INL/MIS-24-81441

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Risk-Informed Systems Analysis
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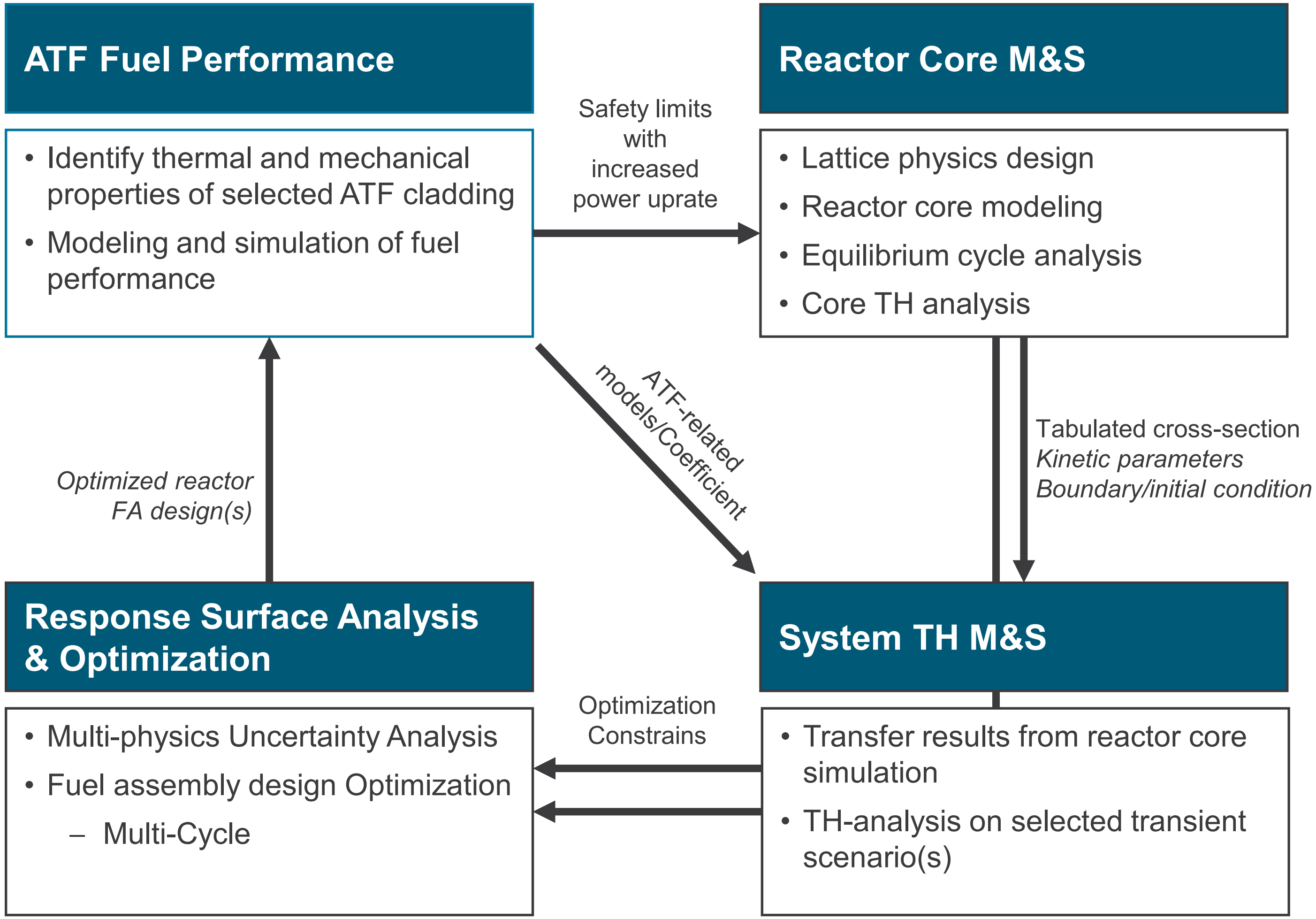
Power Uprates Core Design and Safety Analyses

U.S. NRC – INL RISA Project Status Update

LWRS POWER UPRATE – Project Overview

Project Description

- **Identify engineering design criteria of ATF for power uprate**
- **ATF + Extended Enrichment + (possibly) HBU**
  - ATF can…
    - reduce oxidation kinetic
    - reduce hydrogen production & hydrogen pick up
    - improve post-quench ductility
    - improve corrosion resistance
  - Dopped Pellets has…
    - higher density
    - higher burnup support
    - higher plasticity at high temperature
    - better fission gas retention
    - improved PCI resistance
- **Utilize existing data/models/methods first for ATF safety evaluation**
  - Additional experiments in need can be performed



Overall Approach for Sizable PWR Power Uprates

- **Approach:** ATF + EE (extended enrichment) + (possible) HBU
- **Near-term ATF concepts:** existing data, models, and methods can be used for its safety evaluations
  - (Primarily) Cr-coated Zr cladding
    - Significantly reduced oxidation kinetic
    - Significantly reduced hydrogen production and hydrogen pick up
    - Improved post-quench ductility
    - Improved corrosion resistance
  - (Optionally) Doped pellets
    - Higher density, can support higher burnup
    - Higher plasticity at high temperature
    - Better fission gas retention
    - Improved PCI resistance

LWRS-Developed Framework

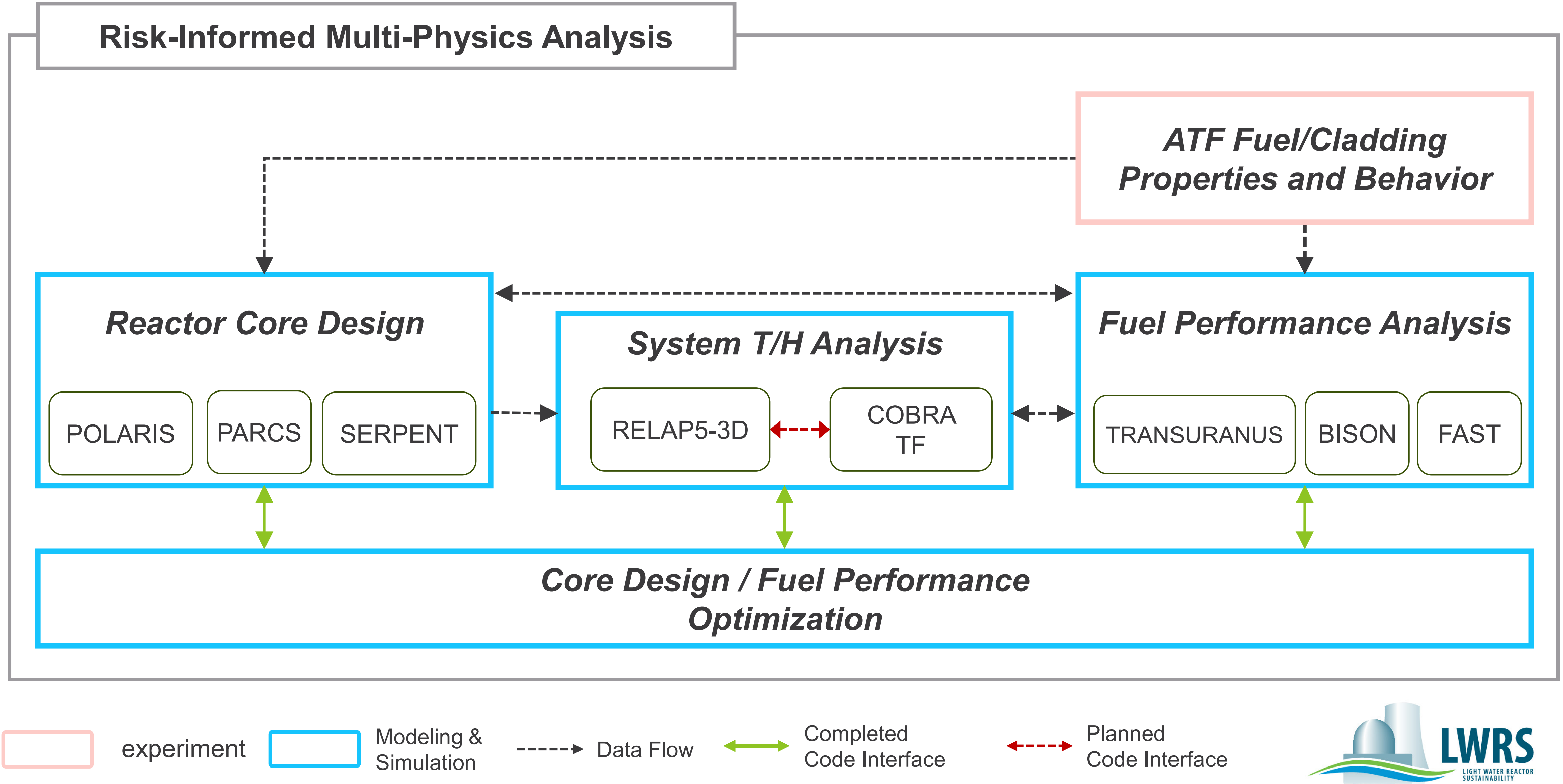
- **Polaris/PARCS for multi-cycle depletion simulation**
  - Fuel operating history
  - Accident initiation
- **RELAP5-3D for core/system thermal-hydraulics analysis**
  - RELAP5-3D allows the user to implement new ATF thermal material properties **including oxidation reaction rates**.
  - Clad deformation model available for Zircaloy clad.
  - **3D neutronics** included (no coupling needed).
- **FAST will be coupled in the later phase of the project to provide steady-state and transient analysis**
- **Serpent for neutronics model verification**
- **RAVEN for response surface analysis and optimization**

LWRS-Developed Framework (cont'd)

Two possible approaches to evaluation of power uprates:

1. **Staged optimization approach ✓**
   - Core → system, steady-state → transient, single-physics → multiphysics
   - Pros: computationally efficient; no complicated coupling scheme
   - Status: multi-objective optimization of core design has been demonstrated; working on core-to-system informing scheme
   - Needs:
     - Relatively accurate surrogate safety limit (e.g., hot channel factors Fq & FΔH)
2. **Holistic multi-physics optimization approach**
   - Pro: incorporation of experimentally determined safety limits (e.g., peak temp. during transient); avoid use of surrogate limits
   - Needs:
     - Experimentally determined safety limits
     - ML surrogate model to accelerate optimization

LWRS Power Uprate – Workflow



Reactor Core and System Design Problem

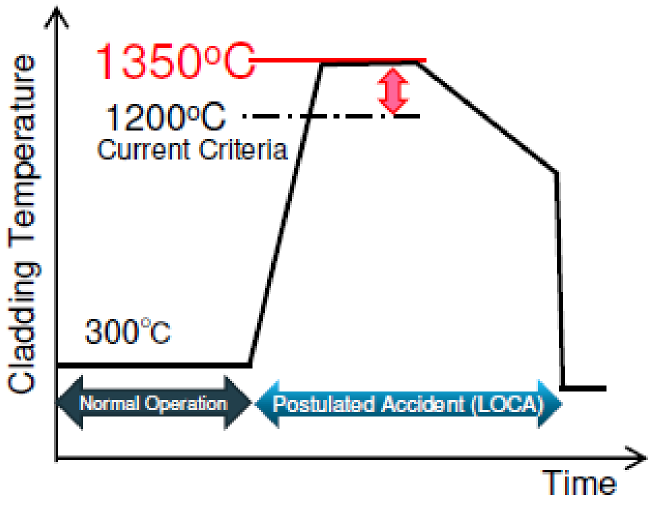
- **Design objectives:**
  - Sizable (~20%) power uprates for a generic PWR plant with minimal increase in fuel cost
- **Design variables**
  - Core reloading scheme
  - Fuel assembly (enrichment, rod dimension, lattice configuration, etc.) and control rod design
  - Plant operating conditions: flowrate, temperature, etc.
- **Design constraints**
  - Safety: hot channel factors, critical boron concentration, reactivity feedback coefficients, shutdown margin, etc.
  - Performance: burnup, enrichment, etc.
  - Economics: reloading cycle length, component upgrade, etc.

Connections to Ongoing R&D Activities

- **LWRS framework provides holistic core/system analysis capability to support power uprates**
  - Flexible design perturbations and efficient multi-scale and multiphysics calculation
  - Provide best-estimate fuel operating history for experimentalist
  - **Staged design and optimization option** is preferred at this moment due to low computational budget and development status
- **On-going work**
  - Small adjustments to enable power uprate optimization
- **Planned work**
  - Transition to holistic multiphysics optimization approach to enable further optimization and potentially larger uprates

Connections to Ongoing R&D Activities (cont.)

- **Need input from ATF (and HBU) experimental campaign**
  - Obtain and update correlations used in fuel performance code
  - How to translate new thermal and mechanical limit of ATF failure to constraints used in core optimization
  - For example:
    - Increase power output leads to reduced margin for hot channel factors (F_q & $F_{\Delta H}$)
    - ATF can help maintain the margin due to elevated temperature criterion
    - How to correlate temperature criterion during LOCA with linear heat generation rate (LHGR) and hot channel factors?



Murakami (2023)*

* Murakami (2023) Development of Cr coated zirconium alloy fuel cladding: Progress in 2022. Annual EPRI/DOE/INL Joint Combined Workshop on Accident Tolerant Fuel and Higher Burnup. April 2023.

Reactor Core Design

ATF Fuel Performance Experiments

	Thermal / Mechanical Properties	Chemical Behavior
SiC	• Thermal conductivity as a function of irradiation temperature • Fuel-clad averaged gap thermal conductivity • Brittle fracture Weibull parameters for the range of (expected) clad. temperature in the uprated-core. • Validation of effective thermal parameters	• Determination of precise oxidation reaction parameters
Cr-coated Zr	• Rod bowing correlation; Variation on coating thickness; Steady-state strain limits; High temperature burst stress and ballooning strain studies; • Post-quench ductility limit • Fretting wear tests (Ex-reactor) • Grid elements tests to ensure they are not damaged by the hard coating. • Coating delamination mechanisms	• Determination of precise oxidation reaction parameters
FeCrAl	• Radiation effects including both single variable and integral tests	• Determination of precise oxidation reaction parameters

New ATF Safety Limits →

Reactor Core Modeling & Simulation

Plant ReLoad Optimization (PRLO) *with* ATF

SERPENT	• Neutronics model verification
PARCS (or POLARIS)	• Multi-cycle depletion simulation – Fuel operating history • Analyze RIAs with a new uprated-core
FAST	• steady-state and transient fuel performance analysis
RAVEN	• Response surface Analysis • Optimization

Reactor Core Design

Design Objective	• Sizable (~20%) power uprates for a generic PWR plant with minimal increase in fuel cost
Design Variable	• Core reloading scheme • FA (enrichment, rod dimension, lattice configuration…) and control rod design • Plant operating conditions: flowrate, temperature, etc.
Design Constraints	• Safety – FΔH, critical boron concent., reactivity feedback coefficients, shutdown margin, etc. • Performance – burnup, enrichment, etc. • Economic: reloading cycle length, etc.

System Thermal hydraulics Analysis

Reactor Core Modeling & Simulation

Tabulated cross-section

Kinetic parameters

Boundary/ initial condition

System T/H Modeling & Simulation

Transfer results to T/H System code

- Update core cross sections, control rod positions *(On-going)*
- Update clad deformation model with ATF data *(Planned)*
- Update oxidation correlation for ATF data *(Planned)*

- Compute steady state temperature distribution in reactor core

ATF Models and Coefficients

Fuel Performance Modeling & Simulation

ATF-integrated TH System Model

TH-analysis on selected transient scenarios

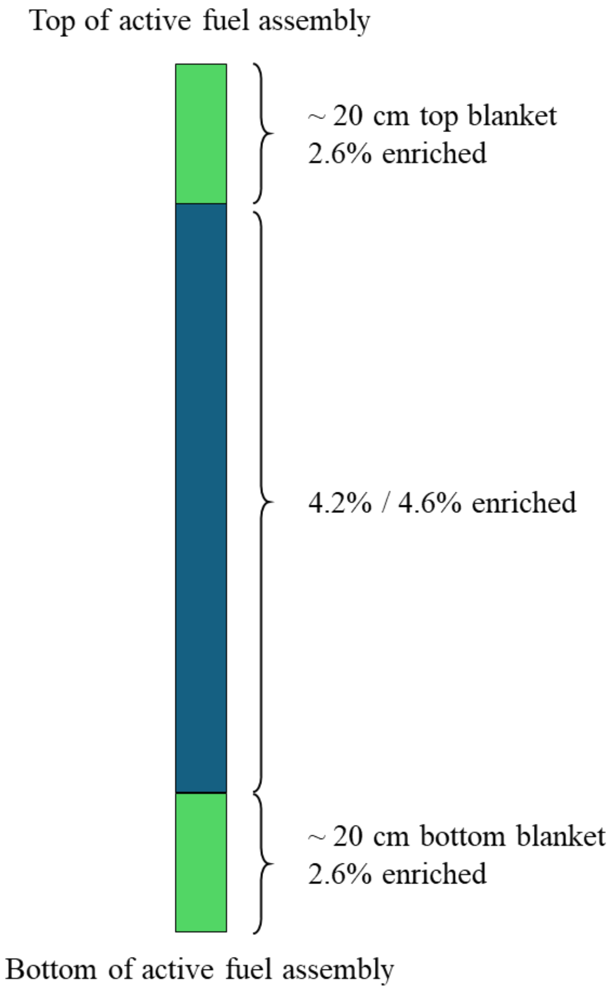
- Monitor ATF parameters – thermal & mechanical limits
- Determine post-quench ductility limit
- Vary accident initiation and safety system response parameters
  - Sufficient degree of margin?
  - “Hotspots” or vulnerable scenarios?

Neutronic design

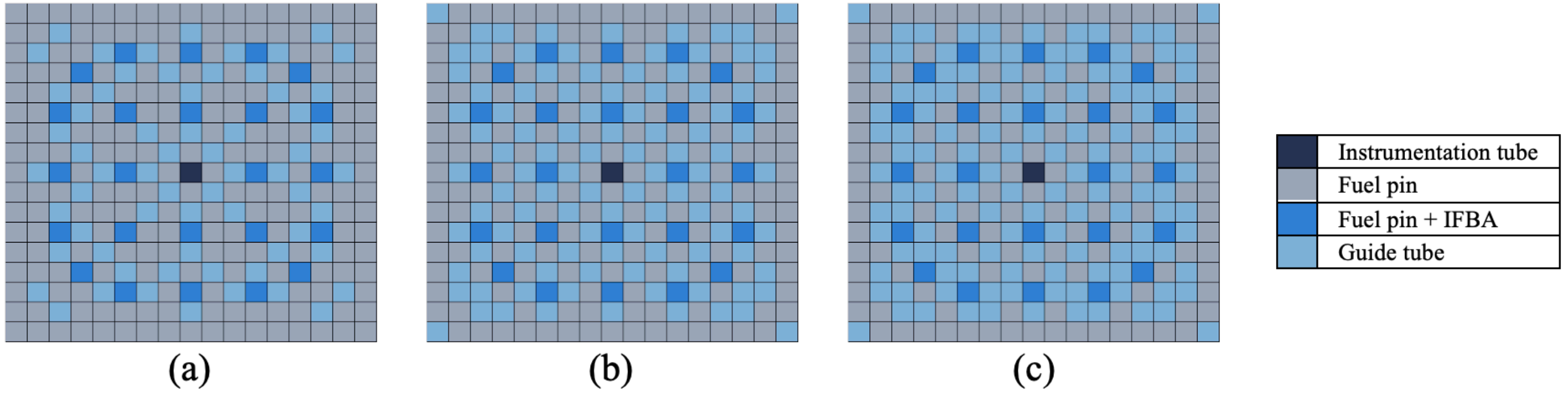
- Core developed based on publicly available information
- Reference reactor: South Texas Project
  - Westinghouse 4 loops
  - 14 ft core
  - 3.85 GW_{th}
- Development of core model based on previous works
- Scope:
  - extend core cycle to >18 months (2 years)
  - increase reactor power

Neutronic design

- Fuel rods with different axial enrichments
- Four types of 17x17 fuel assemblies (FA), with different number of Integrated Fuel Burnable Absorbers (IFBA) and enrichment

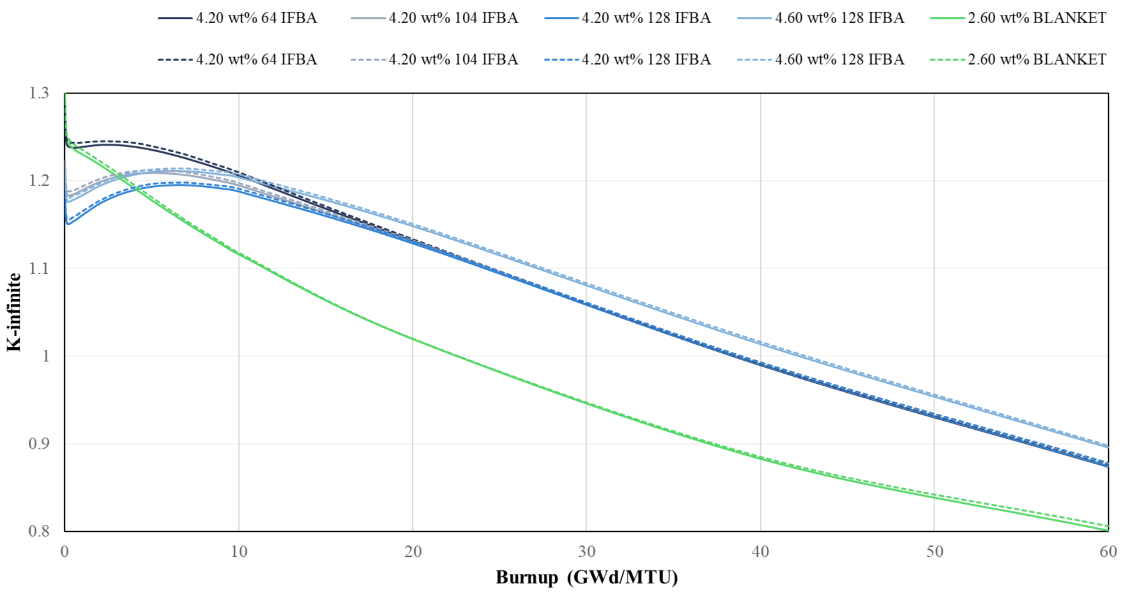


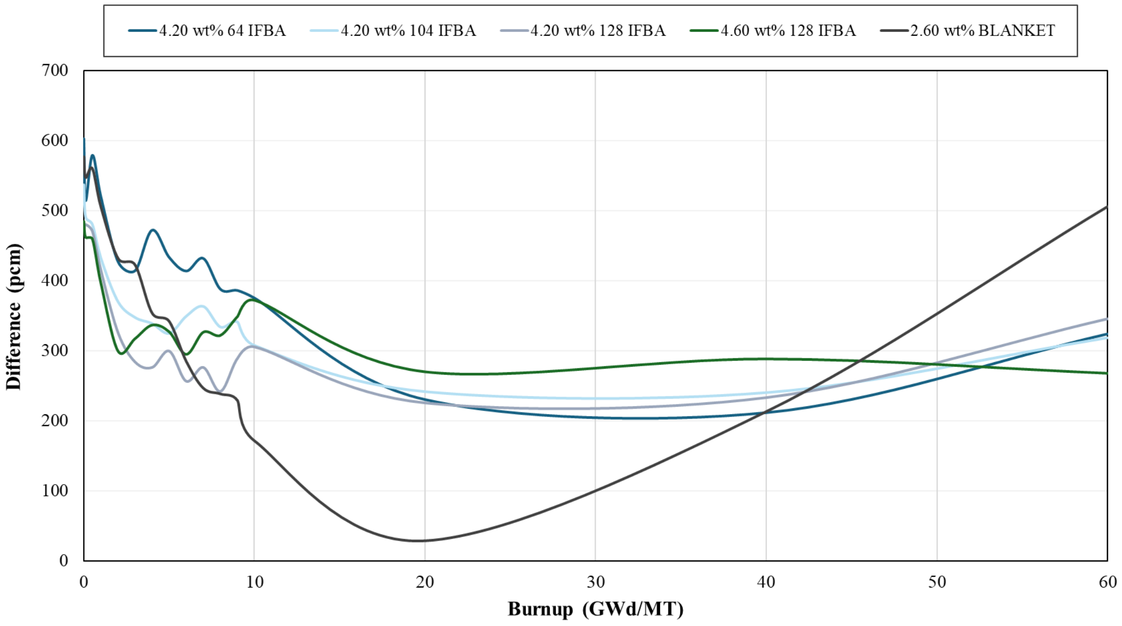
Assembly (ID)	Enrichment (wt.%)	Burnable poison loading (IFBA)
A194	4.2	64
A195	4.2	104
A196	4.2	128
A197	4.6	128



Neutronic design

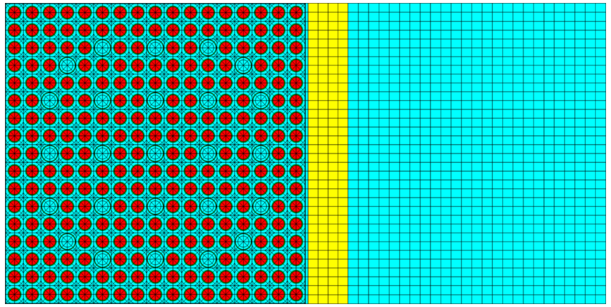
- FA cross-sections (Xsec) database calculated using 2D transport code POLARIS (part of SCALE code package)
- Models benchmarked against continuous-energy Monte Carlo code SERPENT2
- Deviations acceptable: ~100s of pcm





Neutronic design

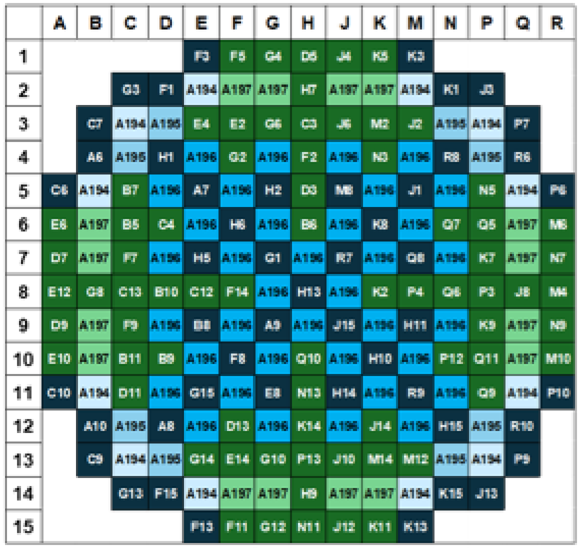
- Reflector modeled using colorsets



- Xsec parametrization

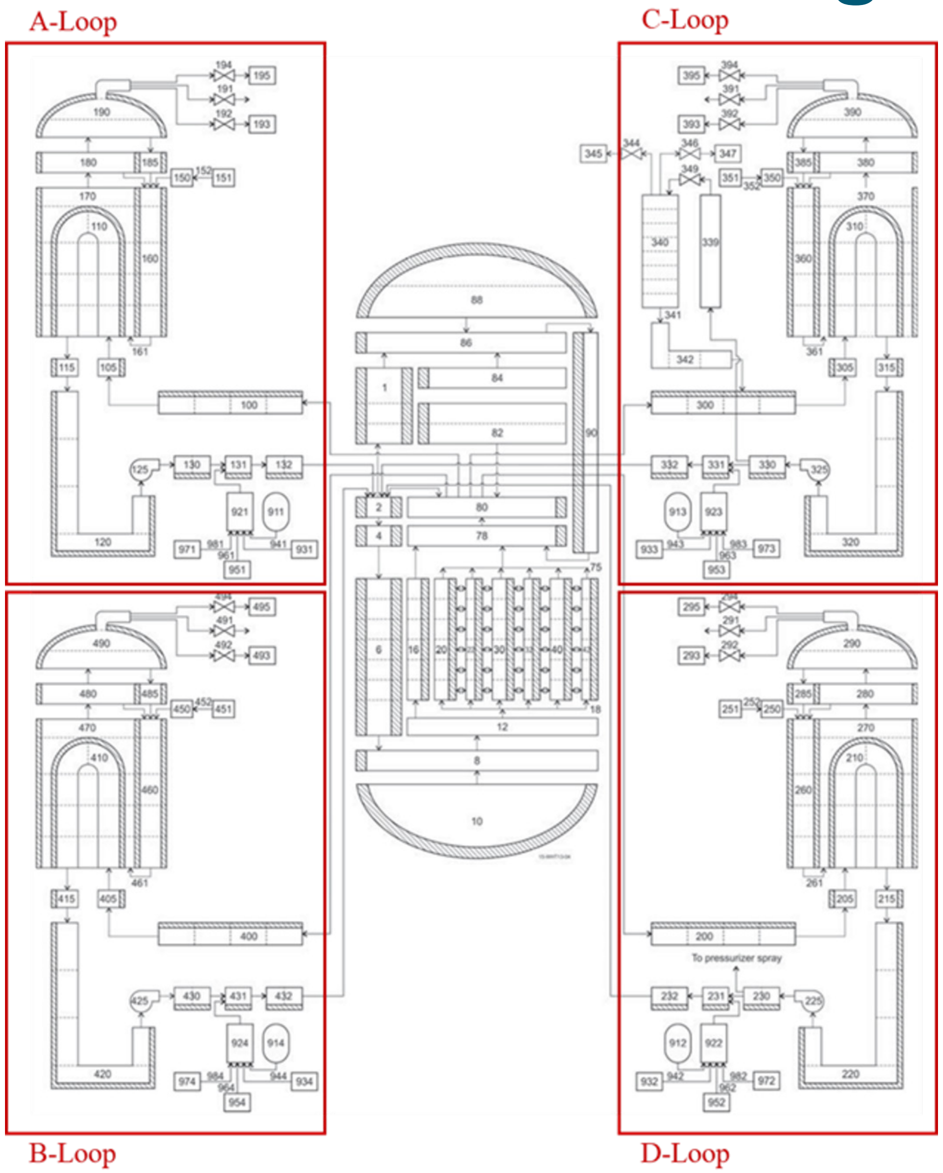
State Variable	Values				
Fuel Temperature (K)	560	900	1200	2000	
Coolant Density (g/cm^3)	0.102	0.200	0.450	0.653	0.740
Boron Concentration (PPM)	0	500	1500		
Control Rods Inserted	Fully Removed	Fully Inserted			

- Equilibrium core



Key	Description
A194	4.2 wt.% + 64 IFBA
A195	4.2 wt.% + 104 IFBA
A196	4.2 wt.% + 128 IFBA
A197	4.6 wt.% + 128 IFBA
OB	Once burned
TB	Twice burned

RELAP5-3D modeling



- Initial PWR model from a previous LWRS work (INL/EXT-16-39805)
    - Reactor vessel
    - 4 RCS loops
    - Pressurizer in the C-loop
    - Partial BOP: SG, steam lines, steam valves
    - Main & aux feedwater systems
    - ECCS: high & low pressure
- Key model adjustments:
    - Core hydrodynamic volumes
    - Core heat structures
    - Reactor kinetics computation model

LWRS LIGHT WATER REACTOR SUSTAINABILITY

Core Model

	A	B	C	D	E	F	G	H	J	K	M	N	P	Q	R	S	T
1					7	7	7	7	7	7	7	7	7				
2			7	7	7	6	6	6	6	6	6	6	7	7	7		
3		7	7	6	6	6	5	5	5	5	5	6	6	6	7	7	
4		7	6	6	5	5	5	4	4	4	5	5	5	6	6	7	
5	7	7	6	5	5	4	4	4	3	4	4	4	5	5	6	7	7
6	7	6	6	5	4	4	3	3	3	3	3	4	4	5	6	6	7
7	7	6	5	5	4	3	3	2	2	2	3	3	4	5	5	6	7
8	7	6	5	4	4	3	2	2	1	2	2	3	4	4	5	6	7
9	7	6	5	4	3	3	2	1	1	1	2	3	3	4	5	6	7
10	7	6	5	4	4	3	2	2	1	2	2	3	4	4	5	6	7
11	7	6	5	5	4	3	3	2	2	2	3	3	4	5	5	6	7
12	7	6	6	5	4	4	3	3	3	3	3	4	4	5	6	6	7
13	7	7	6	5	5	4	4	4	3	4	4	4	5	5	6	7	7
14		7	6	6	5	5	5	4	4	4	5	5	5	6	6	7	
15		7	7	6	6	6	5	5	5	5	5	6	6	6	7	7	
16			7	7	7	6	6	6	6	6	6	6	7	7	7		
17					7	7	7	7	7	7	7	7	7				

- 17x17 core assemblies
  - 7 hydrodynamic volumes
  - 6 multiple-junctions for channel cross-flows
  - 7 heat structures

HS	No. of assemblies
1	5 fuel assy.
2	16 fuel assy.
3	28 fuel assy.
4	40 fuel assy.
5	48 fuel assy.
6	56 fuel assy.
7	64 “reflector assy.”

193 FAs (HS 1–6)

@ 19 axial nodes (HS 1–7)

- Heat structure power generation is calculated using reactor kinetics model.

Connecting Core Model to Reactor Kinetics

- Assign thermal-hydraulic zones to kinetics nodes using zone figures (Cards 3002ZZ01-3002ZZ99). Sample:

30020101	0	0	0	0	7	7	7	7	7	7	7	7	7	0	0	0	0
30020102	0	0	7	7	7	6	6	6	6	6	6	6	7	7	7	0	0
30020103	0	7	7	6	6	6	5	5	5	5	5	6	6	6	7	7	0
30020104	0	7	6	6	5	5	5	4	4	4	5	5	5	6	6	7	0
30020105	7	7	6	5	5	4	4	4	3	4	4	4	5	5	6	7	7
30020106	7	6	6	5	4	4	3	3	3	3	3	4	4	5	6	6	7
30020107	7	6	5	5	4	3	3	2	2	2	3	3	4	5	5	6	7
30020108	7	6	5	4	4	3	2	2	1	2	2	3	4	4	5	6	7
30020109	7	6	5	4	3	3	2	1	1	1	2	3	3	4	5	6	7
30020110	7	6	5	4	4	3	2	2	1	2	2	3	4	4	5	6	7
30020111	7	6	5	5	4	3	3	2	2	2	3	3	4	5	5	6	7
30020112	7	6	6	5	4	4	3	3	3	3	3	4	4	5	6	6	7
30020113	7	7	6	5	5	4	4	4	3	4	4	4	5	5	6	7	7
30020114	0	7	6	6	5	5	5	4	4	4	5	5	5	6	6	7	0
30020115	0	7	7	6	6	6	5	5	5	5	5	6	6	6	7	7	0
30020116	0	0	7	7	7	6	6	6	6	6	6	6	7	7	7	0	0
30020117	0	0	0	0	7	7	7	7	7	7	7	7	7	0	0	0	0

17x17, iterate for 19 axial heights

- Assign heat structure compositions to kinetics nodes using composition figures (Cards 3003CC01-3003CC99). Sample:

30030201	0	0	0	0	1	1	1	1	1	1	1	1	1	0	0	0	0
30030202	0	0	1	1	1	3	4	5	6	5	4	3	1	1	1	0	0
30030203	0	1	1	7	8	9	10	11	12	11	10	9	8	7	1	1	0
30030204	0	1	13	14	15	16	17	18	19	18	17	16	15	14	13	1	0
30030205	1	1	20	20	21	22	23	24	25	24	23	22	21	20	19	1	1
30030206	1	26	27	28	29	30	31	32	33	32	31	30	29	28	27	26	1
30030207	1	34	35	36	37	38	39	40	41	40	39	38	37	36	35	34	1
30030208	1	42	43	44	45	46	47	48	49	48	47	46	45	44	43	42	1
30030209	1	6	12	19	25	33	41	49	50	49	41	33	25	19	12	6	1
30030210	1	42	43	44	45	46	47	48	49	48	47	46	45	44	43	42	1
30030211	1	34	35	36	37	38	39	40	41	40	39	38	37	36	35	34	1
30030212	1	26	27	28	29	30	31	32	33	32	31	30	29	28	27	26	1
30030213	1	1	20	20	21	22	23	24	25	24	23	22	21	20	19	1	1
30030214	0	1	13	14	15	16	17	18	19	18	17	16	15	14	13	1	0
30030215	0	1	1	7	8	9	10	11	12	11	10	9	8	7	1	1	0
30030216	0	0	1	1	1	3	4	5	6	5	4	3	1	1	1	0	0
30030217	0	0	0	0	1	1	1	1	1	1	1	1	1	0	0	0	0

17x17, iterate for 19 axial heights

NESTLE Reactor Kinetics Model

- NESTLE (**Nodal** Eigenvalue, Steady-state, Transient, Le core Evaluator) solves few-group **neutron diffusion**
- NESTLE uses a nested non-linear iterative solution strategy that requires thermal-hydraulic feedback
  - From hydrodynamic volumes (Cards 31ZZZZ1N1-31ZZZZ1N9). Sample:

310001111	20010000	1.0	1.0	1.0

  - From heat structures (Cards 31ZZZZ2N1-31ZZZZ2N9). Sample:

310001211	7000001	1.0

- Process: TH model → temperature & density → correct cross sections.

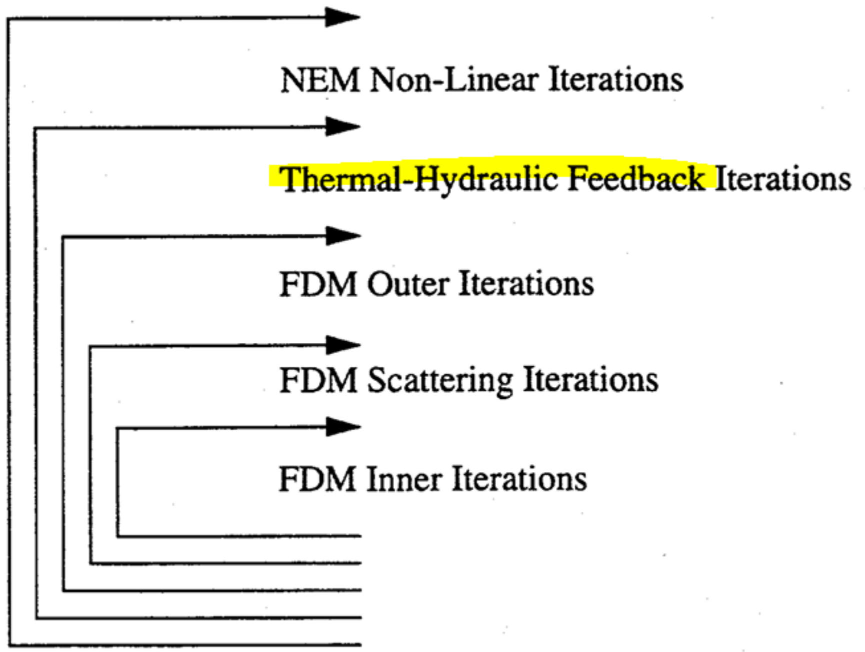


Figure 1: Overview of NESTLE nested iterative solution strategy

NESTLE Neutron Cross Section

Core assembly configuration

				R	R	R	R	R	R	R	R	R				
		R	R	R	64	128A	128	128A	128	128A	64	R	R	R		
	R	R	128A	128A	64	128A	128A	104	128A	128A	64	128A	128A	R	R	
	R	128A	64	104	128A	64	128A	64	128A	64	128A	104	64	128A	R	
R	R	128A	104	128A	128A	128A	128	128A	128	104	128A	128A	104	128A	R	R
R	64	64	128A	128A	128	128A	104	104	104	128A	128	128A	128A	64	64	R
R	128A	128A	64	104	128A	128A	128A	128A	128A	128A	128A	128A	64	128A	128A	R
R	128	128A	128A	128	104	128A	128	104	128	128A	104	128	128A	128A	128	R
R	128A	104	64	128A	104	128A	104	64	104	128A	104	128A	64	104	128A	R
R	128	128A	128A	128	104	128A	128	104	128	128A	104	128	128A	128A	128	R
R	128A	128A	64	128A	128A	128A	128A	128A	128A	128A	128A	104	64	128A	128A	R
R	64	64	128A	128A	128	128A	104	104	104	128A	128	128A	128A	64	64	R
R	R	128A	104	128A	128A	104	128	128A	128	128A	128A	128A	104	128A	R	R
	R	128A	64	104	128A	64	128A	64	128A	64	128A	104	64	128A	R	
	R	R	128A	128A	64	128A	128A	104	128A	128A	64	128A	128A	R	R	
		R	R	R	64	128A	128	128A	128	128A	64	R	R	R		
				R	R	R	R	R	R	R	R	R				

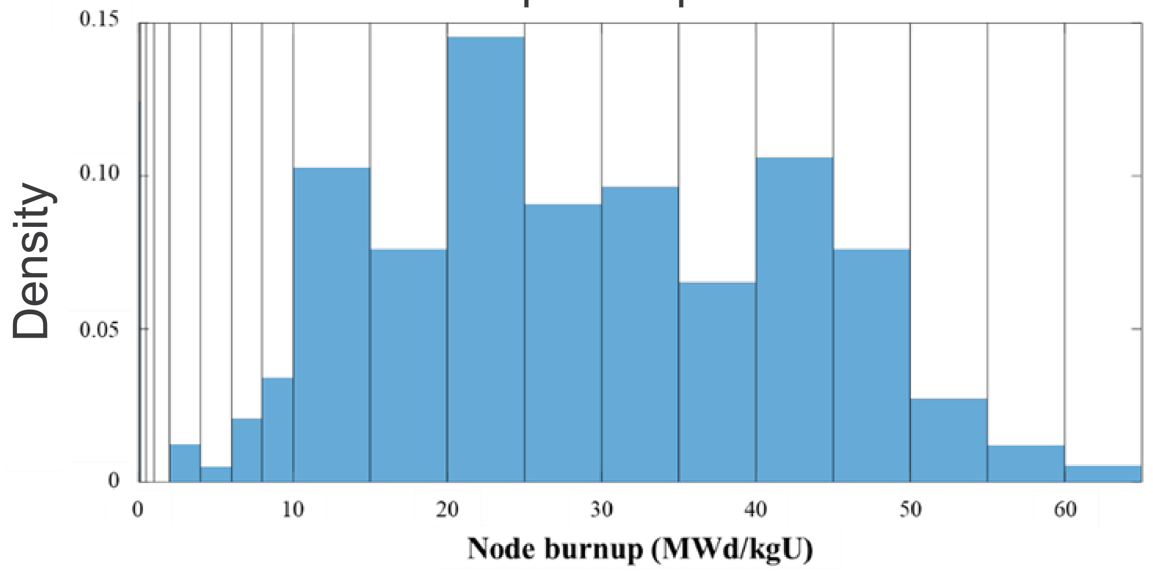
64 : 64 IFBA + 4.2 w.t.%
104 : 104 IFBA + 4.2 w.t.%
128 : 128 IFBA + 4.2 w.t.%
128A : 128 IFBA + 4.6 w.t.%
R : Reflector

- Cross sections as functions of:
  - Assembly ID (see figure)
  - fuel temperature
  - coolant temperature
  - coolant density
  - control rod state (in or out)
  - poison density
  - burnup
- Case matrix: 24 cases @ 19 depletion points

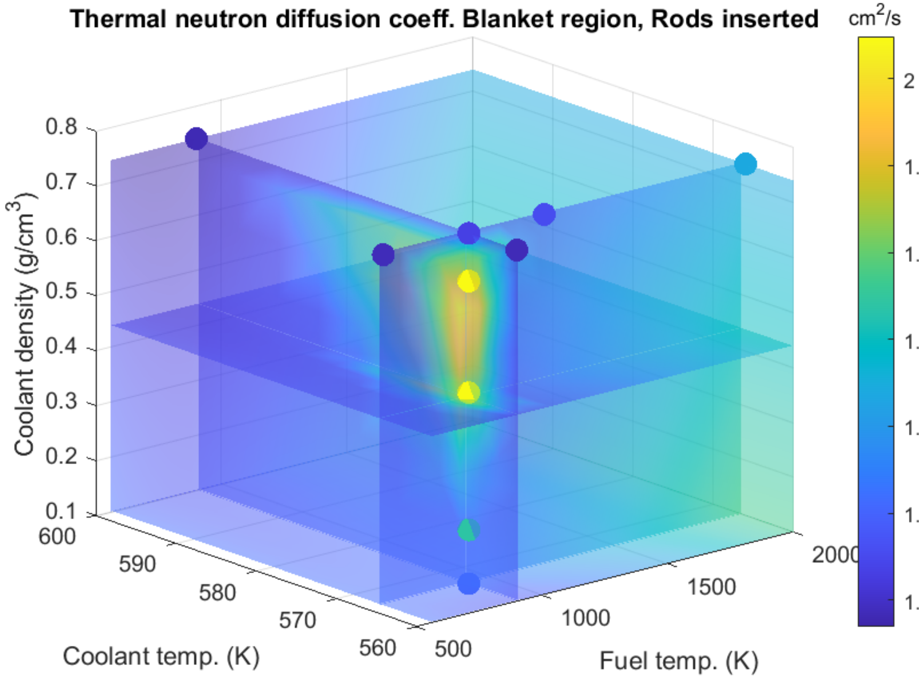
NESTLE Neutron Cross Section

Obtain cross sections for each node using scattered multivariate interpolation techniques

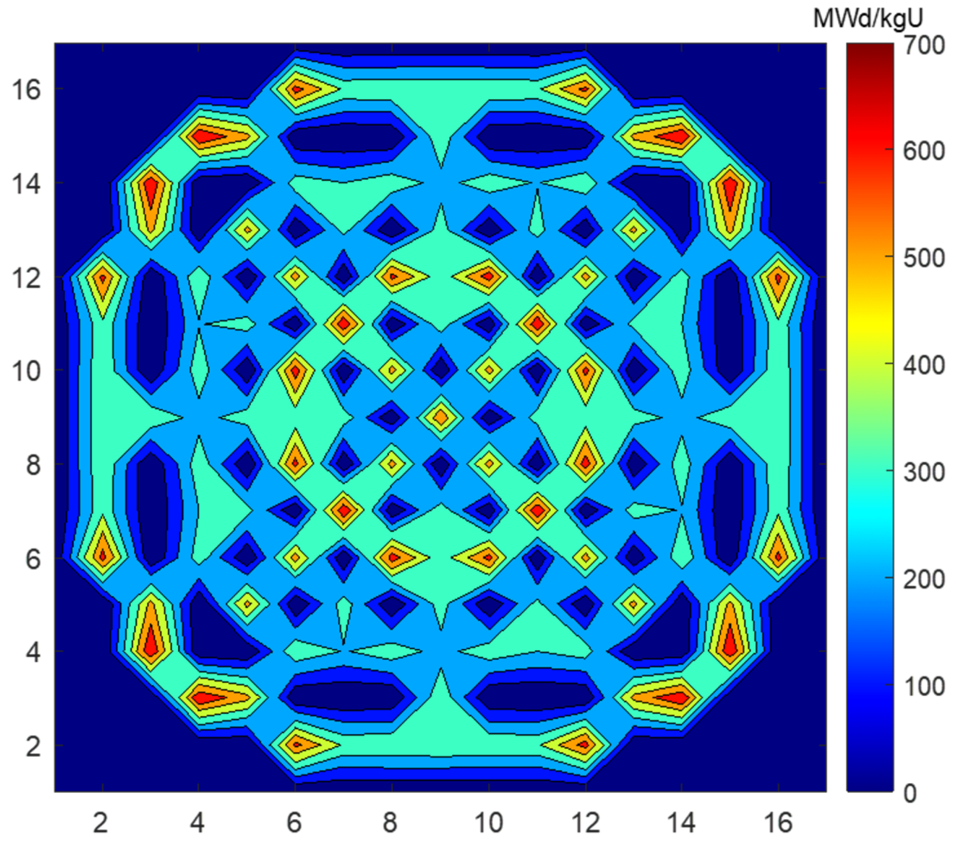
BOC node burnup histogram and sampled depletion points



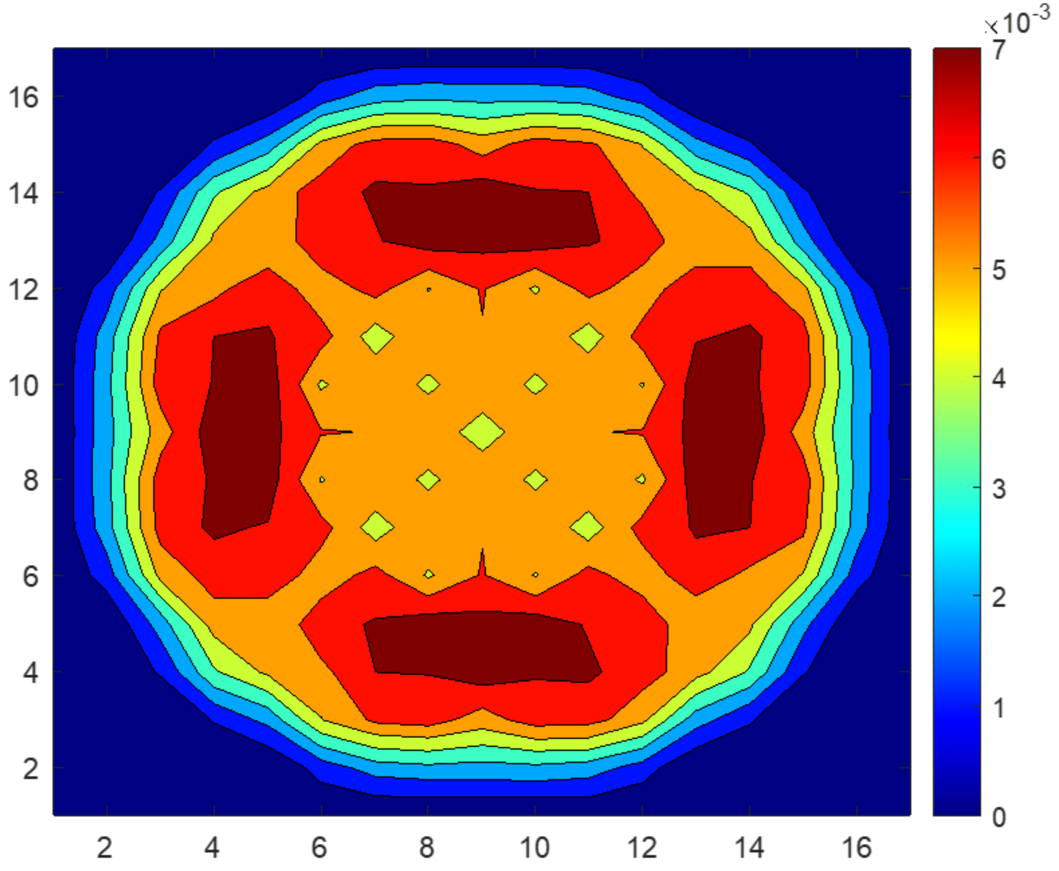
Sample interpolated response surface & known data points



Model Initialization



NESTLE neutron cross section is interpolated from POLARIS using burnup data (BOC visualized above) and steady-state TH data.



NESTLE zone power distribution is initialized using relative power fraction (RPF). BOC RPF is visualized above.

RELAP5-3D Modeling Status

- Modified core nodalization
- Verified nodalization closure
- Verified TH output using a simplified core power module
- Automated the import of POLARIS neutronic results into RELAP5-3D input deck
- Interpolate cross-section to various TH data → **in progress**
- Simulate steady-state using NESTLE nodal kinetics model
- Develop and simulate safety transients and AOO
  - Rod ejection accident
  - Pump rundown
  - Other RIAs

LWRS LIGHT WATER REACTOR SUSTAINABILITY

Sustaining National Nuclear Assets

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