
    - **NCERC (LANL) will receive new HALEU TRISO fuel in FY26**
    - SPRF/CX (SNL) will (hopefully) start to receive 20 wt% UO2 fuel in FY26
    - SPARC (INL) will have 5% RPI fuel rods for commissioning (FY28) and potentially HALEU UMo plates – assuming HST gets funding, will accept experiment proposals in Call #3
- Proposals must be **lab led**
- Priority will be given to proposals which **demonstrate similarity of the experiment to application models** representing validation gaps—LANL, LLNL, ORNL, PNNL, SNL all know exactly what this means

Some of the partners for this effort…

U.S. DEPARTMENT OF ENERGY
Office of Science

NCSP
NUCLEAR CRITICALITY SAFETY PROGRAM

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

Argonne NATIONAL LABORATORY

INL Idaho National Laboratory

Los Alamos NATIONAL LABORATORY

Pacific Northwest NATIONAL LABORATORY

Sandia National Laboratories

Brookhaven National Laboratory

Lawrence Livermore National Laboratory

OAK RIDGE National Laboratory

BWXT BWX Technologies, Inc.

Kairos Power

orano

Westinghouse

TerraPower

Berkeley UNIVERSITY OF CALIFORNIA

Idaho State University

UNIVERSITY OF MICHIGAN

S&T MISSOURI University of Science & Technology

Rensselaer

THE UNIVERSITY OF TENNESSEE KNOXVILLE

OAK RIDGE National Laboratory

DNCSH

DOE/NRC Criticality Safety for Commercial-Scale HALEU for Fuel Cycle and Transportation

For additional information:

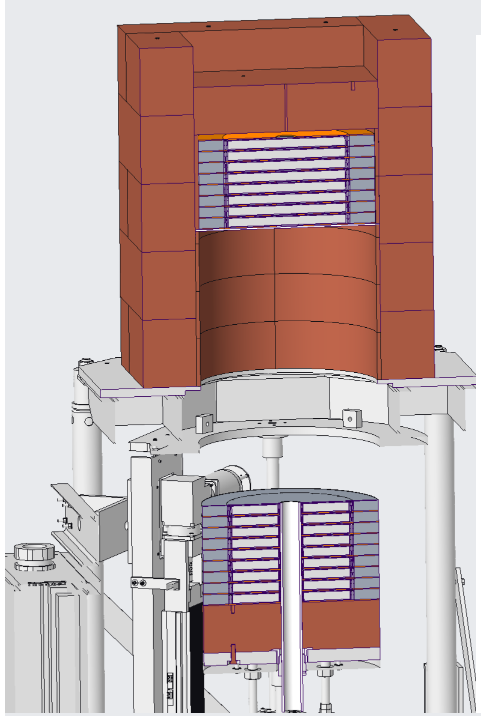
DNCSH@ornl.gov

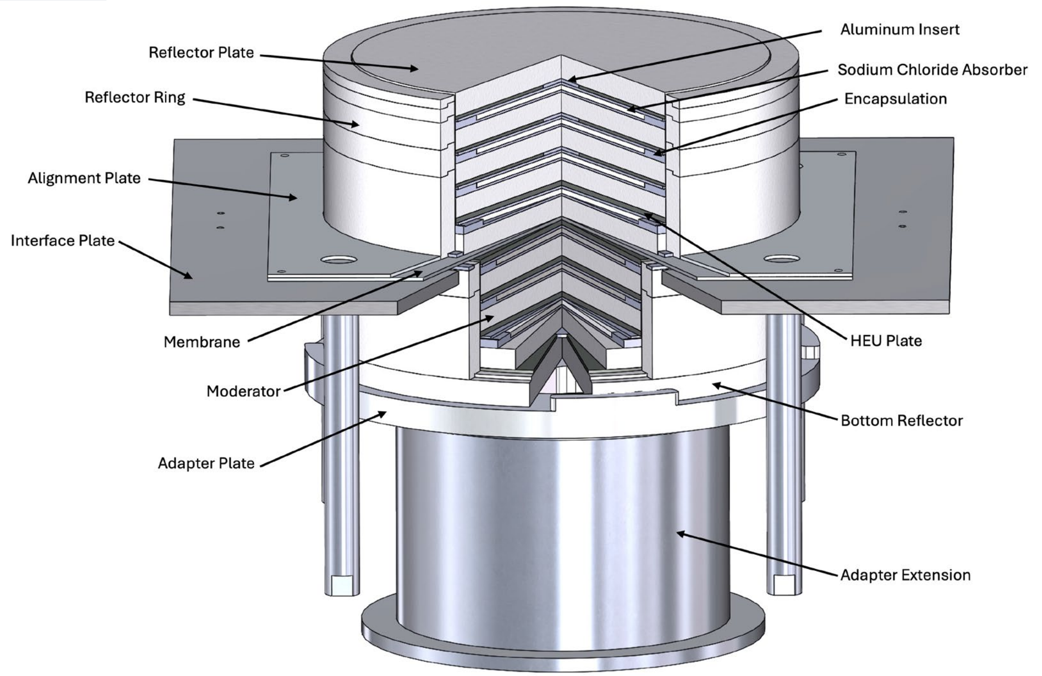
www.ornl.gov/dncsh

www.energy.gov/ne/criticality-benchmarking

Background Slides

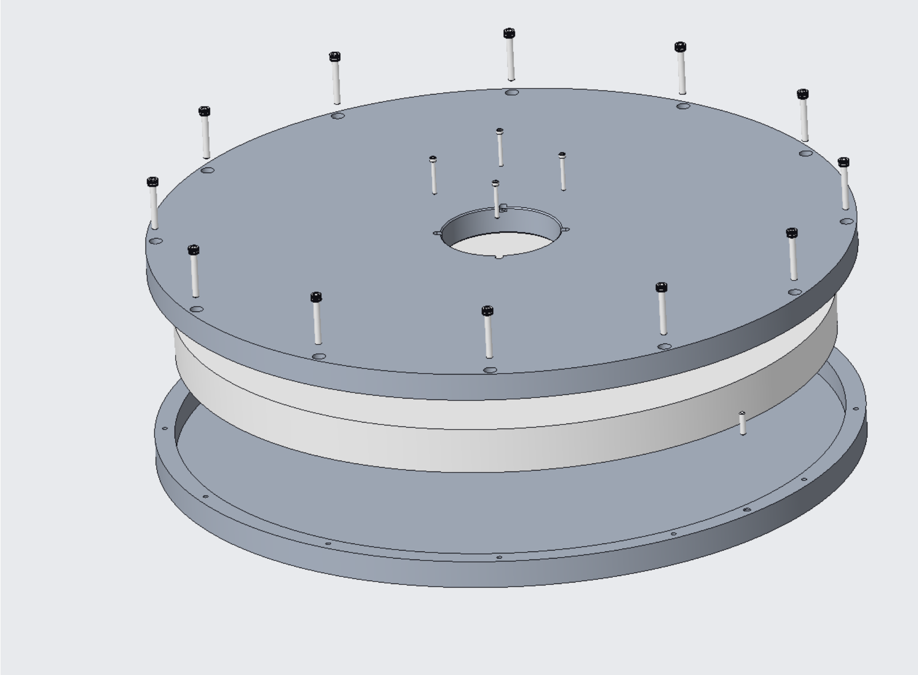
Thermal/Epithermal eXperiments (TEX) Additional Chlorine Configurations to Provide Validation for TerraPower's Molten Chloride Salt Fuel



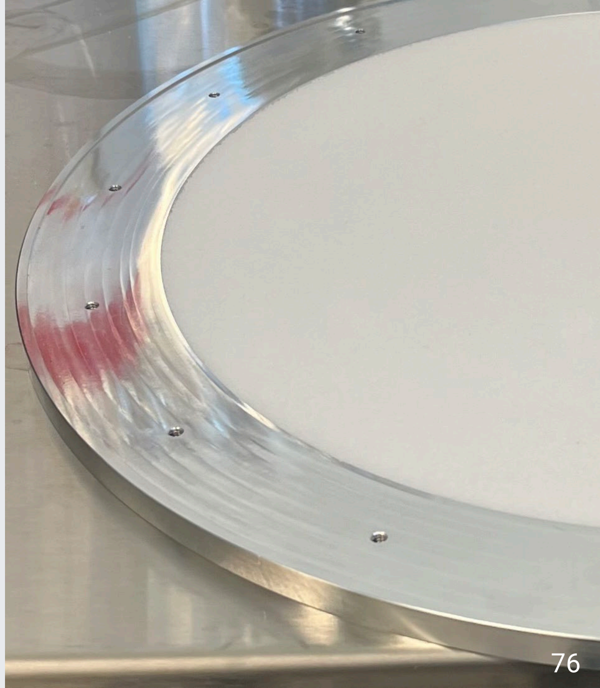


The new TEX experiments are designed to address the need for ^{35}Cl(n,*) validation for the molten chloride fuel fabrication process of TerraPower's MCRE and MCFR. Specifically, these experiments target the thermal ^{35}Cl(n,gamma) and the fast ^{35}Cl(n,p) cross sections, both of which are crucial for adequate criticality safety analyses for fuel fabrication, staging/loading, and transportation.

New Salt Plates



Previous Salt Plates



Lead Lab: LLNL
Partners: LANL and TerraPower

Oak Ridge National Laboratory

world nuclear news

Home · Energy & Environment · New Nuclear · Regulation & Safety · Nuclear Policies · Corporate · Uranium & Fuel

HOME / NEW NUCLEAR / LANL RESEARCHERS COMPLETE HALEU CRITICALITY EXPERIMENT

LANL researchers complete HALEU criticality experiment

Friday, 29 November 2024

The Deimos experiment at Los Alamos National Laboratory is the first criticality experiment using high assay low-enriched uranium fuel to be carried out in the USA in more than 20 years, and will help to develop public data and criticality benchmarks for the material.

NUCLEAR ENGINEERING INTERNATIONAL

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News

LANL conducts first HALEU fuel experiment in two decades

Staff Writer November 29, 2024

November 25, 2024

Nuclear SmartBrief

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TOP STORY

LANL performs 1st HALEU experiment in over 20 years

Los Alamos National Laboratory has conducted the first critical experiment using high-assay low-enriched uranium ceramic fuel in more than two decades. The Deimos experiment at the National Criticality Experiments Research Center tested TRISO particle fuel at room temperature and at more than 200 degrees Fahrenheit, generating critical safety data. The Department of Energy and the Nuclear Regulatory Commission are collaborating on the project. **Full Story:** Department of Energy (11/21)

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Headlines For You

TRU waste storage shrinks at Savannah River Site
22h ago

How can the U.S. make nuclear waste a nonissue?
Tue, Nov 26, 2024, 6:01AM

Senate committee advances NRC nominee Matthew Marzano
Mon, Nov 25, 2024, 2:06PM

RESEARCH & APPLICATIONS

LANL's Deimos—the first critical experiment with HALEU fuel in over 20 years

NuclearNews Daily ANS

In this issue: LANL's Deimos HALEU experiment, funding for Radiant supports microreactor testing in INL's DOME, Senate EPW Committee advances NRC nominee Matthew Marzano, and more.

National Lab Conducts First Critical Experiment Using HALEU-Based Fuel in Decades

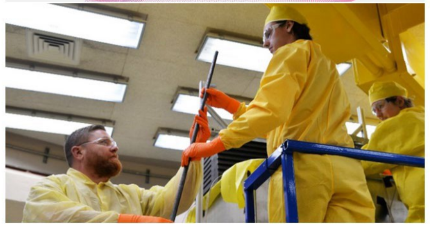
For the first time in more than two decades, Los Alamos National Laboratory researchers performed the nation's first critical experiment using a ceramic fuel required by some advanced reactor designs.

Office of Nuclear Energy

November, 21 2024

3 min

NUCLEAR MILESTONES



LDRD

CNPS Repack

- Arrange fuel into needed configuration for the experiment and future experiments
  - Secondary goal: Improve vault space usage
    - 6 old containers worth fit into 1 new container
  - Protects structural integrity of the fuel, as the dense packing makes it susceptible to chipping
  - Fuel will be loaded in these graphite “cups” for the experiment
    - One graphite cup is similar to one Jemima plate (or more closely, a fuel rod in SPRF/CX)
  - Allows for efficient loading of fuel during experiments
  - Fuel becomes more universally useful, namely for advanced reactor work

LDRD

Deimos: Advanced Reactor Testb

National Lab Conducts First Critical Experiment Using HALEU-Based Fuel in Decades

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Office of Nuclear Energy

November, 21 2024

3 min

NUCLEAR MILESTONES

November 25, 2024

Nuclear SmartBrief

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TOP STORY

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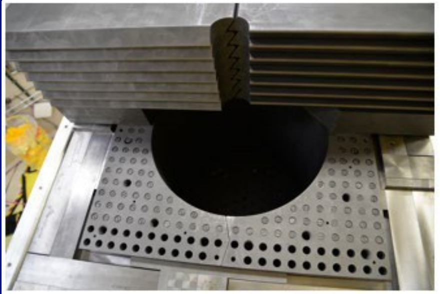
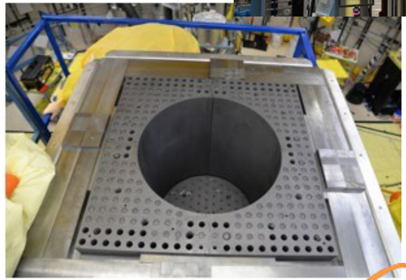
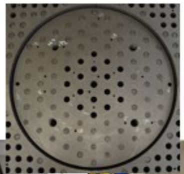
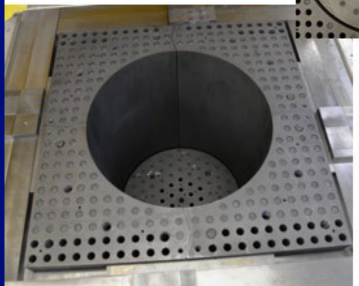
- First HALEU-TRISO fueled critical experiment in 4
  - November 2024 on the Comet critical assembly at NC
  - 5 unique configurations
- Used *existing* fuel in NCERC inventory
  - Full characterization of fuel completed

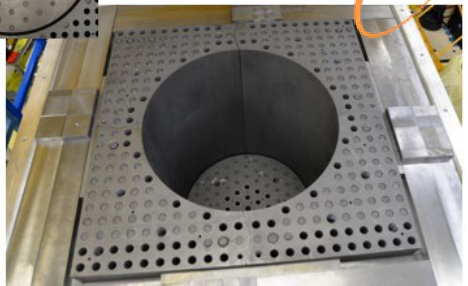
COMET

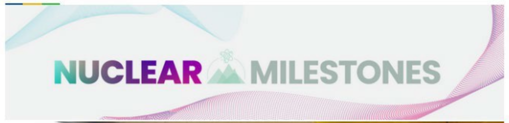






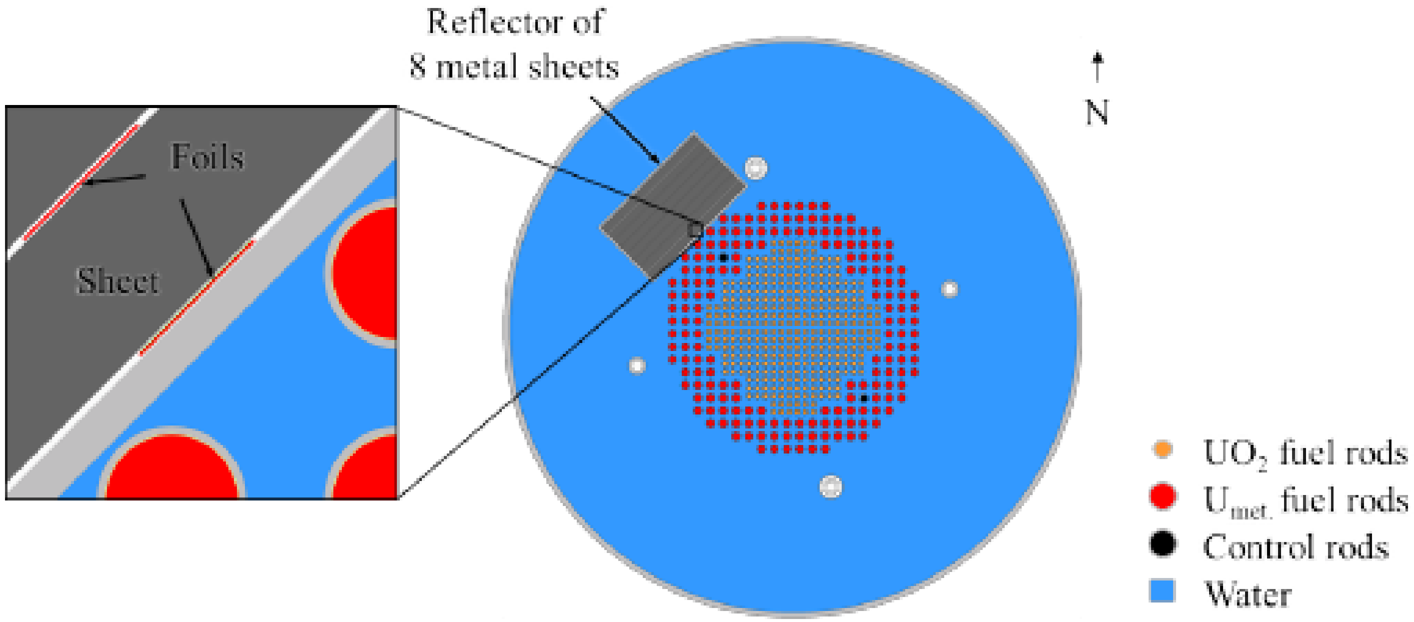






NCERC
NATIONAL CRITICALITY EXPERIMENTS RESEARCH CENTER

PETALE Benchmark



Top view of the Serpent2 model of CROCUS with the metal reflector set on the north-west corner of the reactor core. Activation foil dosimeters were embedded at various depths in the metal reflector.

- Experiment completed in 2020 by the Ecole Polytechnique Federale de Lausanne (EPFL) in Switzerland
- CROCUS is a well characterized zero power (100 W) teaching reactor with an existing benchmark
- Four configurations were measured, using large reflectors of 304L stainless steel and its components (iron, nickel, and chromium)
- Criticality measurements as well as neutron transmission via activation foils at various depths in the reflector blocks
- Lead Lab: LLNL
- Partners: University of California, Berkeley and EPFL

Evaluation and Benchmark Development of Reactor Critical Experiments of the RPI Reactor Critical Facility with Noteworthy Non-fissile SS Element Sensitivity

The Rensselaer Polytechnic Institute (RPI) Reactor Critical Facility (RCF) is a pool-type reactor fueled with SPERT-type 4.807 wt% enriched UO_2, stainless-steel clad fuel rods.

Evaluations of critical experiments involving a stainless-steel pipe placed at the center of the core between 2016 and 2018 will be submitted for inclusion in the ICSBEP handbook.

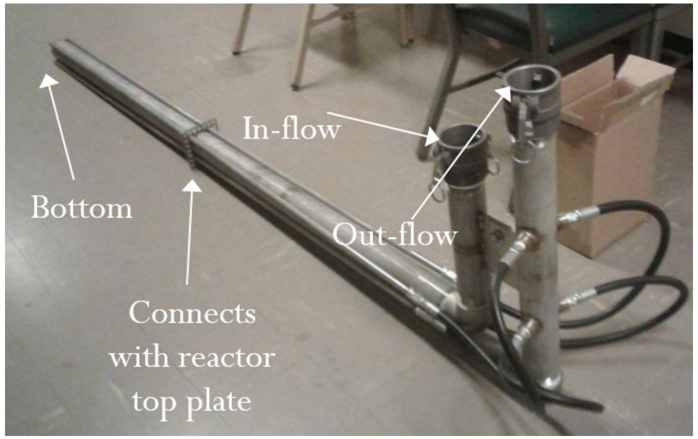
Lead Lab: Oak Ridge National Laboratory

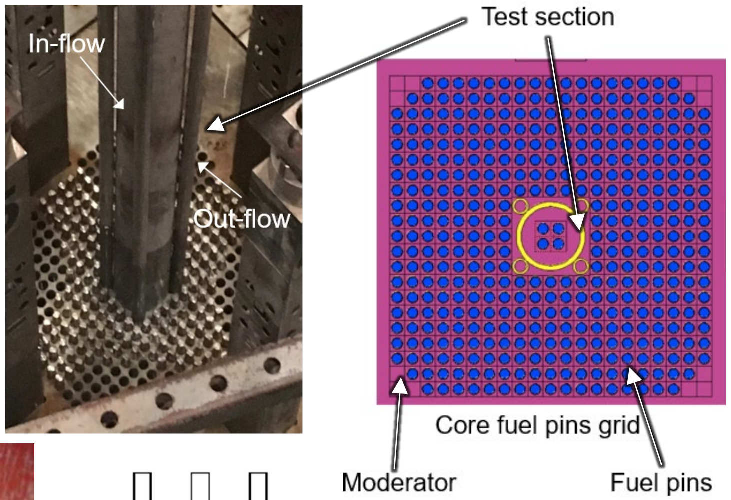
Partners: Rensselaer Polytechnic Institute, Idaho National Laboratory

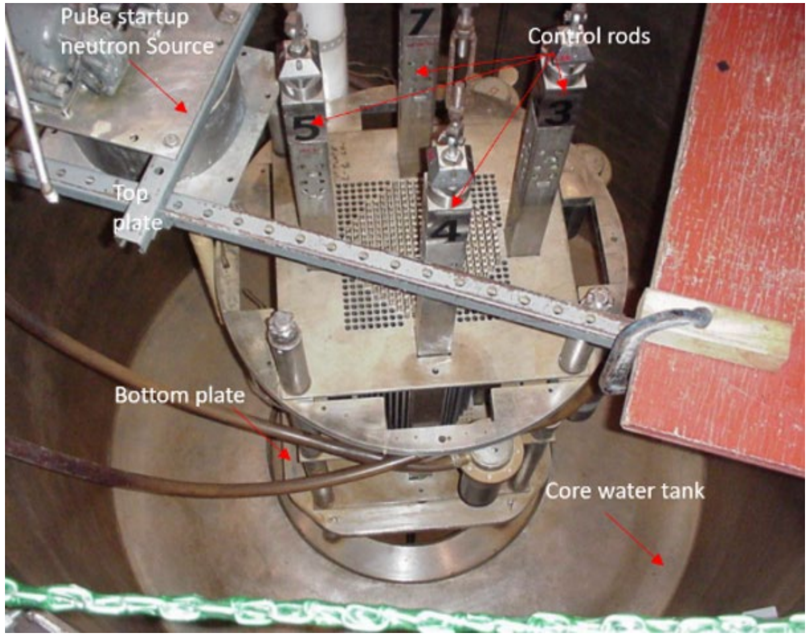
DNCSH

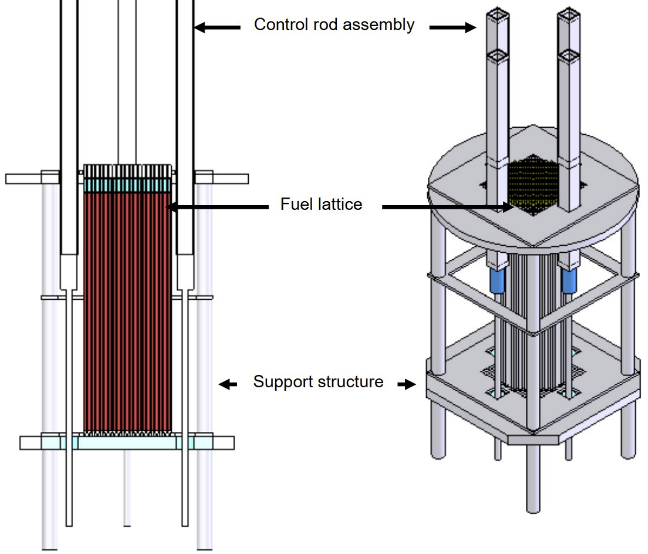
DOE/NRC Criticality Safety for Commercial-Scale HALEU for Fuel Cycle and Transportation

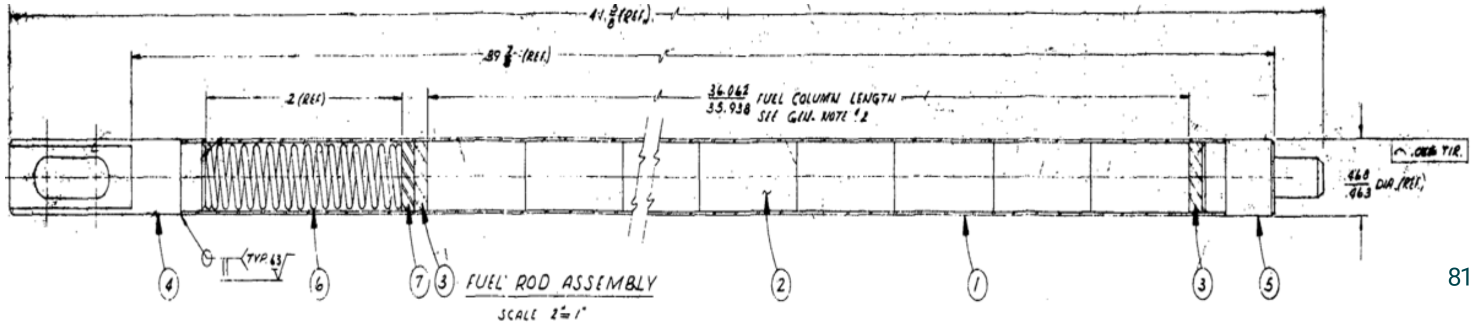
Oak Ridge National Laboratory · Rensselaer · INL Idaho National Laboratory











Evaluation of Critical Configurations of the Missouri S&T Reactor

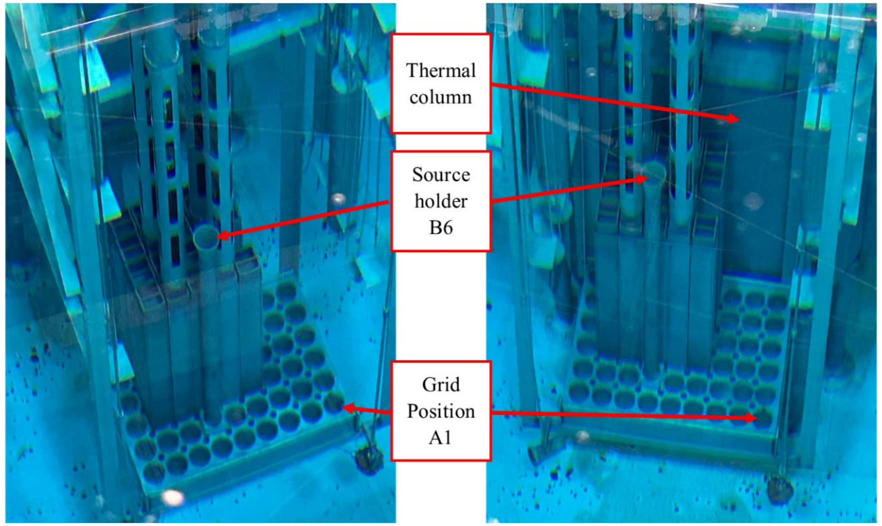
Missouri S&T Reactor (MSTR) is a light-water moderated, 200 kW pool-type research reactor fueled with U_3Si_2-Al fuel with a nominal enrichment of 19.75 wt% ^{235}U. The facility license and design allows for flexible core configurations/loadings.

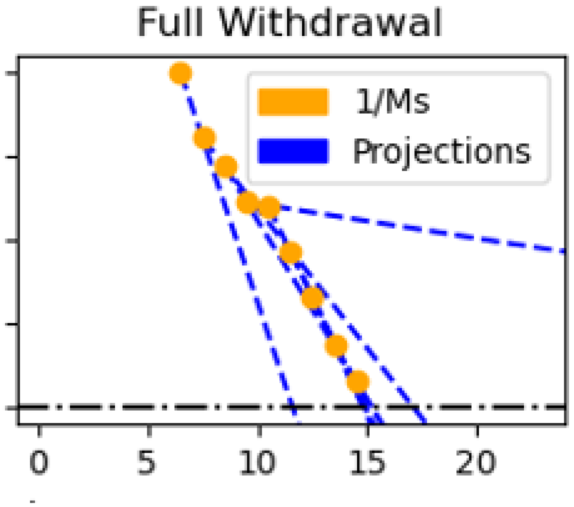
MSTR can be operated in a fully light-water reflected mode, or adjacent to a graphite thermal column and with graphite-lined experimental facilities.

Two evaluations of critical operations of the MSTR will be submitted for inclusion in the ICSBEP handbook.

Lead Lab: ORNL

Partners: Missouri S&T





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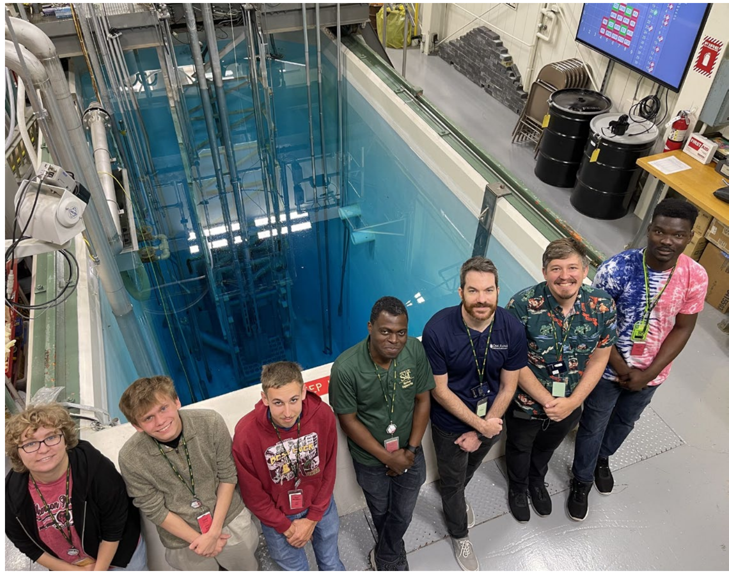
A									
B						S			
C				F6	C5	F18	C3	F4	
D					F16	IF1	F11	F9	
E				HR1	C2	F17	C1	F10	
F				F5	F15	F14	F13	F12	



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Evaluation of SLOWPOKE-2 Refuel Measurements

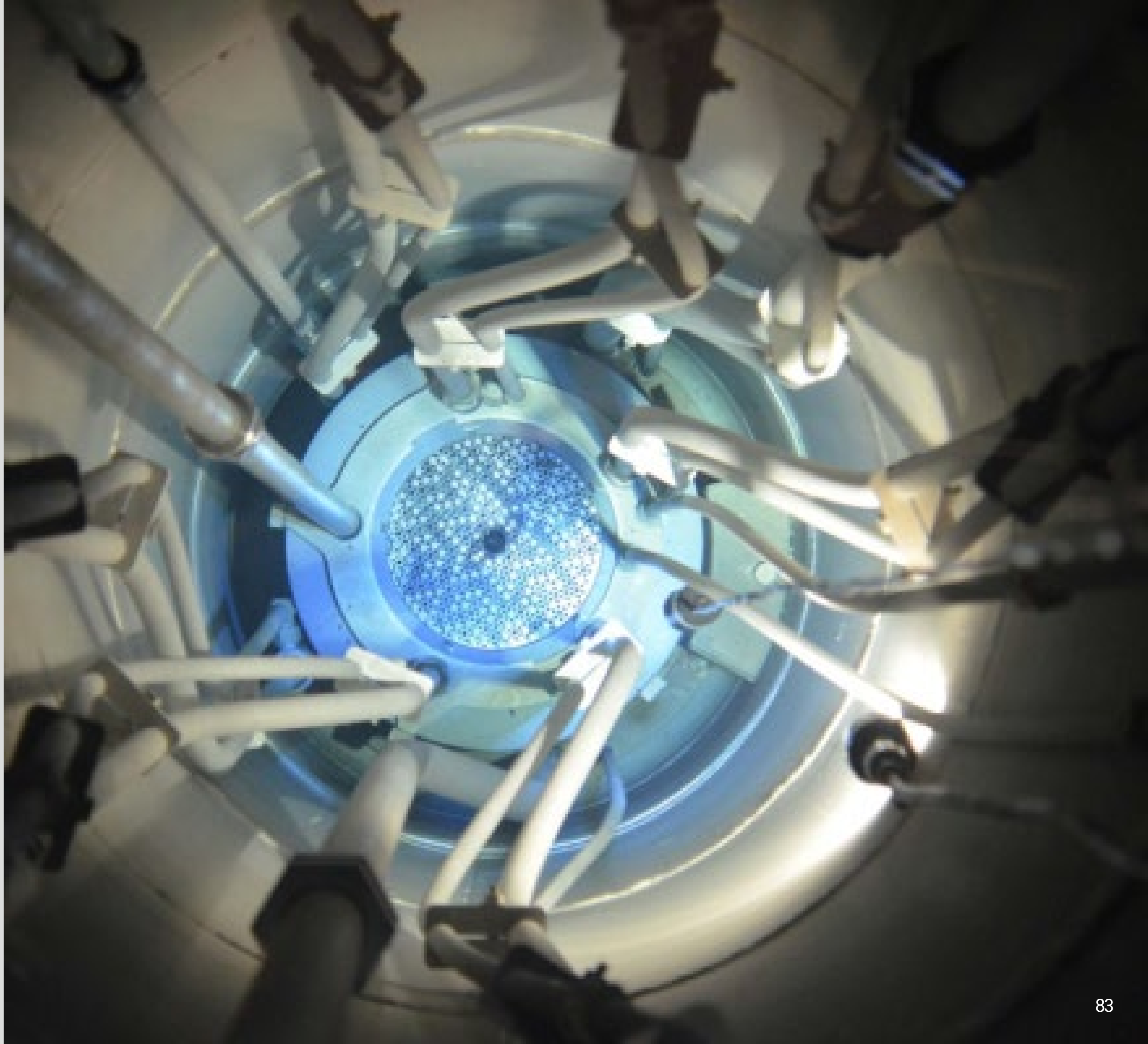
Existing data on criticality measurements taken at the Safe LOW POwer (K) critical Experiment (SLOWPOKE-2) reactor at the Royal Military College of Canada will be evaluated for a benchmark to be included in the ICSBEP handbook.

In 2021, the entire reactor core was removed, replaced with a core of fresh HALEU (19.75 wt%) uranium dioxide fuel pins, and taken to criticality at very low power.

Lead Lab: ORNL

Partners: Canadian Nuclear Laboratories

Oak Ridge National Laboratory



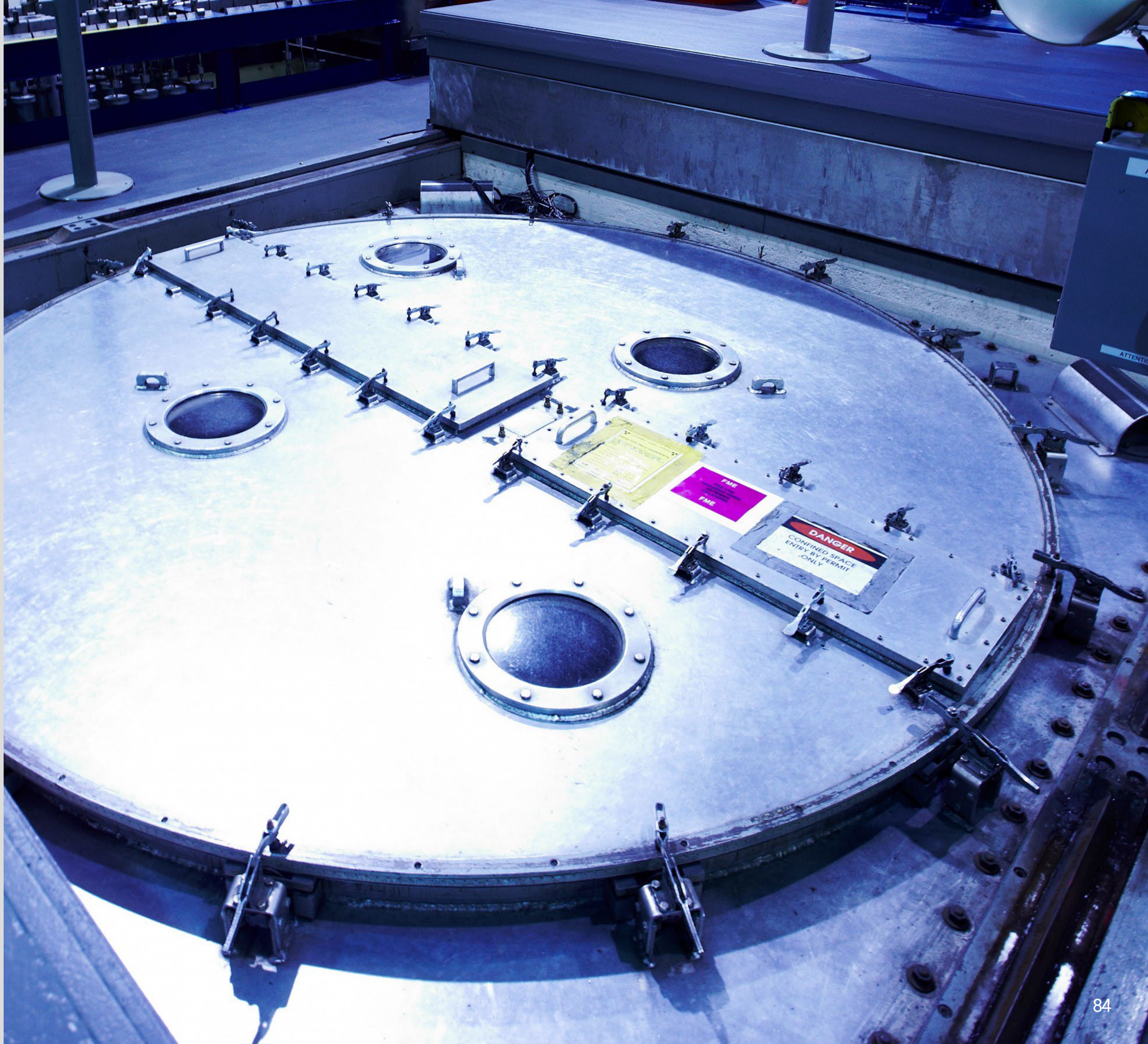
ZED-2 Measurements with In-Core Absorbers

New criticality measurements with large samples of non-fissile structural materials (e.g. Ni, Ta, Mo, Fe) will be performed in the Zero Energy Deuterium (ZED-2) Reactor in Chalk River to be submitted for inclusion in the ICSBEP handbook, using fuels already available in the ZED-2 inventory.

The absorbers will be exposed to a thermal neutron spectrum and their sensitivity will be maximized during the experimental design phase to increase their use for validation.

Lead Lab: ORNL

Partners: Canadian Nuclear Laboratories



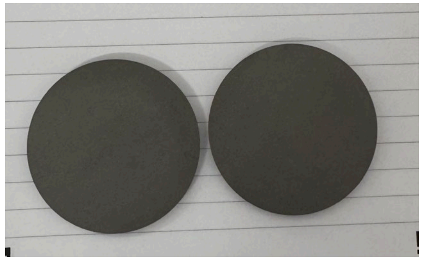
Temperature Dependent Transmission and S(alpha, beta) Measurements of Advanced Nuclear Moderators

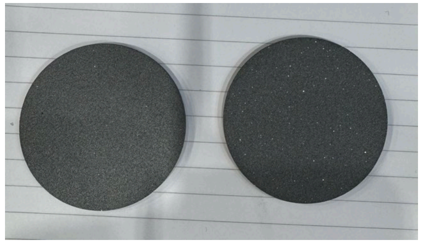
Differential INS measurements have been performed at the Spallation Neutron Source at Oak Ridge National Laboratory, while transmission measurements are being performed at the ISIS Neutron and Muon Source in the UK, as well as at the Spallation Neutron Source.

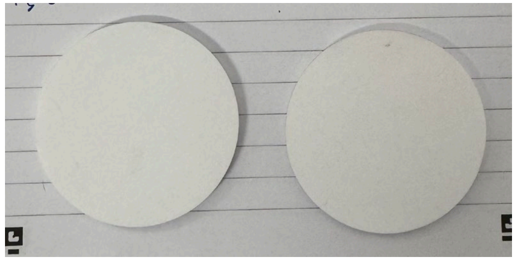
Machine-learned potentials have also been developed to guide temperature-dependent TSL evaluations for various advanced moderator materials.

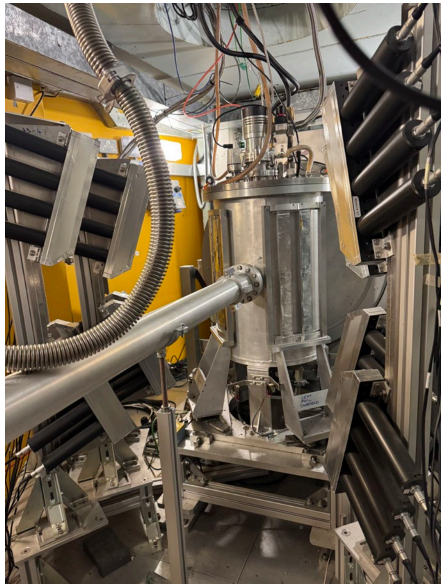
Lead Lab: ORNL

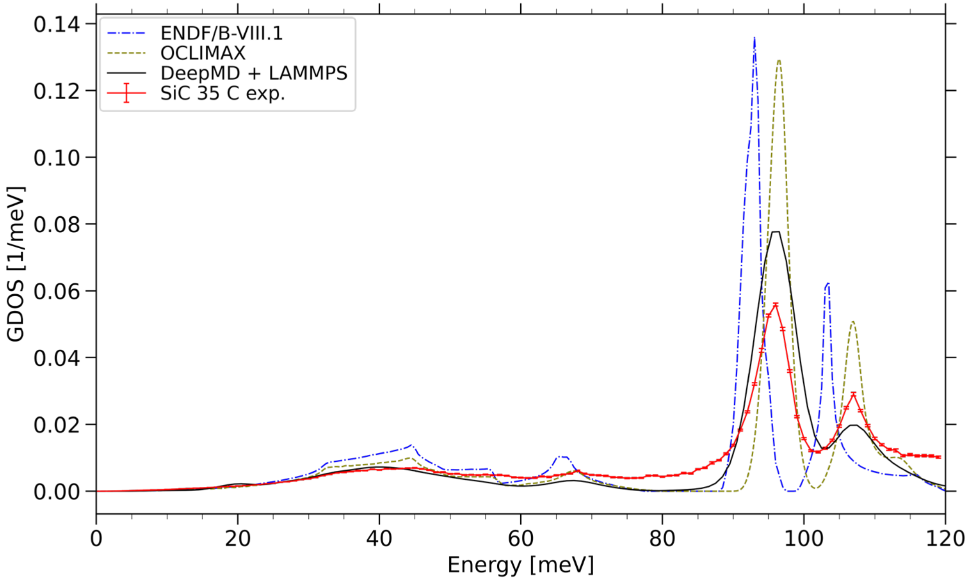
Partners: X-energy

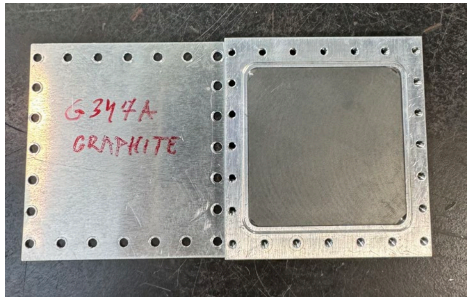












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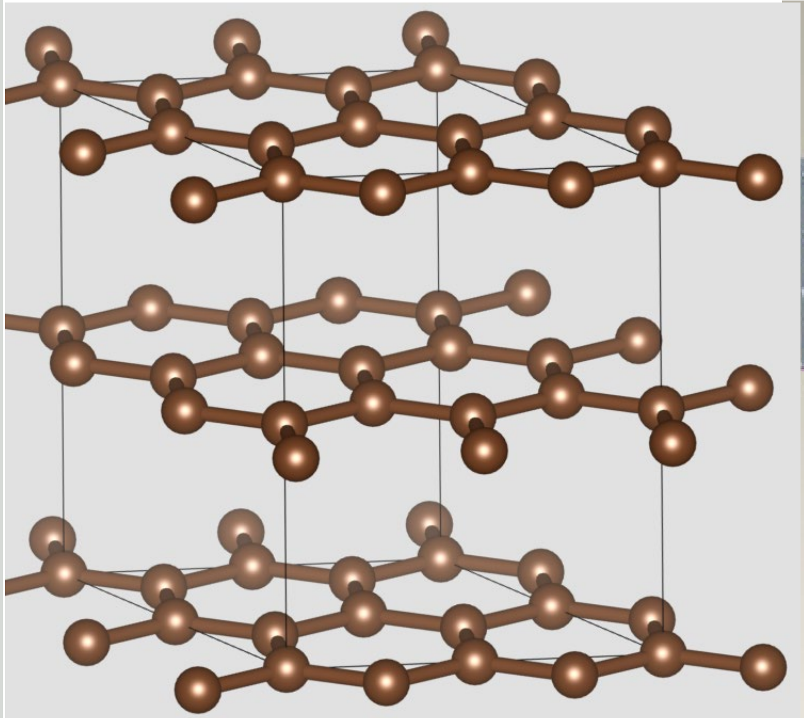
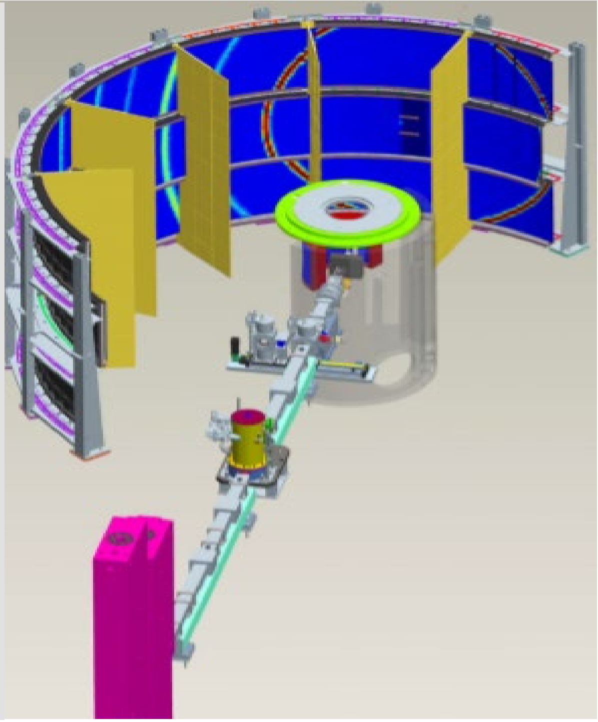
High-Temperature Graphite Double Differential Scattering Cross Sections: Measurement, Evaluation, and Validation

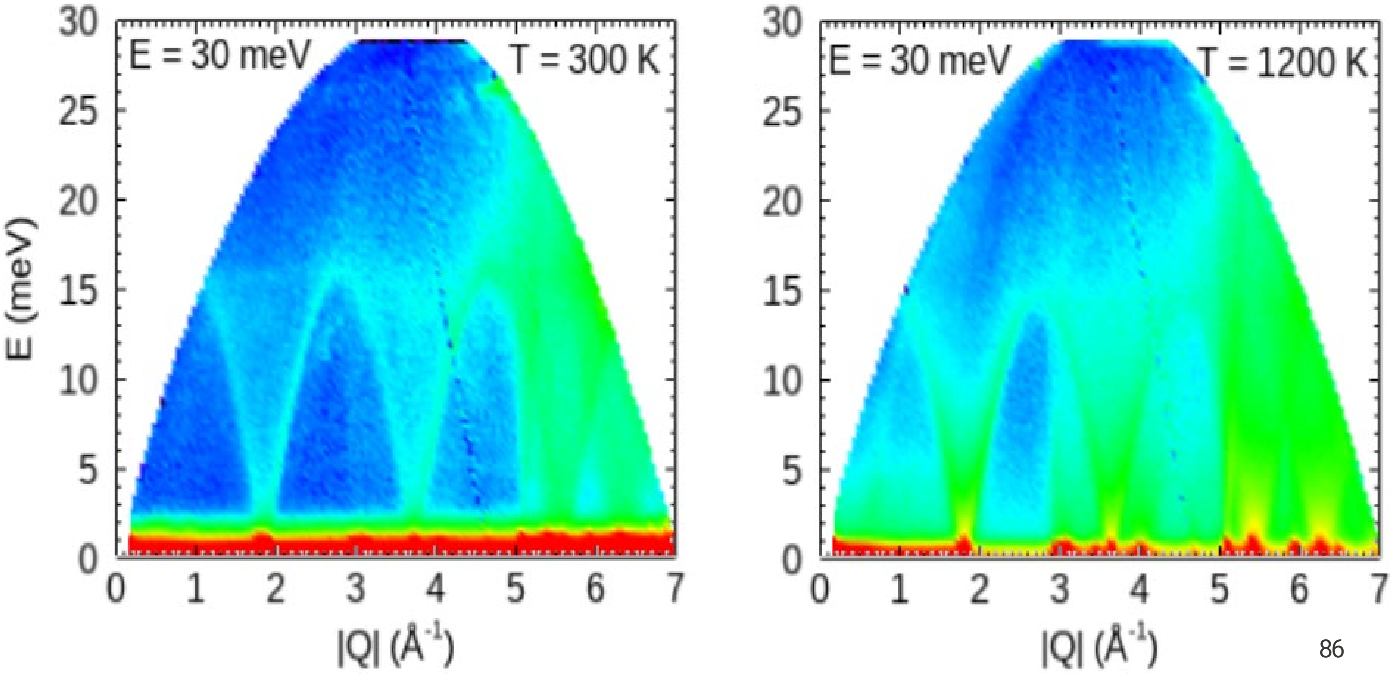
Inelastic neutron scattering technique utilizing ARCS instrument at SNS/ORNL is used to study thermal neutron interactions with the atomic vibrations of carbon atoms in graphite at different temperatures between 10 and 1200 K

Lead Lab: ORNL

Partner(s): Yarmouk University-Jordan







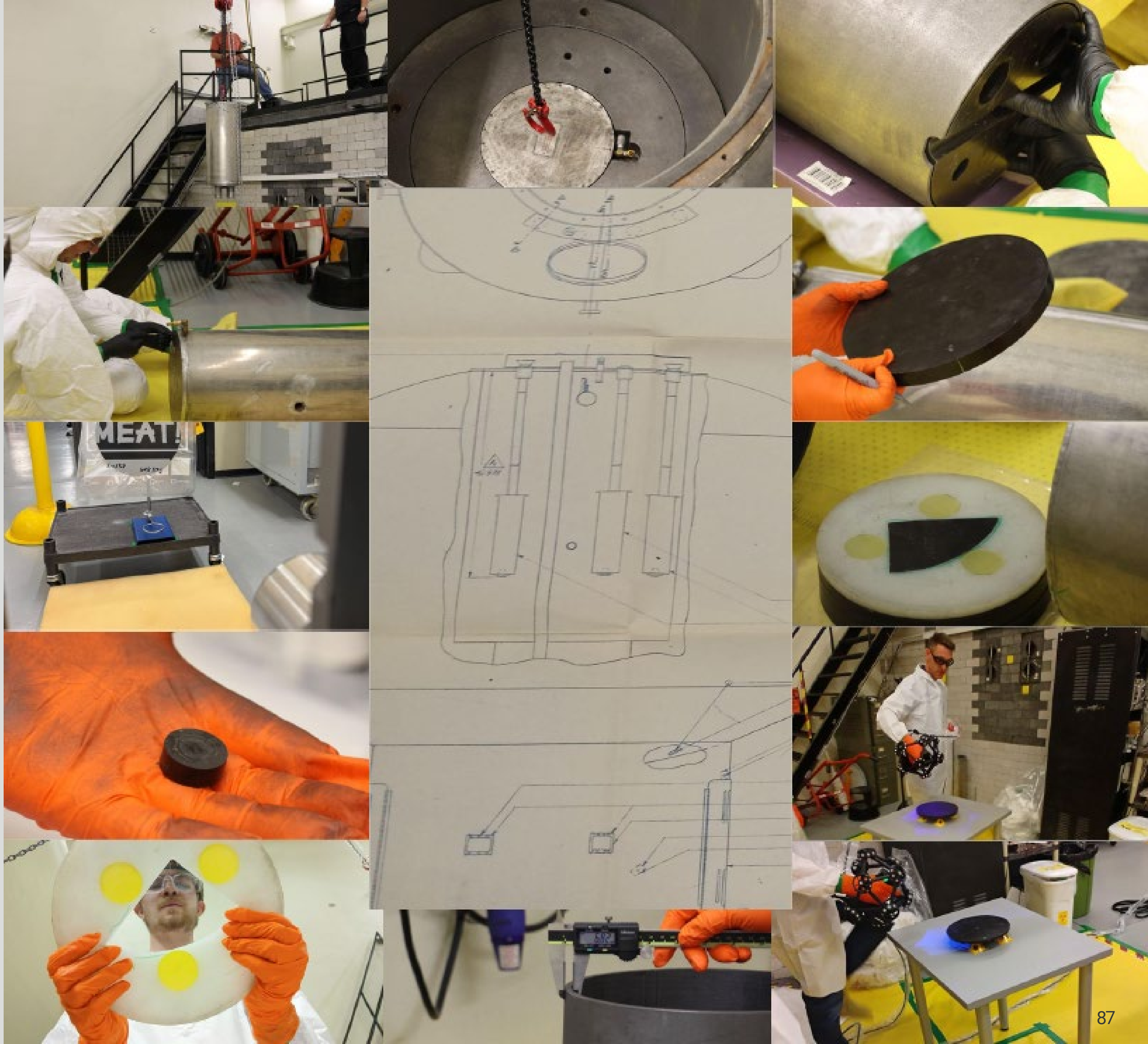
Characterization of ISU’s AGN-201 Reactor for Qualification as an ICSBEP Benchmark

The AGN-201 reactor has a graphite reflected core comprised of a fuel system with UO2 enriched to approximately 20 wt% U-235 that is matrixed in polyethylene.

Additional configurations with non-fissile structural materials are also planned.

Lead Lab: ORNL

Partners: GE Vernova, Idaho State University



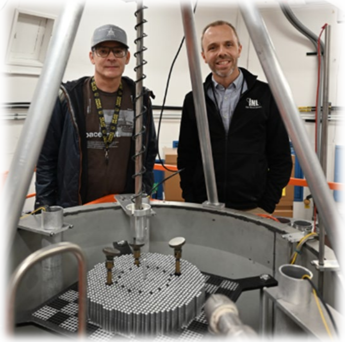
OAK RIDGE
National Laboratory

Bottom Line Up Front

- Large-scale critical experiments are crucially needed, the United States (and western world) has lost this capability
  - Support licensing, deployment, and optimization of LEU+ fuel current fleet and HALEU-fueled advanced reactors
  - Strong industrial support, direct relevance to executive orders
  - Enhance criticality safety (especially intermediate neutron energy benchmark experiments)
- DNCSH supported INL to find location and develop plan to deploy a multipurpose split table type “zero-power reactor”
  - Team evaluated user needs, developed requirements, and assessed over 20 candidate INL facilities
  - PBF-613 (former SPERT-IV building) uncontested winner for HALEU critical experiments facility
  - System Physics Advanced Reactor Critical facility (SPARC)
- Early project planning estimates
  - Simple system, historic precedents, existing building, expertise continuity with smaller machines at NCERC & SPRF/CX ⟶ SPARC has low risk profile (compared to other reactor projects)
  - Point estimate $34M and 3 ½ years to complete (uncertainty range $30-45M, 3-5 years)
  - INL is standing by for an earnest decision to move forward and make SPARC one their highest priority reactor projects
    - We have direct support from INL management through to LD.

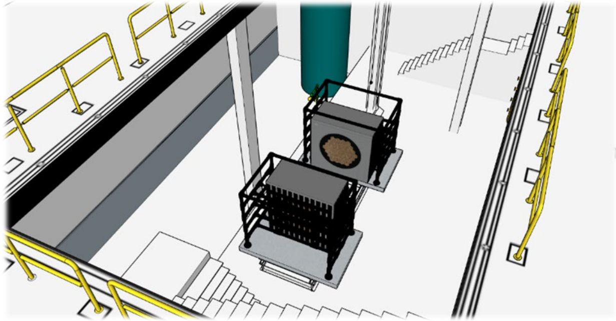
Historic ZPPR split table (now gone)

Project Kickoff at NCERC (LANL)



Technical tour of SPRF/CX (Sandia)

PBF-613

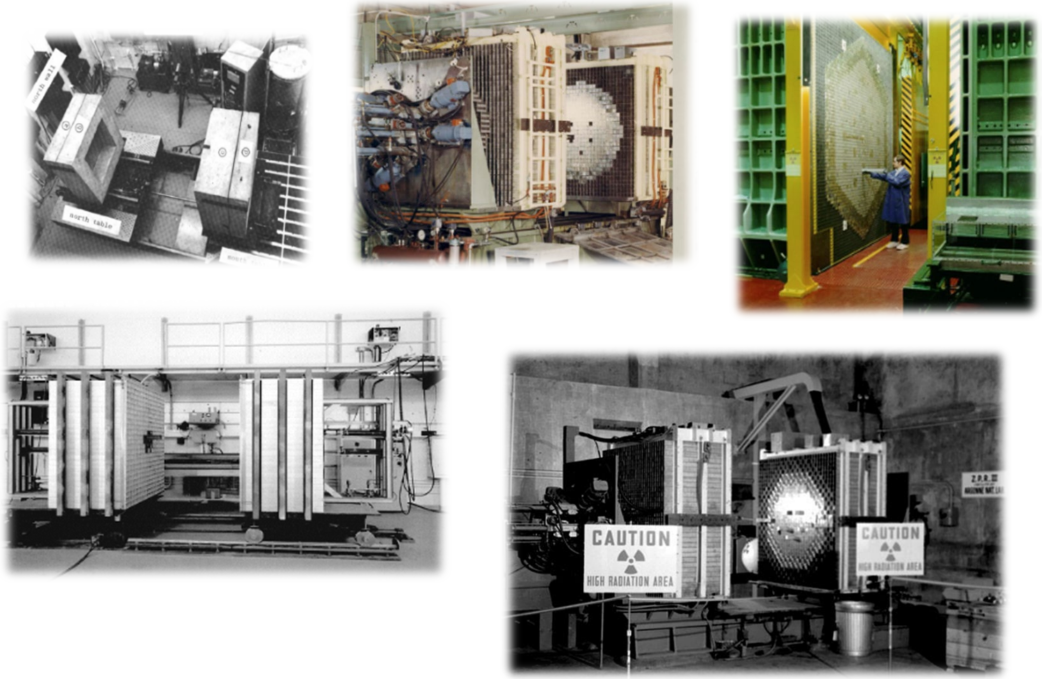


Rendering of Split Table in PBF-613

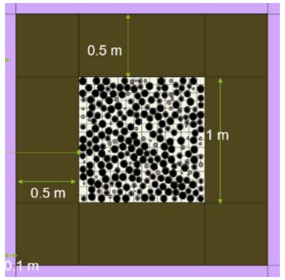
Advanced Fuels Campaign

What's an HST, and Why is One Needed?

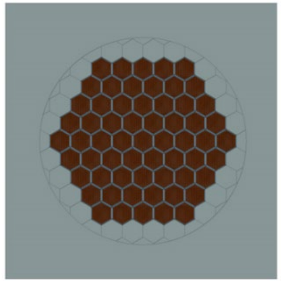
- The Swiss army knife of zero power reactors
  - Horizontal arrangement enables motions of massive core sections
  - Stack fuels and materials in two separate and subcritical sections
    Evacuate the assembly area and operate the reactor
  - Close gap to achieve criticality, power low (~10W), run time short
  - No heat removal system, flux too low to activate materials, assembly area not a radiation area when reactor is off
  - Gap separates to shutdown (including SCRAM or loss of power)
- Crucially needed to mockup large critical systems
  - "Large and leaky" intermediate neutron energy tests for criticality safety data gaps
  - Represent reactor cores and criticality safety scenarios for advanced reactors
  - Large enough (2m cube, 24 MT) to support various needs and decades of experimental programs

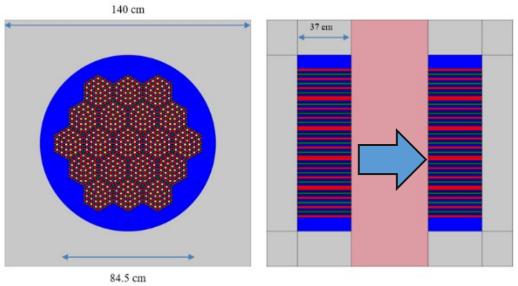


Historic Split Tables in the US (all now decommissioned)

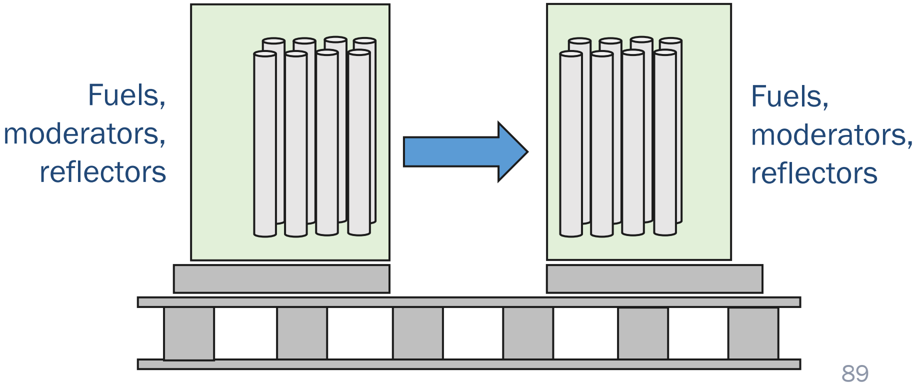


TRISO Pebble Bed in Graphite Reflector

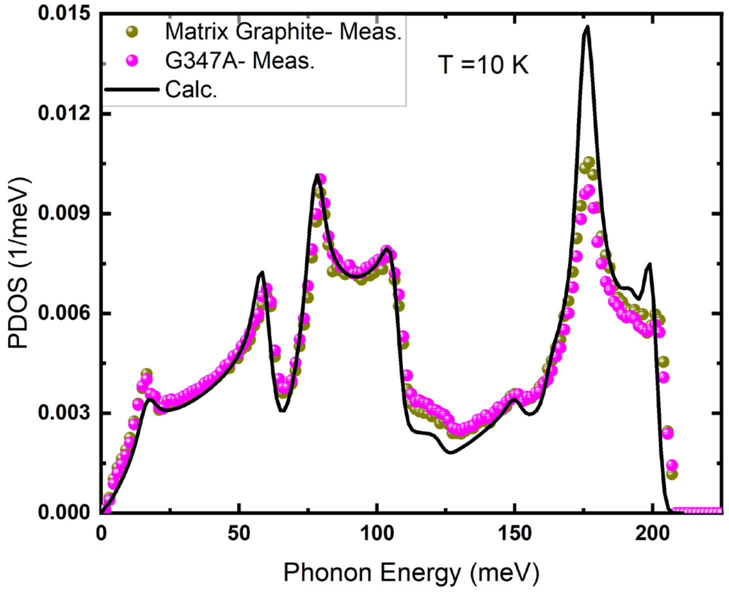

U-10Zr Metallic Fuel in SST Reflector, Fast Spectrum



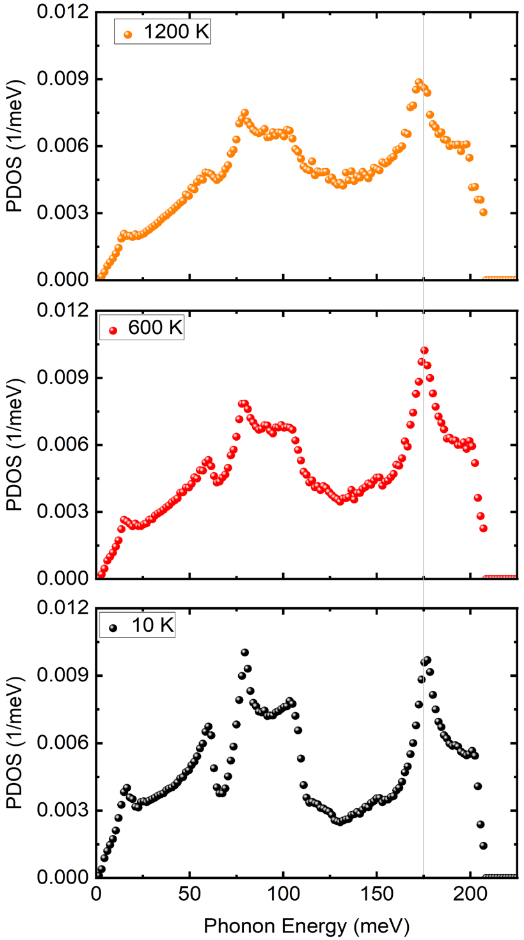
Metal Hydride Heat Pipe Microreactor



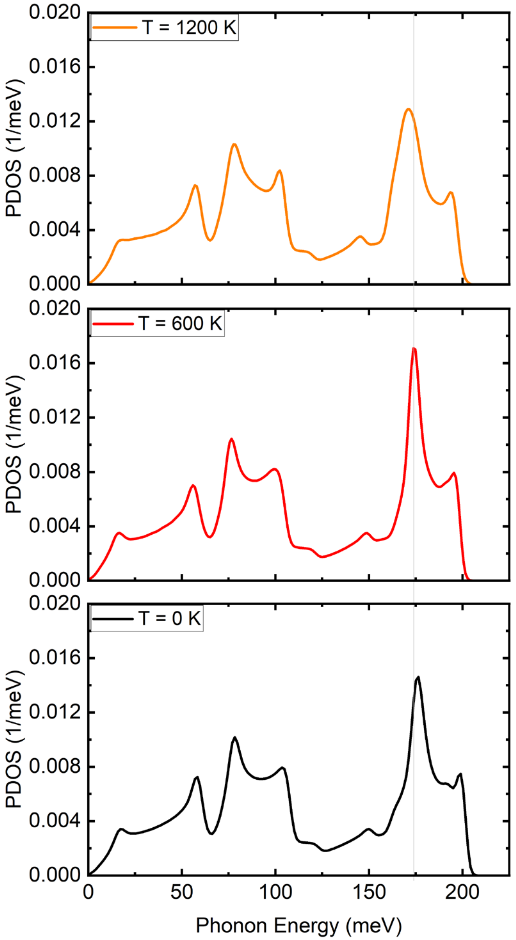
Graphite

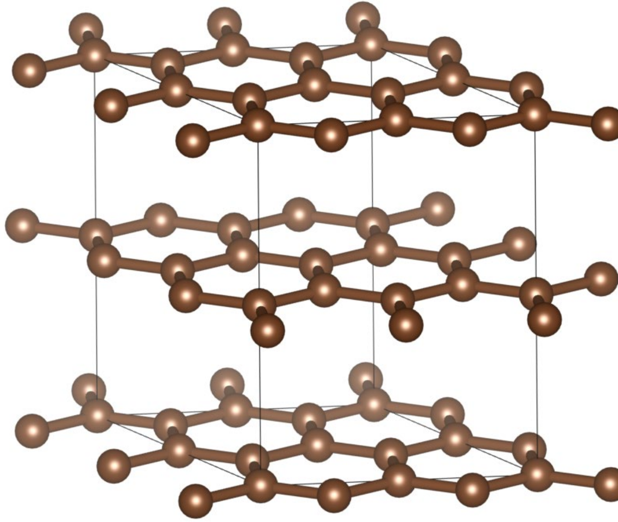


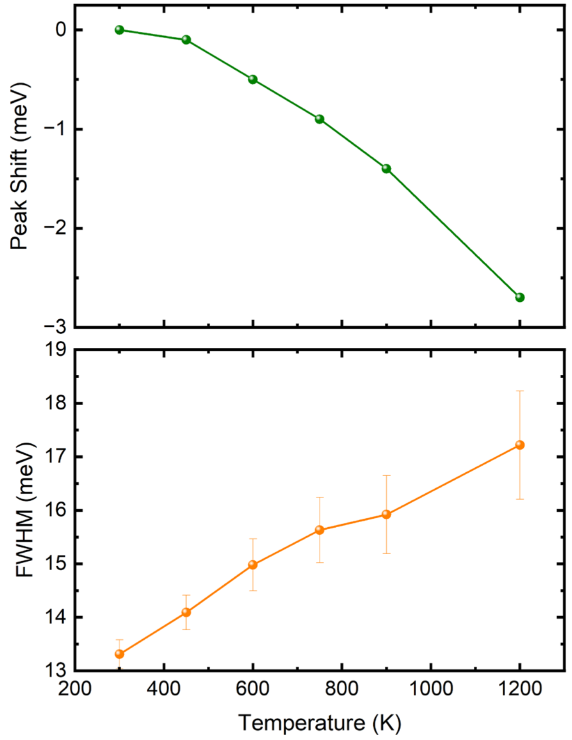
ARCS/SNS Measurements



AILD Calculations







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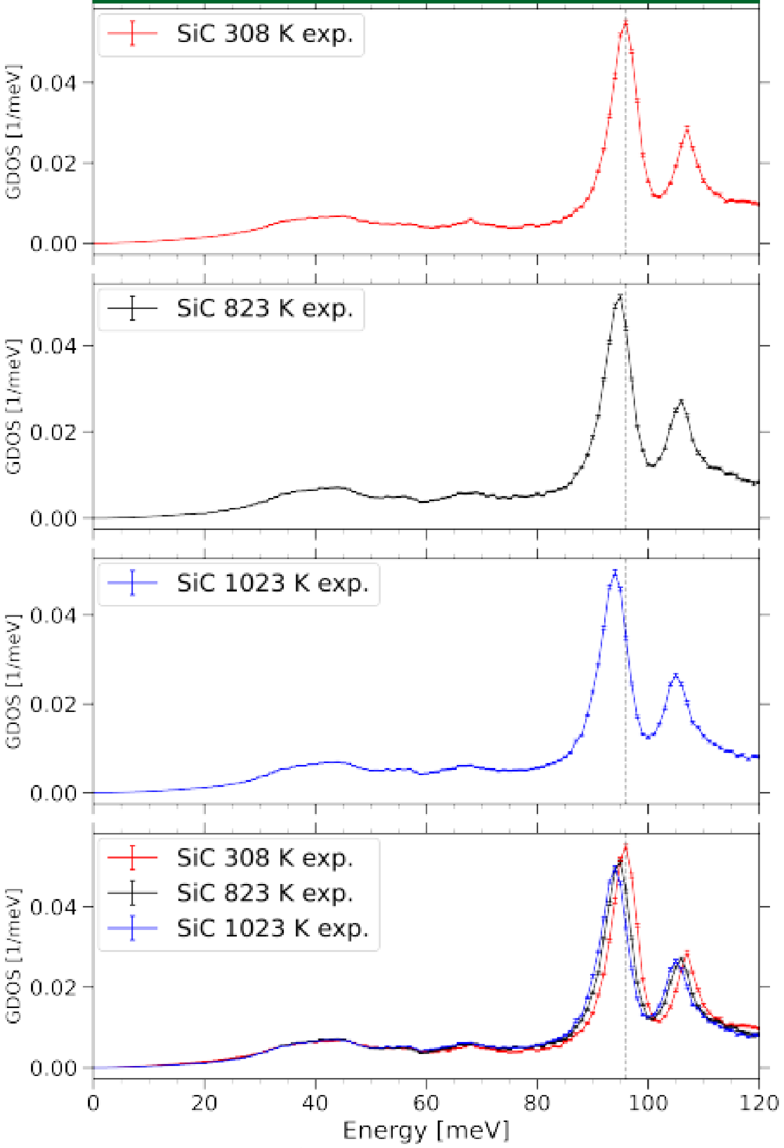
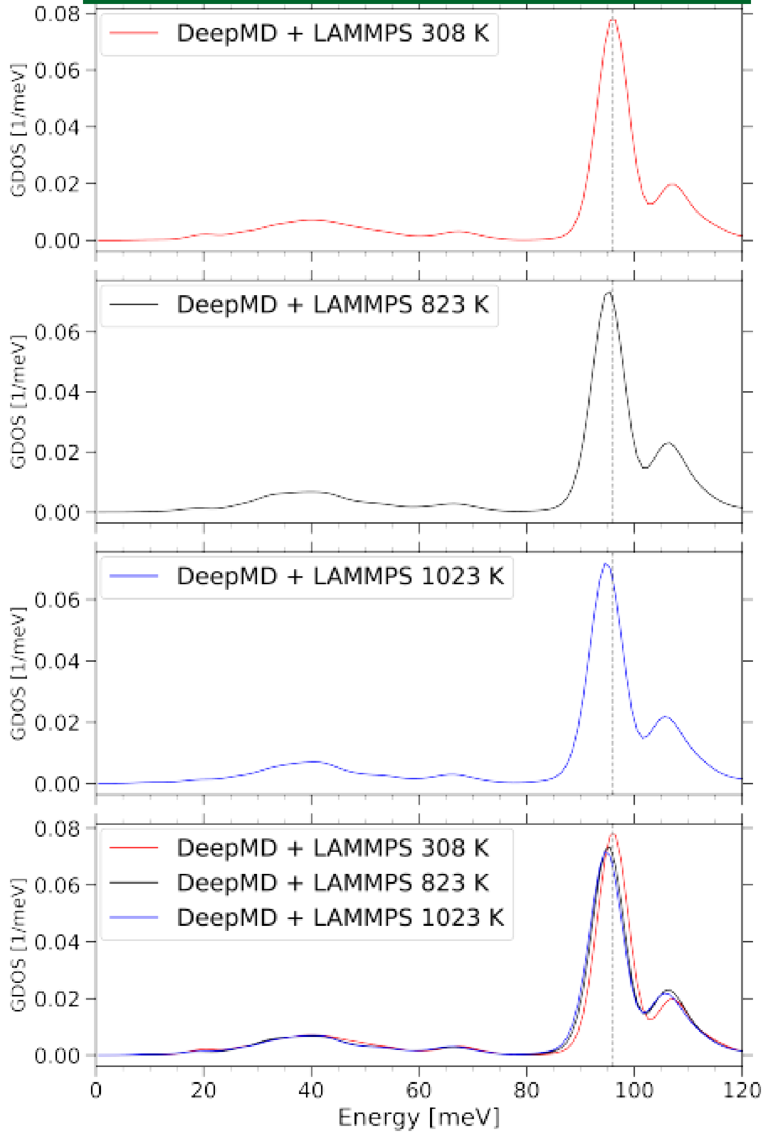
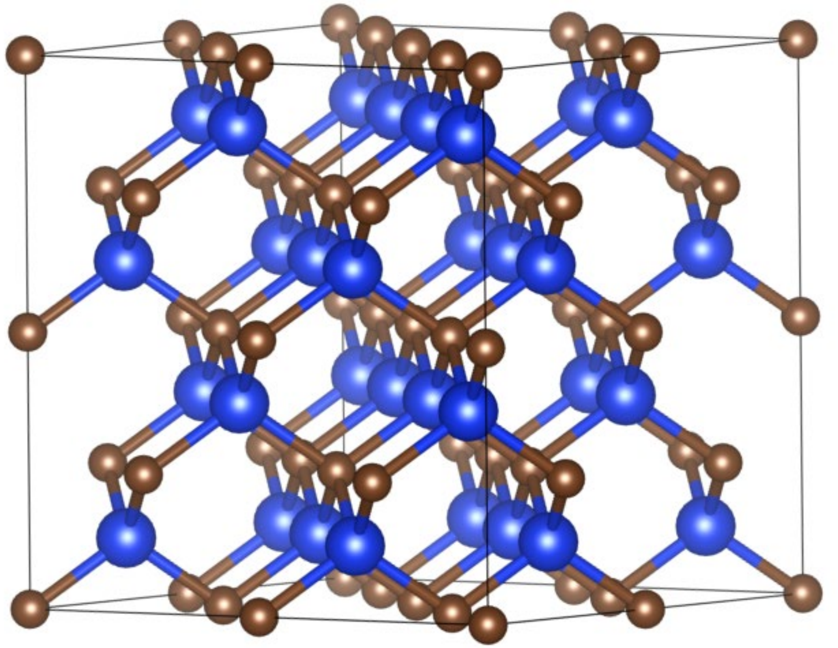
SiC

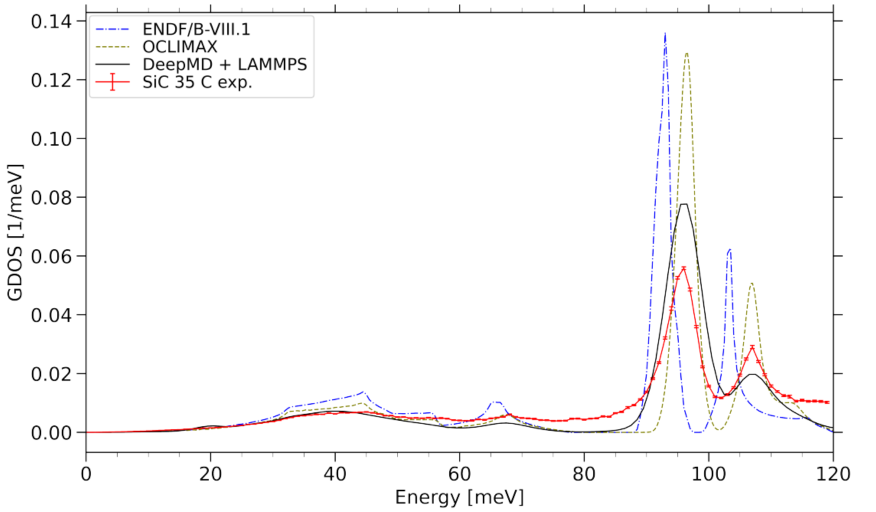
ARCS/SNS Measurements



MLMD Calculations

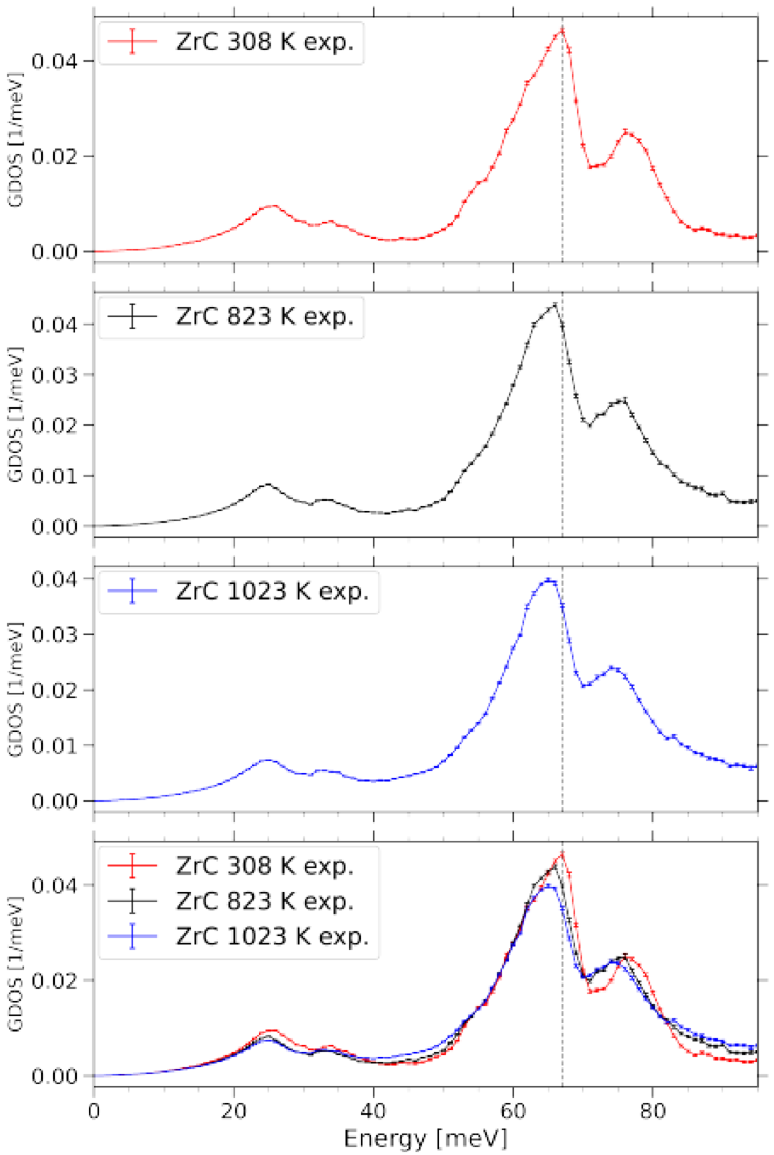




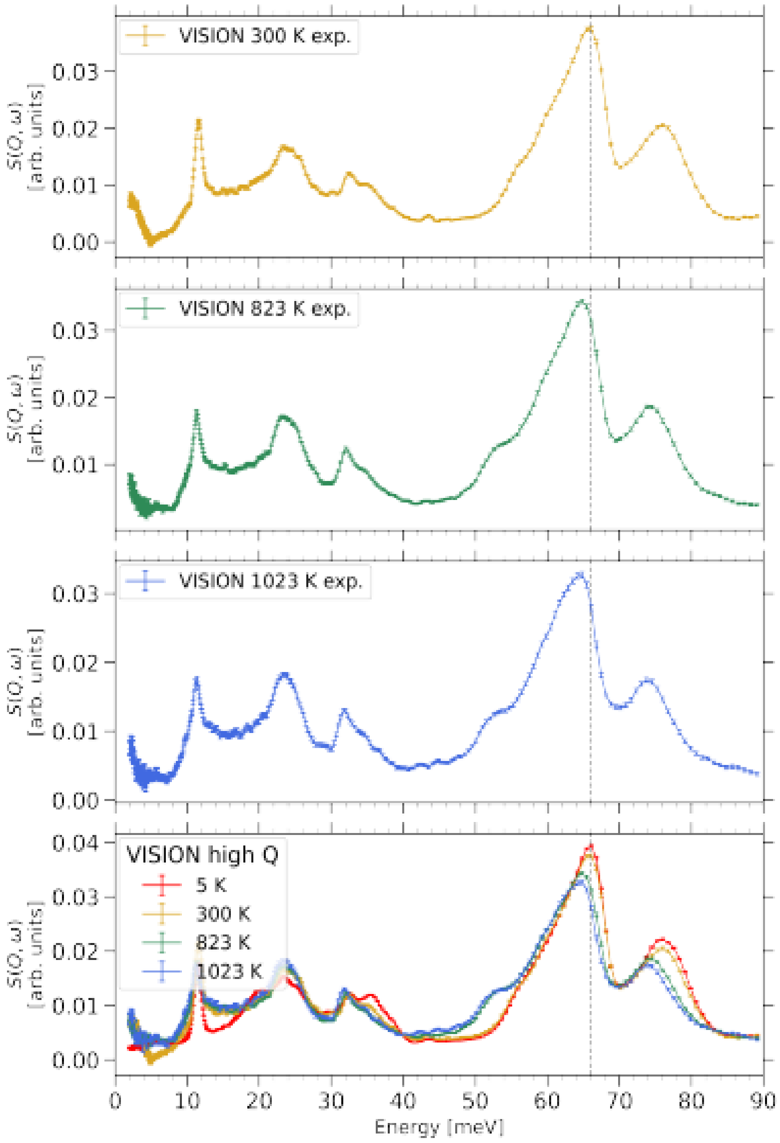


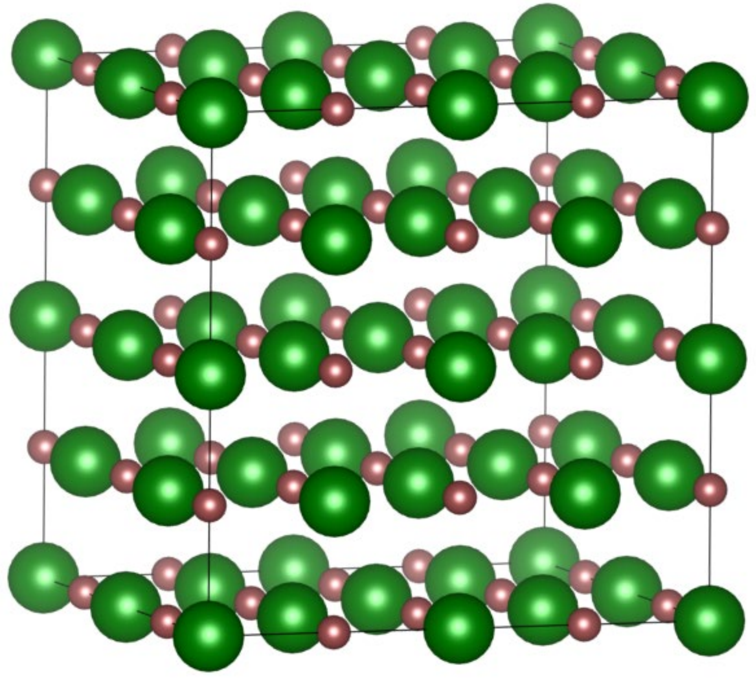
ZrC

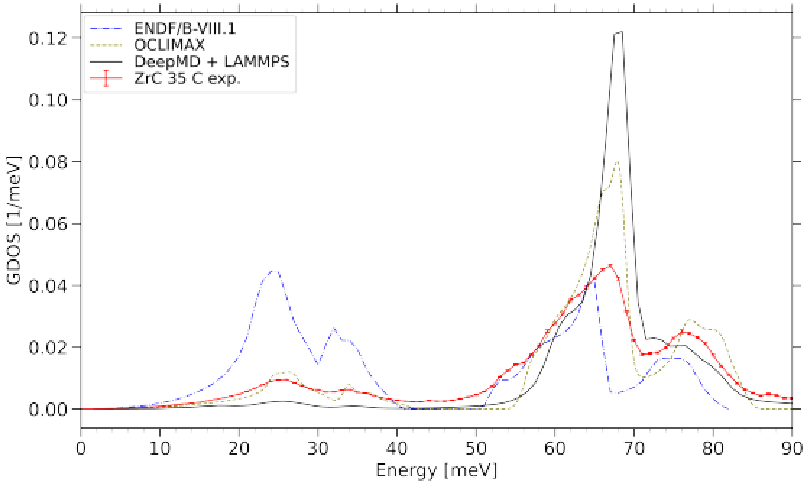
ARCS/SNS Measurements



VISION/SNS Measurements

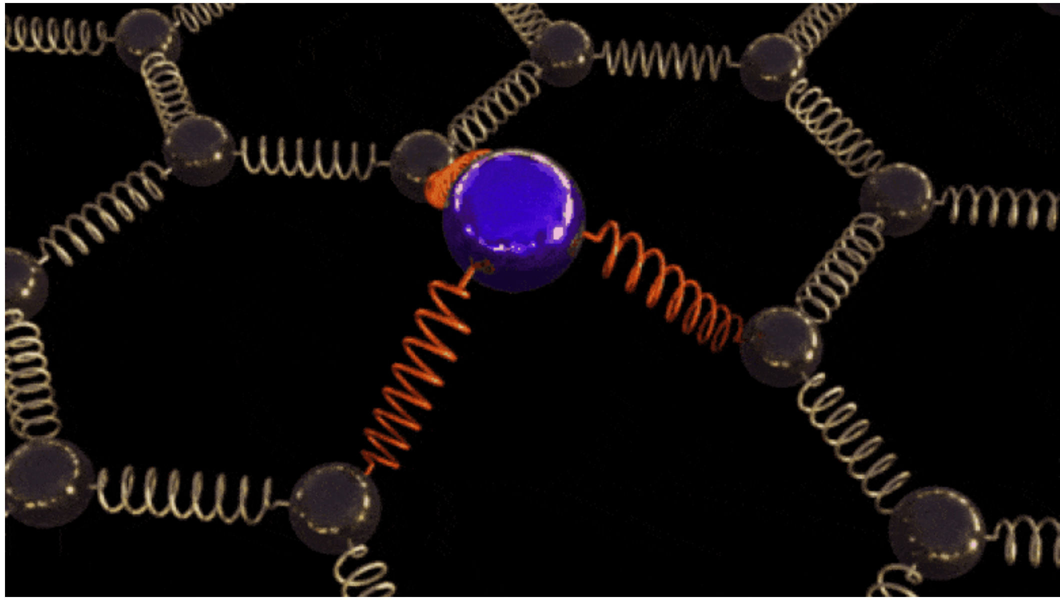






Thermal Neutrons

- Thermal neutrons have wavelengths (~Å) comparable to the separation distances of atoms in solids.
- **Hence, the thermal motion of atoms or molecules in the scattering medium can no longer be ignored**
- **Thermal neutron scattering law (TSL) describes the neutron scattering intensity as a function of energy and momentum transfer between the thermal neutron and the atoms of the scattering medium**



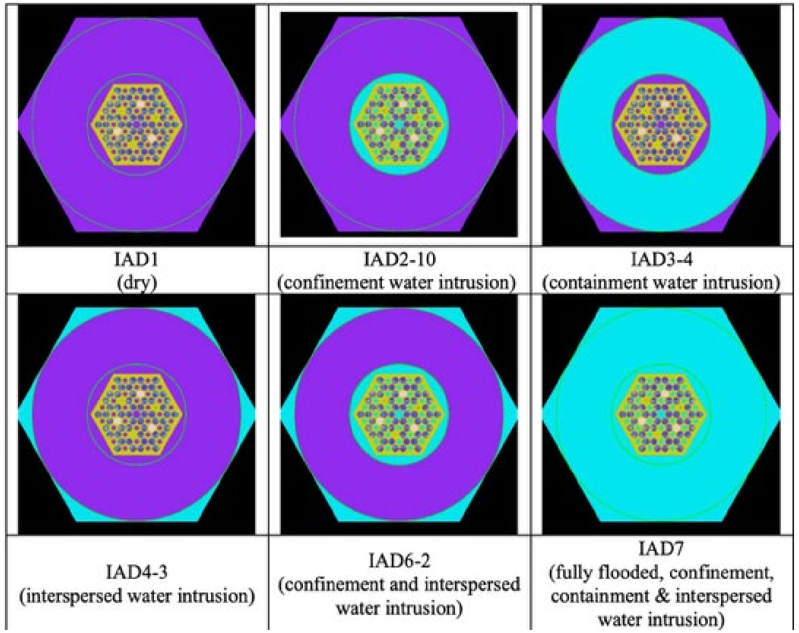
Models in review: GCMR in a cask

Gas-cooled Microreactor (GCMR) fuel assembly in a cask

- 19.75 wt% ^{235}U UCO TRISO fuel
- modified ES-3100 package

Investigated scenarios with nominal (void and Kaolite) and damaged (deformed and flooded with water)

SCALE models are in review



GCMR assembly models in ES-3100[1]

A subset if similarity analysis with LEU benchmarks[1]

ICSBEP Experiments		Undamaged cask, dry scenario	Damaged cask, no Kaolite, optimal water moderation
	k_{eff}	*0.57712*	*0.77990*
LEU-COMP-XX (1263 files)	$c_k > 0.8$	0	9
	$c_k > 0.9$	0	0
	Maximum c_k	**0.7061** [1]	**0.8098** [1]
LEU-MET-XX (84 files)	$c_k > 0.8$	0	0
	$c_k > 0.9$	0	0
	Maximum c_k	0.2853	0.4325
LEU-MISC-XX (60 files)	$c_k > 0.8$	0	0
	$c_k > 0.9$	0	0
	Maximum c_k	0.3105	0.4289
LEU-SOL-XX (113 files)	$c_k > 0.8$	0	0
	$c_k > 0.9$	0.3505	0.4877

1) Experiment LEU-COMP-THERM-084-001, Critical Loading Configurations of the IPEN/MB-0 Reactor with a Central Cruciform Rod

[1] Z. Zhong, A. A. Abdelhameed, N. E. Stauff and C. Celik, "Microreactor Assembly Transportation Cask Model for Criticality Safety Basis Assessment", Argonne National Laboratory, ANL/NSE-25/9

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Models in review: MSR fuels in GBC-32 cask

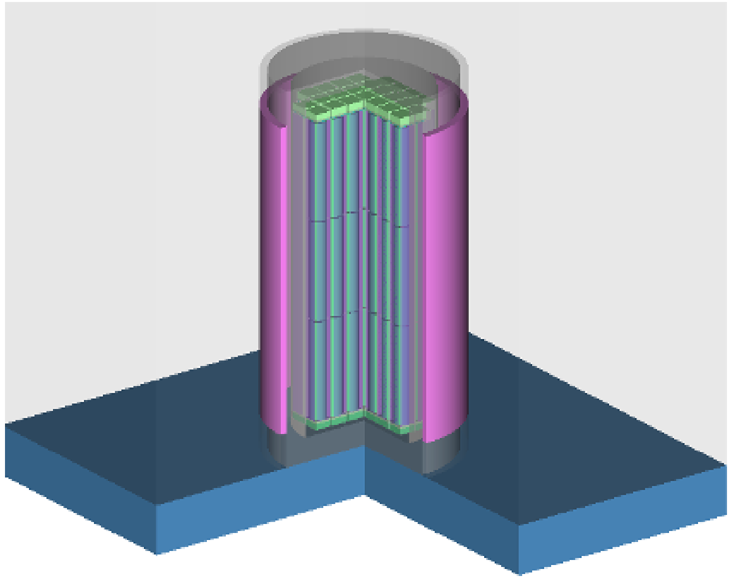
Generic 32 PWR-assembly burnup credit (GBC-32) cask with HALEU fuel salts (19.9 wt% ^{235}U)

- $LiF\text{-}BeF_2\text{-}UF_4$ (FLiBe)
- $LiF\text{-}BeF_2\text{-}ZrF_4\text{-}UF_4$ (FLiBeZr)
- $NaCl\text{-}UCl_3$ (NaCl)
- $KCl\text{-}UCl_3$ (KCl)

Investigated scenarios with nominal and water intrusion

Investigated similarity index for both **fresh** and **depleted** fuel salts

SCALE models are in review



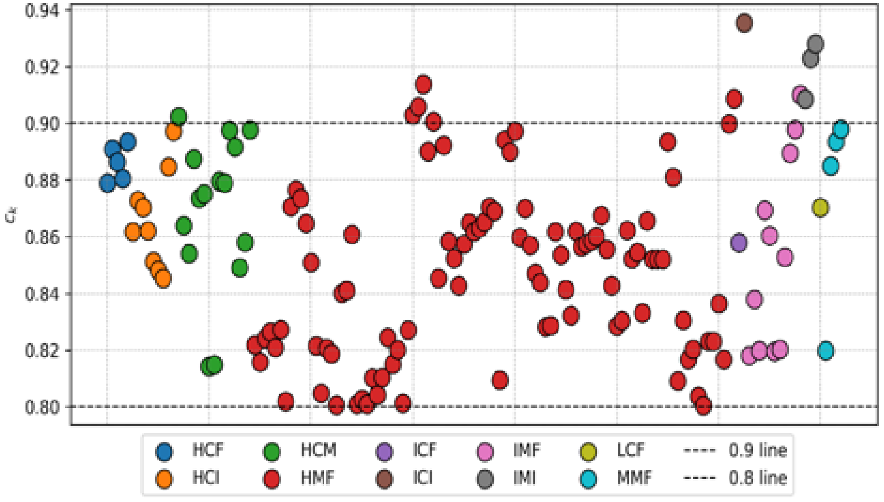
GBC-32 cask model[1]

Similarity analysis of FLiBe fuel at a power level of 100 MW

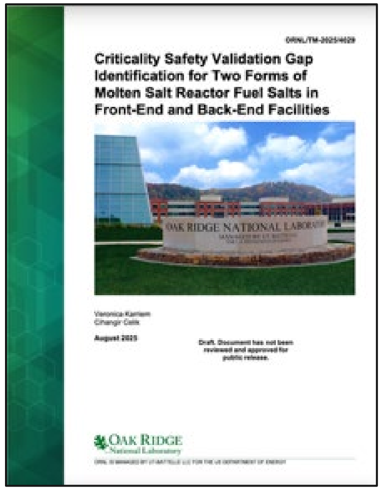
DAYS	Burnup [GWd/MTIHM]	Keff 5yr cool	EALF [eV]	c_k>0.8
1	0.006	1.003578	55.16	61
10	0.059	1.003418	55.41	61
50	0.295	1.00275	55.75	61
100	0.590	1.001909	56.18	61
200	1.180	1.000257	57.03	61
500	2.949	0.995464	59.49	61
1000	5.898	0.987788	63.40	58
2000	11.796	0.973462	70.69	53
3000	17.694	0.960391	77.36	50

Fuel salt-loaded GBC-32 similarity analysis

Fuel	k_{eff}	#Exp c_k>0.8	Highest correlating Experiment (c_k)
UF_4	1.1253	208	ICT-005-001 (0.99)
FLiBeZr	0.8477	146	ICT-005-001 (0.94)
FLiBe	0.6934	63	ICT-005-001 (0.88)
UCL3	0.9819	0	ICT-005-001 (0.76)
NaCl-UCl3	0.7606	0	MMF-011-004 (0.99)
KCl-UCl3	0.5235	0	MMF-011-003 (0.99)



FLiBeZr similarity analysis[1]



[1] *Veronica Karriem, Cihangir Celik (2025), Criticality Safety Validation Gap Identification for Two Forms of Molten Salt Reactor Fuel Salts in Front-End and Back-End Facilities, ORNL/TM-2025/4029, Oak Ridge National Laboratory, Oak Ridge, TN.*

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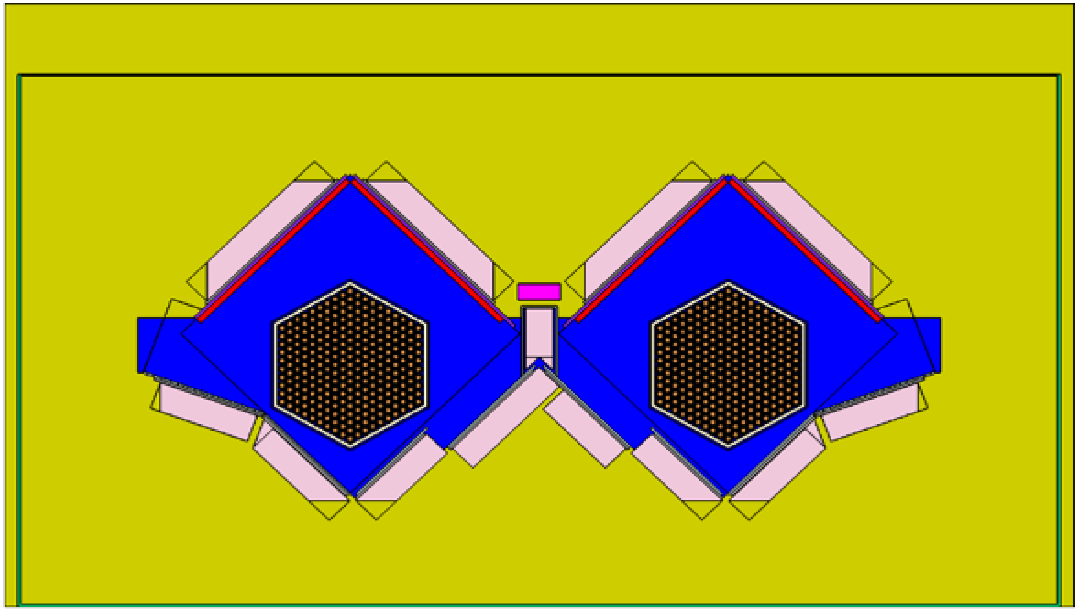
Models in review: MAP-12 shipping container

AREVA MAP-12 package with Advanced Burner Reactor (ABR) fuel assemblies

- SFR fuel (16 wt% ^{235}U in U-10Zr)

Investigated scenarios with nominal and water intrusion

SCALE models are in review



Model of MAP-12 package with ABR fuel assembly[1]

Similarity analysis for benchmarks with c_k>0.8[1]

Reflector around array	Moderator in assembly	Pin pitch	k_{eff}	VALID $C_k > 0.8$	ICSBEP $C_k > 0.8$
No	No	Touching	0.20641	0	0
No	No	Half-nominal	0.20076	0	0
No	No	Nominal	0.19612	0	0
No	Yes	Touching	0.57850	84	651
No	Yes	Half-nominal	0.63340	102	711
No	Yes	Nominal	0.69200	159	668
Yes	No	Touching	0.20712	0	0
Yes	No	Half-nominal	0.20108	0	0
Yes	No	Nominal	0.19722	0	0
Yes	Yes	Touching	0.57850	84	651
Yes	Yes	Half-nominal	0.63340	102	711
Yes	Yes	Nominal	0.69200	159	668


ORNL/TM-2025/3997

Validation Gap Assessment of a MAP Package Containing Fresh, Metallic Sodium-Cooled Fuel

Riley Cumberland

July 2025

OAK RIDGE National Laboratory

[1] Riley Cumberland (2025), *Validation Gap Assessment of a MAP Package Containing Fresh, Metallic Sodium-Cooled Fuel, ORNL/TM-2025/3997, Oak Ridge National Laboratory, Oak Ridge, TN.*

OAK RIDGE National Laboratory

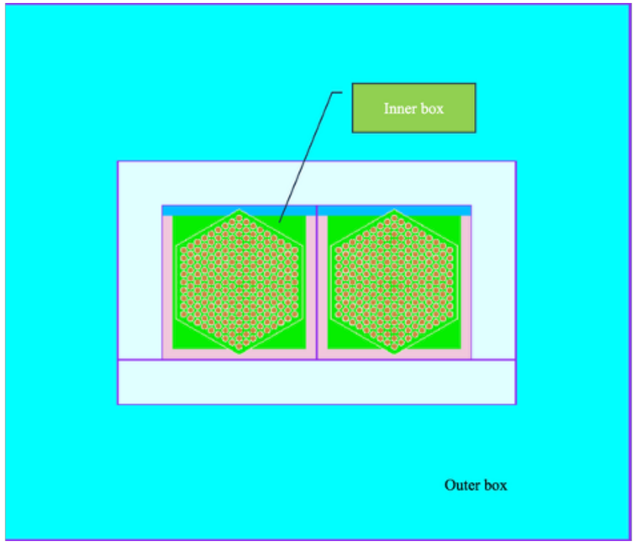
Models in review: TN B1 package

AREVA Transnuclear B1 (TN B1) package with Advanced Burner Test Reactor (ABTR) fuel assemblies

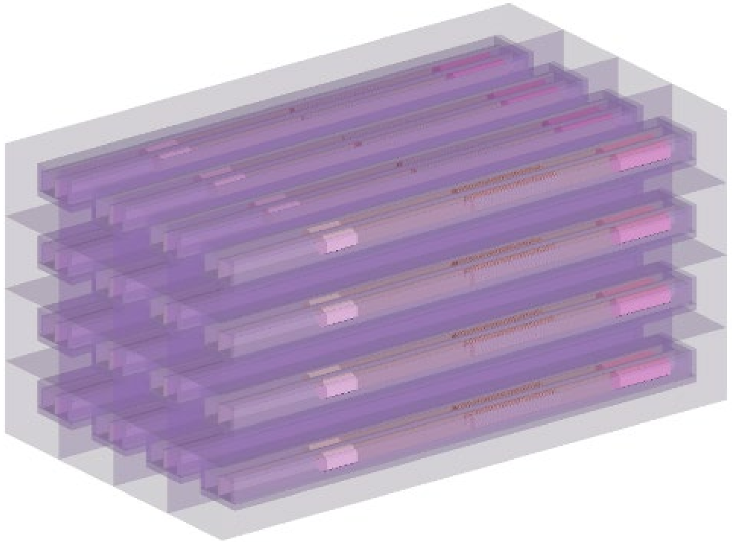
- SFR fuel (20 wt% ^{235}U in U-10Zr)

Investigated scenarios with nominal and water intrusion

SCALE models are in review



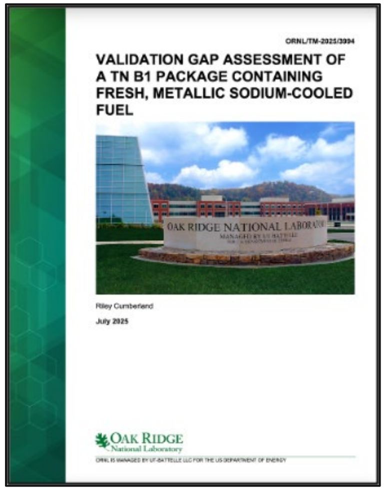
Single TN B1 package with ABTR fuel[1]



4 x4 array of TN B1 packages[1]

Similarity analysis for benchmarks with c_k>0.8[1]

Pin pitch	Moderator in assembly	Reflector around array	Outer box volume fraction water	k_{eff}	EALF	VALID $C_k > 0.8$	ICSBEP
Touching	No	No	0%	0.21745	5.61×10^5	0	0
Touching	No	Yes	0%	0.3246	2.10×10^3	0	0
Touching	Yes	No	0%	0.6402	21.8	64	448
Touching	Yes	Yes	0%	0.6708	16.3	64	565
Nominal	No	No	0%	0.20942	5.51×10^5	0	0
Nominal	No	Yes	0%	0.32287	1.62×10^3	0	0
Nominal	Yes	No	0%	0.6859	15.4	66	501
Nominal	Yes	Yes	0%	0.7123	11.3	73	641
Spread to channel	No	No	0%	0.20256	5.28×10^5	0	0
Spread to channel	No	Yes	0%	0.32046	1.33×10^3	0	0
Spread to channel	Yes	No	0%	0.7225	10.5	77	553
Spread to channel	Yes	Yes	0%	0.7574	7.90	77	669
Spread to channel	Yes	Yes	100%	0.7275	5.33	82	709
Spread to channel	Yes	Yes	31%	0.7294	5.47	88	718
Spread to channel	Yes	Yes	62%	0.7246	5.54	109	713



[1] *Riley Cumberland (2025), Validation Gap Assessment of a TN B1 Package Containing Fresh, Metallic Sodium-cooled Fuel, ORNL/TM-2025/3994, Oak Ridge National Laboratory, Oak Ridge, TN.*

Repository Server

Linux server that hosts the repository and runs models

Needed for model reviewing process and generating derived data

Interfaces with the GitLab repository

Server installation/setup is in progress