

Design and Safety Analysis Codes and Methods for HALEU Conversions and New Build Research Reactors in the U.S. and Beyond



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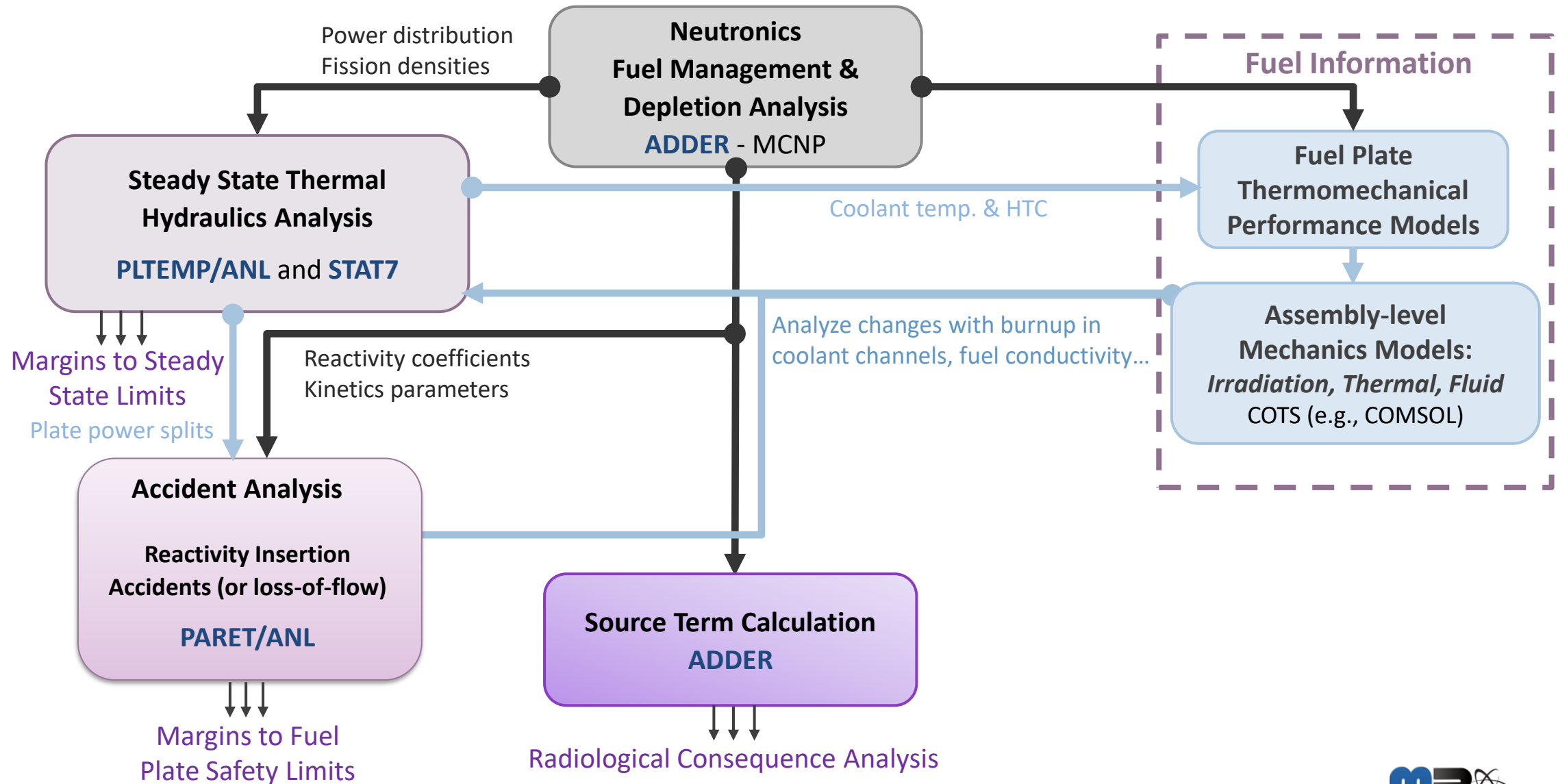
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Conversion Codes and Methods Framework

- **Established, validated analytical tools**
 - **Demonstration of computational codes** for safety and performance analyses that are **verified/validated** (V&V) for a range of application useful for many research reactors
 - Capabilities for predicting reactor behavior under normal and off-normal conditions
- **Tailored for research reactors**
 - **Describes the reactor analysis methods** used for calculation of margin to safety limits and core operating limits.
 - Supported many reactors for safety and key operational parameters
- **Enable consistent and efficient efforts**
 - Supported by international and U.S. high-performance research reactor (USHPRR) HALEU conversions
 - **Available and used** for analysis or verification **in 30+ reactors in the U.S. and beyond** including in licensing submittals
- **Support sequence of regulatory submissions for USHPRR HALEU Conversions**
 - NNSA – NRC engagement working to **accelerate readiness for HALEU** deployment in USHPRR through evaluation of Research Reactor **Conversion Codes and Methods**
 - **Initial deployment of U-10Mo alloy** and easily integrates **thermomechanical** effects for **very high-density fuels**

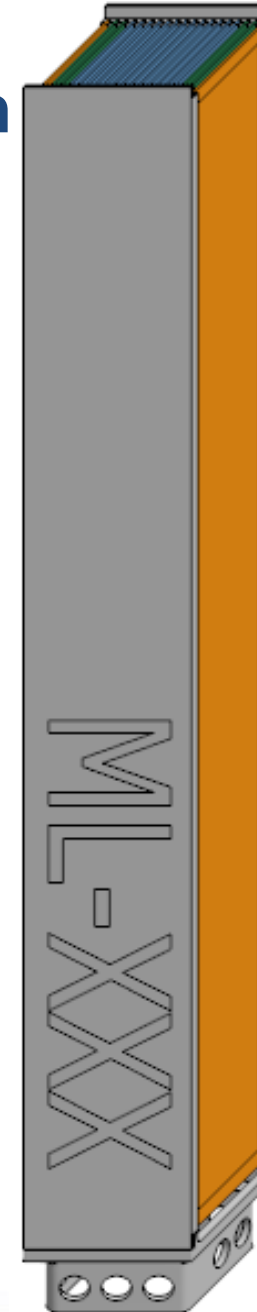


RTR Codes and Methods Framework for HALEU Conversions



Conversion Assembly Flow Testing – Validation Tests Support HALEU Conversion

- HALEU U-10Mo fuel element USHPRR design changes:
 - Fuel density (all 5 USHPRR)
 - Added multiple fuel thicknesses per element (4)
 - Change number of plates (2)
 - Thinner plates (2)
 - Remove fins (1)
 - Coolant channel gap thickness (2)
 - Increased flowrate or limits (2)
- Hydraulic flow testing complements irradiation testing
- Hydraulic performance evaluation provides measurements for validation of flow distribution and pressure effects
- Supports analysis of thermomechanical effects predictions that feed into safety analysis
 - Irradiation swelling, creep, thermal, flow & pressure
- Provides response of swaged joint

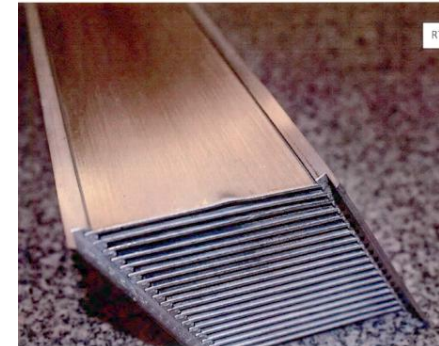
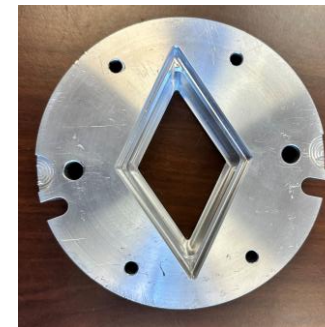


MITR Elements	HALEU	HEU
Fuel	U-10% Mo monolithic alloy	UAl _x dispersed in Al powder
Enrichment	19.75%	93%
U-235 Density	3 g/cc	1.5 g/cc
Fuel plates	19 / flat	15 / flat
Plate Thickness	0.049 inch	0.080 inch w/fins
Fuel Thickness	0.013 -0.025 inch	0.030 inch
Cladding & Interlayer	AA6061 Zr (0.001")	AA6061 -
Cladding Thickness	≥ 0.011 inch	0.015 inch



Flow Testing of MITR Nonfueled Conversion Element

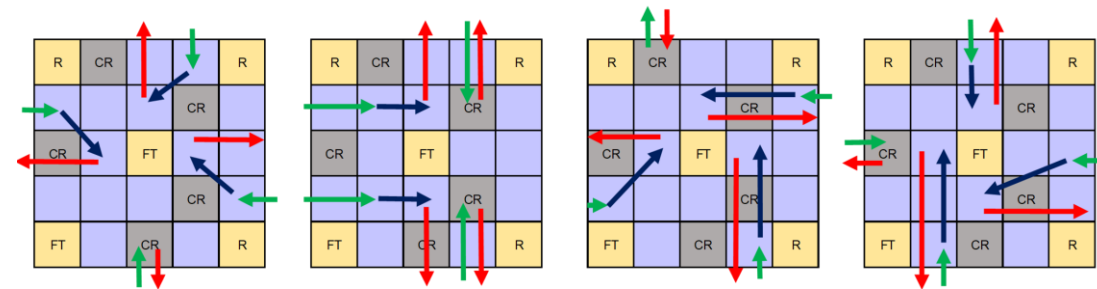
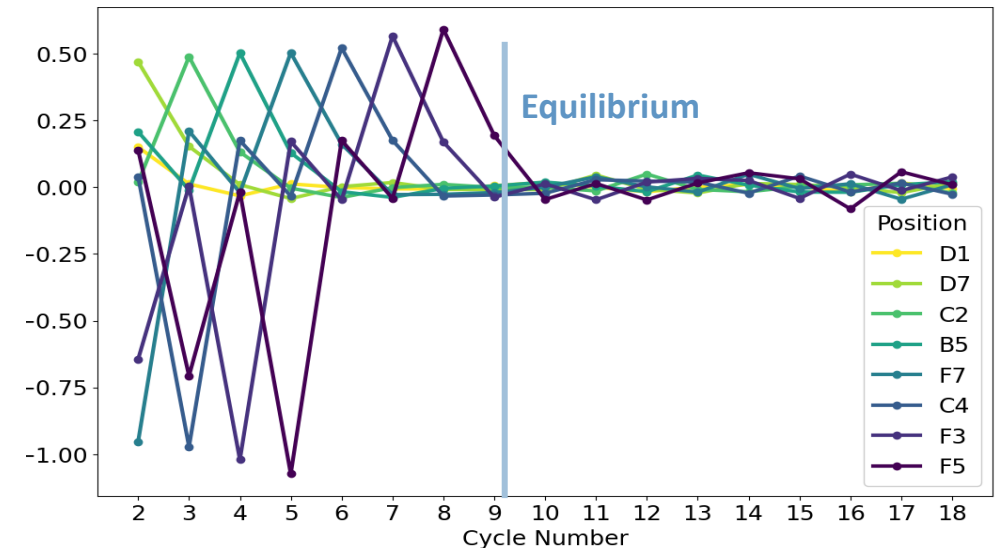
- 1st MITR nonfueled conversion elements fabrication complete
 - Commercially fabricated
 - End fitting on inlet and outlet Pitot tube array for all channels
 - Measure channel flow distribution and fuel element pressures and effects
- The test vehicle accommodating the element has been fabricated
 - Key hydraulic features in-core with combined end channels are part of vehicle
 - Flexibility in testing at both nominal and limiting fuel element configurations
 - Integration into OSU flow loop underway
- Generates validation data supporting codes and fuel conversion



ADDER Overview

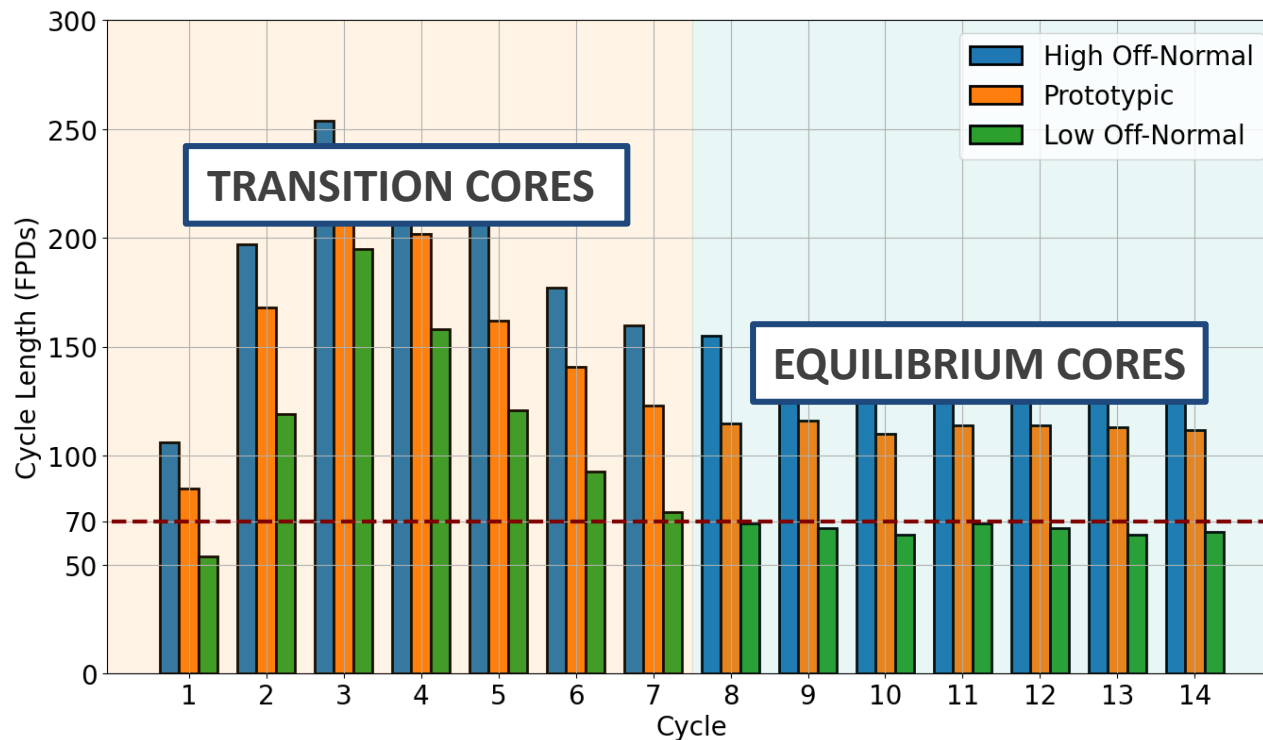
Summary: The Advanced Dimensional Depletion for Engineering of Reactors (ADDER) is Argonne-developed open-source software providing a flexible fuel cycle, fuel management, and depletion analysis tool. ADDER provides scalability to simply handle many depleting materials in a single model and increases automation of common reactor analysis tasks.

- **Current NQA-1 release:** ADDER v1.1.0
- **Key capabilities**
 - Fuel management (shuffling, moving, storing fuel)
 - Automated geometry transformation
 - Criticality search capability
 - Automatic cross section generation
- **Application Examples**
 - USHPRR and PRO-X analysis and verifications
 - AFIP-7 U-10Mo experiment depletion solver benchmark
- **Recent Development (ADDER 2.0)**
 - Enhanced features and speed of critical search
 - Local power distribution and cumulative fission density in fuel components
 - Interfaces with latest MCNP6 version
 - User tallies follow components in fuel management



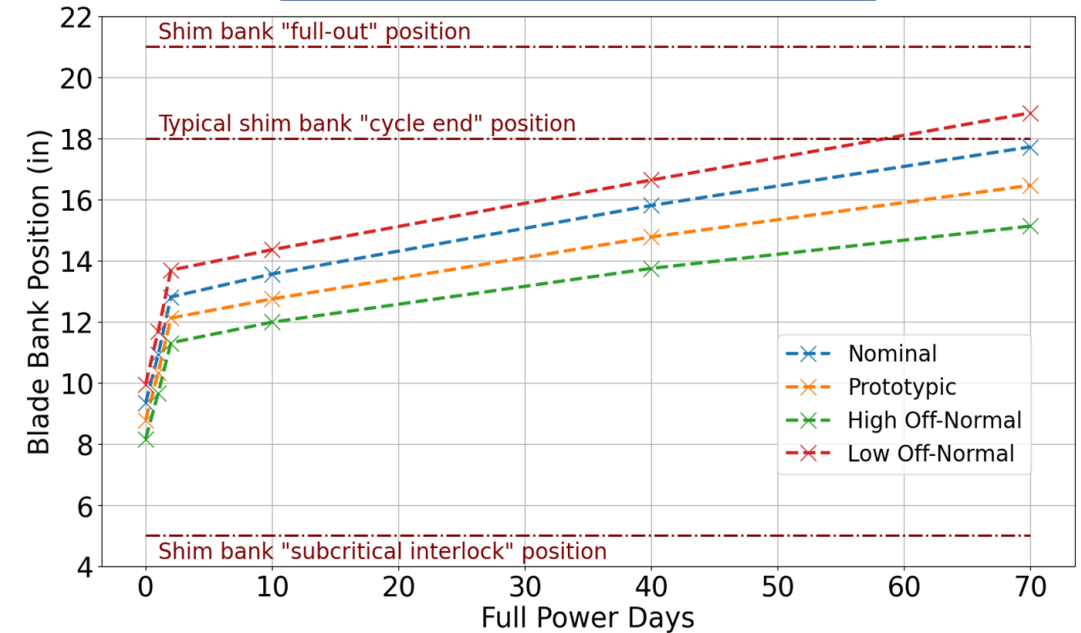
ADDER: MITR LEU Fuel Specification Impact All-LEU Transition Cores

- The impact of the U-10Mo LEU fuel specification on select neutronics characteristics was assessed using ADDER



*Cycle #1 contains 22 elements, #2 23 elements, #3 and onward 24 elements

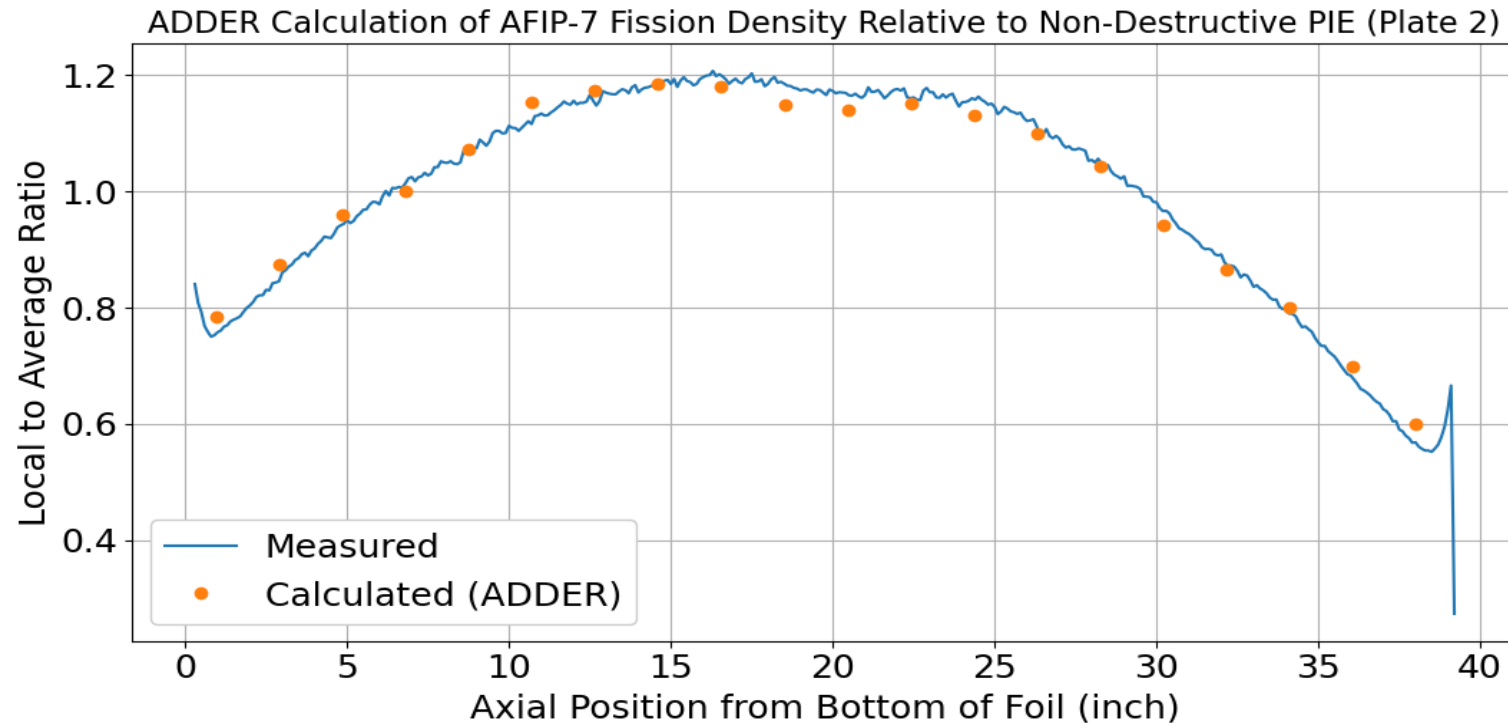
CYCLE LENGTH SENSITIVITY



- Includes many operations such as **fuel elements rotations and flips** that are **easily handled by ADDER**
- Fuel **depletion** always performed **at critical blade position**
- Acceptable results for all combinations** shown: Nominal, Prototypic, High Off-Normal, and with the ability to custom-select for Low Off-Normal

AFIP-7 Experiment Benchmark with ADDER

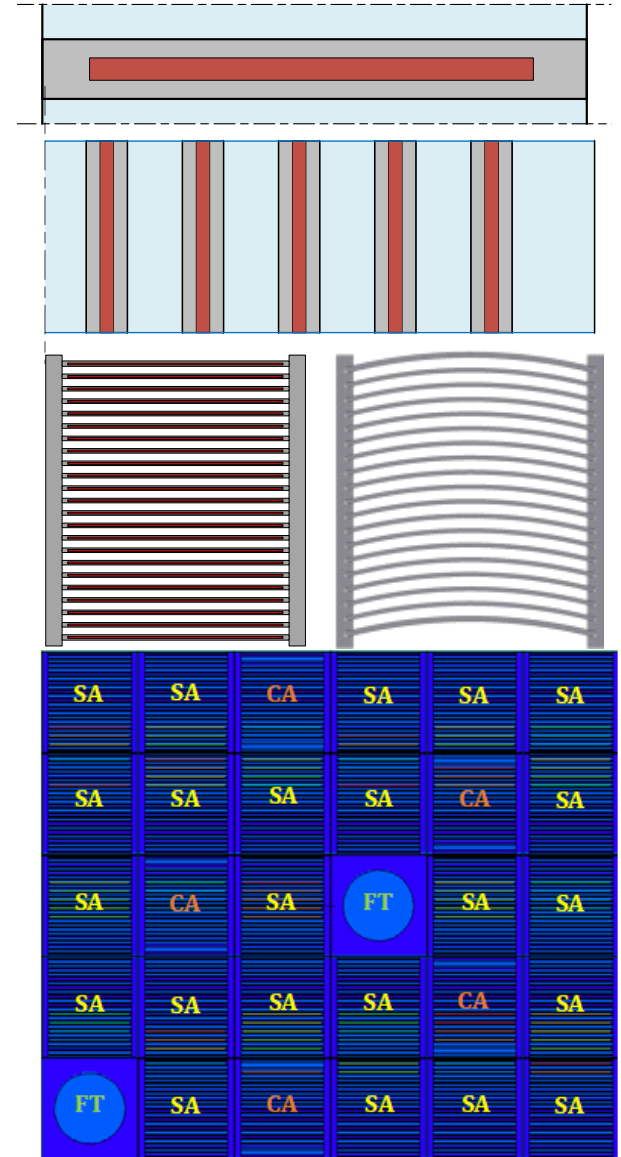
- AFIP-7 was an experiment in the center flux trap of ATR to evaluate **high assay LEU U-10Mo** fuel performance in **geometry and irradiation conditions prototypic of USHPRR fuel plates**
- A benchmark model of the experiment was developed using ADDER to predict fission densities in 19.75% enriched U-10Mo. The model includes a **detailed power history and control element movement** over two cycles of ATR.
- The benchmark shows that **ADDER can properly predict fission densities in LEU U-10Mo fuel**, showing good agreement with destructive and non-destructive post-irradiation examination (PIE) measurements



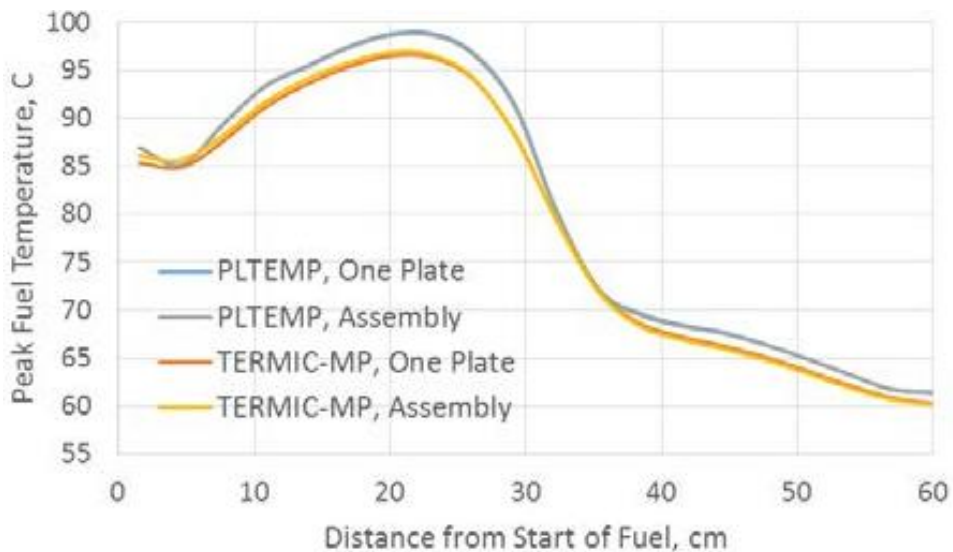
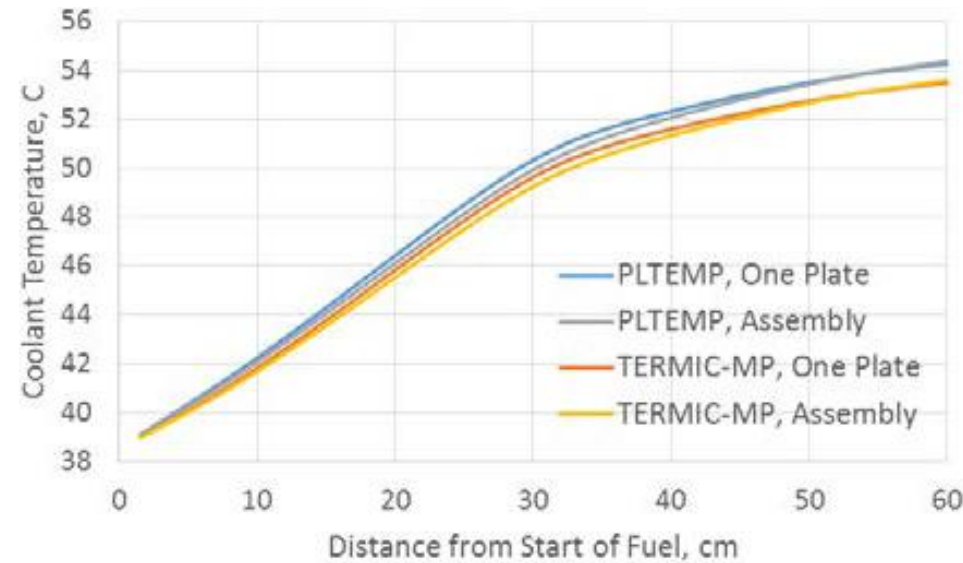
PLTEMP/ANL Overview

Summary: PLTEMP/ANL is used to predict steady-state thermal hydraulics performance and safety margins of research reactors, cooled by H₂O or D₂O, using hot channel factors to account for uncertainties including manufacturing tolerances

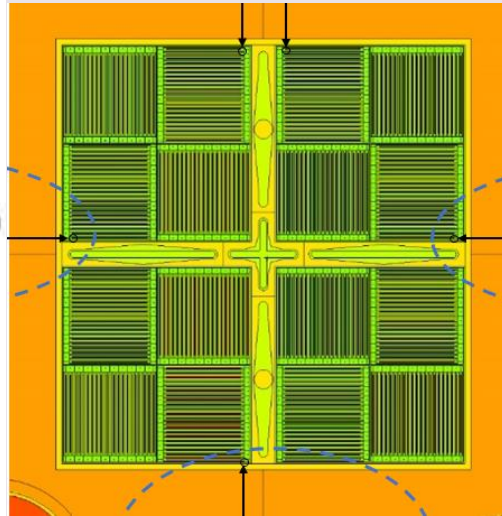
- **Current NQA-1 release:** PLTEMP/ANL v4.4.2
- **Key capabilities**
 - Application to research reactors with plate, tube, or pin-type fuel
 - Steady-state thermal hydraulics in thermally coupled parallel coolant channels with up to 3-D heat conduction within flat or curved fuel plates
 - Determination of margins to various boiling phenomena, such as onset of nucleate boiling (ONB), onset of flow instability (OFI), and critical heat flux (CHF) with multiple correlations for each
- **Application**
 - US: MURR (Missouri), UFTR (Florida), PUR-1 (Purdue), RINSC Reactor (RI), etc.
 - International: RHF (France), OPAL (INVAP), MNR (McMaster), RPI (Portugal), IRT-1 (Libya), DNRR (Vietnam), etc.
- **Recent Enhancements**
 - 2-D or 3-D heat conduction in fuel plates; Enhanced water properties; Saha-Zuber and the Moritz OFI correlations



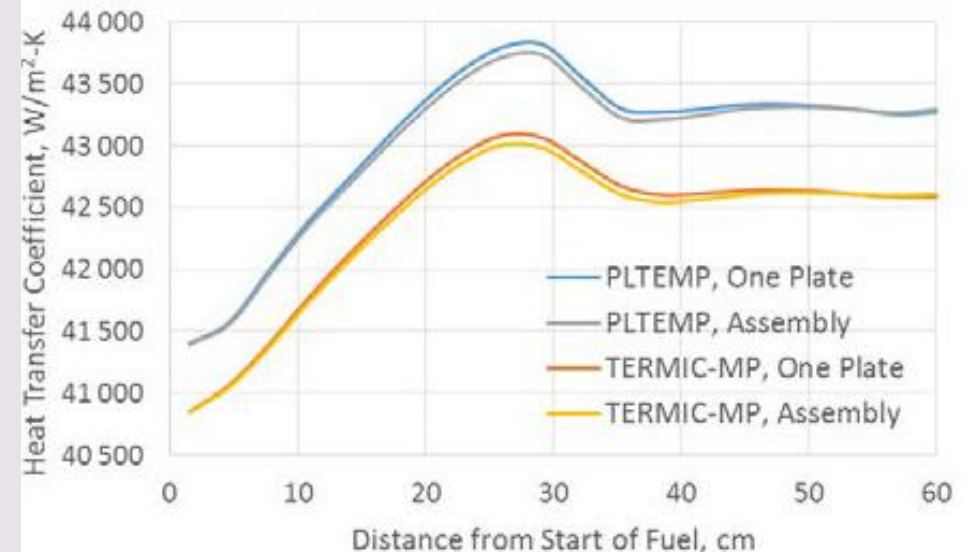
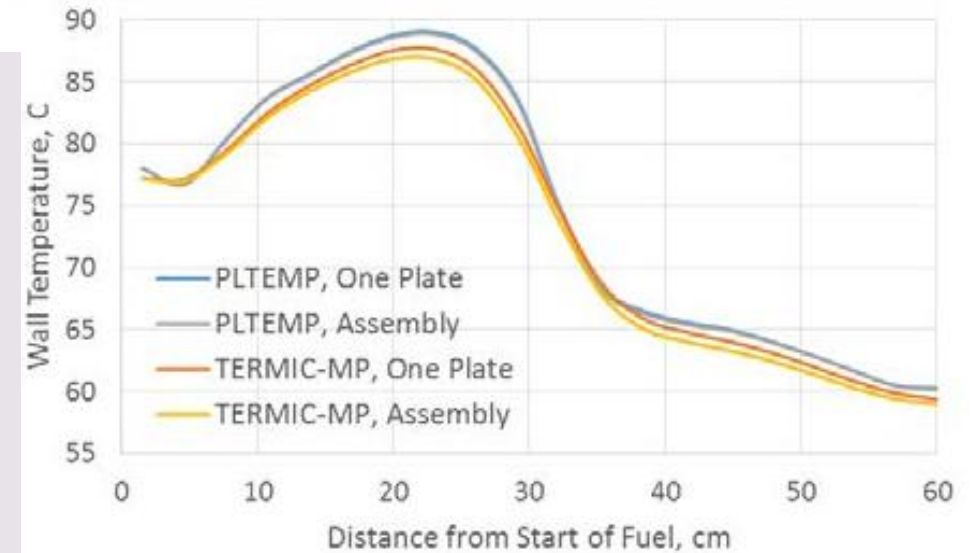
OPAL: PLTEMP & INVAP TERMIC – Very Different Codes, Good Agreement



- Whole core analysis
- 16 MTR plate-type FA
- Control Blade Motion



- Detailed burnup and power distributions
- High-performance research reactor results compare very well

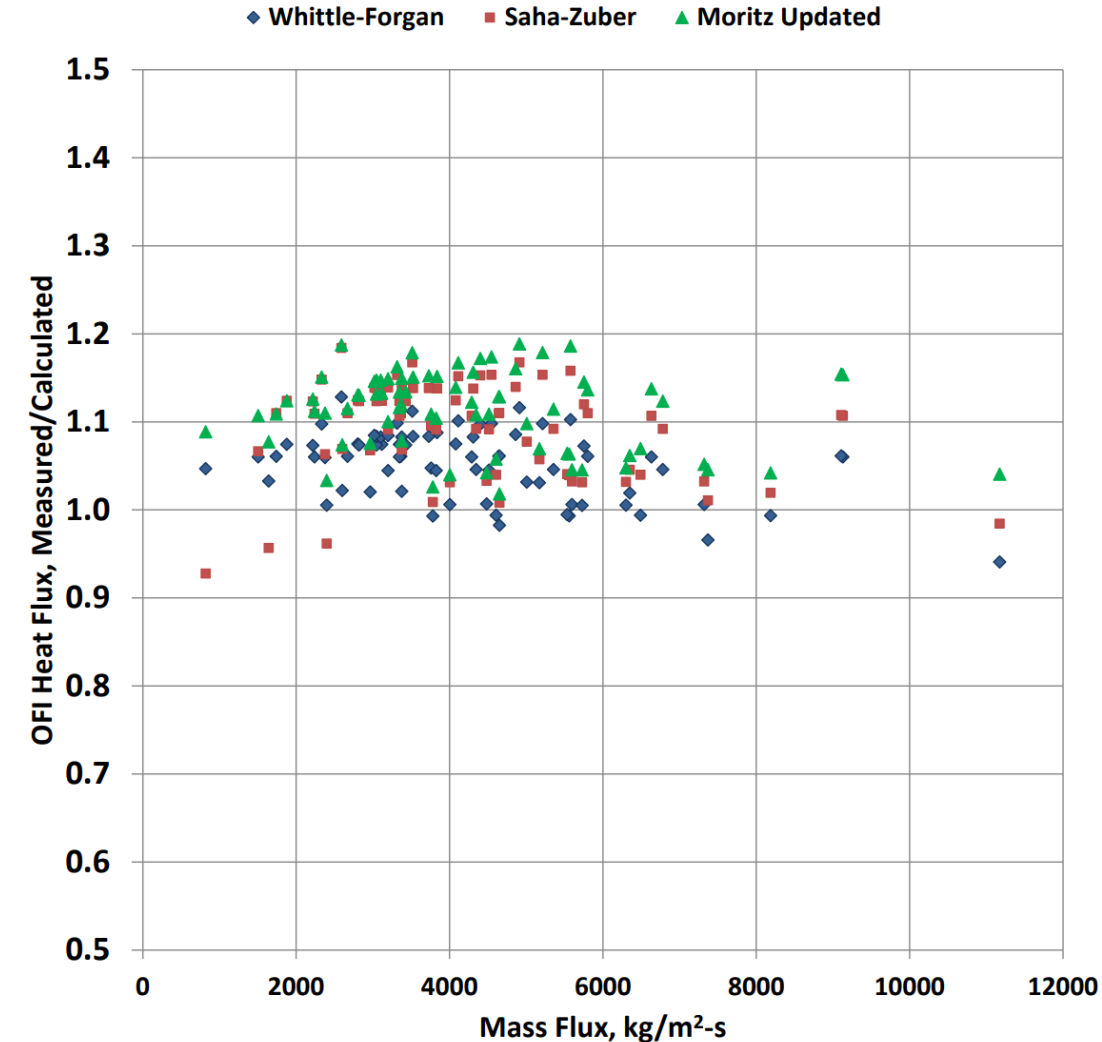


PLTEMP/ANL Flow Instability Correlations

- Onset of Flow Instability is a key safety parameter for many RTR
- PLTEMP/ANL predicts OFI well compared to measurements
- OFI correlations tend to conservatively predict OFI before it was measured

	Measured/Calculated OFI Heat Flux Ratio		
Correlation	Whittle & Forgan	Saha-Zuber	Moritz Updated
Mean	1.05	1.09	1.11
Standard Deviation	0.04	0.05	0.04
Ratio, with example 99.95% confidence, to avoid OFI	1.09	1.10	1.04

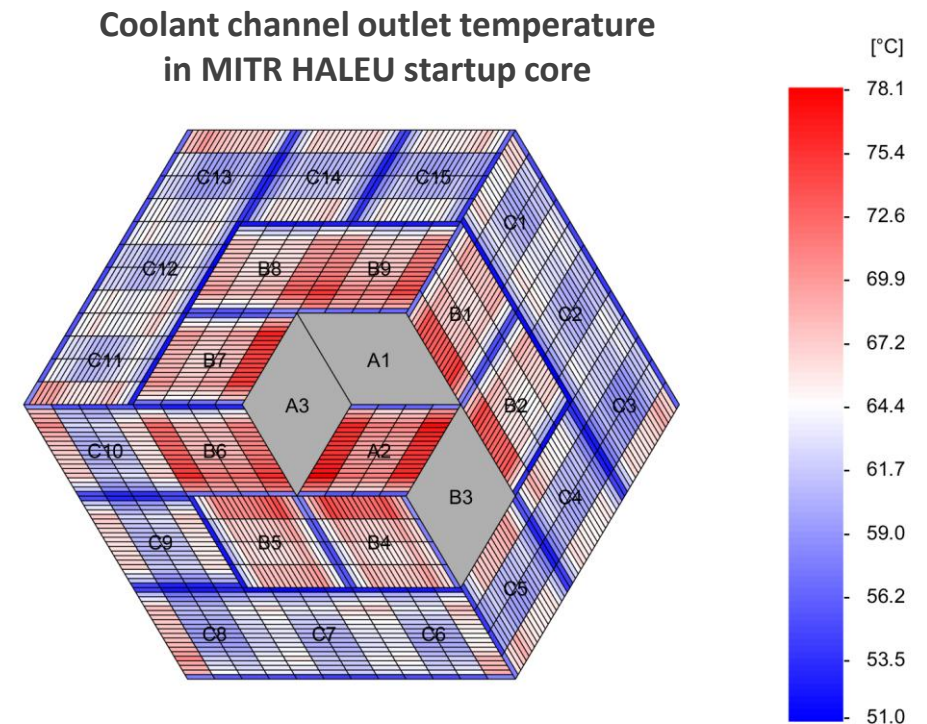
Comparison of Measured-to-Calculated OFI Heat Flux Ratio for Three Correlations for 75 Tests



STAT7 Overview

Summary: Developed jointly by Argonne and MIT, STAT7 automates many of the steady-state thermal hydraulics safety calculations for research and test reactors with plate-type fuel. A Monte Carlo statistical propagation approach is used to treat uncertainties of key reactor parameters and fuel fabrication tolerances in the analysis.

- **Current NQA-1 release:** STAT7 v2.0
- **Key capabilities**
 - Steady-state thermal hydraulics in thermally coupled fuel element parallel channels with 1-D heat transfer in fuel plates
 - TH solver can simulate an entire reactor core
 - Monte Carlo histories run per power with random parameters
 - Search for power at which Onset of Nucleate Boiling (ONB) and Onset of Flow Instability (OFI) occurs at a specified probability
- **Application**
 - Fuel specification impact assessments
 - Mixed HEU-LEU transition cores
 - Ex-core fission converter analyses
- **Recent Enhancements**
 - Whole-core modeling capability with hydraulic flow network (burnup-dependent flow modeling, mixed fuel transitions, etc.)
 - Non-normal sampling



STAT7: MITR Conversion Fabrication Specification Impacts

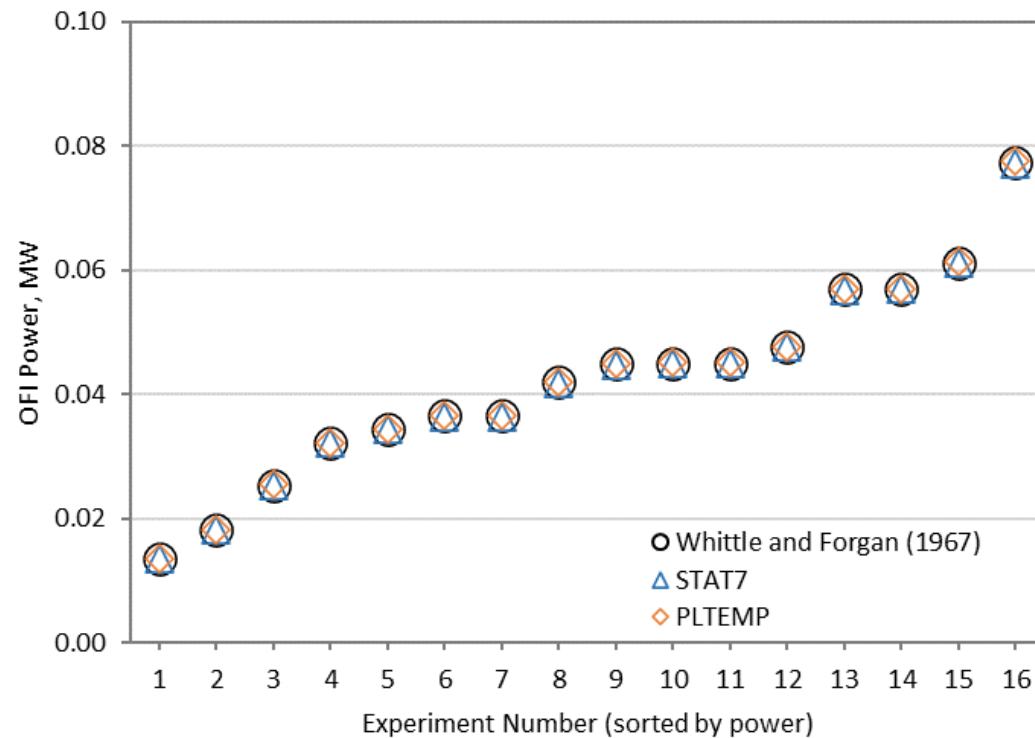
- Statistical uncertainty propagation capability allows for explicit treatment of fabrication and operational uncertainties to evaluate LSSS and operational margins to the safety limits
 - Uncertainties of modeling, measurements, and fabrication tolerances analyzed using a Monte Carlo method
- Input Parameters to STAT7 Code
 - Statistical parameters
 - Power Distribution (heat flux) from ADDER
 - Coolant channel gap thickness and spec tolerance
 - Plate thickness (fuel core & cladding)
- STAT7 Output Variables
 - Coolant temperature at each node (1-D)
 - Fuel temperature at each node (2-D)
 - Saturation temperature at each node (Tsat)
 - Nodal ONB temperature
 - Power at which a specified ONB probability occurs (P_{ONB})
 - Most limiting channel & axial node per stripe

By studying the sensitivity to fabrication tolerances, we analyze the impact on:

- Location of the most limiting element, stripe, channel, and axial node at the LSSS power ($P_{LSSS} = 8.68$ MW)
- Location of the most limiting element, stripe, channel, and axial node at the safety limit (@ P_{ONB} , other conditions were maintained as LSSS conditions)
- The parametric study showed P_{ONB} remained above 9 MW
- Transition cores with HEU and LEU will next be analyzed

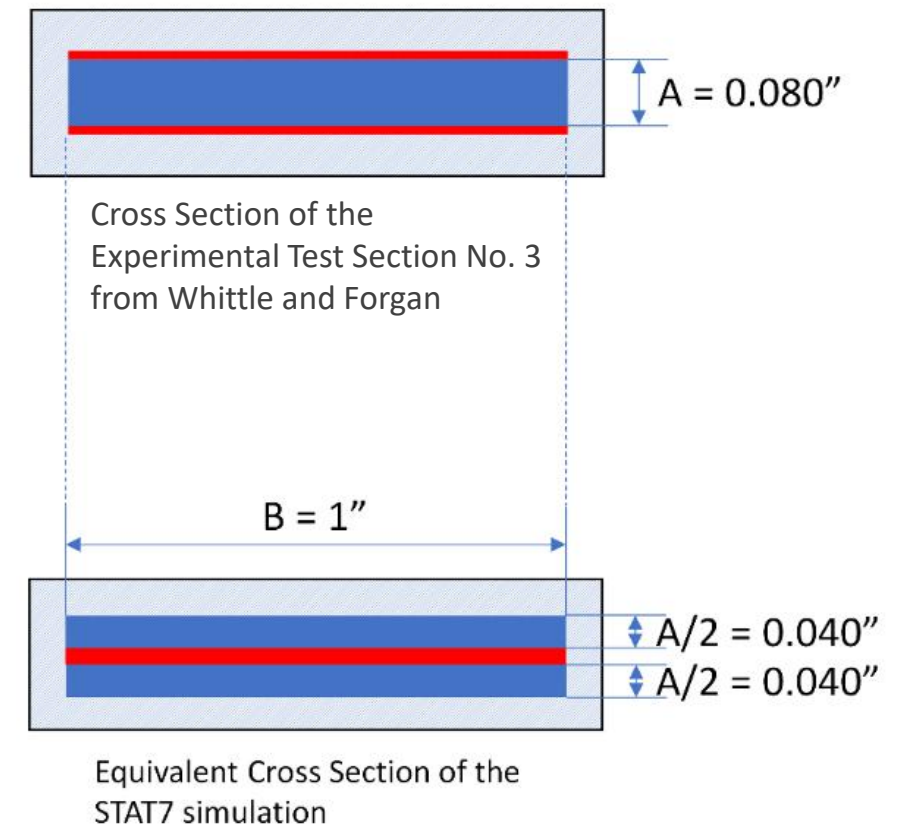
STAT7 OFI Correlation Validation

- Whittle and Forgan OFI correlation implemented in STAT7
- STAT7 agrees well with the Whittle and Forgan experimental data (as does PLTEMP/ANL)



Whittle and Forgan Test Section Model

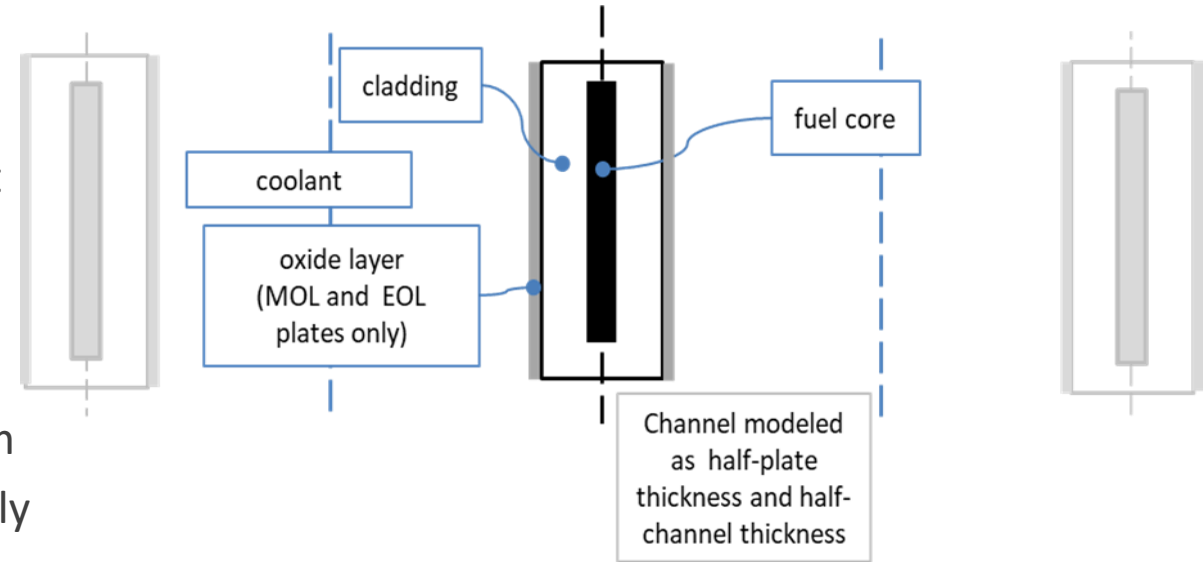
- Heated wall
- Coolant Channel
- Adiabatic Boundary



PARET/ANL Overview

Summary: PARET/ANL is used to determine expected transient behavior following reactivity insertions or loss-of-flow accidents in research reactors

- **Current NQA-1 release:** PARET/ANL v7.7
- **Key capabilities**
 - Coupled point kinetics and hydrodynamics transient analysis, including rapid transient heat transfer
 - Natural convection and forced flow
 - Reactivity feedback effects
 - Reactor trip settings can be adjusted to model scram
 - Simple input for multiple channel DBA analysis, easily modified for RTR
- **Application**
 - Calculates maximum fuel and cladding temperatures during transients for comparison to safety limits
 - Many conversions completed worldwide and used in licensing
 - MURR and MITR safety analyses
 - Multiple channels can be used for sensitivity analysis to uncertainties of key properties & performance data
- **Recent Enhancements (PARET/ANL v7.7)**
 - Improved thermal hydraulics modeling capabilities that includes modeling of recoverable pressure drop

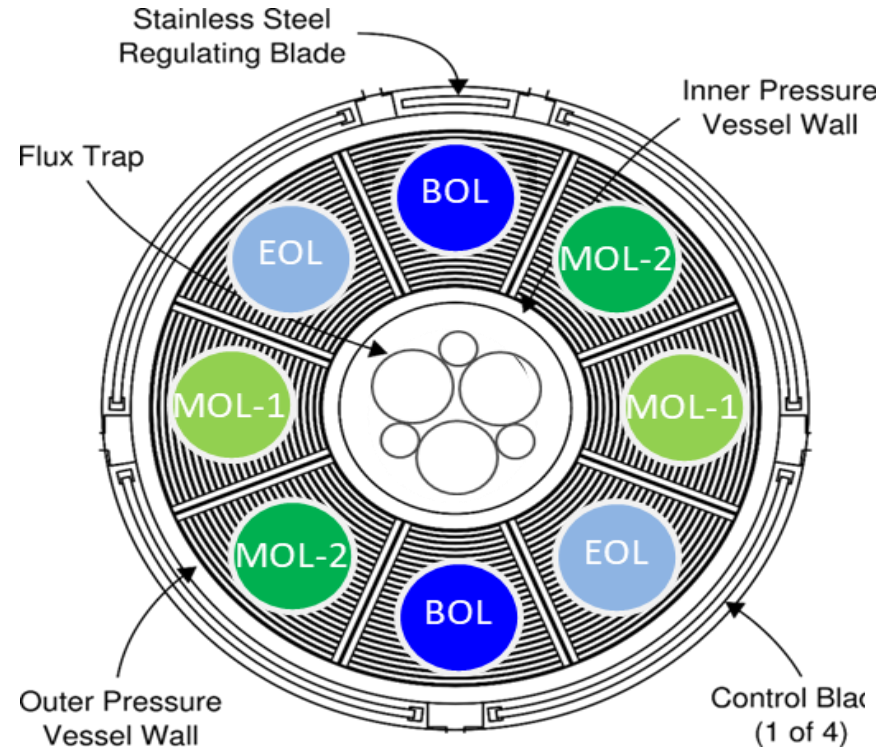


Modeling of Plate Geometry in PARET/ANL

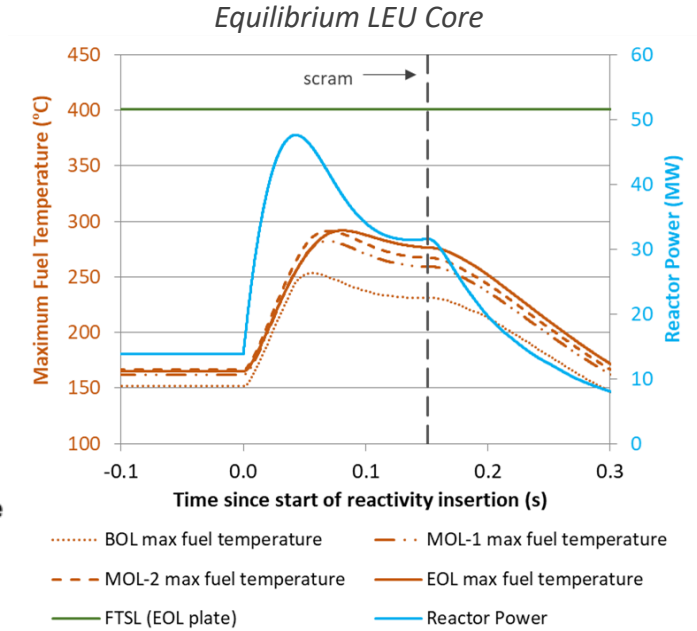
PARET/ANL: Safety Analysis for MURR LEU Conversion

(Stillman, et al., ANL./RTR/TM-19/18, <https://www.osti.gov/biblio/1708833>)

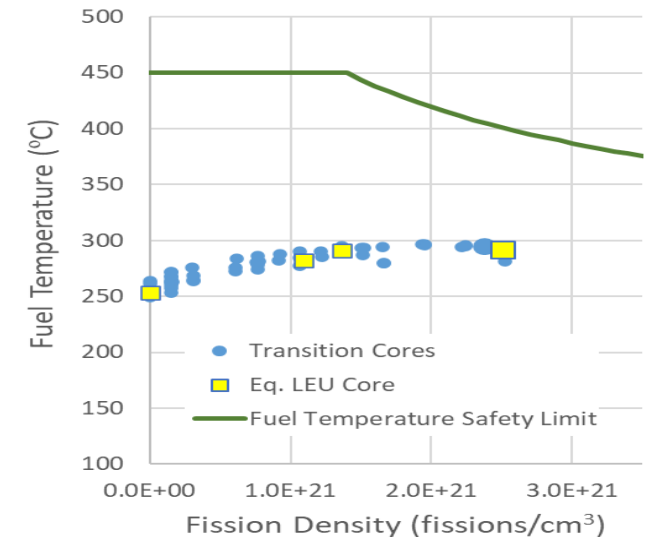
- Pressurized, reflected, open-pool type, light-water moderated and cooled
- Fuel elements with curved plate geometry; HEU and LEU fuel have similar geometry
- Safety analysis completed for 12 MW LEU fuel
 - Rapid insertion of 0.6% $\Delta k/k$ is most limiting RIA
 - PARET/ANL model composed of five channels; limiting plate in each of four elements (LEU P23)
- Greater than 100 °C margin to preliminary safety limit



0.6% $\Delta k/k$ step RIA in MURR LEU Core
Maximum Fuel Temperature for Limiting Plates

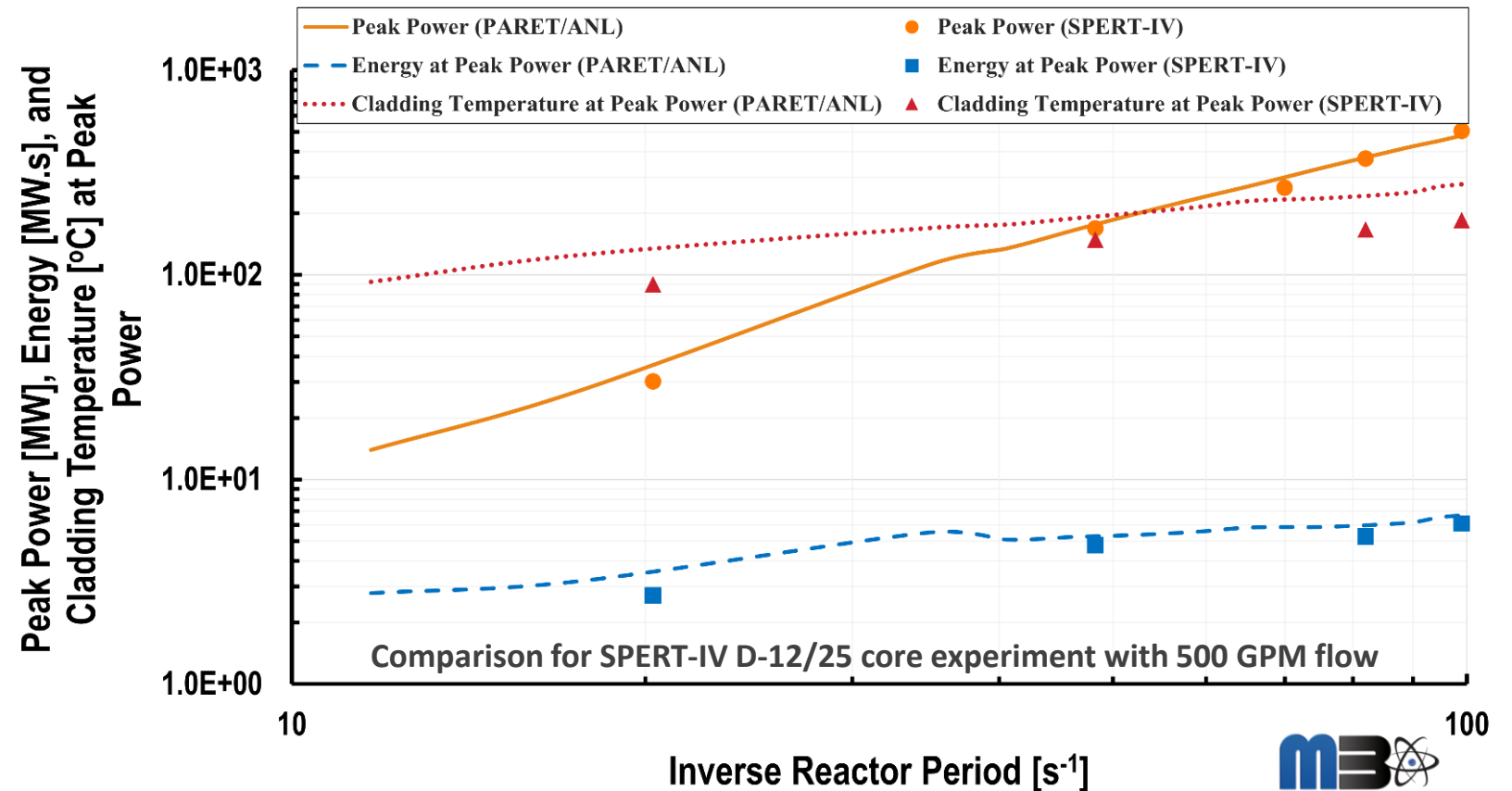
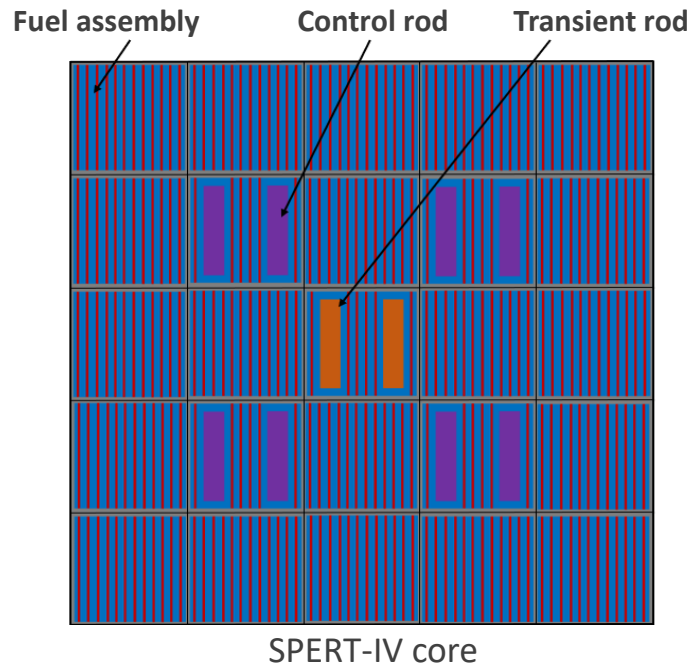


Transition Planning and Equilibrium LEU Cores



SPERT-IV Experiment Benchmark with PARET/ANL

- The SPERT-IV D-12/25 core consists of 25 fuel assemblies in a rectangular grid structure cooled by light water
- The experiments aim to determine the consequences of accidental reactivity insertion produced in the core
- Rapid ejection of the transient rod starting from very low initial power levels
- PARET/ANL agrees well for power excursions including reactivity feedback effects



Research and Test Reactor Codes Support HALEU Deployment in Current and New Reactors

- **Validated analytical tools** provide an analytical basis for HALEU conversions
- Steady-state/transient core analysis
- **Support Tech Specs/operating limits**
- **Simple whole-core analysis** approach
- Scaled to relevant phenomena for efficient accurate results
- **Successfully deploying HALEU** fuels to replace HEU
 - 2 conversions in Japan, complete FY26
 - 74 reactor converted from HEU, next Belgium
 - Over 7 metric tons of material removed
- Supporting continued research reactor operations



KUCA HALEU
Silicide Fuel Element

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