Idaho Falls, ID

2019 TRTR Annual Meeting

September 22 – 26

Idaho National Laboratory

Welcome and INL Overview

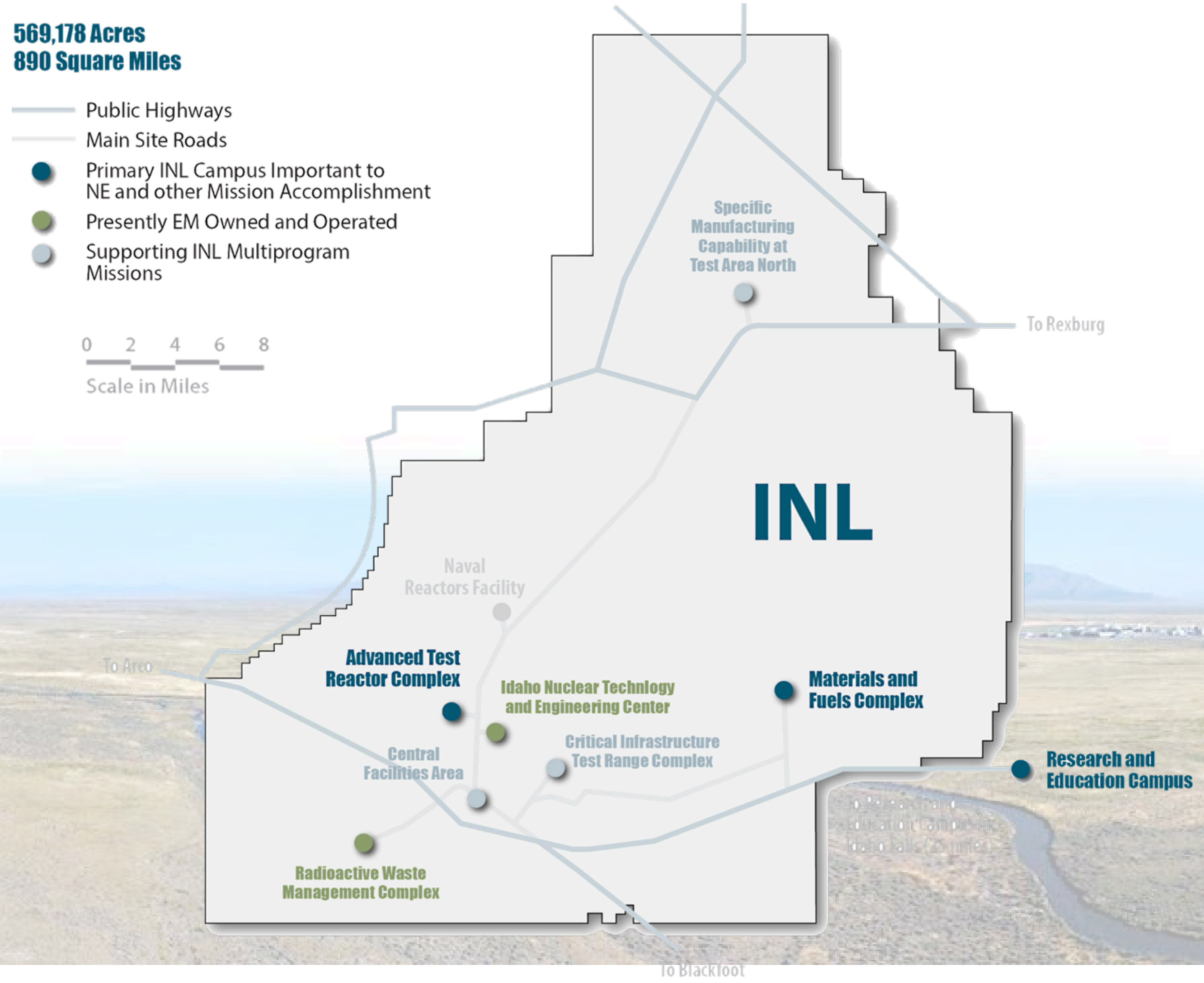
Marianne Walck, PhD
Deputy Laboratory Director S&T,
Chief Research Officer

September 23, 2019

www.inl.gov

Idaho National Laboratory

Idaho National Laboratory provides a unique capability for the Nation



16 Nuclear facilities (Haz Cat 1, 2 & 3)

44 Radiological facilities

4 Operating reactors

17.5 Miles railroad for shipping nuclear fuel

40 Miles primary roads (125 total)

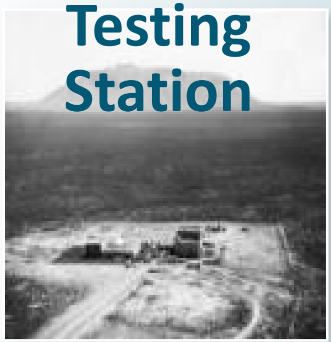
7 Substations with interfaces to three power providers

112 Miles high-voltage transmission lines

3 Fire stations

Idaho National Laboratory
Evolving to Meet the Nation's Needs for 70 Years

National Reactor Testing Station



Idaho National Engineering Laboratory

Energy Mission – Reactor Science, Safety and Sustainability Solutions

Idaho National Engineering and Environmental Laboratory

Environmental Management Mission

Idaho National Laboratory

INEEL & ANL-W combined to create the new Idaho National Laboratory

Nuclear Energy

National and Homeland Security

Energy and Environment

Argonne National Laboratory

Advancing Nuclear Energy

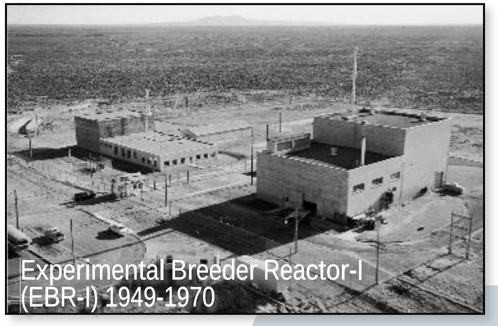
Securing & Modernizing Critical Infrastructure

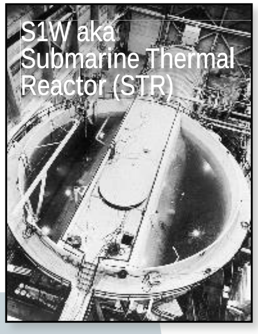
Enabling Clean Energy Systems

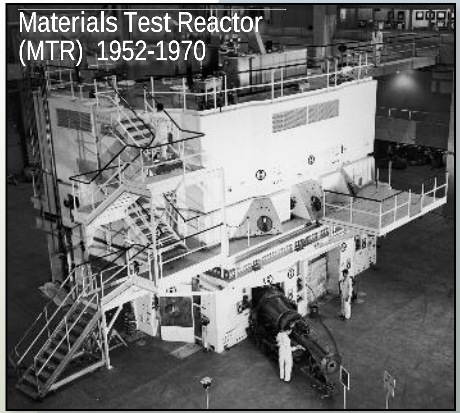
1949 | **1974** | **1997** | **2005** | **2019**

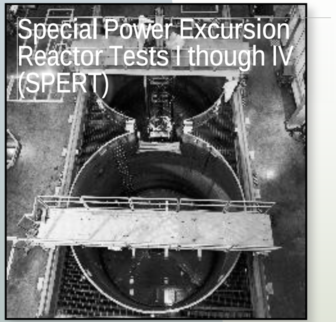
INL's beginning as the National Reactor Testing Station

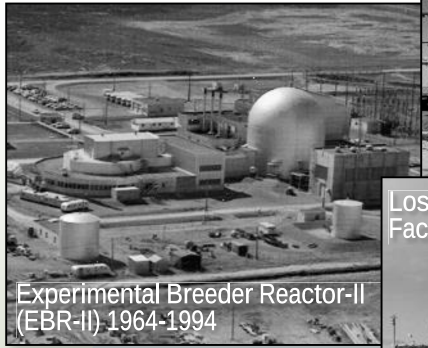
- Established in 1949 on 890 square miles of remote federal land
- Argonne's EBR-I was the first reactor for the nation's new test bed
- Materials Test Reactor (MTR) in 1952 to provide irradiation testing of fuels and materials for other reactors in planning stages
- Additional reactor concepts explored transient and other safety testing and demonstration
- **Over 52 reactors have operated on the INL**


Experimental Breeder Reactor-I (EBR-I) 1949-1970

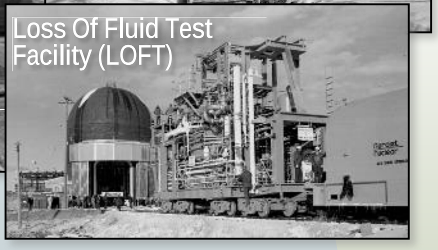

S1W aka Submarine Thermal Reactor (STR)

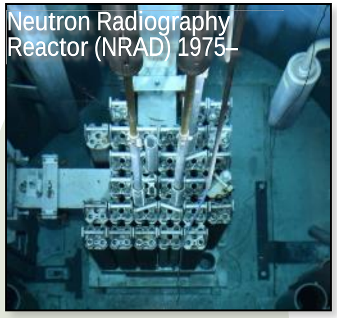

Materials Test Reactor (MTR) 1952-1970


Special Power Excursion Reactor Tests I though IV (SPERT)

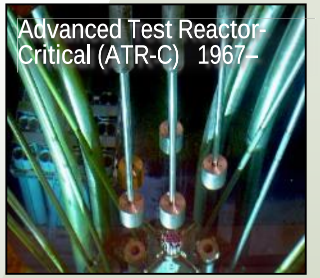

Experimental Breeder Reactor-II (EBR-II) 1964-1994

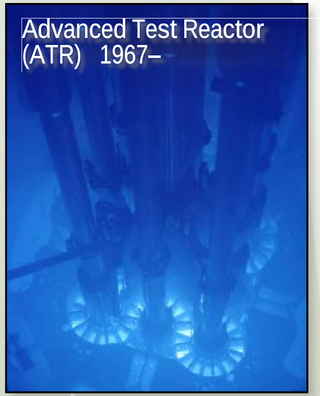

Power Burst Facility (PBF) 1972-1985


Loss Of Fluid Test Facility (LOFT)


Neutron Radiography Reactor (NRAD) 1975–


Transient Reactor Test Facility (TREAT) 1959-1994 2017--

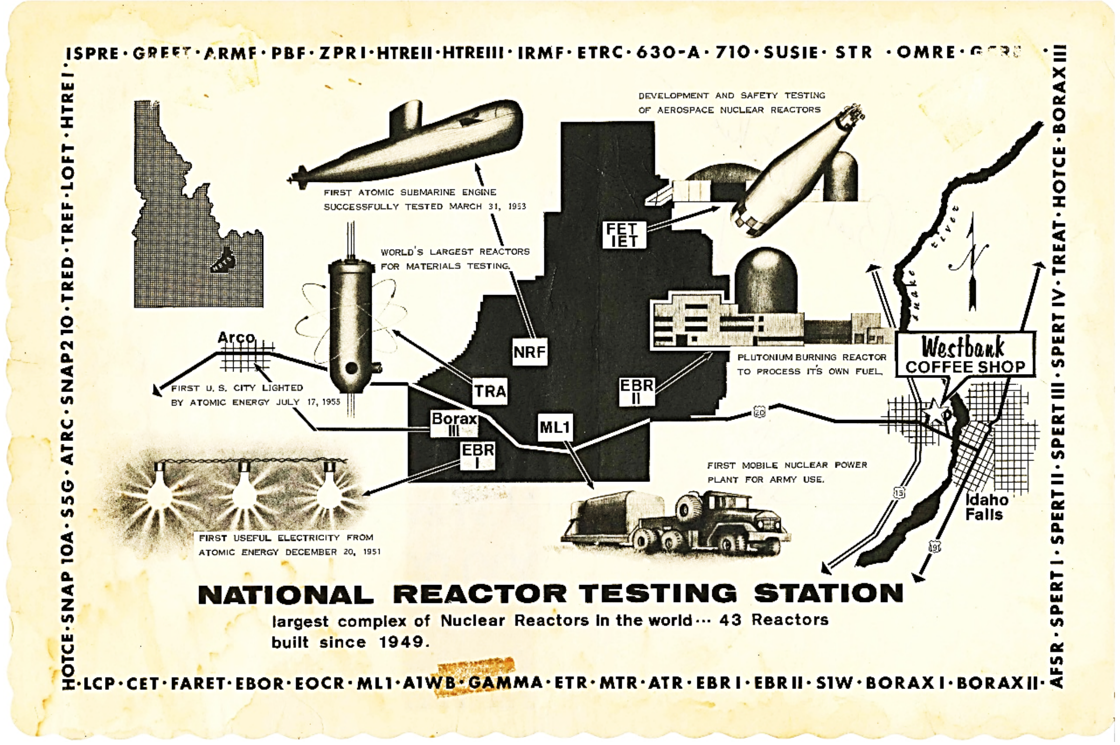

Advanced Test Reactor-Critical (ATR-C) 1967–


Advanced Test Reactor (ATR) 1967–

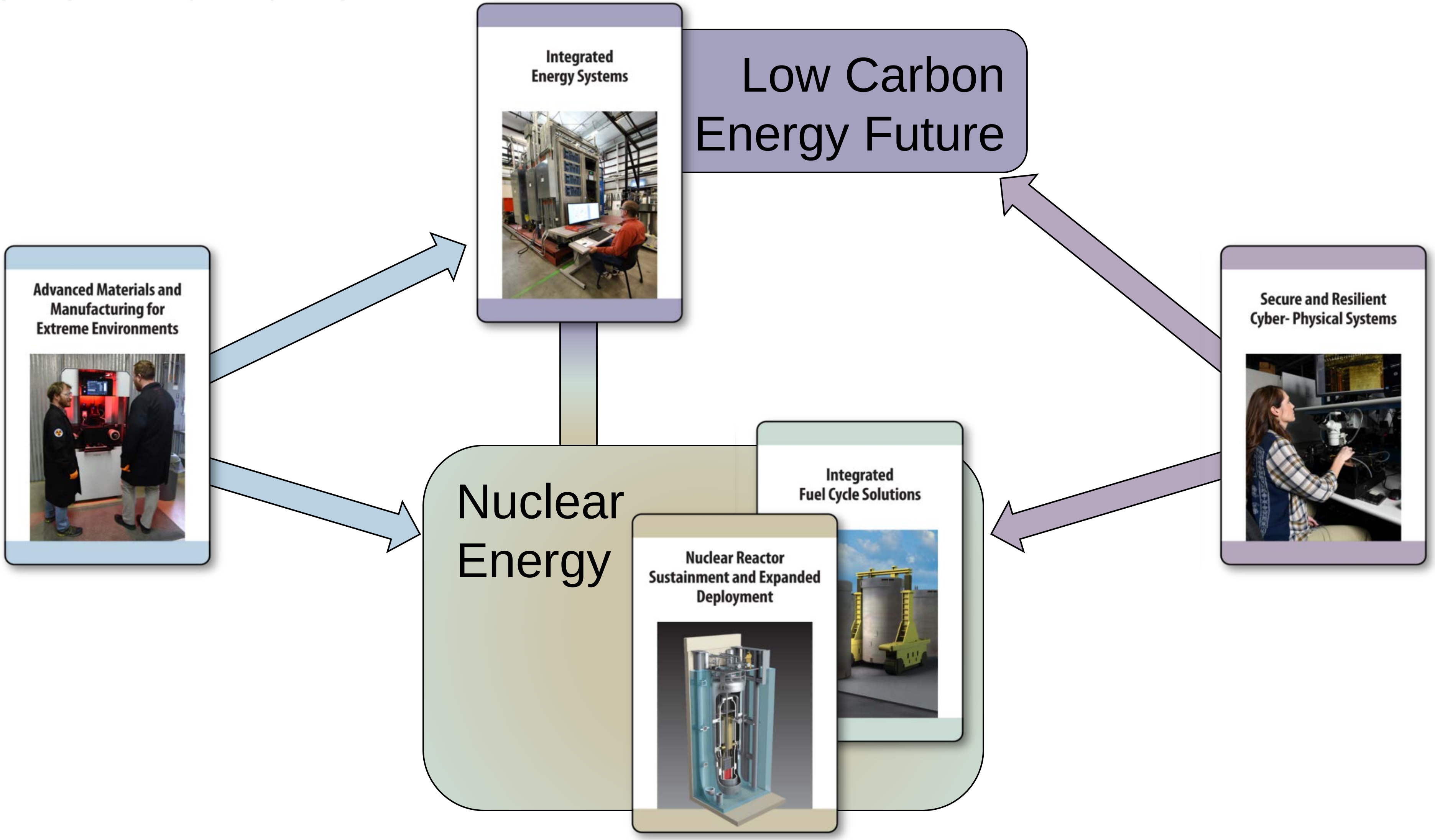
1949 | 1950s | 1960s | 1970s | 1980s | 1990s | 2000s | 2010s

The NRTS Provided Capabilities That Drove Nuclear Innovation

- **First** nuclear power plant
- **First** U.S. city to be powered by nuclear energy
- **First** submarine reactor tested
- **First** mobile nuclear power plant for the army
- **First** materials testing reactor
- Demonstration of self sustaining fuel cycle
  - EBR-II
- Basis for LWR reactor safety
  - LOFT, BORAX, SPERT
- Aircraft and aerospace reactor testing



INL's Current Strategic Focus Will Advance Energy and Security Goals for the Nation

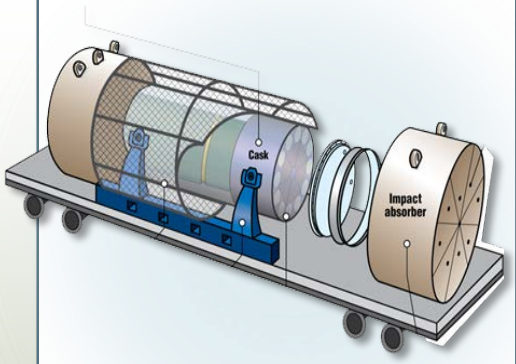


Creating the Next-Generation National Reactor Testing Station:

Advanced Reactor Pipeline Vision at Idaho National Laboratory

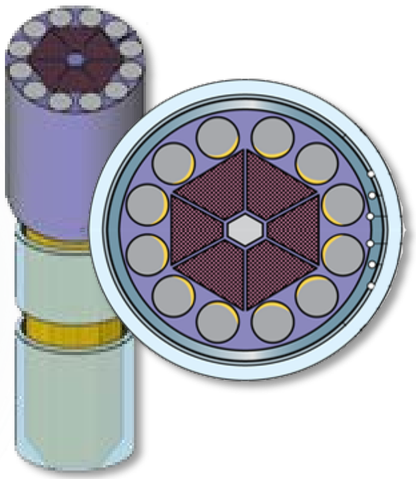
Microreactor (<10MW) demonstration by early 2020s

- Resolve key advanced reactor issues
- Open new markets for nuclear energy
- Provide a ‘win’ to build positive momentum



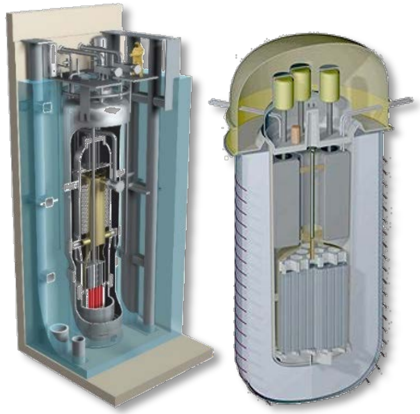
Commercial microreactors deployed

- Support deployment of of micro-reactors for key remote site power and process heat customers



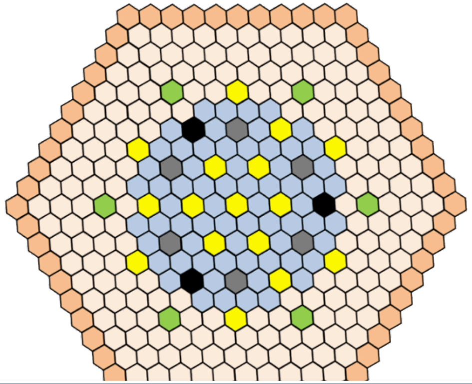
SMR(s) operating by 2026

- Enable deployment through siting and technical support
- Joint Use Modular Plant leased for federal RDD&D



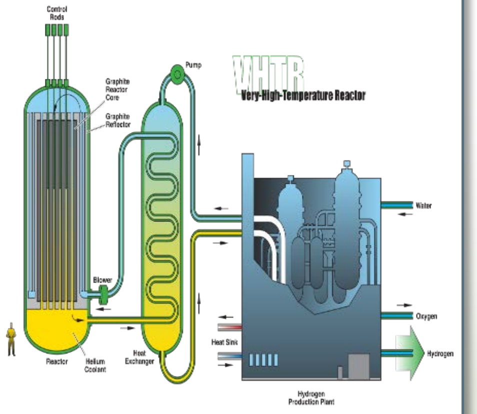
Versatile Test Reactor (VTR) operating by 2026

- Establish fast-spectrum testing and fuel development capability
- Support non-LWR advanced reactor demonstration



Non-LWR advanced demonstration reactor by 2030

- Demonstrate non-LWR technology replacement of US baseload clean power capacity



2023

2030

Idaho National Laboratory

August 15, 2019: The National Reactor Innovation Center Established at INL

1	2	3	4	5	6	7	8

Proof-of-Concept | *Proof-of-Performance* | *Proof-of-Operations*

R&D to Address Technical Feasibility

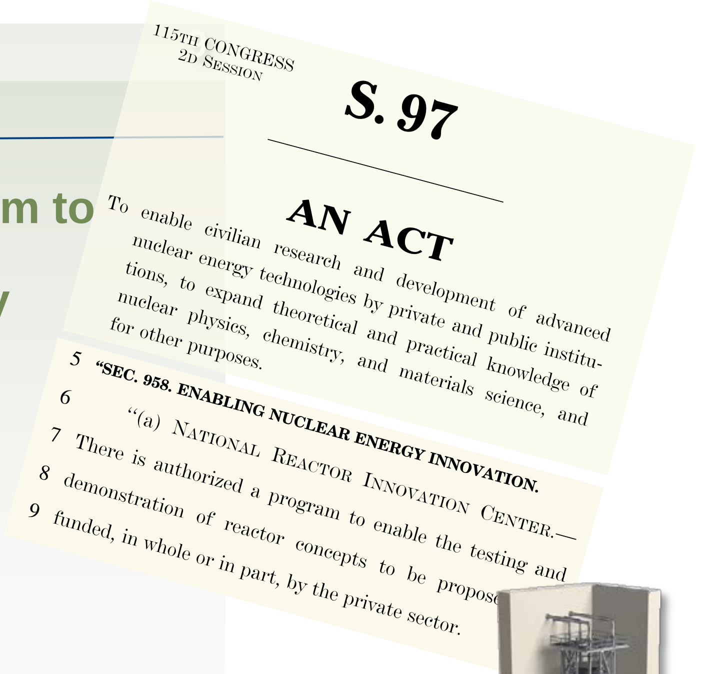
- Materials and Fuels
- Validated predictive modeling and simulation capabilities
- Experimental Capabilities

Establish Performance of Nuclear Technologies

- Validation data
- Irradiation and transient testing
- Irradiated materials characterization

Demonstration Platform to Address Economic/ Operational Feasibility

- Sites for demonstration
- Licensing Support
- Integrated energy systems support

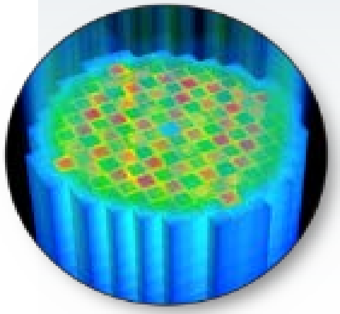

115TH CONGRESS
2D SESSION

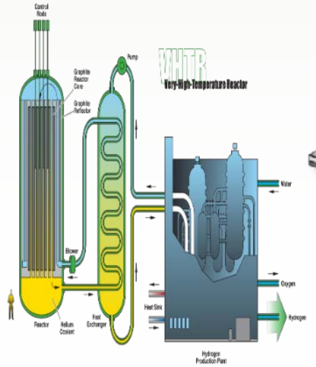
S. 97

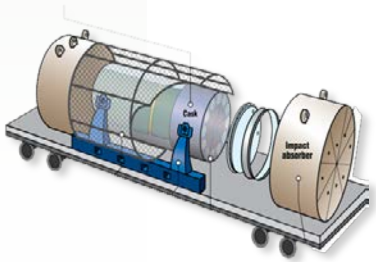
AN ACT

To enable civilian research and development of advanced nuclear energy technologies by private and public institutions, to expand theoretical and practical knowledge of nuclear physics, chemistry, and materials science, and for other purposes.

5 "SEC. 958. ENABLING NUCLEAR ENERGY INNOVATION.
6 "(a) NATIONAL REACTOR INNOVATION CENTER.—
7 There is authorized a program to enable the testing and
8 demonstration of reactor concepts to be propose
9 funded, in whole or in part, by the private sector.







NuScale Power

Thank You

Image credit: Third Way and Gensler

INL/MIS-19-55475

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A Life Cycle and Aging Management investment strategy to sustain long term strategic irradiations at ATR

Hans Vogel
Director, Strategic Irradiation Capabilities
Advanced Test Reactor

Idaho National Laboratory

Test Research and Training Reactors
Annual Conference
September 22 – 26, 2019
INL Meeting Center
Idaho Falls, ID

Advanced Test Reactor Complex History

- The ATR Complex has been host to Materials Test Reactor (MTR; 1950 - 1970), Engineering Test Reactor (ETR; 1956 - 1981), and the Advanced Test Reactor (ATR)

ATR:

- Conceptual design late 1950s
- Construction began early 1960s
- First criticality 1967
- Now in our 52nd year of operation
- Department of Energy has requested a study to sustain ATR capability for 60+ years…
- How do we better anticipate long term needs and make plans to identify and address them?

Aging Test Reactor Challenges

- Many of the designs were unique to the application
  - Systems were designed for very specific functions, often with one-of-a-kind or first-of-a-kind components
- Many of the original manufacturers no longer make the original components, or the original manufacturer is no longer in business
  - Example: original relays – were not readily available for replacement

- Additional challenges are reactor cycle times, with frequent start ups and shut downs puts additional stress on the equipment
  - A typical commercial PWR runs approximately 18 months between refueling outages
  - ATR runs 2 weeks to 2 months between fueling cycles

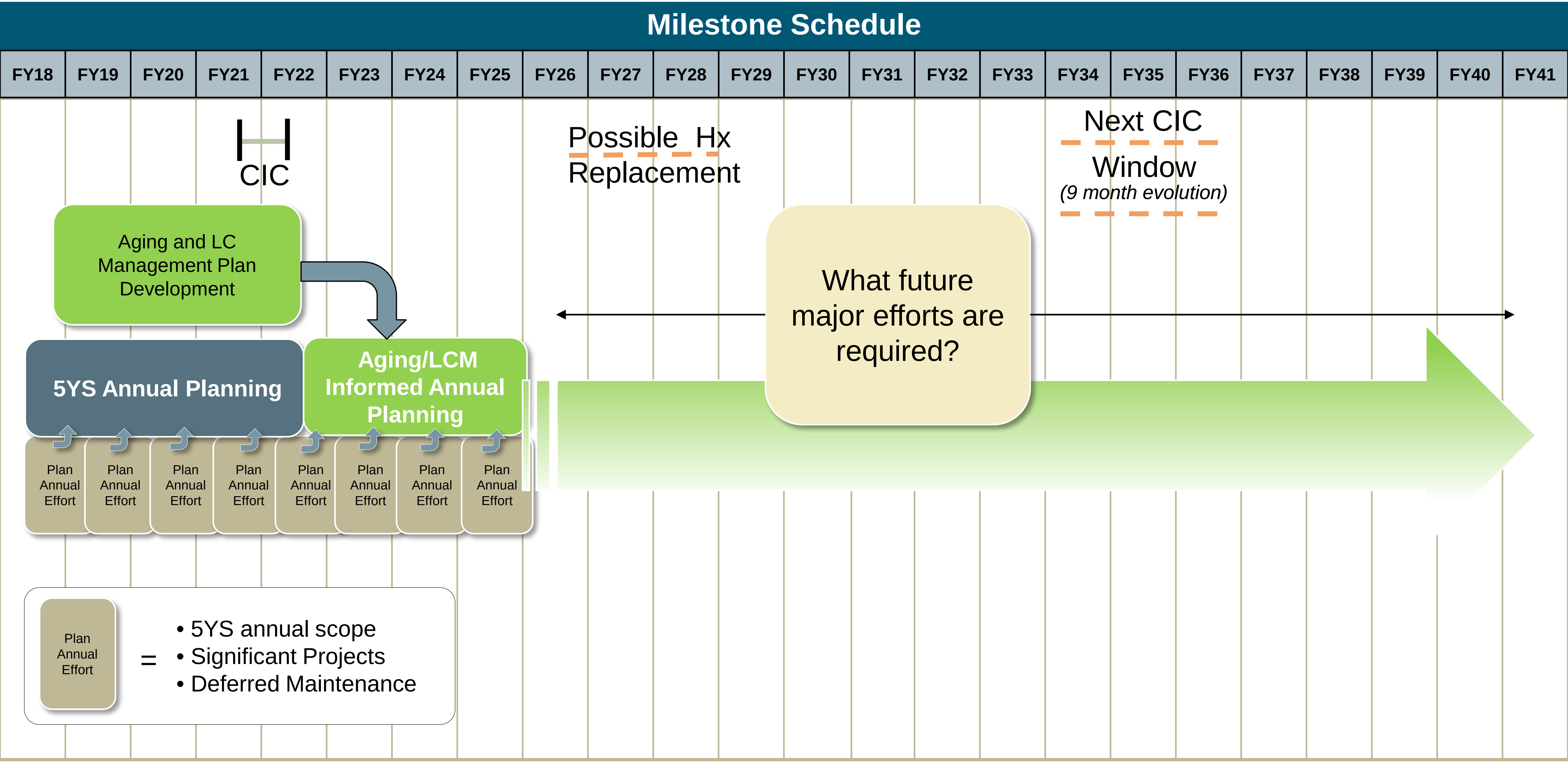
Methods to Deal with Obsolescence and Aging

- System Health monitoring
- Equipment Reliability Index
- Plant Health Committee
- Long Range Strategy

Plant Health Strategy

- A strategy was developed to address long term reliability at ATR
  - The written strategy for systematic replacement/ refurbishment of components critical to ATR operation
    - Documents all the major maintenance and project choices
  - Gives stakeholders and sponsors an opportunity to review and understand the long term strategy for sustainability and resource investment
  - Dynamic; can be adjusted, based on emergent needs
    - Sometimes problem equipment self identifies!

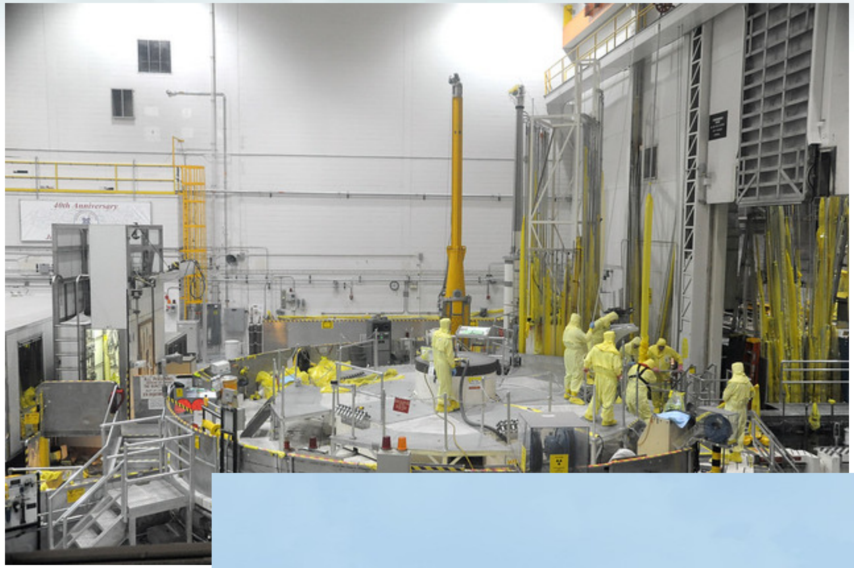
Sustain ATR Capabilities - Notional Schedule / Milestones



Integrated Aging/LC Management -- Resources and Expertise

- Identify additional resources who can lead the Aging/LC Management effort:
  - Consistent with IAEA-TECDOC-792 “Use of Experts”
  - Minimize impact to ATR system engineers current workload.
  - Outside expertise provides necessary guidance and experience.
    - ENERCON – subcontractor experienced in Aging and Life Cycle Management.
      - On-site team performs the day to day legwork of scoping / screening / Aging Management and LC Management Reviews and development.
    - Rely on ATR system engineer expertise for review / feedback and technical check.

Outcomes



- A “methodical” approach (TECDOC)
  - Scoping
  - Screening
  - Aging Management Reviews / Plans
  - Life Cycle Management Plans

- Based on commercial nuclear industry Information and equipment

- Economic planning and feasibility
  - A 20+ year look at effort and costs
    - Ongoing inspections and analyses associated with Aging/LC Management plans
    - Consideration and early identification of large / out-year reactor system updates
    - These costs can then be “escalated” for additional time increments

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Breazeale Reactor Beam Lab Refurbishment Progress Report

Jeffrey A. Geuther, Daniel B. Beck, Maksat Kuatbek, Alibek Kenges, Bryan Eyers, Amanda M. Johnsen

Introduction

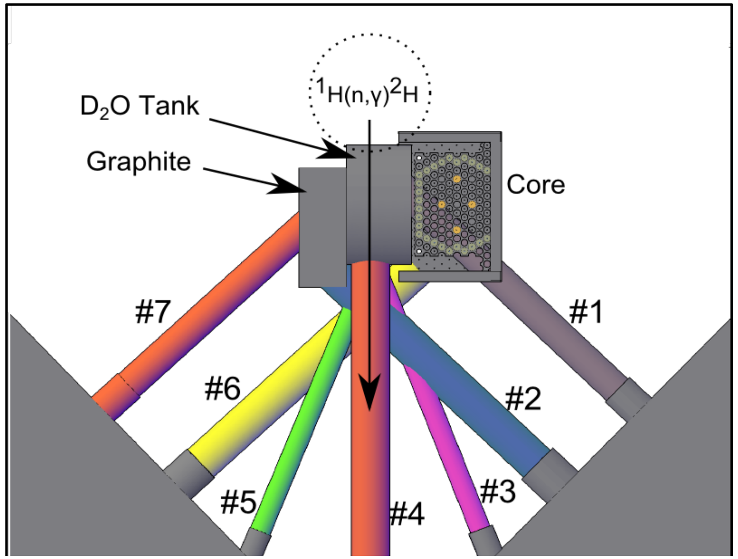
- PSBR is a TRIGA conversion that has the unique ability to move along two axes and rotate 180 degrees.
- This allows versatility – the reactor can be coupled to a variety of experiments.
- A new D_2O tank and beam ports were installed in 2018 to allow increased utilization of neutron beam facilities. *("PSU Breazeale Nuclear New Core-Moderator Assembly and Neutron Beam Port Installation," TRTR 2018).*
- This project was a major FY2014 NEUP-funded infrastructure enhancement. *DE-NE0000640*
- Work remains to characterize and utilize the beam lab experiments.

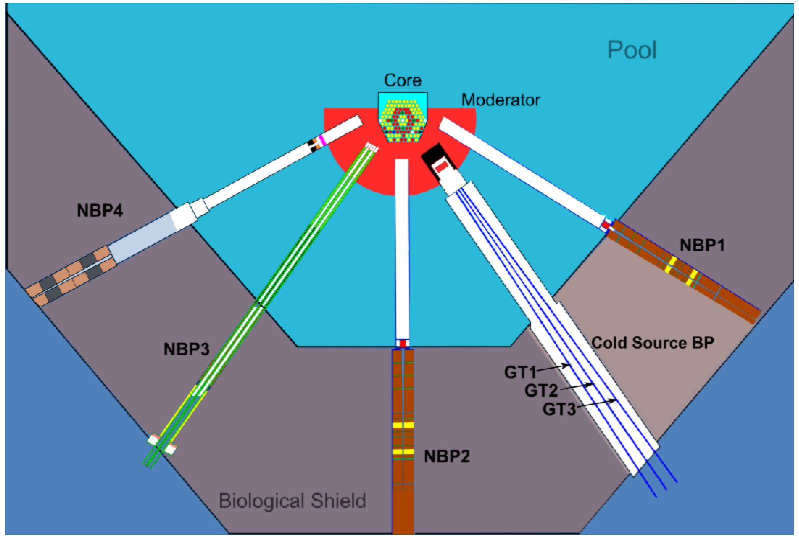
Introduction

- This presentation will summarize:
  - Characteristics of new moderator tank and beam ports;
  - Operational status of beam ports;
  - Flux measurements at various experimental facilities;
  - Radiography system status and test images;
  - Plans for installation of cold source;
  - Plans for beam lab expansion.

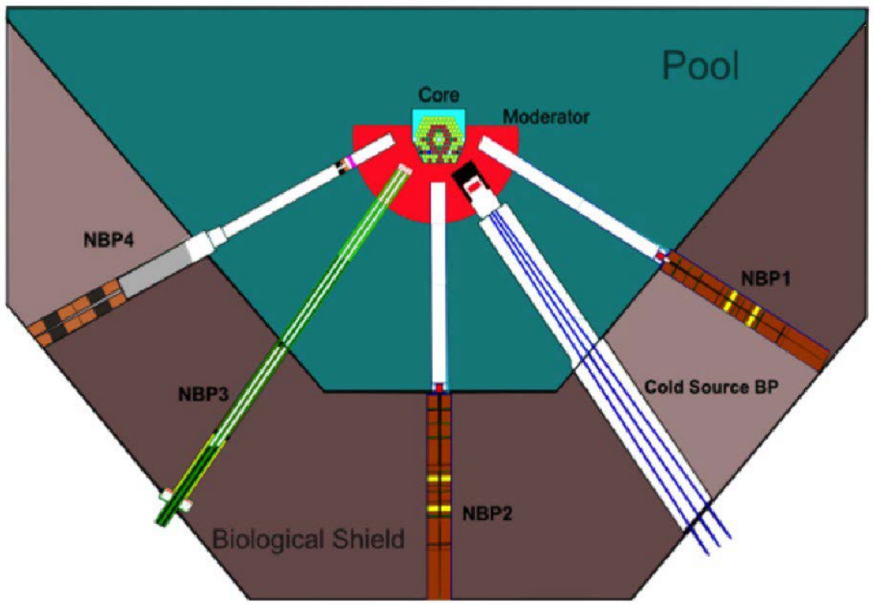
Beam Port / D_2O Moderator Upgrade

- \$1.36 M (DOE FY2014 NEUP reactor infrastructure grant)
- Replace core grid plates, support tower, D_2O moderator tank, and beam ports
- Enables use of five radial beam ports, vs. two tangential beam ports in prior design





PSBR New Neutron Beam Ports



NBP1 : Triple Axis Student Spectrometer

NBP2 : Thermal Neutron Beam Port for Exploratory Research Projects

NBP3 : Neutron Transmission (Service Activities)

NBP4 : Neutron Imaging

GT1 : TOF Neutron Depth Profiling

GT2 : Neutron Powder Diffraction / SANS

GT3 : Prompt Gamma Activation Analysis

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Shutter

Un-stopped beams are controlled with a rotary shutter

- Three positions
- Fails closed
- Motor and chain driven
- 9” lead

Flush to wall, preventing door from interfering with shield caves

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Operational Status of Beam Ports

NBP1: Plugged, no experiments installed

- More space for work following renovation

CS: Plugged, no experiments installed

-Cold source under development

NBP2: Plugged, no experiments installed.

NBP3:

-Collimated and shielded

-Used frequently for service work

NBP4:

-Shield under construction

-Plugged when not in use

-Used for radiography. Shield under construction. No collimation or filters.

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Foil Activation Measurements

- Thermal and resonance flux at certain locations was measured using bare and Cd-covered gold foils
- The foils were placed on aluminum frames and were slid inside the beam tubes to the point of interface with the D_2O tank.
- These measurements indicate:
  - Soft neutron spectra, ~50:1 to 100:1 Cd ratio
  - Up to ~1E9 n / cm^2 / s at the exit of the biological shield

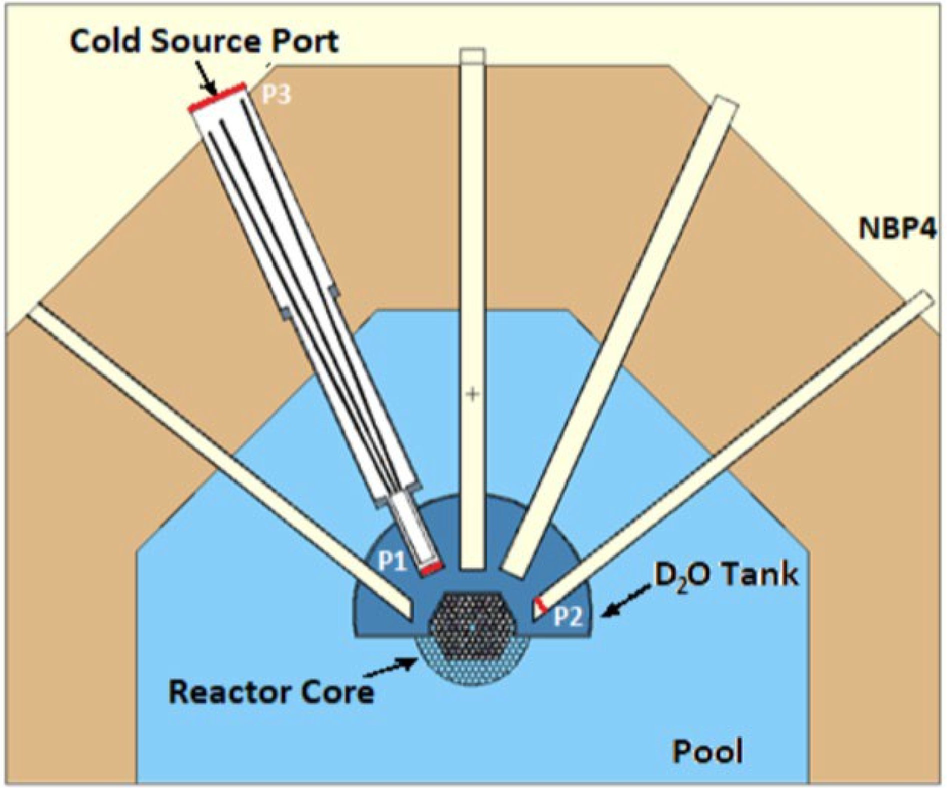
Beam Port Neutron Flux

Table 1. The neutron flux profile results at the entrance of the cold source port at 100 kW

Foil Number	Foil Weight	Activity per Au Atom	Thermal Flux	Resonance Flux
1	0.04748	1.42E-10	N/A	4.443E+09
2	0.04715	1.00E-10	N/A	4.636E+09
3	0.04710	3.38E-11	2.24E+11	4.44E+09
4	0.04765	3.50E-11	1.60E+11	3.18E+09

Table 2. The neutron flux profile results at the entrance of the NBP4 at 100 kW

Foil Number	Foil Weight	Activity per Au Atom	Thermal Flux	Resonance Flux
1	0.04779	8.11E-12	N/A	5.24E+09
2	0.04762	3.39E-11	2.61E+11	5.24E+09
3	0.04773	3.31E-11	2.55E+11	5.12E+09



NBP4 at bio shield exit:
3.11E7 thermal / 4.98E5 res. flux at 1 MW

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Neutron Flux at CS Exit

Table 5. The estimated neutron flux profile values at the exit of the CS port at 1MW

Foil Number	Thermal Flux	Resonance Flux
1	N/A	8.94E+06
2	N/A	9.62E+06
3	N/A	1.016E+07
4	N/A	1.007E+07
5	8.72E+08	8.94E+06
6	8.59E+08	9.62E+06
7	1.02E+09	1.02E+07
8	1.04E+09	1.01E+07
9	1.03E+09	1.02E+07

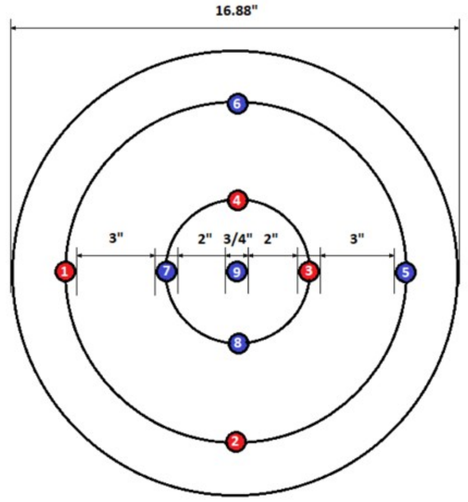

Figure 4. A pattern of the cadmium covered (red) and bare (blue) gold foils. The outer circle represents the size of the CS port's exit flange

Est. 8.2E8 n / cm^2 s thermal flux at bio shield exit with a 1” mesitylene moderator (Eyers)

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Radiography Shield Cave

- Old shield cave was demolished.
- Blocks were repurposed for new cave.
- Door resigned using air casters in place of cog / rail system
- Roof is shielded with plastic resin, BPE, and lead

Radiography Shield Cave

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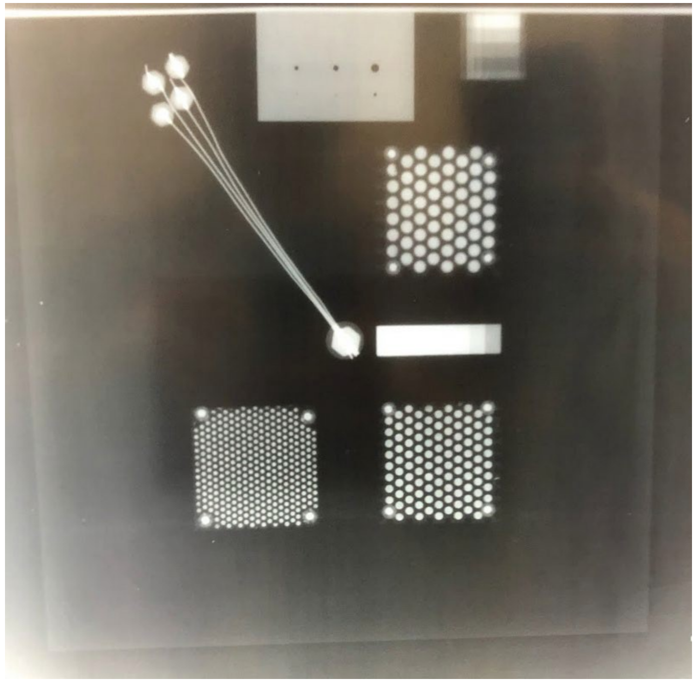
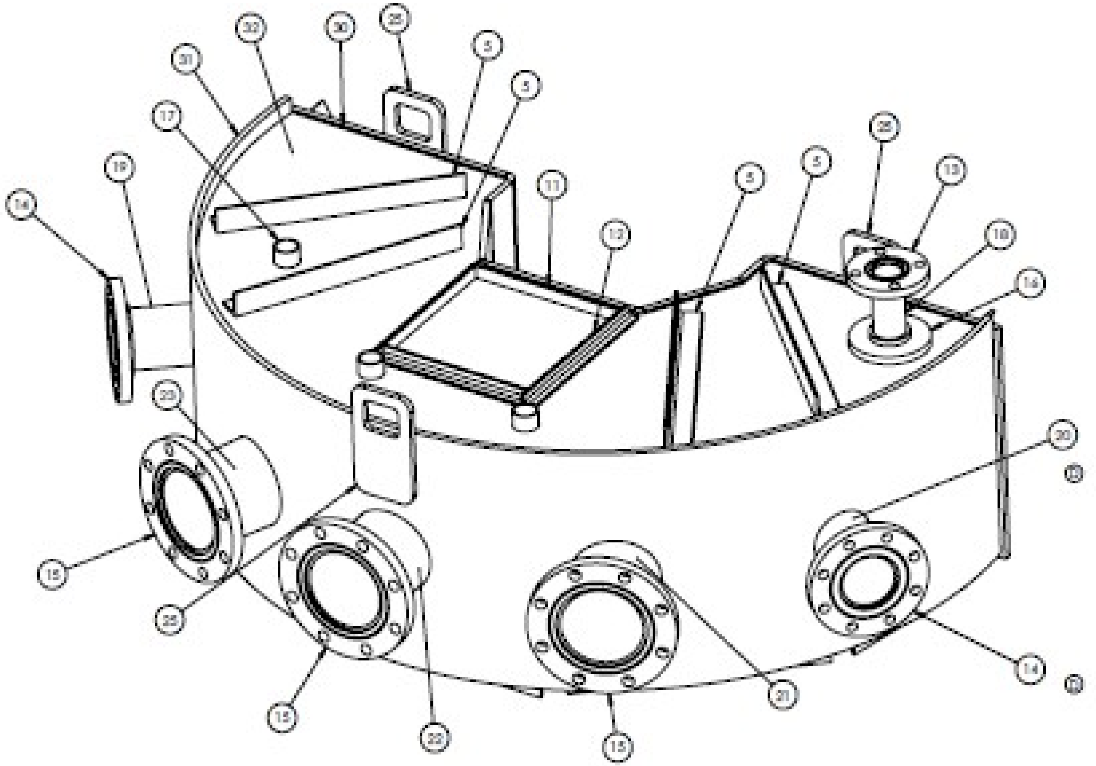
Neutron Radiography

- Un-collimated, un-filtered radiography beam can be used below 50 kW
- Useful images have been produced for customers (not shown)
- Work will continue to improve beam characteristics

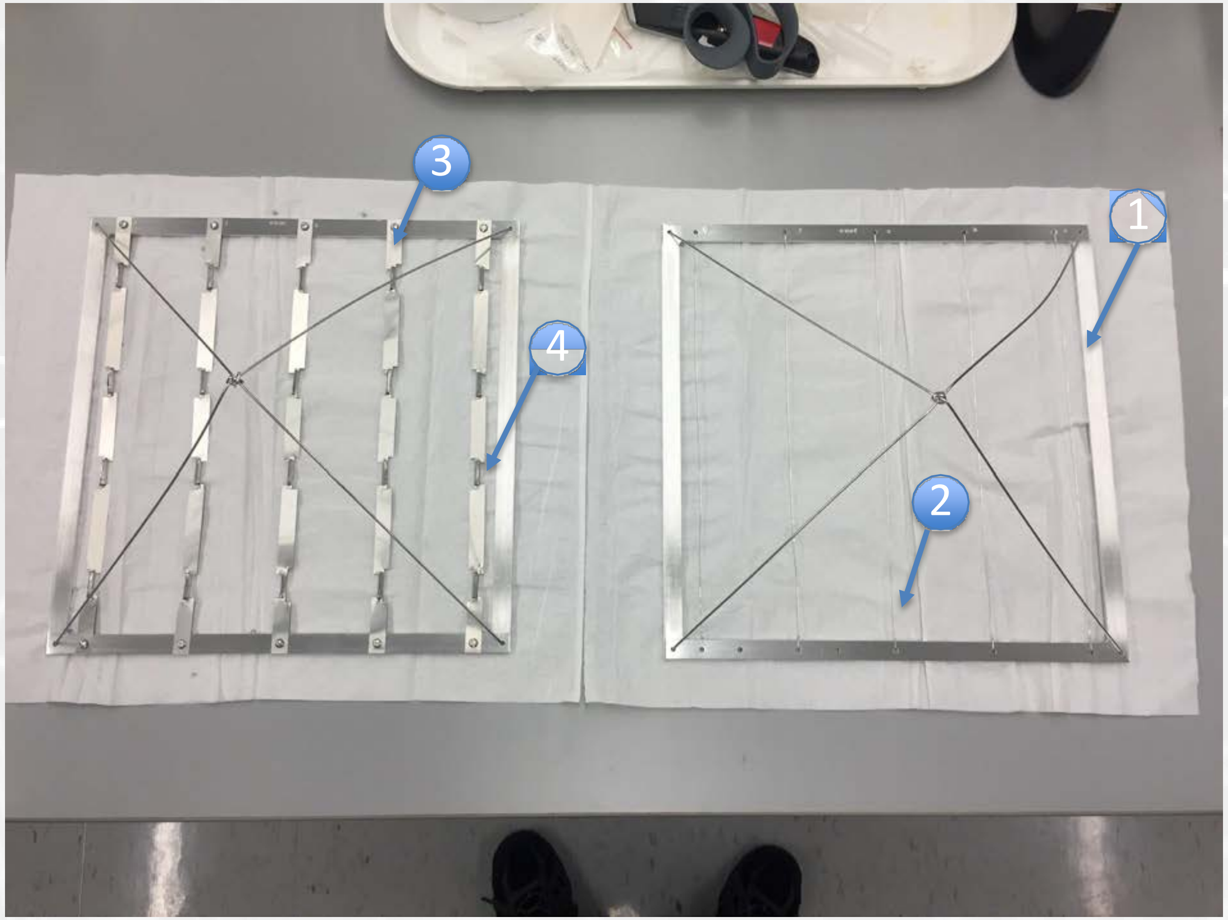


Moderator Tank Basket



- Basket is 12.25” x 12.25”, may be used for long-term thermal irradiation
- Basket is being considered for Si doping

The Sample Holder



1. Aluminum frame,
   12”x 12” (1100 alloy)
2. 5 x 12” AlAu wires
   (D=0.02”) with 0.12%
   gold concentration
3. Aluminum strips,
   0.157”x 12” (6061 alloy)
4. Cadmium sleeves,
   20 pcs, each 0.6” long

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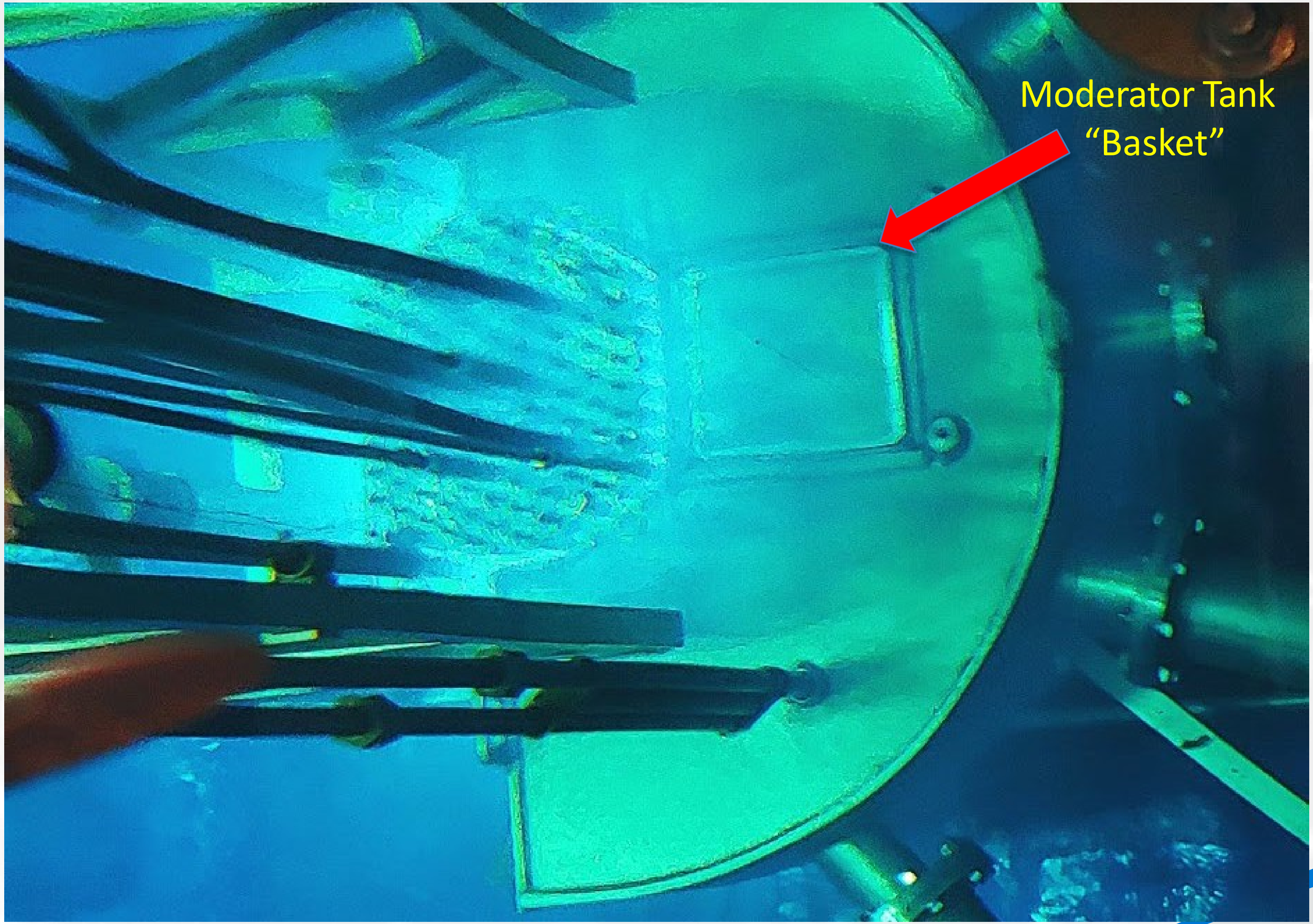
Reactor Operating at D_2O Tank



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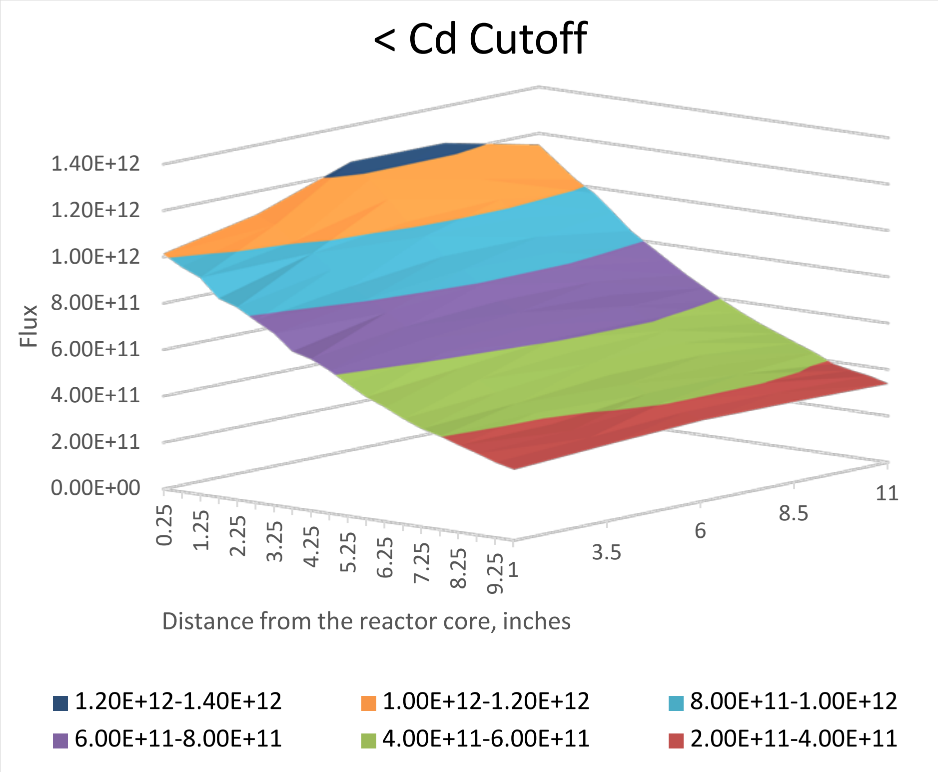
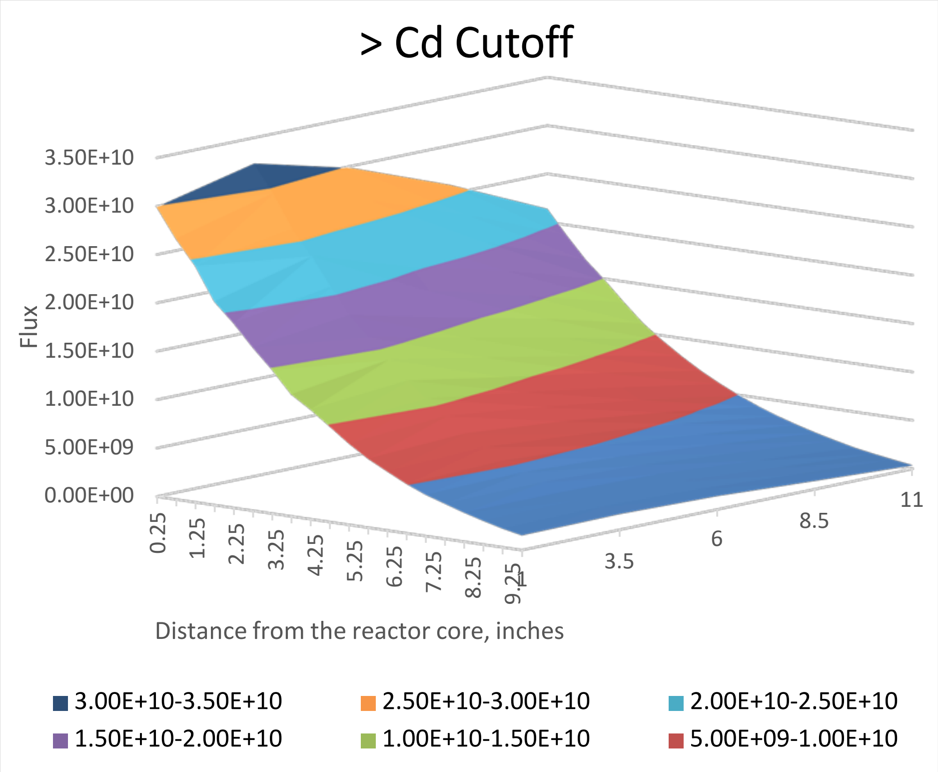
Flux Measurement at Basekt





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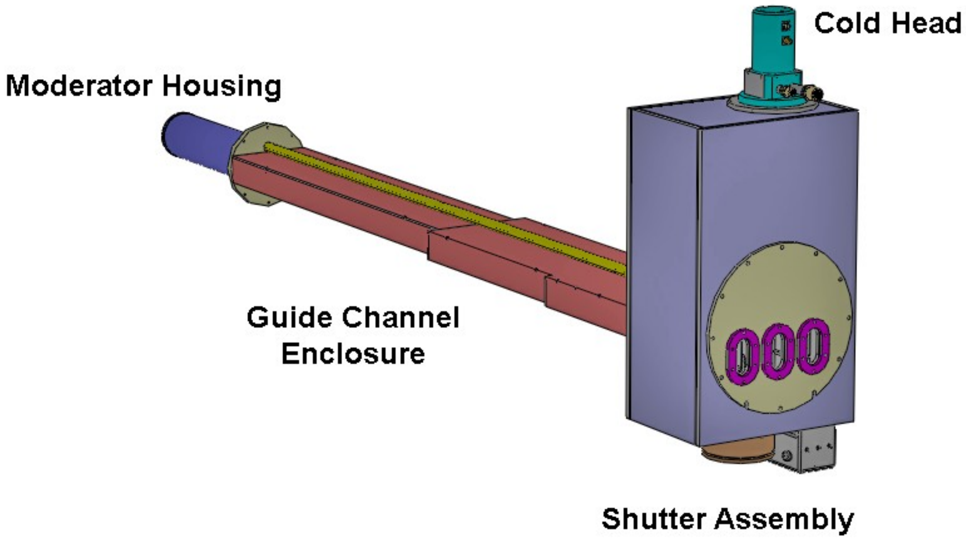
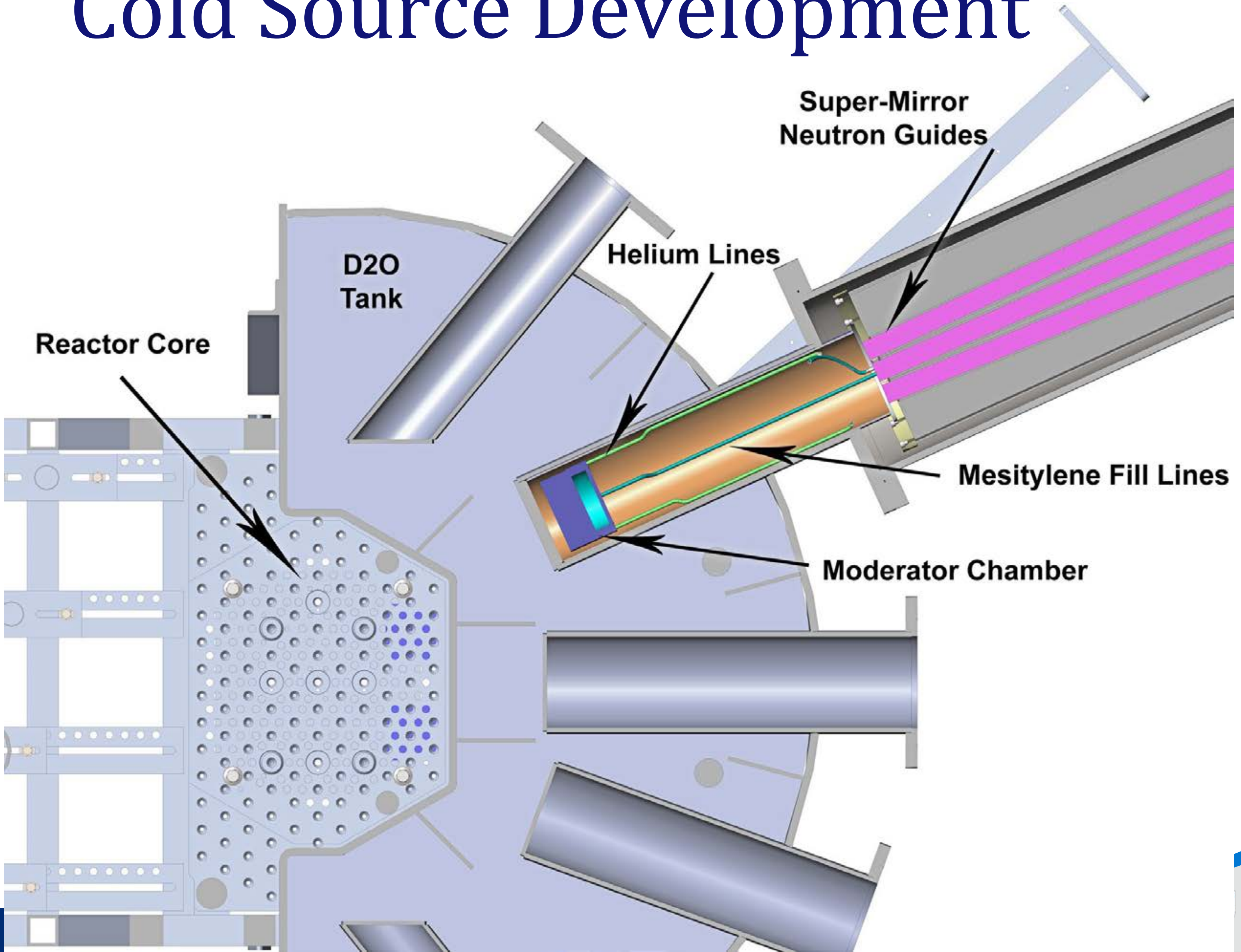
Cold Source Development

- One BP will house a cold neutron source
  - 20 K
  - Mesitylene $C_6H_3(CH_3)_3$
  - 4.25” dia. x 1” thick
- Intended for use in:
  - NDP
  - PGAA
  - SANS



Cold Source Development



Beam Lab Expansion

- In order to accommodate additional beam lab experiments / cold source experiments, ~8000 sq. ft. of lab and office space will be added to the RSEC
- Project is underway, expected completion in 2020
- Will enable use of cold neutron source and SANS, to be donated by Helmoltz-Zentrum Berlin

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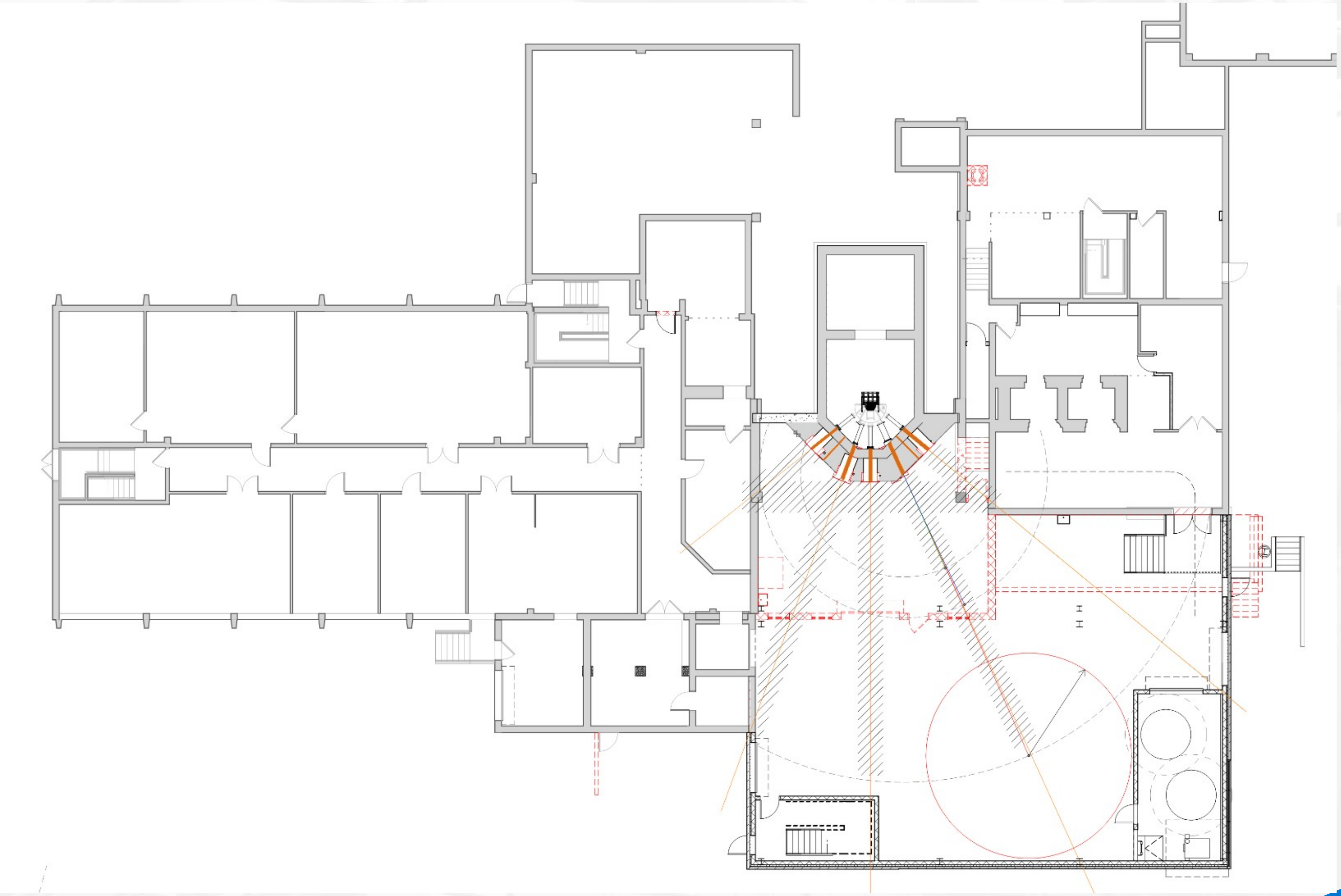
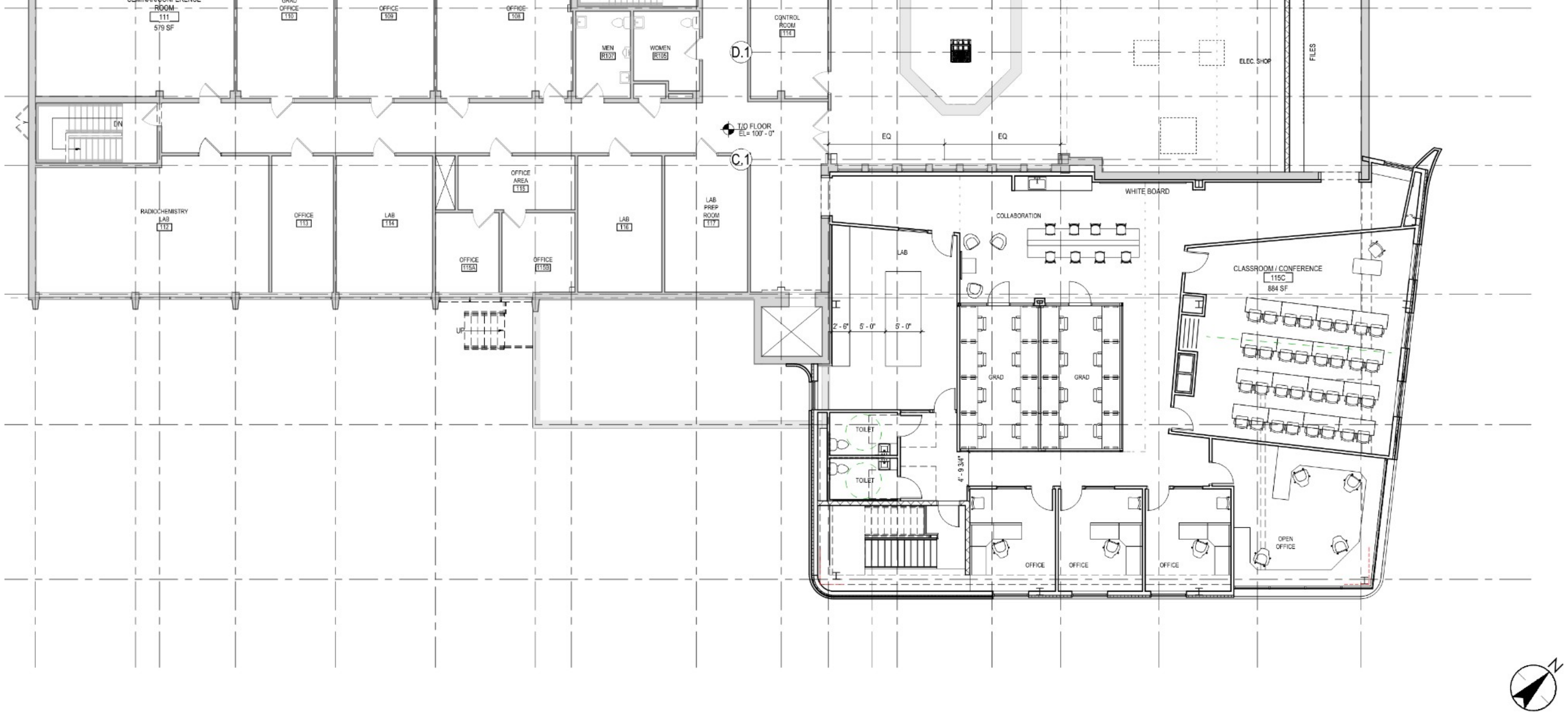
Beam Lab Expansion





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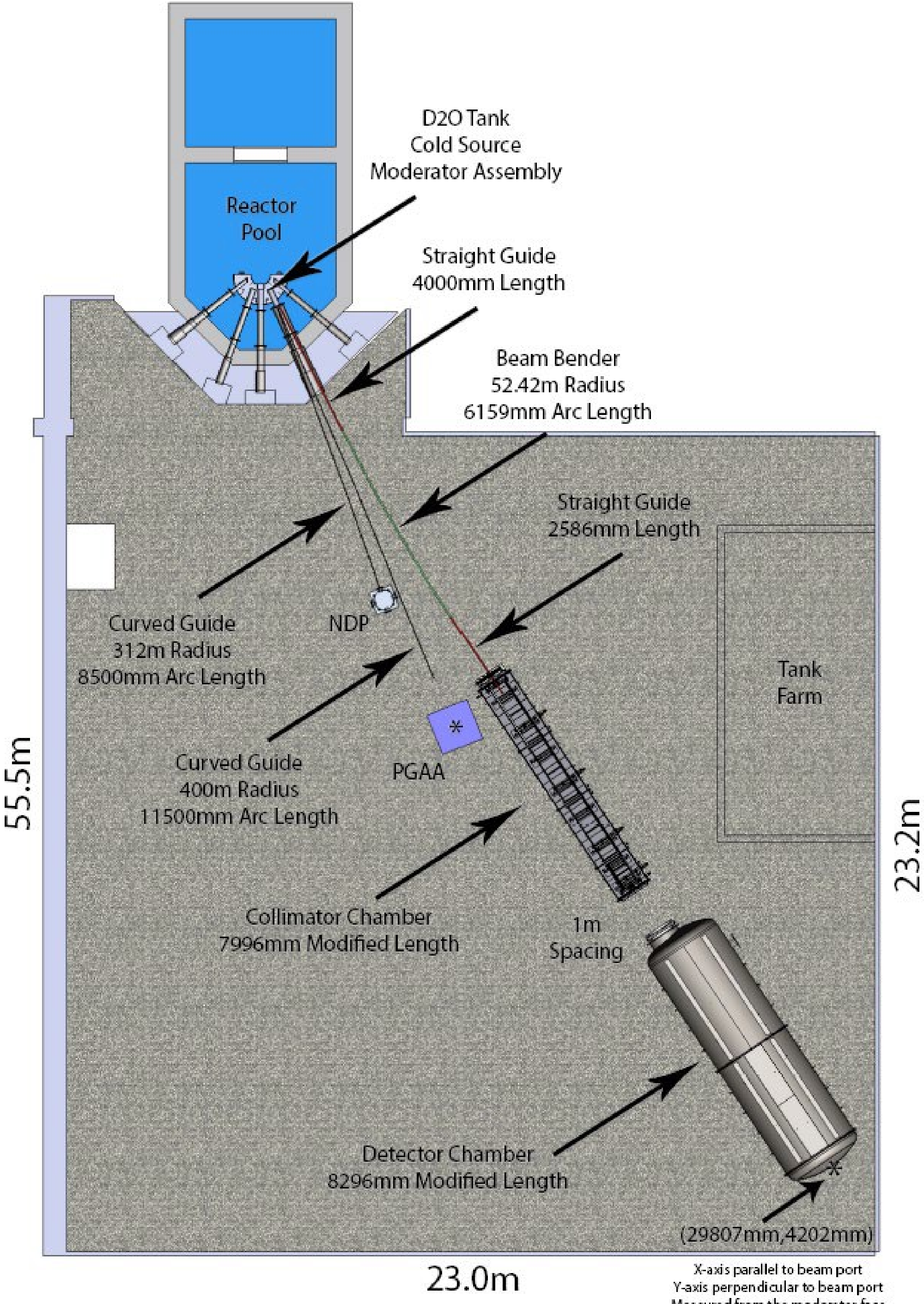

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Future Work

- Complete beam cave
- Install cold source
- Design and install neutron radiography beam filters and collimato
- Expand neutron beam laboratory
- Install SANS and other cold source experiments

PennState College of Engineering

Breazeale Reactor Beam Lab Refurbishment Progress Report

Jeffrey A. Geuther, Daniel B. Beck, Maksat Kuatbek, Amanda M. Johnsen

PennState
College of Engineering

External Review
Deuterium Cold Source Project

NIST Center for Neutron Research
Michael Middleton
Robert Williams, John Jurns,
Mike Rowe

SUMMARY

External Review as a Form of Risk Management

Overview of the Deuterium Project

External Review Committee's Charge

Concerns and Issues Identified

Additional Concerns and Comments

External Review Committee Report and Response

Discussion of Some of the Issues

Risk Associated with Installation and Future Operation of the Deuterium Moderated Cold Source

Identifying and Resolving Critical Issues before they Becomes Serious Problems

Minimizing the Risk of Restarting the Reactor and Deuterium Cold Source after a Year Shutdown and Finding, a Design, and or Installation Discrepancy that would Prevent the Operation of the Deuterium Cold Source and the Reactor

Minimizing the Risk of Operating the Deuterium Cold Source and Peewee Hydrogen Source in Parallel as Designed with the Proper Void Fractions in the Cryostats.

External Independent Review

Control Risk.

Ensure readiness to proceed to subsequent project phase.

Enable identification and resolution of issues at the earliest time, lowest work level and cost.

Functional Integration of project products and effort of organizational components.

Characteristics and Benefits of External Independent Review

Look into Critical Issues before they become serious problems. Review serves as a tool for Risk Management and Mitigation.

Provide Senior management with substantive, independent, Unbiased Assessment of Project Assumptions and Alternatives.

Provides a review as a Means of Adding Value, not just an Audit or Over Sight Function.

Provides Credible Evidence to Management and Funding Agency that Project Funding has been well Founded.

Provides Reporting on the Readiness of the Project to Proceed.

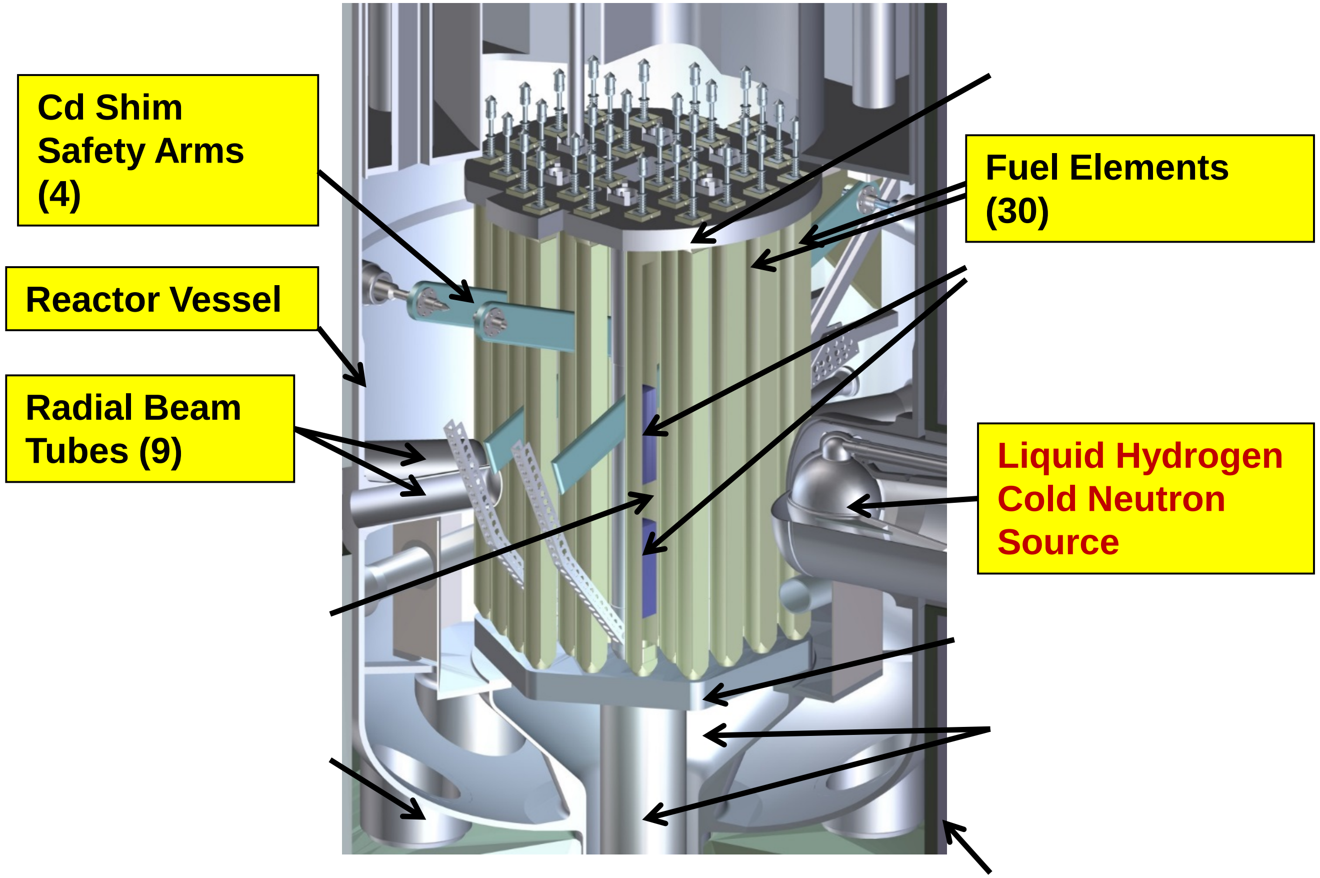
INDEPENDENT REIVEW

Reviewers Who did not Participate in the Planning and Execution of the Project is Vital for Objectivity and for Increasing Confidence in Decisions on Major, Complex Projects with Significant Inherent Risks.

The more Detached the Reviewers are from Economic, Political, Social or Other Influences, the more Independent they can be.

Recommendations made, and practices identified during the Review can also be used to Improve the Project Management Process by Identifying Areas that need More Scrutiny or a Different Approach.

Cut-away View of the 20 MW NBSR



History of Deuterium Project

2011, National Nuclear Security Administration To Provide Funding

2016, Original Project Completion, 11 Million Dollars

2014, Eden Defaults on Refrigerator Contact

2014, Cryostat Procurement Cancelled

2014, Cryostat Funds use to Complete the Refrigerator

2016, Six Million Dollars Committed to Complete Project

2022, Planned Project Completion, 17 Million Dollars

Deuterium Project Timeline

6/2013, Ballast Tank Contract
2/2014, Eden Request Delay in Delivery of Refrigerator
2/2014, Received JJ Crewe Compressors
8/2014, Received Eden Coldbox
8/2014, Cryostat Plug Contract
5/2015, Helium Piping Contract
7/2015, Compressor Commissioning Contract
2/2016, Received Additional Funding, 6 Million
8/2016, Condenser and Connection Piping Contract
9/2016, Complete Compressor Commissioning
5/2017, Cryostat Contract, Option 1 Engineering
9/2017, Refrigerator Startup Services Contact
9/2017, Cryostat Contract, Option 2 Proof of Concept
1/2018, Cold Source Operation with New Refrigerator
9/2018, Cryostat Contract, Option 3 Fabricate Four Prototypes
9/2020, Cryostat Contract, Option 4 Fabricate Two Cryostats
9/2020, Ballast Tank Interconnecting Piping Contract
4/2023, Installation of Cryostat

Deuterium Project Major Components and Assemblies

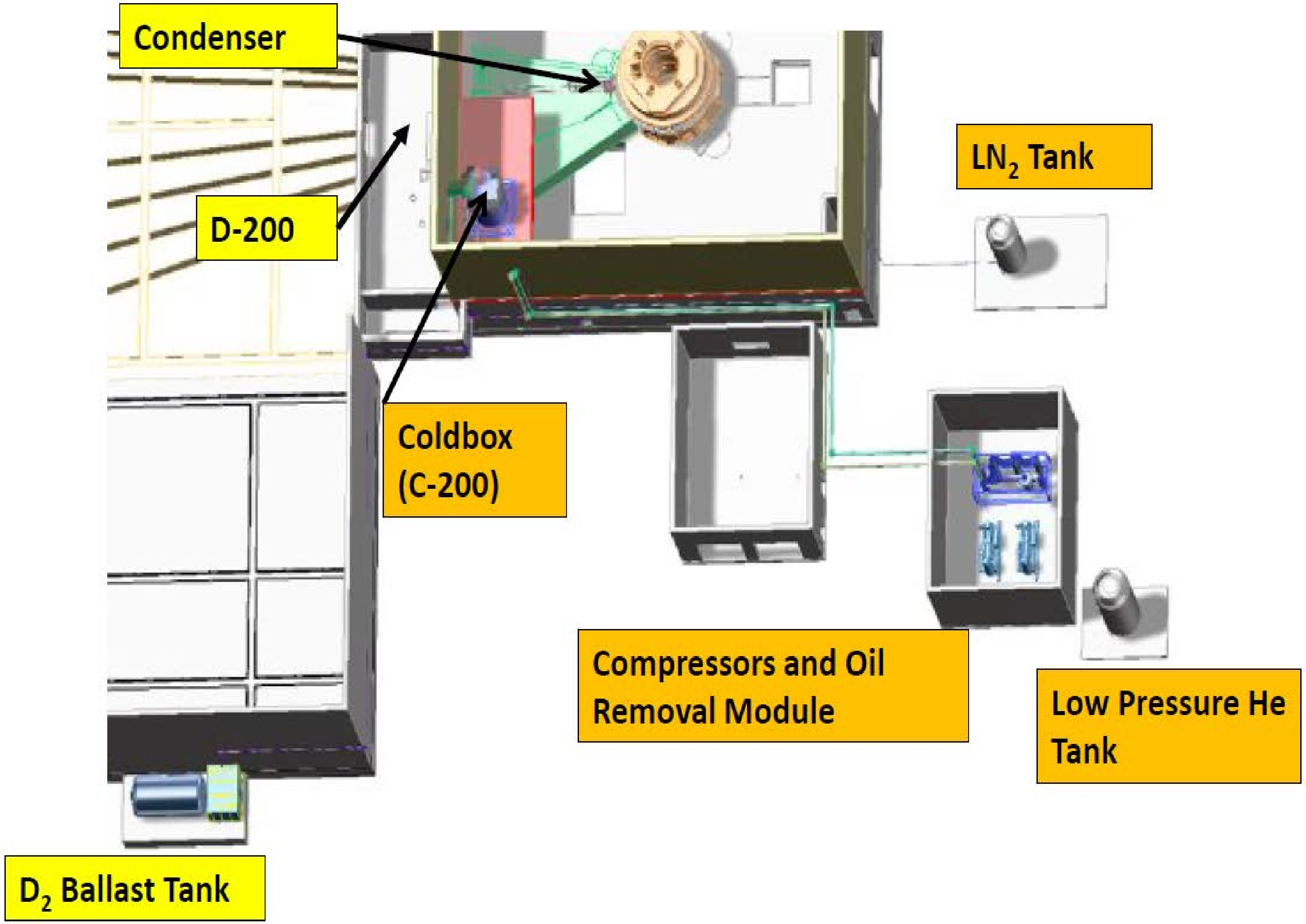
7 KW Helium Refrigerator

Helium/Deuterium Condenser
Condenser/Cryostat Interconnecting Piping
Cryostat Plug Assembly

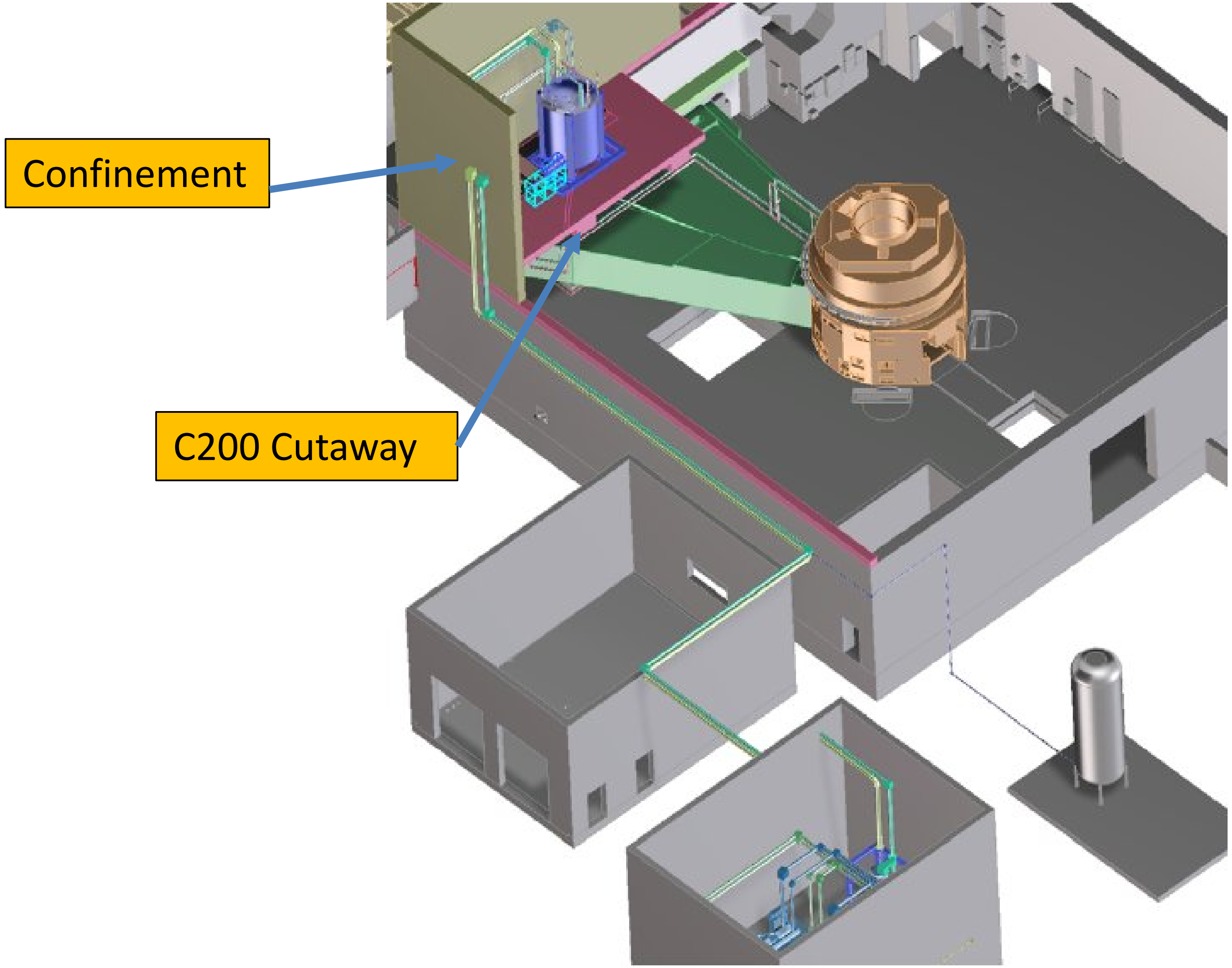
Cryostat Assembly

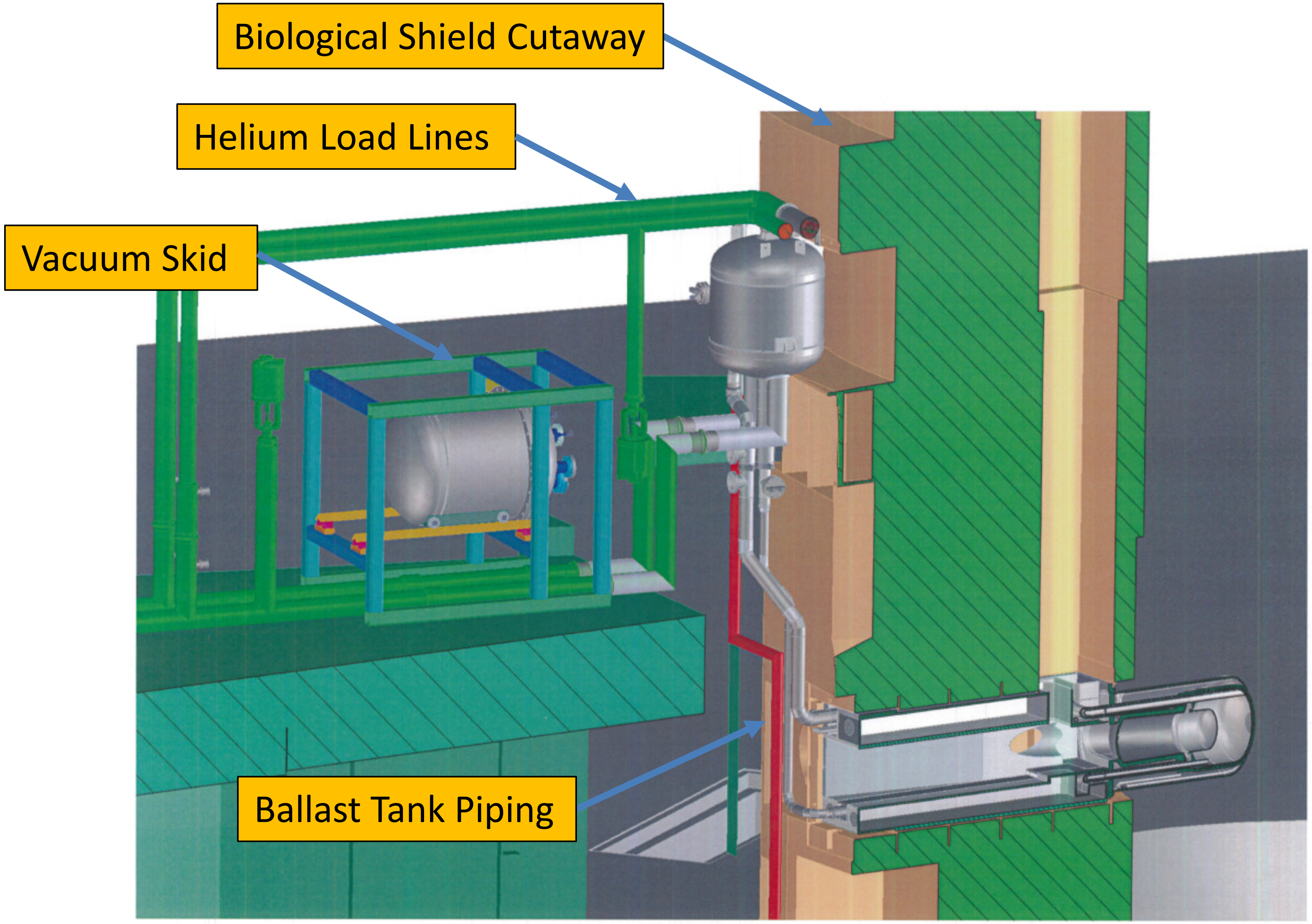
16 Cubic Meter Ballast Tank
Condenser/Ballast Tank Interconnecting Piping

Layout of Major Components

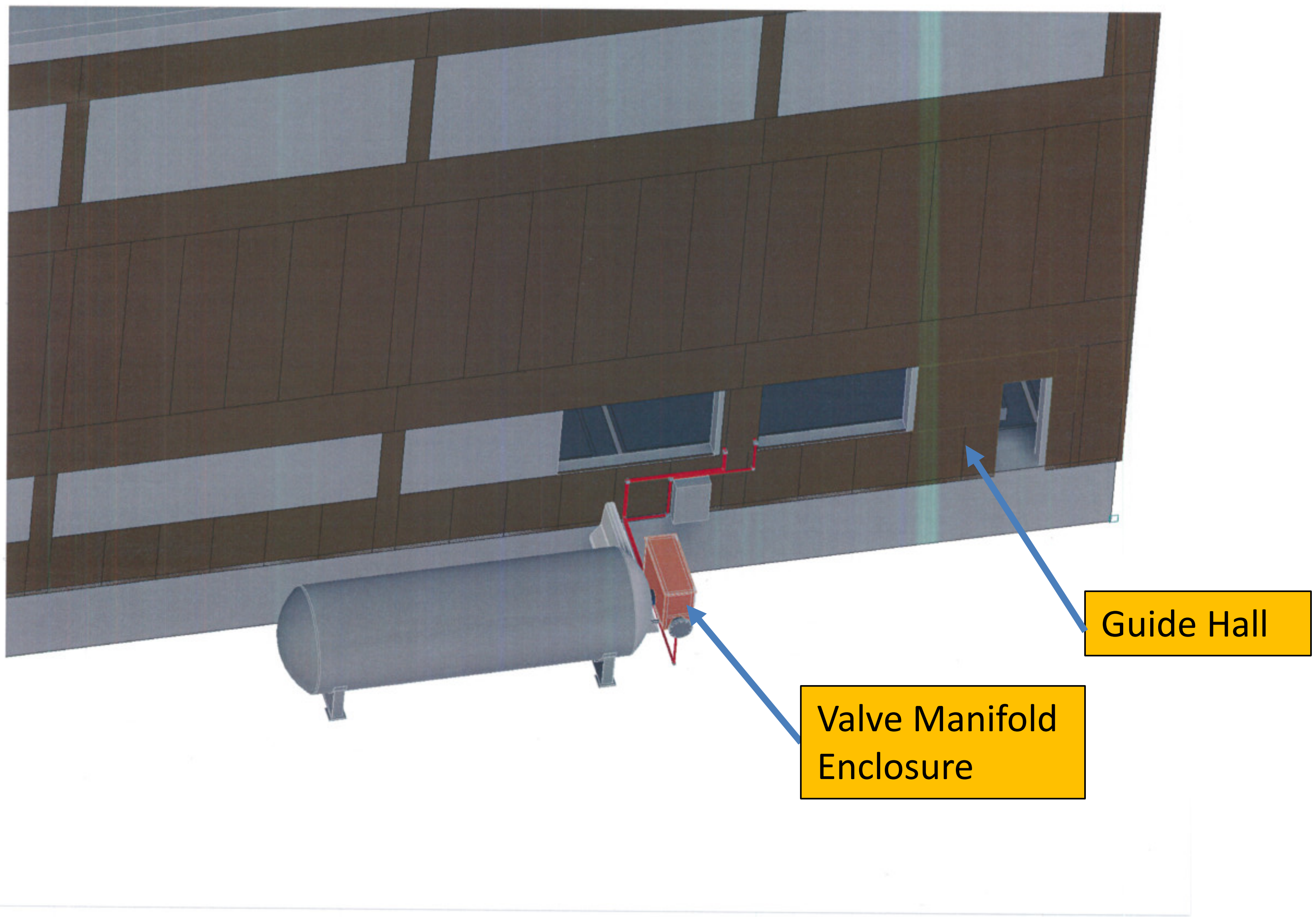


7 KW Helium Refrigerator Layout

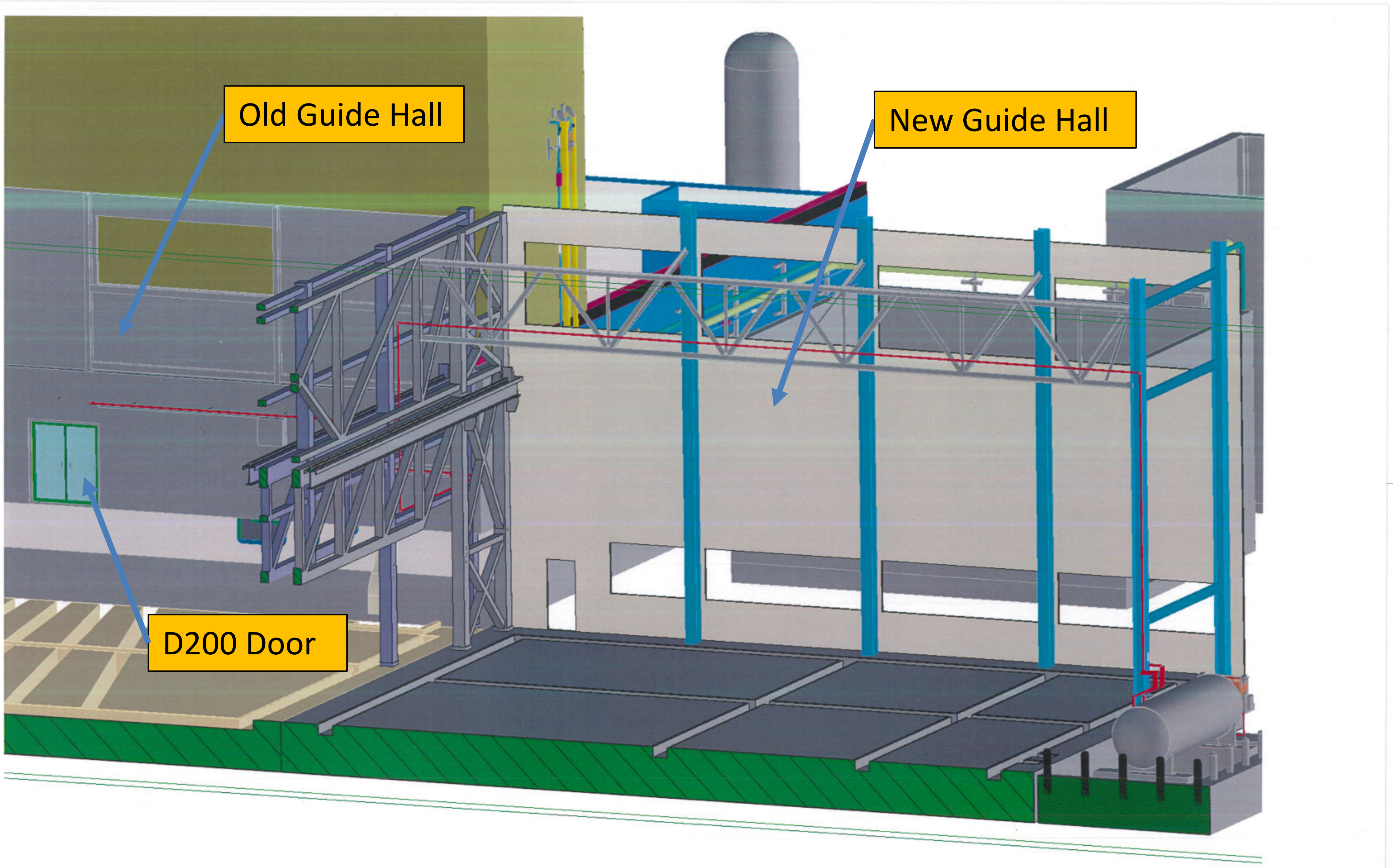




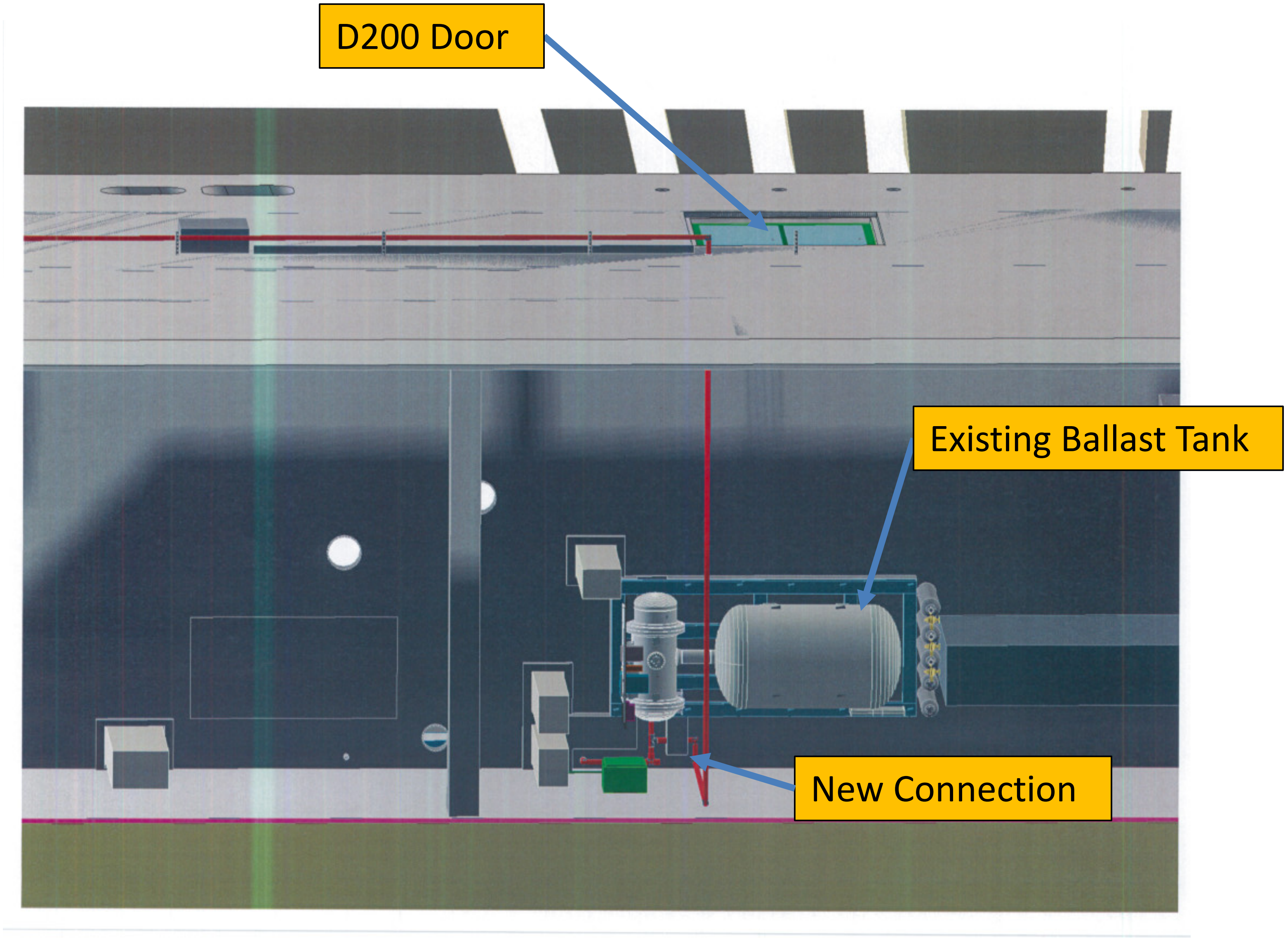
Layout of Deuterium Cold Source in C100



16 Cubic Meter Deuterium Ballast Tank



Connecting Piping between New Ballast Tank and Existing Ballast Tank



Connecting Piping between New Ballast Tank and Existing Ballast Tank

NIST NSBR Deuterium Cold Source Review Committee

Jamie McAllister,
NIST Fire Protection Engineer and Toxicologist

Bertrand Blau,
Paul Scherrer Institut

Erik Iverson,
Oak Ridge National Laboratory

Weijian Lu,
Australia Nuclear Science and Technologies Organisation

Basic Charge for External Review

Basic Design of Cryostat and Proposed Gain in Flux

Safety Issues

Operation of a Hydrogen Cold Source in Parallel with a Deuterium Cold Source

Review of Project Progress

Additional Concerns

Overall Comments

Documents are well written and Presentations excellent with Useful Information.

Project Team has the Expertise and Experience to Successfully Complete the Project.

Project Implementation under Difficult Circumstance are Commendable.

Encourage Funding Authority to Ensure Continued Funding to Project Completion.

External Review Issues/Concerns
Aug 2019

Bench mark Nuclear Heat Loads

Deuterium- Helium Detection Methods

Tritium Release Analysis

Credible Abnormal Events

International Fire Code Section 421

External Review Issues/Concerns(Cont)
Aug 2019

Protection of Ballast Tank Isolation Valves

Location of Ballast Tank

Adequate Temperature Margin to D2 Triple Point

Mass Flow Sensors and Modification of VJ Piping

Temperature Control of Ballast Tank located Outside

Physical Security of Ballast Tank

Additional Interesting Comments

Security of a Tank Filled with Deuterium that can be Recognized from the Air.

Analyze Ballast Tank Piping Integrity in Guide Hall during an Earthquake.

Review location of Electrical Panels Relative to the Ballast Tank.

Lockout/Tagout for Ballast Tank Isolation Valves.

Review of Safety Associated with Deuterium Project

Fire Safety

Oxygen Deficiency Hazard

Tritium Release to Environment

Maximum Hypothetical Accident

Deuterium Release inside Confinement

Fire Safety

NIST Fire and Facilities Safety Group we reviewed several applicable fire codes, particularly, NFPA 2 and NFPA 55.

LD_2 source needs three Exemptions to NFPA requirements:

1. Maximum Allowable Quantities (MAQ)

    Loaded to 500 kPa, we would have 3178 scf, above limit for which sprinklers are required.

    Exemptions for gas enclosure (He confinement) can be granted to raise the limit to 4000 scf.

2. Automatic Emergency Isolation Valve

    Generally applied to cut off and external supply in an emergency.

    There must be no valves blocking the flow of D_2 to the ballast tank.

3. Pressure Relief Valve

    Pressure relief achieved with gas expansion back to ballast tanks.

Exemptions to NFPA requirements must come from the Fire and Facilities Safety Group (“Authority Having Jurisdiction”).

Oxygen Deficiency Hazard (ODH)

New Refrigerator Located Outside Control Room, in C200.

NCNR Hazards Review Committee studied the Consequences of Major releases of LN_2 or He into C-200.

- Rupture of LN_2 supply line would create hazard in 17 minutes.
- ODH monitor to alarm locally and in the Control Room.
- LN2 Isolation Valve close if the oxygen concentration drops below 19%.

Time for Operations to Scram the Reactor and have all personnel evacuate confinement.

There is an Emergency Control Room in the basement.

Tritium Release

After decades of operation the deuterium inventory will include about 1800 Ci (6.7 x 10^{13} Bq) of tritium in the form of DT molecules.

A HotSpot analysis of the off-site consequences of a rapid release of 80 % of the inventory was performed (used larger activity, 2832 Ci).

An individual at the site boundary, 300 m from release point would receive a TEDE (Total Effective Dose Equivalent) of at most 0.5 µSv (0.05 mrem).

An occupational worker would have a maximum TEDE of 6.0 µSv (0.6 mrem).

Both well below regulatory limits.

ICRP dose conversion factor used for DT. It is 10,000 times lower than 10 CFR 20 value for DTO molecules.

Maximum Hypothetical Accident (MHA)

Deuterium Gas Explosion in Cryostat.

Similar to Existing Cryostat, but much larger volume, 7X.

Multiple failures would be required for this accident to occur:

- 350 liters of STP air freeze on moderator vessel (450 g, 104 g O_2).
- Cryostat Vessel leaks LD_2 and Detonation.

Maximum pressure:

P_{max} = 1000 psia × (26 g D/ 9.7 g H) × {(62 kJ/g D)/(121 kJ/g H)} × (33 L / 50 L)= 902 psia.

The Helium Vessel Designed for the Pressure Generated from the MHA and will Provide Protect for the Reactor Vessel thimble that houses the Cryostat.

Deuterium Release into Confinement Building

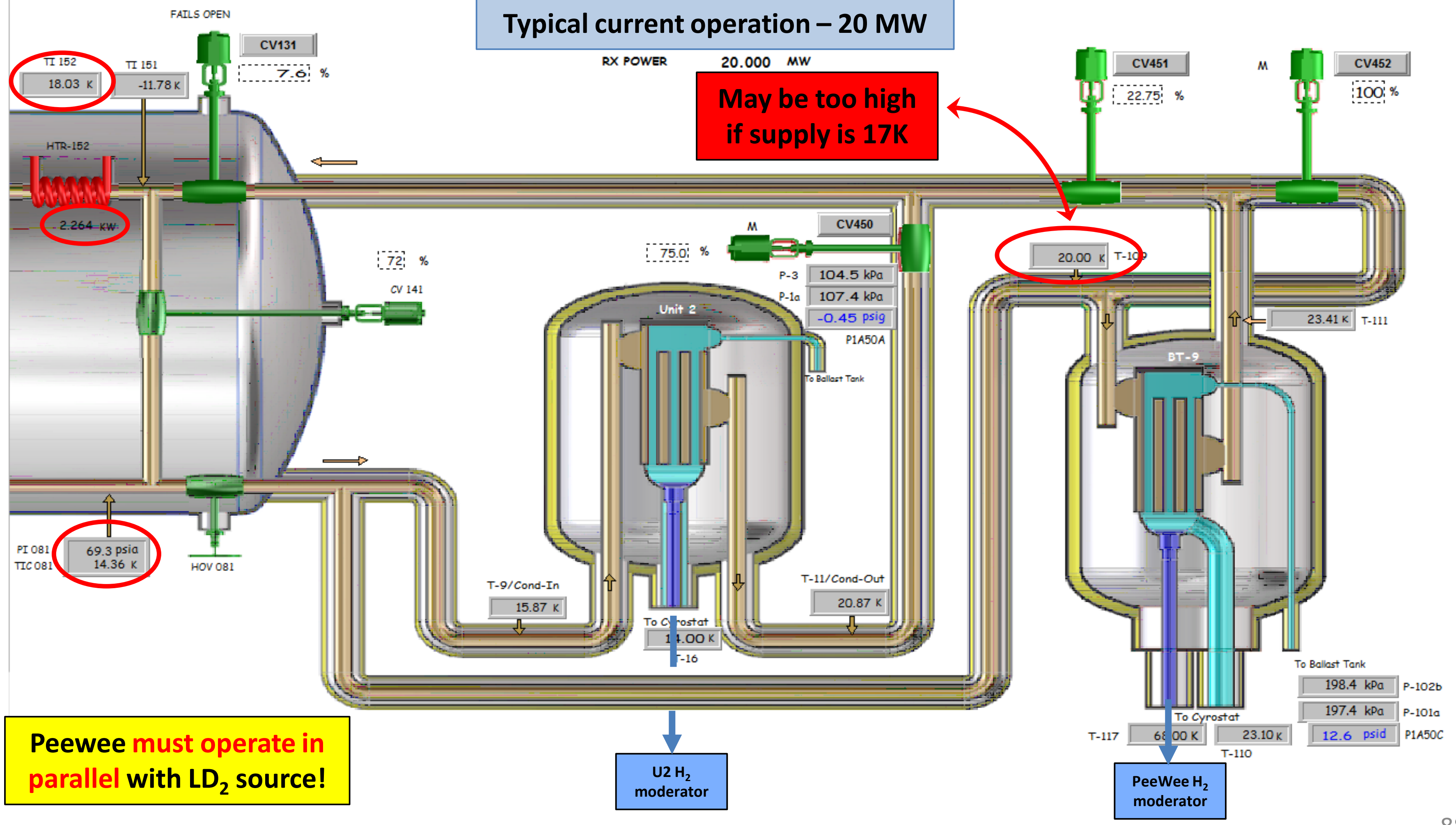
There is no credible scenario for a massive release of D_2 into C-100 while the reactor is operating.

All Deuterium boundaries surrounded by Helium Barrier.

Crane “no fly zone” Near Deuterium Condenser.

Administrative Procedures Require that the Ballast Tank Isolated when Maintenance needs to be Performed.

Typical current operation – 20 MW



Operation of Parallel Hydrogen Cold Source and Deuterium Cold Source

Planned operating parameters for Unit 3 (LD_2) & Peewee (LH_2)

Nuclear cryogenic heat loads – current and planned for LD_2

Radiation source	Unit 2		PeeWee		LD_2	
	H_2	Al	H_2	Al	D_2	Al
Neutrons	104	3	33	1	440	6
Beta Particles		308		29		567
Gamma rays	185	815	25	74	1053	1538
Subtotal	281	1080	58	104	1493	2111
Total cryogenic heat load [w]	**1361**		**162**		**3604**	

Total estimated cryogenic heat loads (static + neutronic) – Planned for LD_2

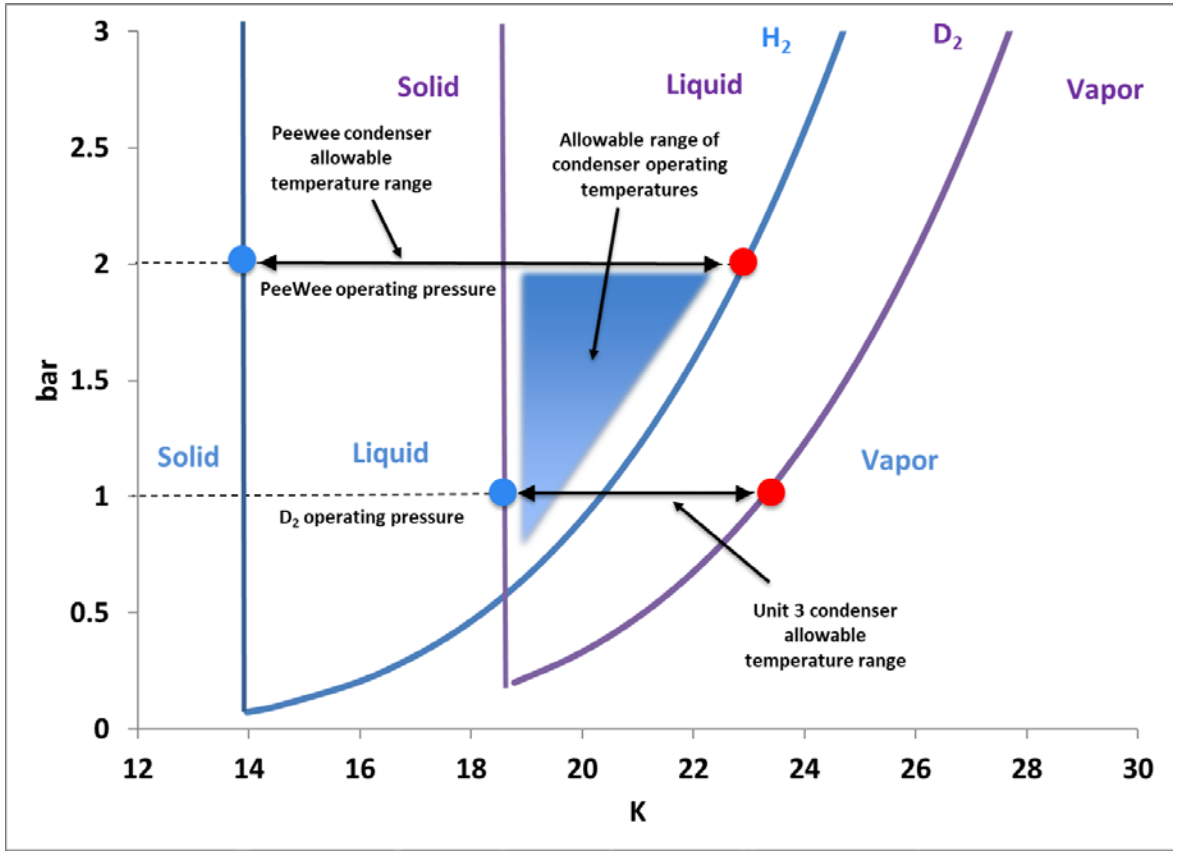
	Static	Neutronic	**Total**
Unit 3	250	3600	3850
Peewee	150	165	315
VJ piping	250	-	250
Cold box heater	2600	-	2600
Total cryogenic heat load	**3250**	**3765**	**7015**

Total estimated heat loads are rough estimates based on review of operation to date, and estimates of flow rates and heat loads based on engineering formulae and industry standards.

Thermodynamic properties, CS geometry

	LH_2 (PeeWee)	LH_2 (Unit 2)	LD_2 (Proposed)
Operating Pressure (kPa)	200	100	100 - 200
B.P. (K)	23.0	20.4	23.2 – 25.9
M.P (K)	14	13.8	18.8 - 19.0
Density (kg/m³)	67.5	70	164 - 157
Geometry	Elliptical	Elliptical Annulus	Cylindrical
Dimensions (cm)	11	32 x 24	40 x 40
LH_2/LD_2 Thickness (cm)	4.5	2.3	3.2
Liquid Volume (L)	0.45	5	35
Mass (kg)	0.03	0.32	5.2
Al Mass (kg)	0.14	2.8	7.2

Parallel Operation of Deuterium CS and Hydrogen CS



Parallel Operation of Deuterium CS and Hydrogen CS

Options/solutions/tests

Additional Cooling to Hydrogen Condenser.

Additional Heating to Deuterium Condenser.

Change Hydrogen Cryostat Operating Pressure.

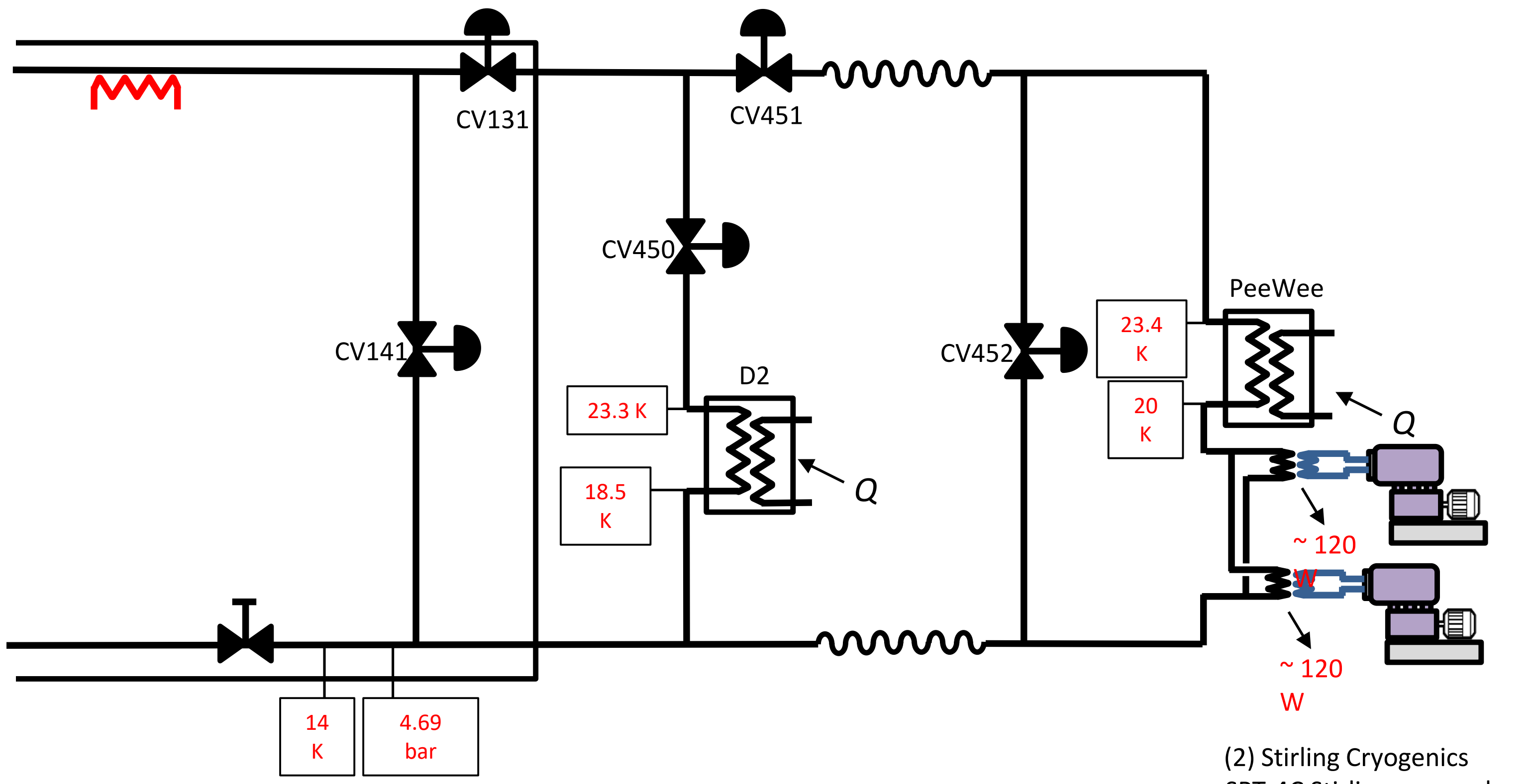
Re-certification of Deuterium/Helium Heat Exchanger.

Plumbing/VJ modifications – Flexible Helium Load Lines

Options/solutions/tests

Additional cooling

Remove heat from Peewee stream

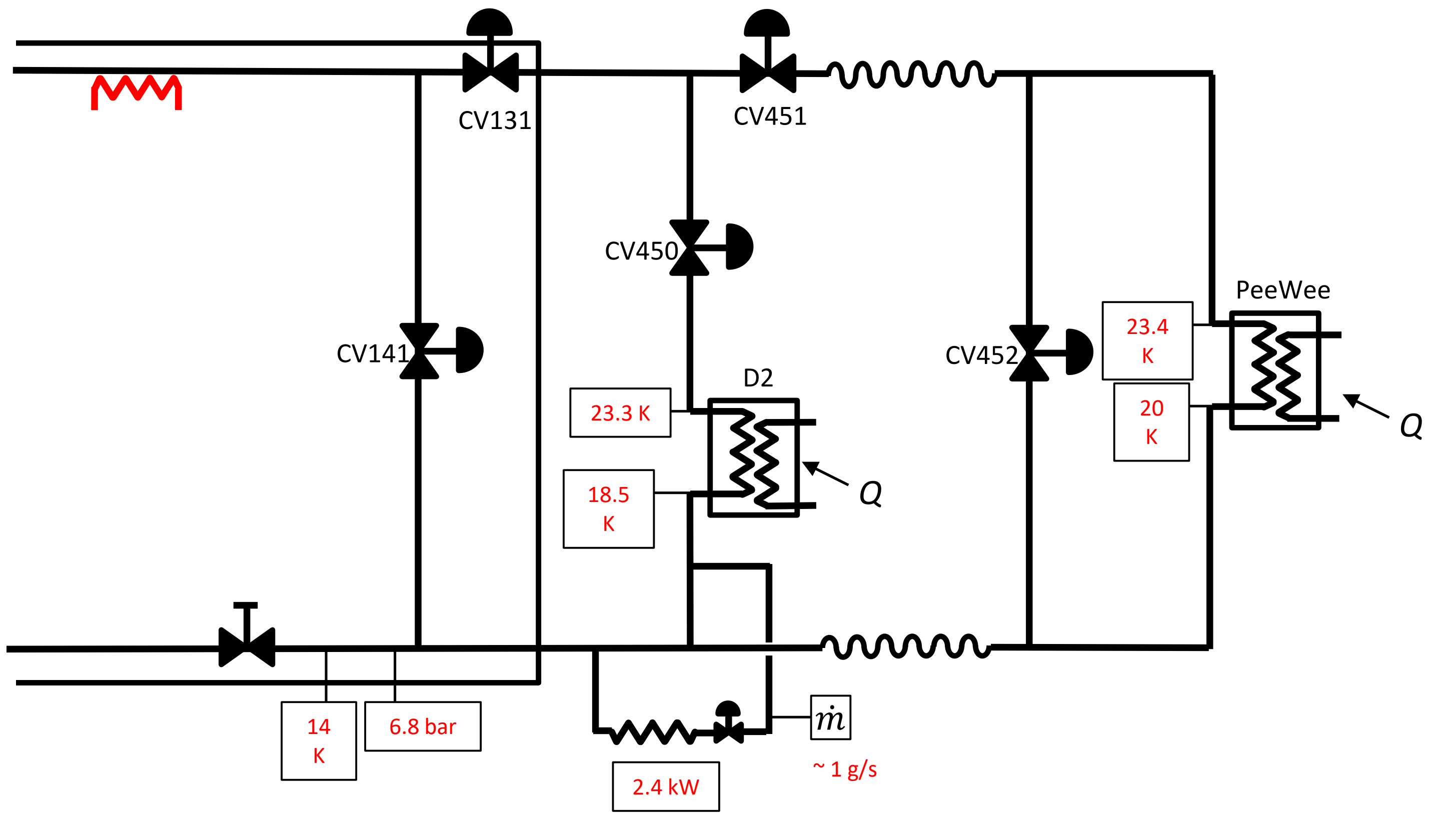


(2) Stirling Cryogenics
SPT-4C Stirling cryocoolers

Options/solutions/tests

Additional heating

Add heat to U3 stream



Change Hydrogen Cryostat Operating Pressure.

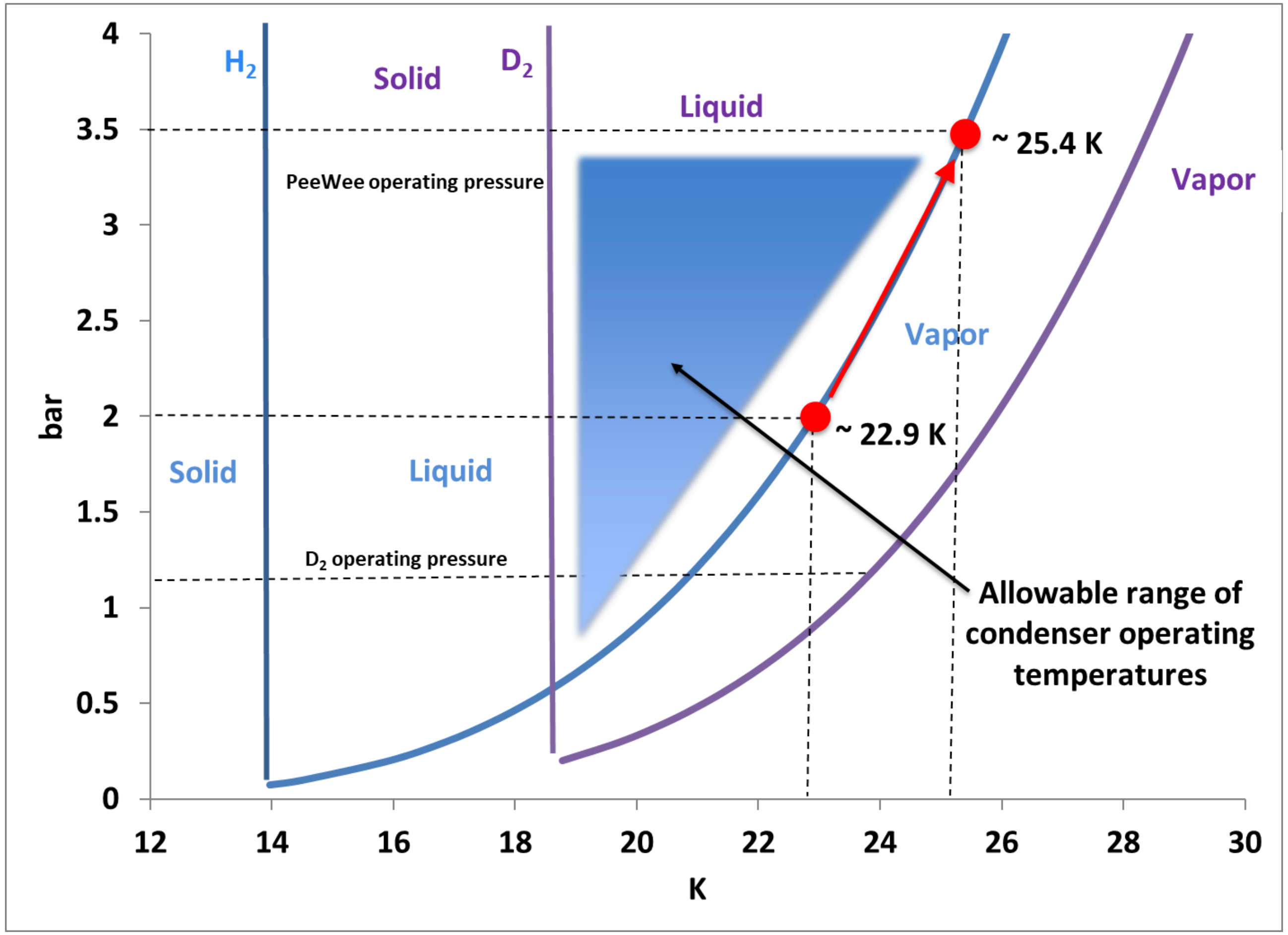
Re-certification of Deuterium/Helium Heat Exchanger.

ASME U-stamp, Currently rated at 500 kPa(`75 psia).

Contract to Obtain R-stamp Increasing Rating to 600 kPa(`90psia).

Increased Operating Pressure would Raise LH_2 Saturation Temperature.

Allow operation at 350 kPa(~52 psia) with a Saturation Temperature of 25.4K.


H2
Solid
D2
Liquid
~ 25.4 K
Vapor
PeeWee operating pressure
Vapor
~ 22.9 K
Solid
Liquid
D2 operating pressure
Allowable range of condenser operating temperatures
bar
K
4
3.5
3
2.5
2
1.5
1
0.5
0
12
14
16
18
20
22
24
26
28
30

Bench Mark Nuclear Heat Loads

Summary of cryoplant tests to date

	Flow [g/s]	Turbine Exhaust Pressure [bar]	Turbine Exhaust Temp [K]	Load Return Pressure [bar]	He Return Heater Temp Set Point [K]	Heat Load [kW]
Current operation	200	4.7	14.4	4.6	18	4.0
7/26/18 test	250	6.32	14	6.2	18	5.8
11/27/2018 test	257	5.2	13.6	5.07	18	6.4
11/29/2018 test	220	5.23	16.8	5.14	22.5	7.0
	267	5.44	13.6	5.3	18	6.7

Temperature Control of Ballast Tank located Outside

89% of the Deuterium Inventory stored Outside, 0 – 100 ^{o}F, (255 – 310 K).

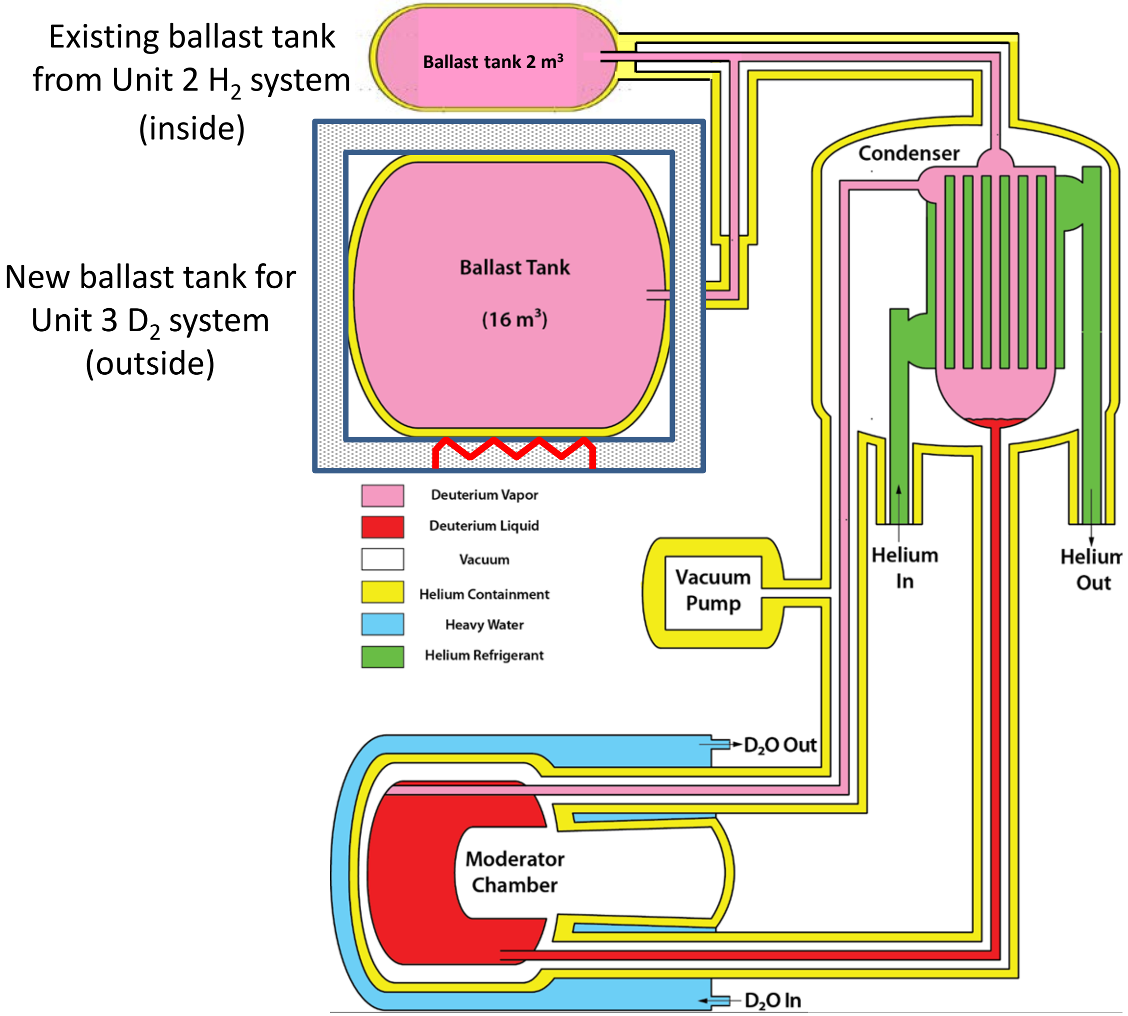
Temperature Fluctuations will Change LD_2 inventory in Cryostat/Condenser.

During Winter Months, More D_2 in the ballast tank and less LD_2 in the Cryostat/Condenser.

Decreasing LD_2 by ~2.5 L, enough to Drain the Pool under the Condenser Plates

Operate LD_2 source at constant pressure by maintaining ballast tank temperature using Heating/Cooling Blanket.

Operate with Variable Pressure to maintain Constant Density.

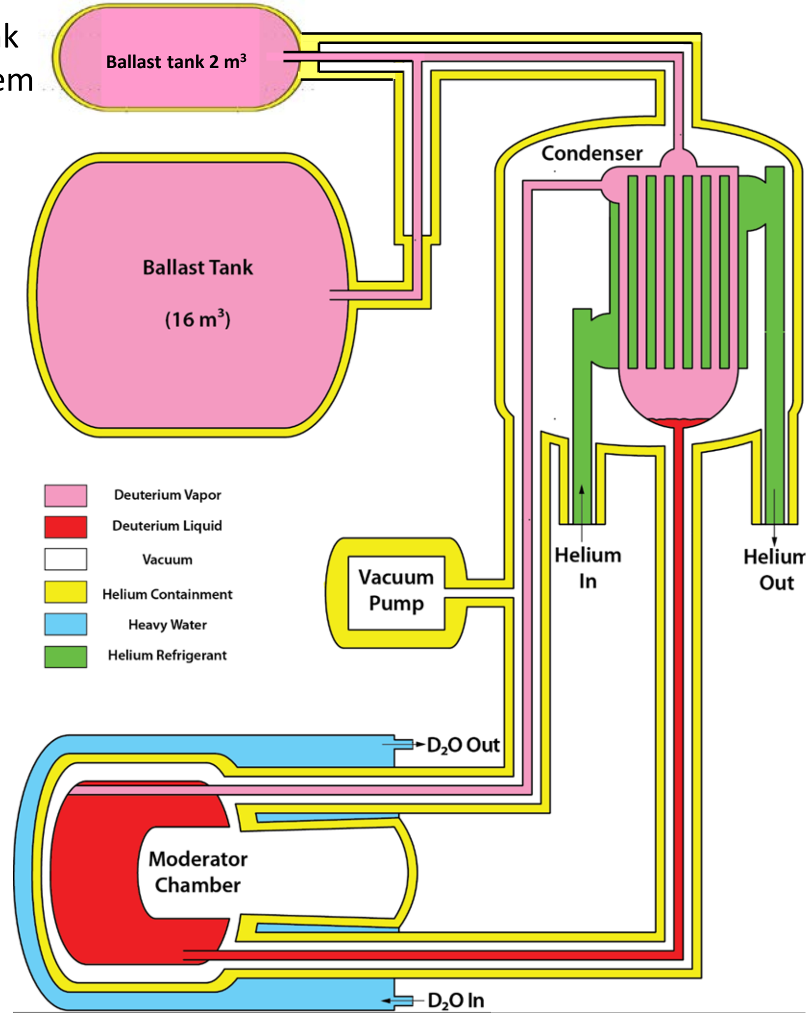


Option 1

Insulate outside ballast tank & provide heater to maintain constant system pressure

Existing ballast tank from Unit 2 H_2 system (inside)

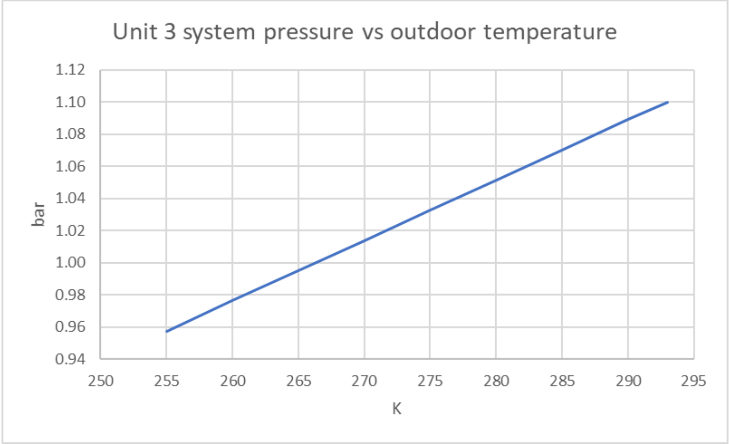
New ballast tank for Unit 3 D_2 system (outside)



Option 2

Vary system pressure to maintain constant density with varying ambient temperature.

A preliminary MCNP calculation has determined that a 20 kPa drop in pressure, which will increase average void fraction from 13% to 16%, resulting in a 5% drop in the cold neutron flux



QUESTIONS

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OSU Radiation Center

Update on Status of OSTR Instrumented Fuel Element

Robert Schickler
Steven Reese

Oregon State University Radiation Center

2019 TRTR Conference
Idaho Falls, ID
September 23rd, 2019

Oregon State University

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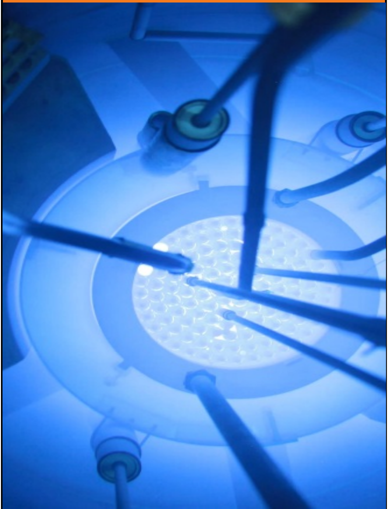


Timeline

May 2018: Performed a $2.20 pulse and noticed a 45^{o}C jump on IFE temperature the next day, from ~340^{o}C to ~385^{o}C.

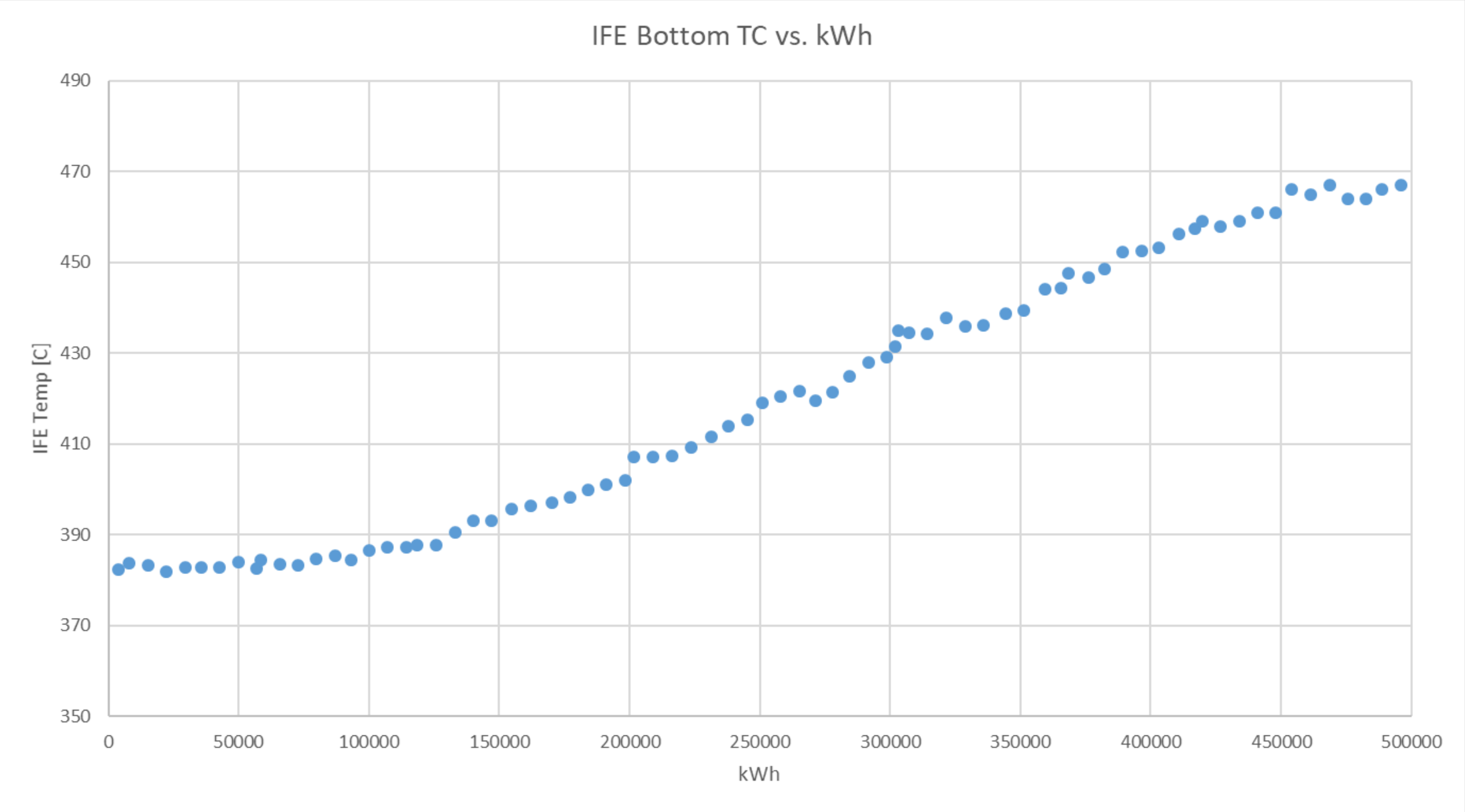
July 2018: Temperature rose to ~410^{o}C. Fuel inspection was performed on IFE and surrounding elements. All found to be acceptable with no visible defects or swell.

October 2018: Temperature rose to ~450^{o}C. Attempted to install spare IFE in core only to find that two of three thermocouples were failed open. Spare IFE was removed and dry-stored for possible immediate replacement.

Timeline

November 2018: Submitted LAR to allow operation without IFE as long as pulsing is precluded.

At this point, fuel temperature reached 470ºC (LSSS of 510ºC).



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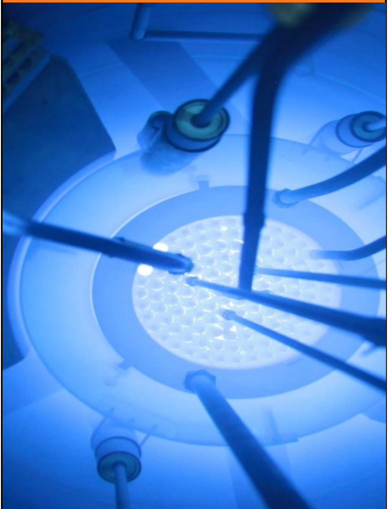

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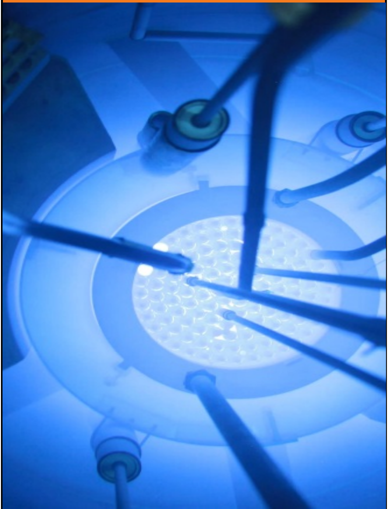


Timeline

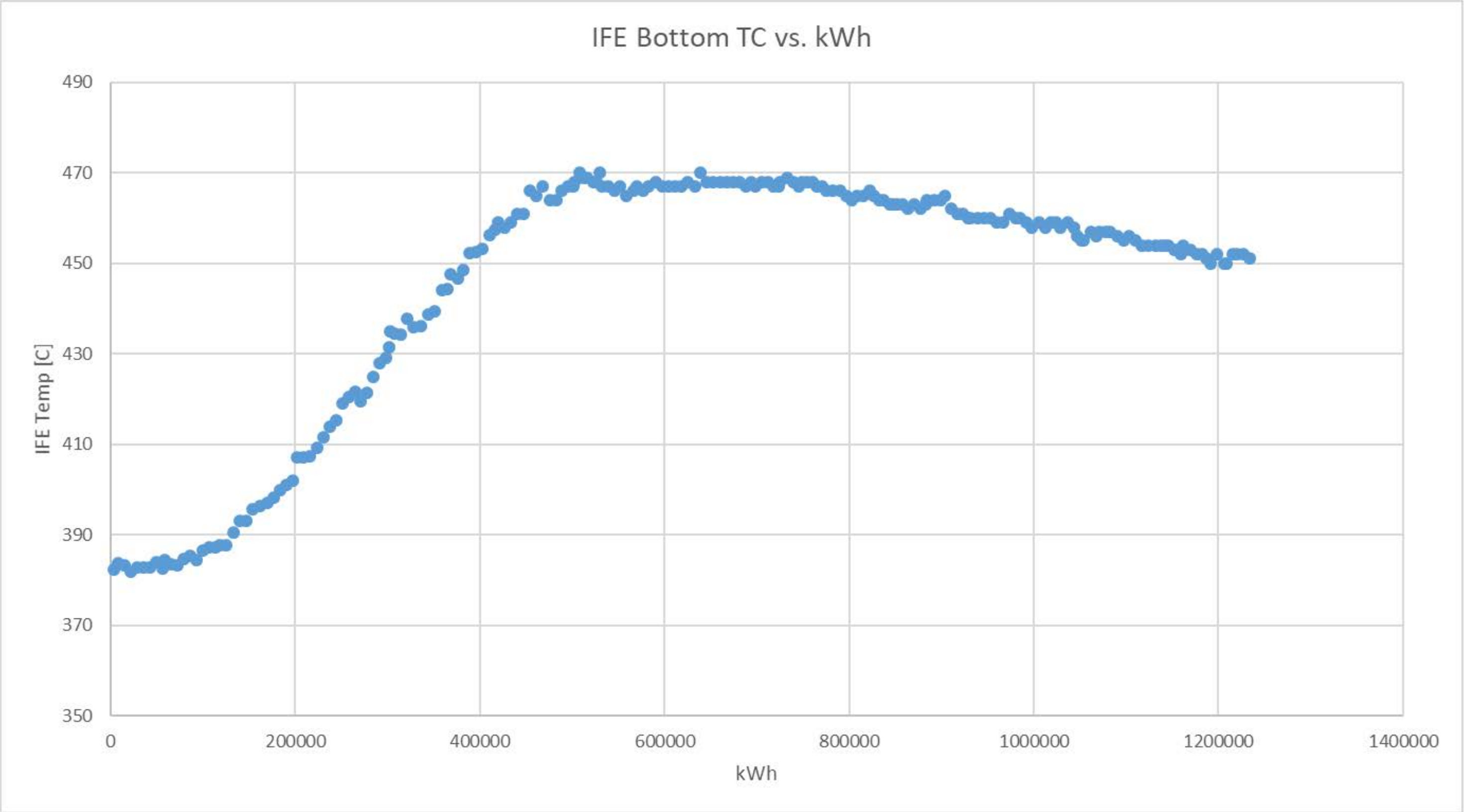
December 2018: Received spare IFE from Penn State (thanks to Jeff Geuther and Doug Morrell!). Tested thermocouples and they were all operable. IFE was dry-stored in anticipation of possible need for immediate installation.

Also, additional analysis is needed before insertion due to differences in erbium content (Penn State IFE has 0.9% erbium, Tech Specs require nominal 1.1%).

Timeline

OSU Radiation Center

April 2019: After peaking at 470^{o}C, temperature gradually decreased to 450^{o}C, reducing immediacy of IFE replacement. Still working with NRC on LAR. End of month NRC Physical Security inspection. Spare IFEs found to be improperly stored. More on that later.



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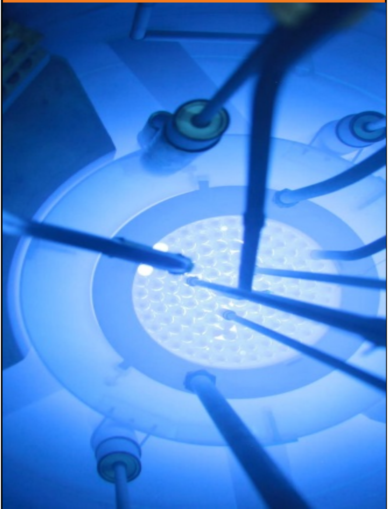

Timeline

June 2019: LAR requested approved! We would like to thank Mike Balazik for being incredibly helpful in getting this completed in a timely fashion.

July 2019: OSTR receives a Level IV violation for improper fuel storage. Confusion between MAA and PSP requirements. Staff commits to clarifying fuel handling procedures and retraining on proper fuel storage.

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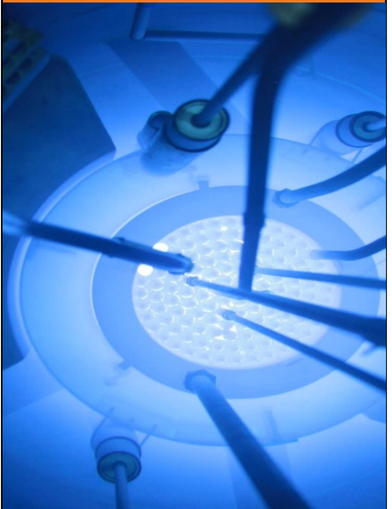
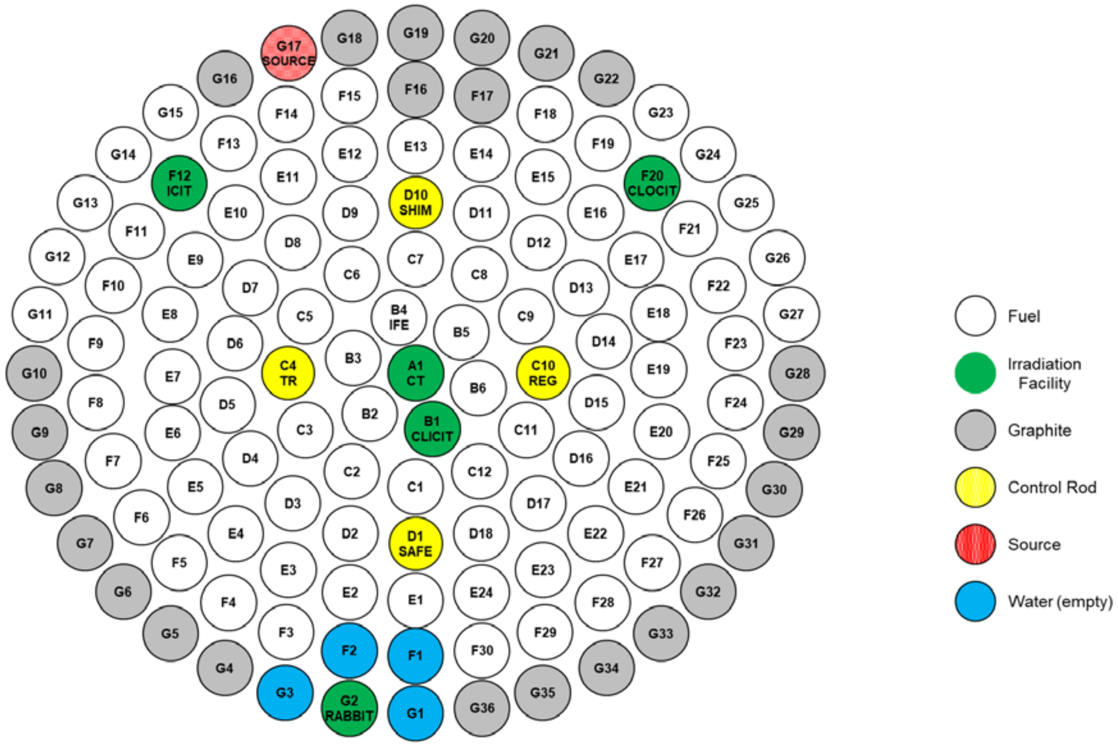


Core Reconfiguration

IFE removed from service on 7/29/19 and fuel temperature meter disconnected. LSSS now based on 1.1 MW on power channels. Core reconfigured for operation without IFE.



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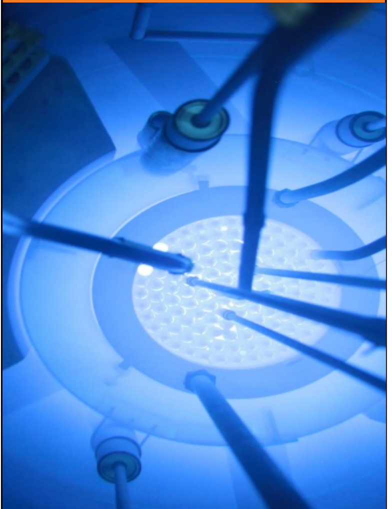

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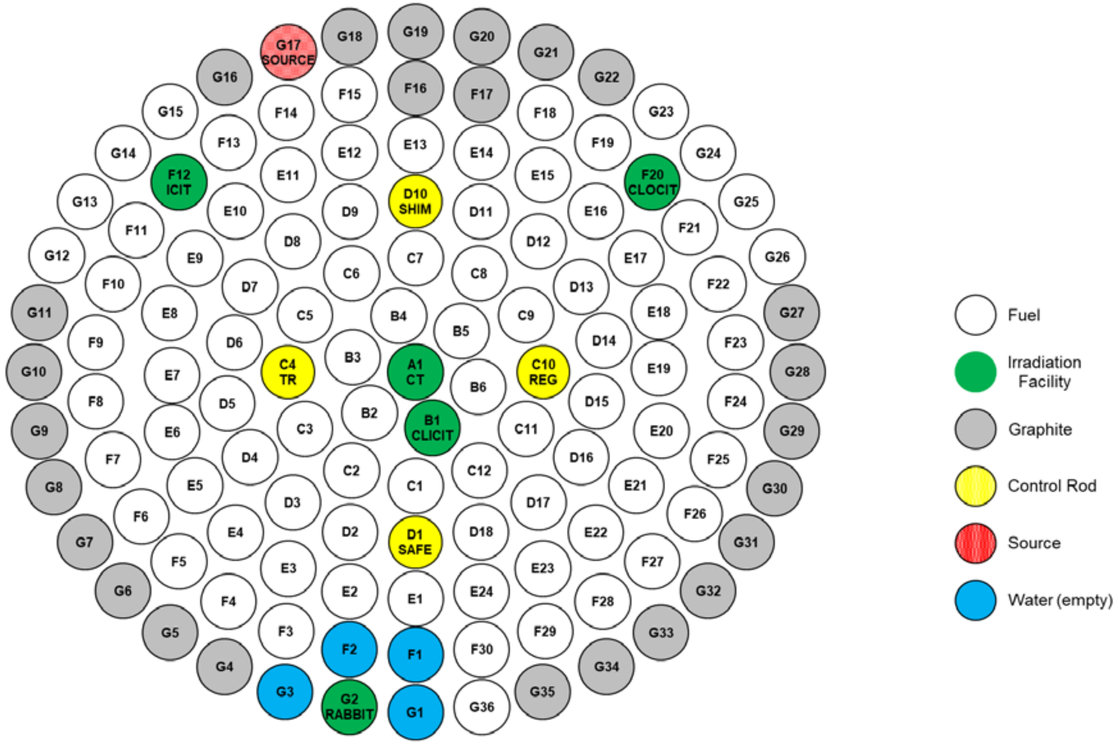

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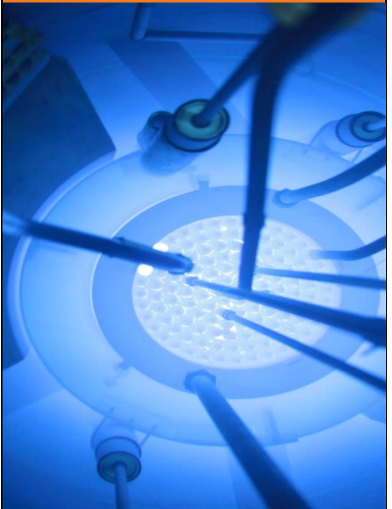


Path Forward

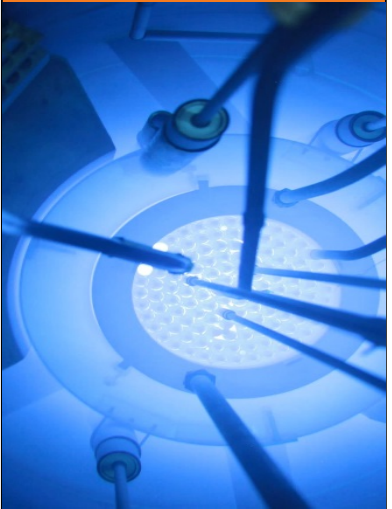
Eventual goal is to regain ability to pulse the reactor, but without the need for an IFE. Plan is to have LAR submitted by the end of 2019.

Analysis was performed during our LEU conversion in 2007, but needs to be updated and incorporated into the Tech Specs to allow for pulsing without IFE.

Neutronic Analysis

In order to perform thermal hydraulic analysis, neutronic analysis (MCNP) must be performed in order to calculate:

- Maximum Power-Per-Element
- Hot Channel
- Hot Channel Peaking Factor
- Axial Peaking Factor
- Radial Peaking Factor
- Effective Peaking Factor (product of three factors)
- Limiting Core Over Fuel Lifetime



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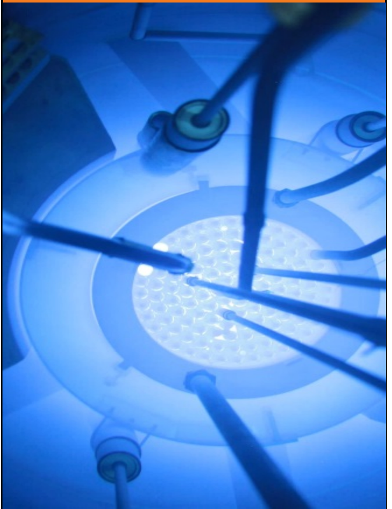
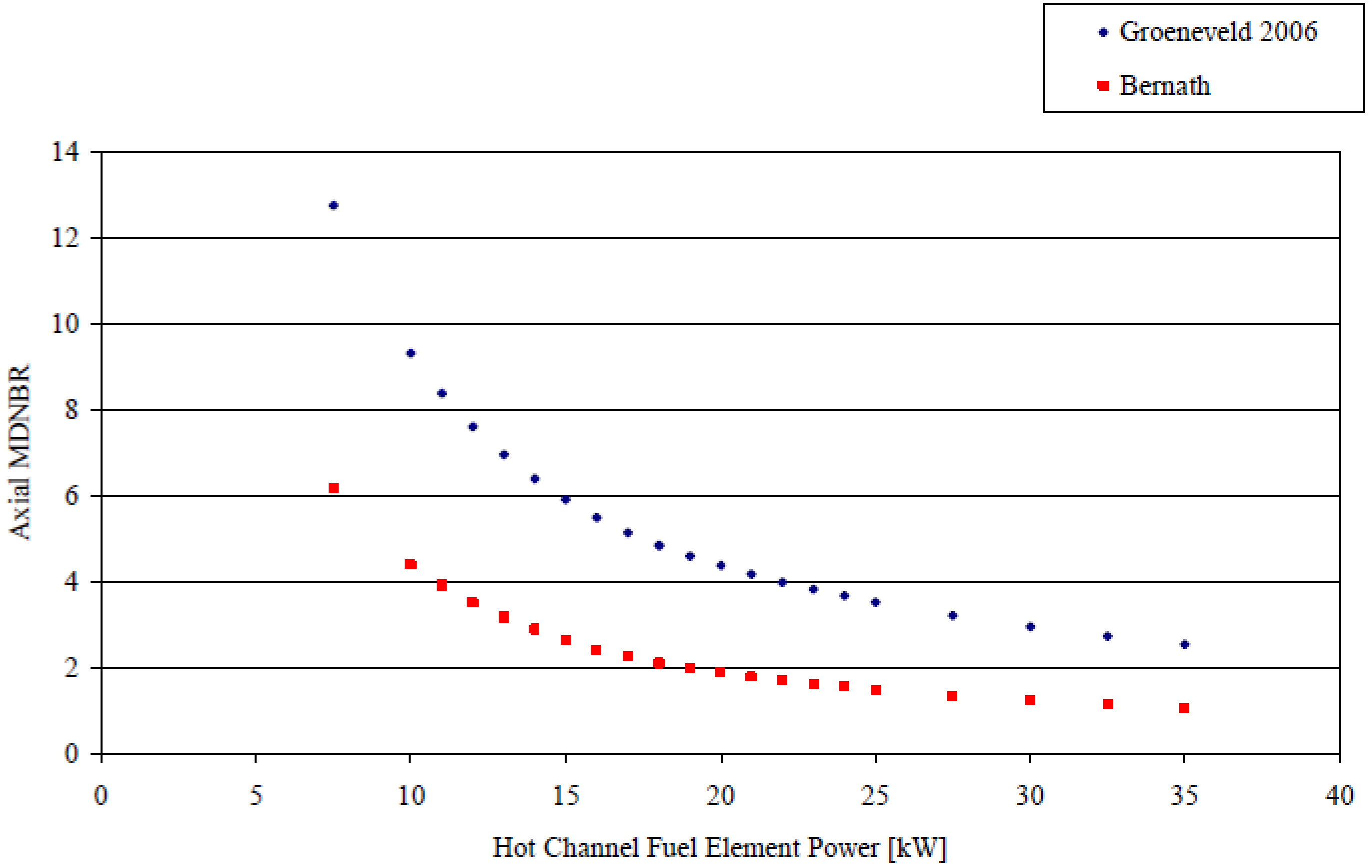
Thermal Hydraulic Analysis

OSU Radiation Center



Once the limiting core configuration is decided, a thermal hydraulic analysis will be performed using RELAP to determine maximum hot channel power to keep DNBR above 2.



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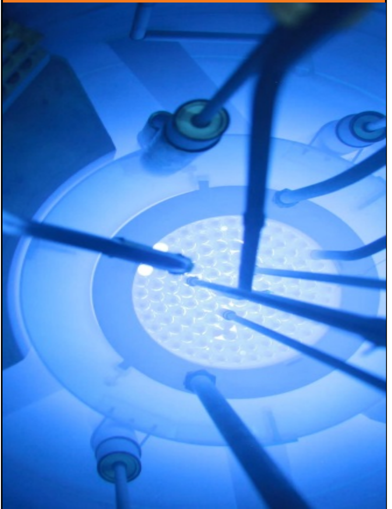
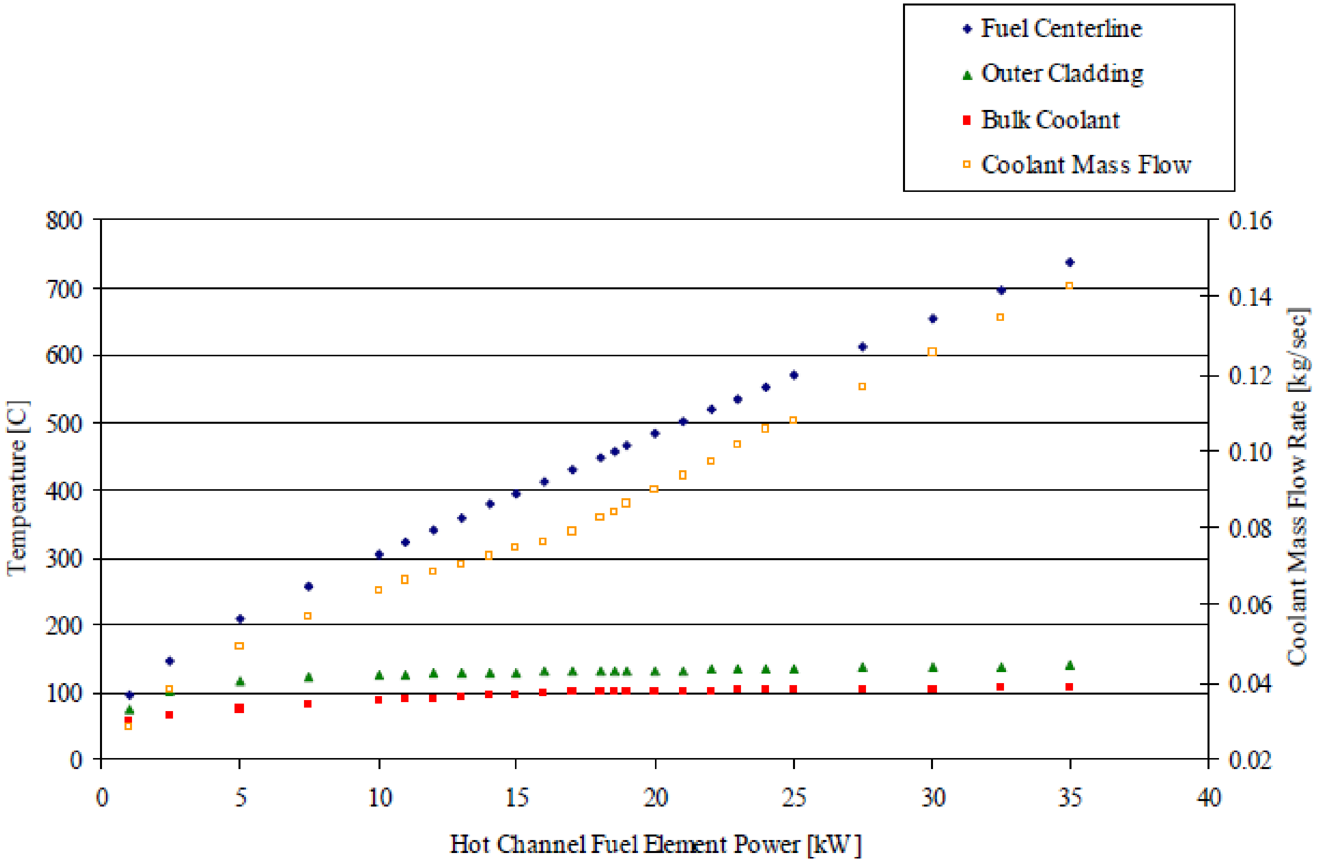


Thermal Hydraulic Analysis

We can also use RELAP to determine the corresponding temperature produced in the hot channel in order to determine limiting fuel temperatures/power-per-element.



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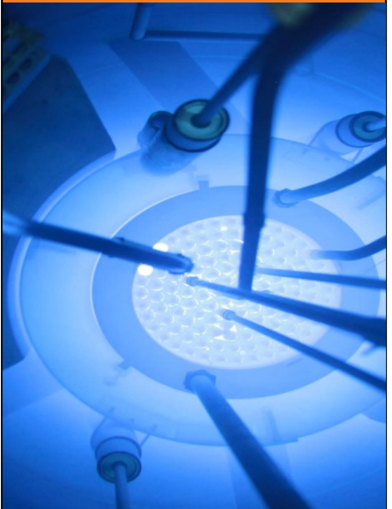
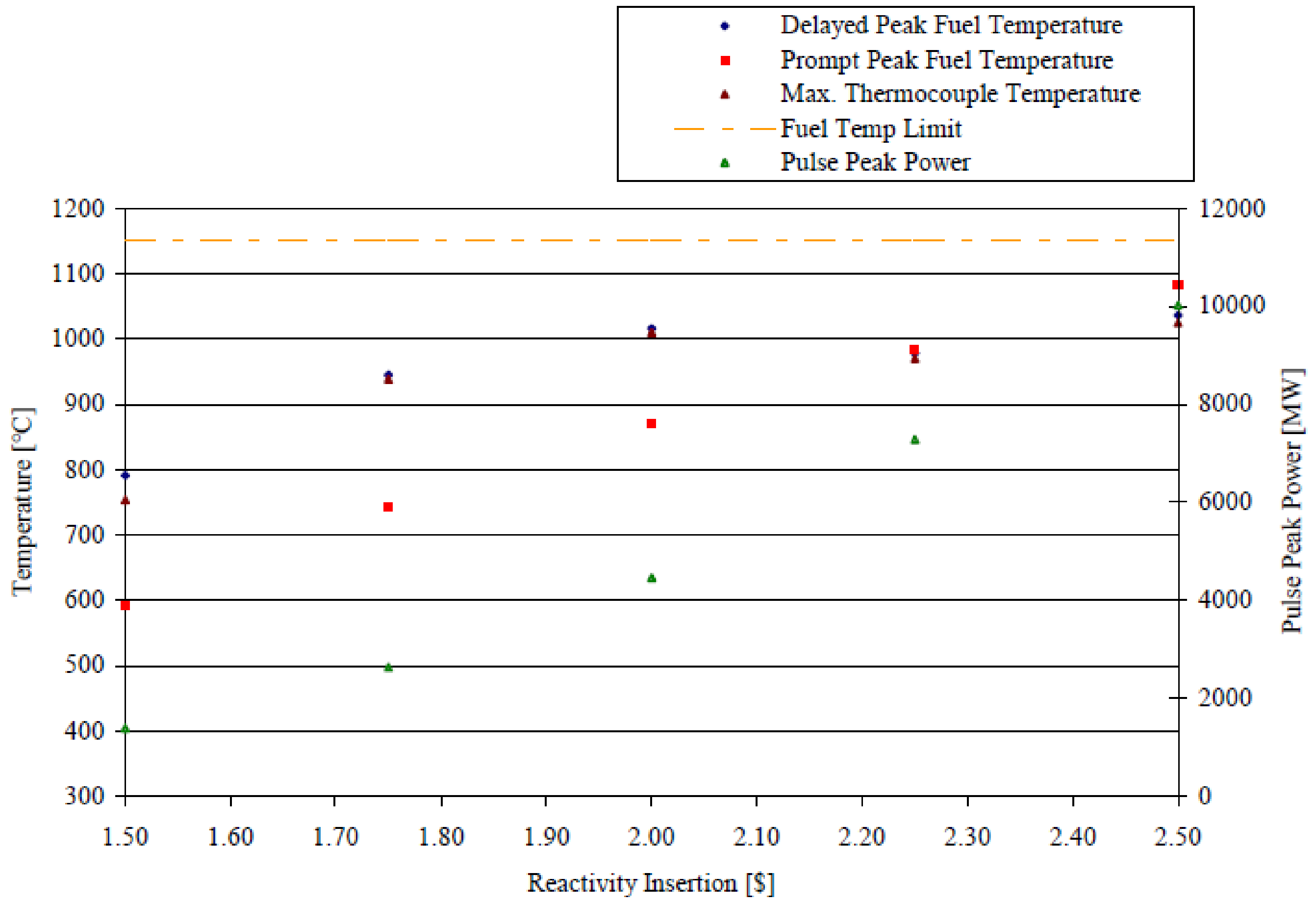
Thermal Hydraulic Analysis

OSU Radiation Center

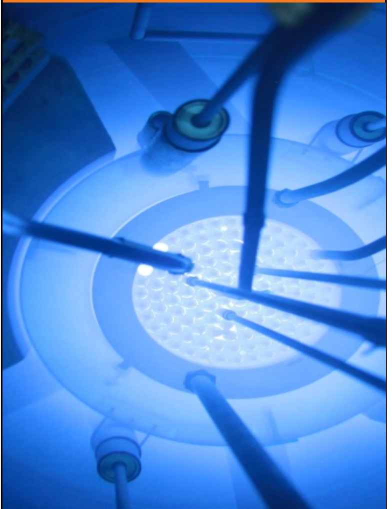


RELAP can also be used to determine the maximum peak fuel temperature during a pulse in order to determine maximum reactivity insertion.



Oregon State University

OSU Radiation Center

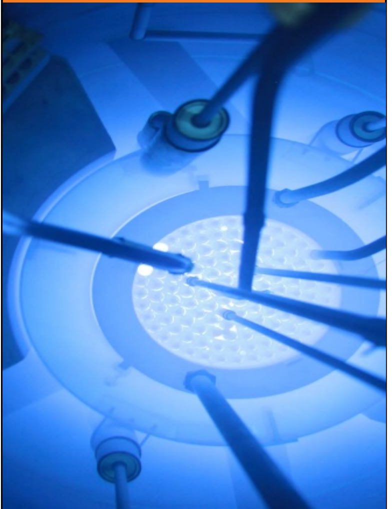


Thermal Hydraulic Analysis

This work was previously performed by Dr. Wade Marcum in support of LEU conversion (Marcum “Thermal hydraulic analysis of the Oregon State TRIGA Reactor using RELAP5-3D”, 2008). There is sufficient analysis to justify the removal of the IFE and the subsequent return of pulsing capability.

Oregon State University

OSU Radiation Center



End Goal

We believe this work will benefit the TRIGA community.

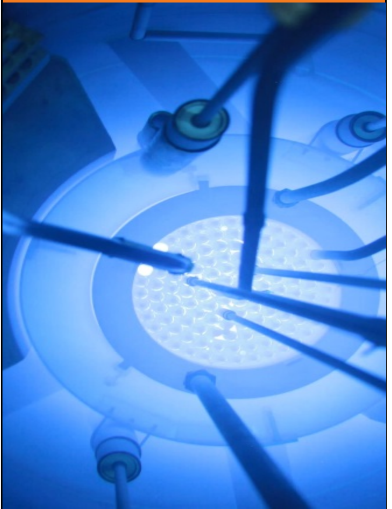
IFEs are more expensive. Removing IFE requirements will save money. Currently 11 TRIGAs utilize IFEs. That could mean significant savings during the next fuel purchasing cycle (assuming one IFE per TRIGA).

OSU Radiation Center

End Goal

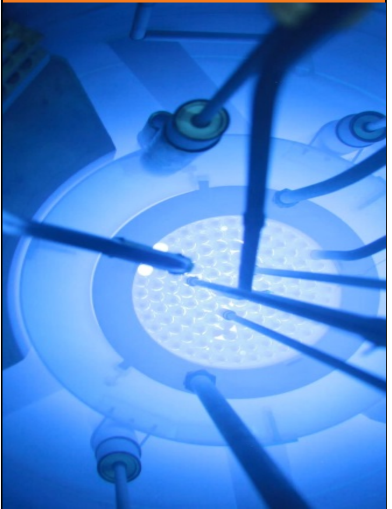
IFEs can be faulty and thereby cause a reactor to remain shutdown. We nearly experienced a shutdown due to our IFE conundrum!

While IFEs are an interesting tool for information, they are ultimately limiting on operation and an unnecessary expense.

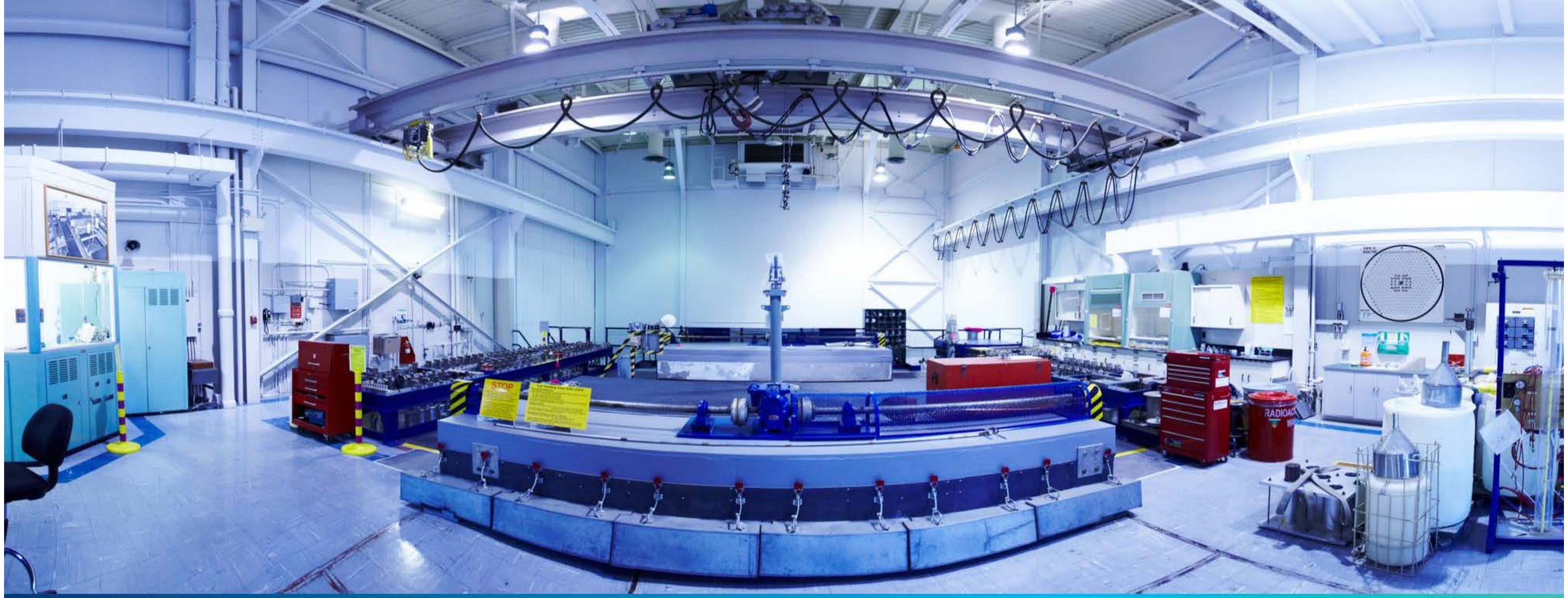


Oregon State University

OSU Radiation Center



Questions?



59 Years and Counting – Ongoing Research Activities in the ZED-2 Reactor

L. R. Yaraskavitch, J. E. Atfield, J. C. Chow, and N. D. Lee

TRTR 2019

Sept. 22, 2019

Acknowledgements

ZED-2 Facility:

D. Trudeau, D. Brushey, J. Horner, S. Mirault,
G. Hamilton, K. Thomson

Applied Physics:

J. Atfield, J. Chow, N. Lee,
L. Li, X. Wang, E. Rand,
S. Livingstone

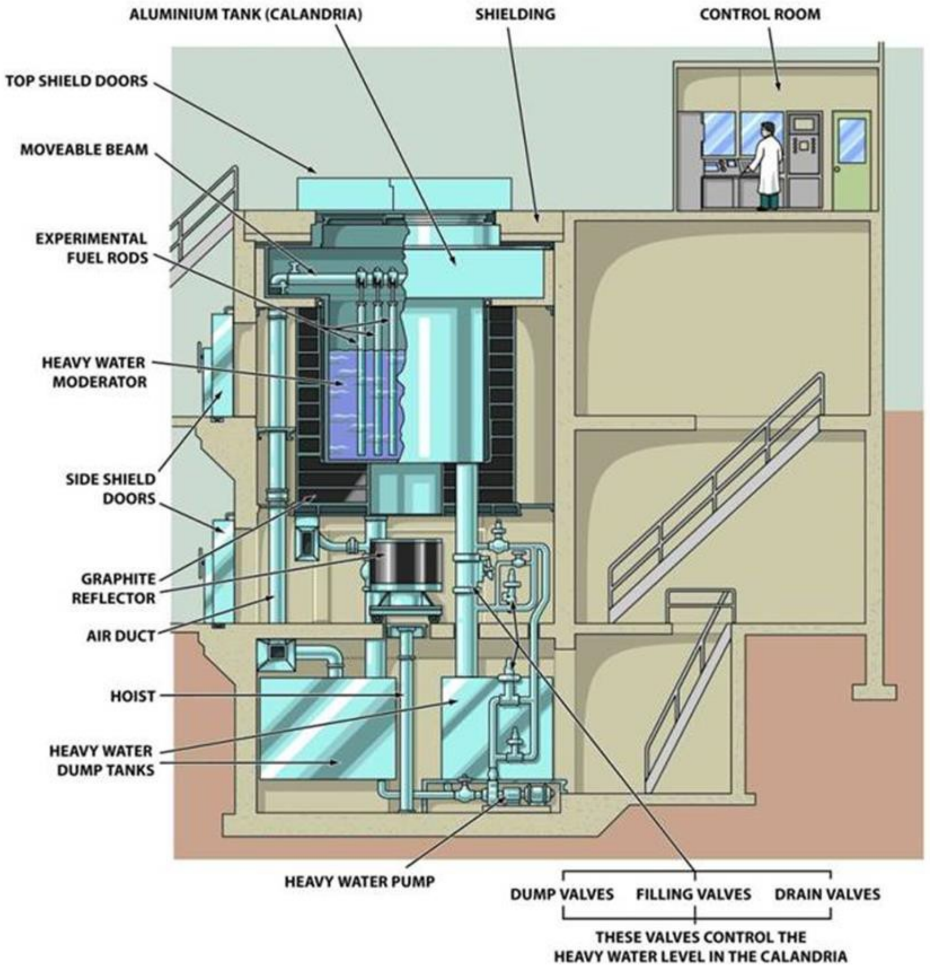
Computational Techniques:

K. Hartling, B. Bromley, F.

ZED-2 - Zero Energy Deuterium

59 years and counting

- Successor to ZEEP – Zero Energy Experimental Pile
- First criticality: 7 September 1960
- Tank type: reactor control via moderator level
- 2524 cores built, 190 of them unique, and counting
- Integral part of the reactor physics design of **all Canadian power reactors**, and much more!



Quick Facts

Power: up to ~200 W (thermal)

Peak Neutron Flux: $1x10^9$ n/cm^2 s thermal, $5x10^8$ n/cm^2 s fast

Calandria: 3.36 m in diameter, 3.35 m in height

Fuel: Various types and assemblies

Moderator: Heavy water (99.8 to 97.5 weight% D_2O), soluble poison capability, temperatures up to 90°C, and variable core height (criticality achieved by pumping moderator into calandria)

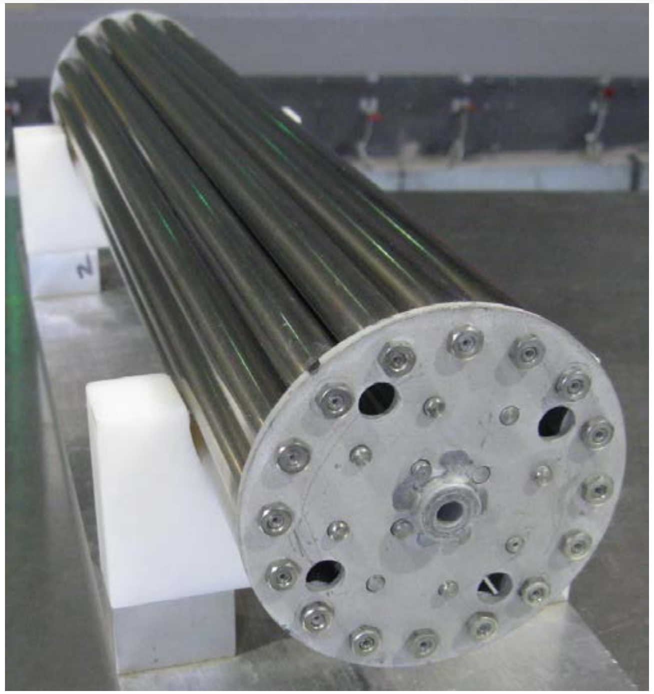
Core Geometry: flexible, typically square and hexagonal lattices, with variable pitch

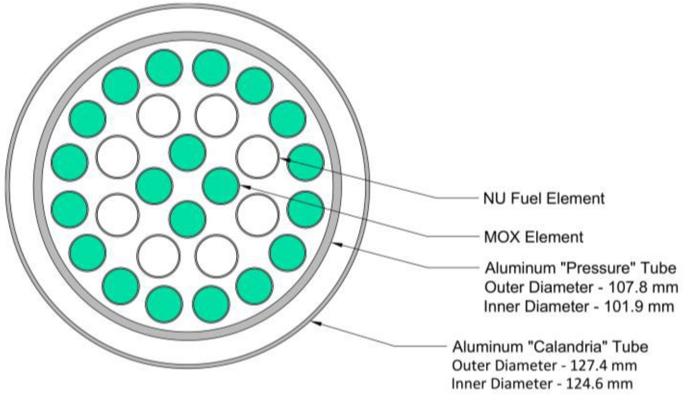
Coolant: Heavy water, light water, air, CO_2, organics, Pb-Bi, etc. (not active). Temperatures up to 300°C in some channels.

Flexibility: We can operate with new fuels/coolants/materials as required

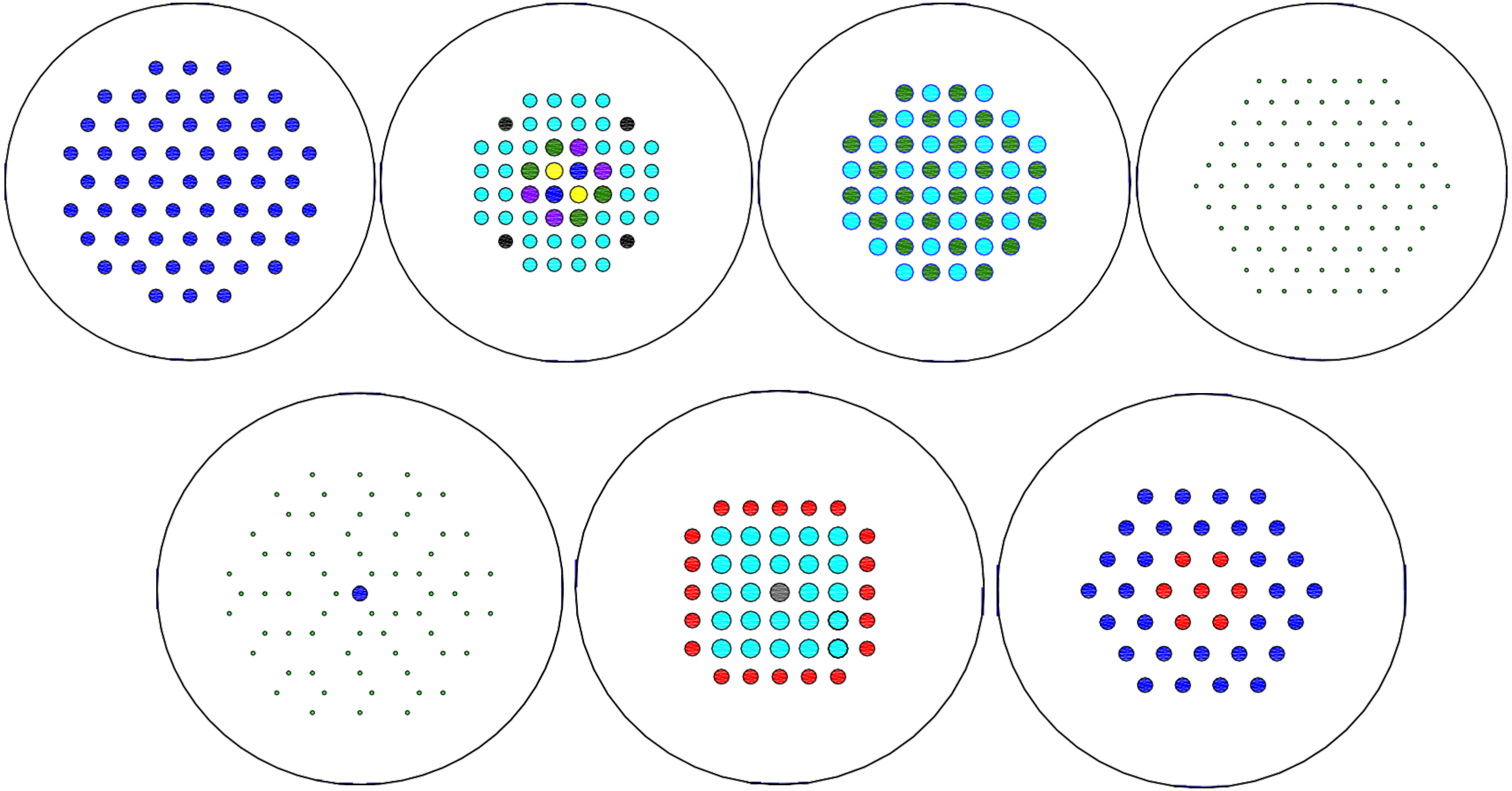
Fuel

- Natural UO_2 Bundles:
  - 7, 19, 28, 37, and 43-element
- Other Natural U flavours:
  - Metal, Carbide, Silicide bundles
  - ZEEP rods
- Mixed oxides
  - Pu-U, ^{233}U-Th, Pu-Th, ^{235}U-Th
- Bundles with burnable absorber (Low Void Reactivity)
- Enriched or reprocessed UO_2 bundles (LEU, RU)
- Assembly geometry: bundles in Pressure Tube/Calandria Tube, clad rods, etc.





Fuel Lattices in ZED-2



Timeline

- 60's – Metal and oxide fuels, D_2O, air, He, organic coolants, pitches from 20 cm to 40 cm (CANDU support)
- 70's – simulated boiling light water (CANDU BLW), enriched U booster rods, liquid absorbers, coupled cores, kinetics measurements, $(Pu, U)O_2$, shutoff rod materials and shapes, reactor regulating systems, Self Powered Flux Detectors, NRU loop site simulation, adjuster rods, 37 el. lattice physics, Th-UO_2, NRX,
- 80's – Co and Cd absorber rods, $(Pu, Th)O_2$, ^{99}Mo for NRU, simulated NRU loop, simulated burned up fuel, $(^{233}U, Th)O_2$
- 90's – Coolant Void Reactivity (fresh and mid-burnup), delayed neutrons, Low Void Reactivity Fuel, 43 element CANFLEX
- 00's – Advanced CANDU Reactor
- 10's – Reactor kinetics, $(Pu, Th)O_2$, $(^{233}U, Th)O_2$

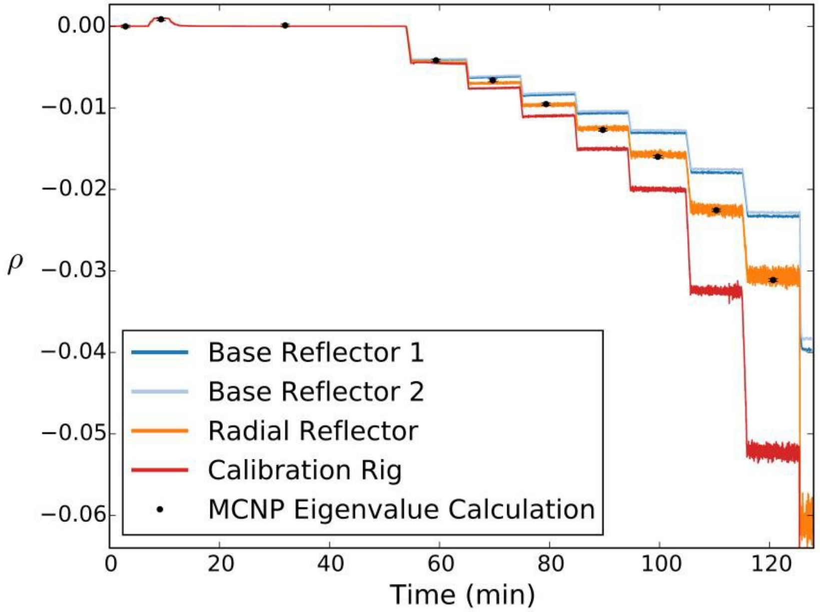
One of **three** of its type operational in the world: MAKET (Russia) and AHWR-CF (India)

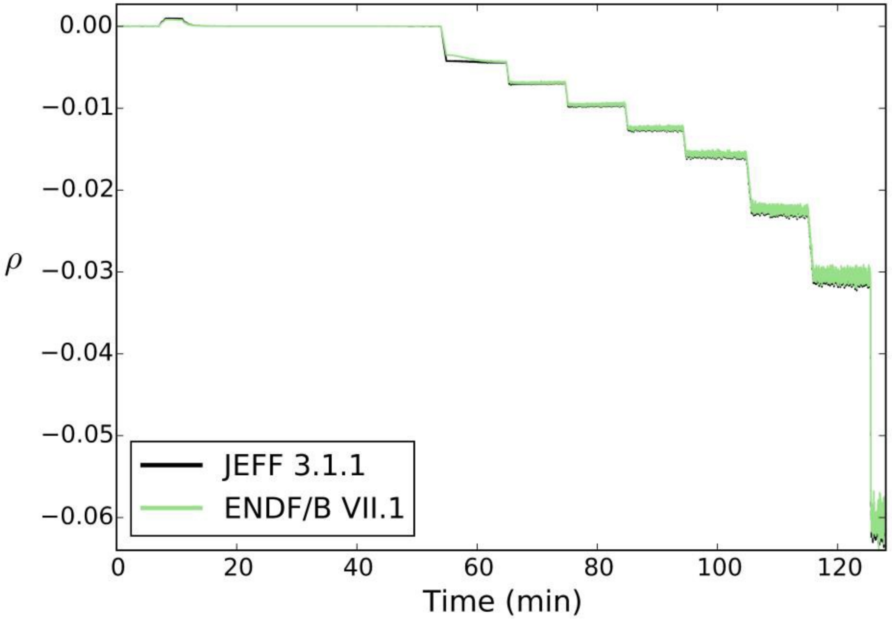
How about now?

Recent history + near future

- Challenge spatial simplification of the reactor
  - Point kinetics versus 3D kinetics
- Challenge quality of delayed neutron data
  - Direct delayed and delayed photoneutron
- Driven by:
  - Advanced fuel cycles programs 2015-2018
    - (U, Pu)O_2, (Pu, Th)O_2, and (^{233}U, Th)O_2
  - ZED-2 research in support of CANDU physics 2018-2021
    - CANDU-relevant, i.e. NU oxide → depl. U, Pu oxide

Advanced Fuel Cycles Programs

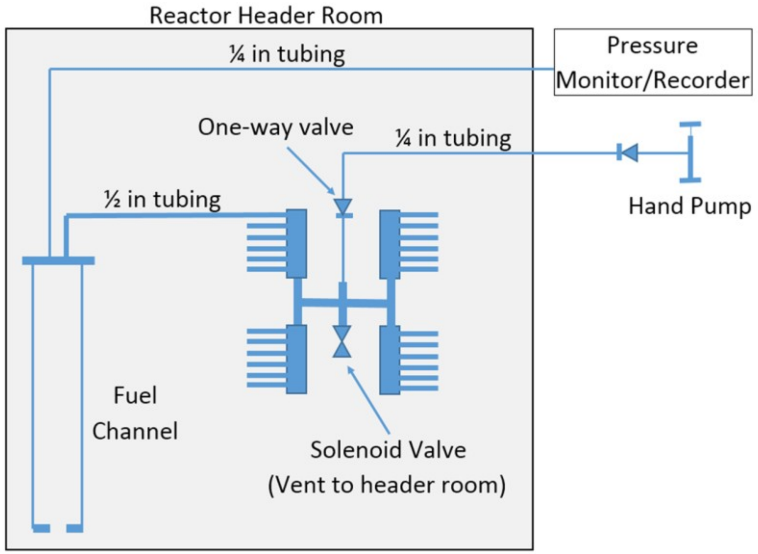


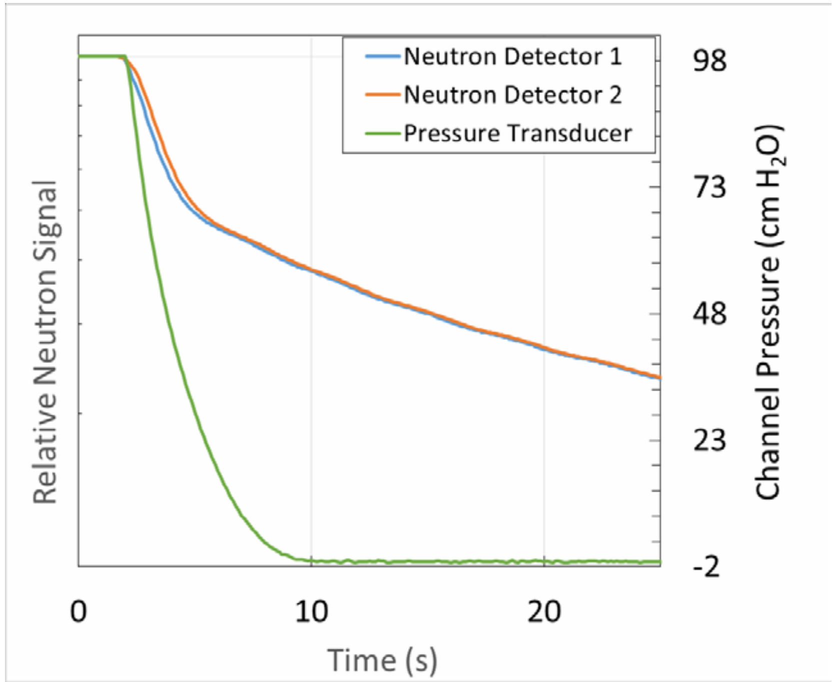


- Transient – First such experiments in ZED-2 with nuclides other than ^{235}U and ^{238}U contributing to fission
- Varied nuclides, varied reactivity insertion (0.2 mk to ~30 mk) but also spatial variation!

"Kinetics experiments in ZED-2 using heterogeneous cores of advanced nuclear fuels", Annals of Nuclear Energy, 121 (2018) 36-49.

ZED-2 Research in Support of CANDU Physics

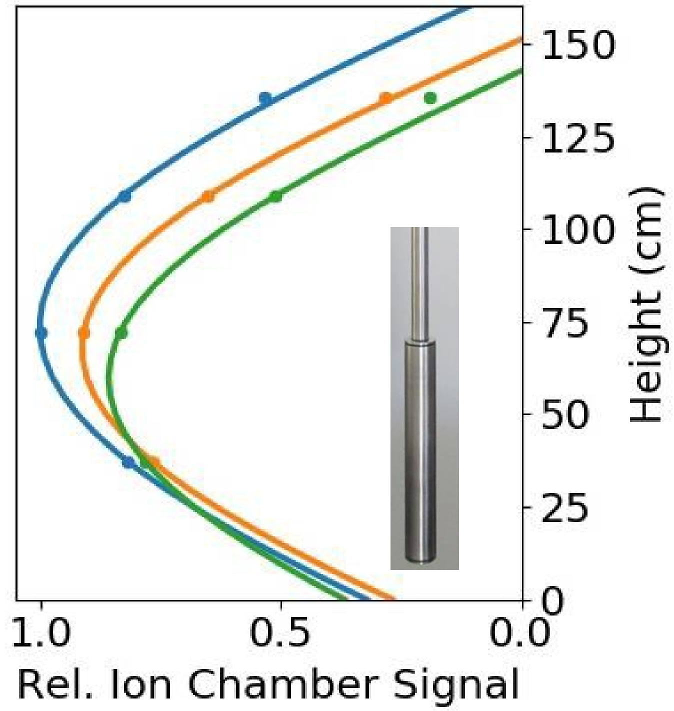




- Ongoing kinetics experiments and technique development (e.g., at-power coolant flood, now up to 24 channels)
- Testing the reverse of CANDU Loss of Coolant Accident – that is, testing opposite negative reactivity insertion

ZED-2 Research in Support of CANDU Physics

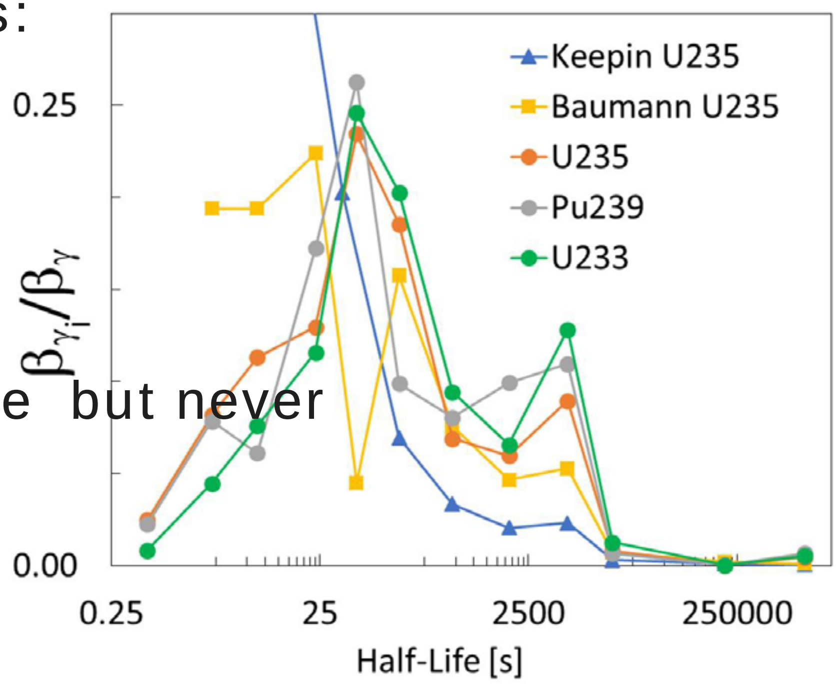
- Carrying on probing spatial variation from Advanced Fuels work
- More in-core detectors to track flux shape real time
- Reactor transfer function measurement and model development
    - Flux perturber(s)
- Even delayed neutron effectiveness carries some spatial dependence
- All good tests for 3D analysis methods



Photoneutron production

- Historically, delayed photoneutron groups in D_2O are stitched together results:
  - Experimental from 1947
  - Calculation from 1951
  - Experimental from 1973
- All are for ^{235}U
- Typically, yield is scaled by nuclide but never any change to group structure.
- Demonstrates what we can test

with our experiments

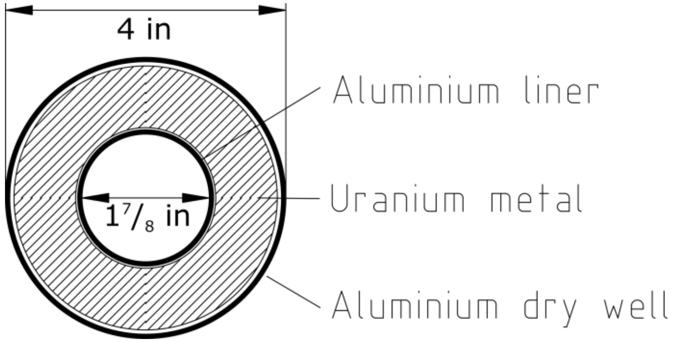


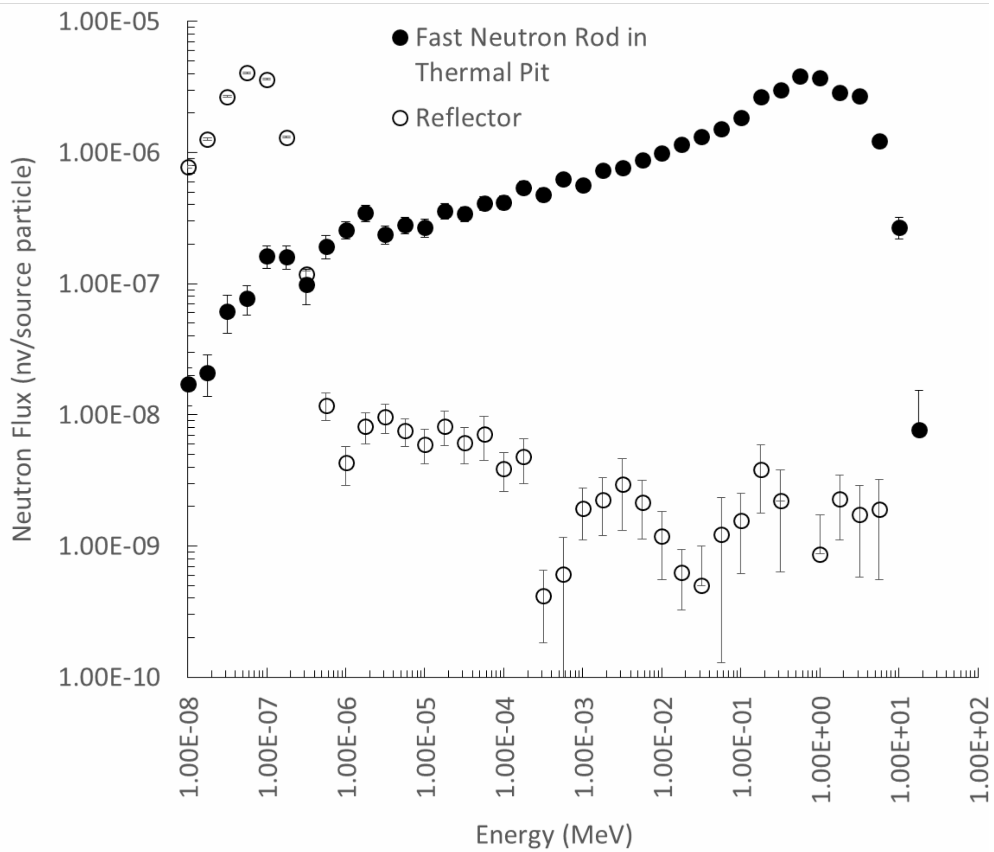
"Microscopic calculation of delayed-photoneutron production in D_2O using Geant4", Annals of Nuclear Energy 129 (2019) 390-398.

Fast Neutron Rod

Recommissioning capability

- ‘Transformer Rod’ of U-metal from 1960’s of interest once more for fast neutron irradiation and spectrum manipulation.
- Fast flux on order of 10^8 nv





- Thermal flux trimmed with Cd
- Appreciable dose to biological samples

ZED-2 for Education

Better aligning educational offerings with the needs of the community.

- Often the overlooked reactor on campus – now the only one, for now!
- Responsibility to community: educate, inspire, and ultimately contribute to training highly qualified personnel
- Recent history: ZED-2 Reactor Safety and Instrumentation School, 2010-2018 with 9 iterations
- **How do we better determine the market pull for what we offer?**
- **How do we reach the most people with limited resources?**

Conclusion

The flexibility of ZED-2 and ongoing investment has ensured relevance and utility into the future.

Ongoing work:

- Federal Science & Technology projects on CANDU reactor physics, including transients, are currently underway
- Commercial work for CANDU Owners Group.
- Flux detector calibrations for commercial clients

Future work:

- exploring zoned capabilities relevant to advanced reactor work – e.g. (thermal) molten salt reactors.

This work was funded by Atomic Energy of Canada Limited, under the auspices of the Federal Nuclear Science and Technology Program

Looking to the future…

- A.G. Ward, The Role of Critical Experiments in the Chalk River Power Programme, Proc., Exponential and Critical Experiments, Amsterdam 2-6 Sept. **1963**, IAEA
  - “Although one may hope for the day when the reactor-physics calculations are confidently based on computer programmes, with no recourse to experimental or critical facilities, it seems likely this happy time will only arrive when new reactor designs are no longer of interest”

ZED-2

EST. 1960

Thank you. Merci.

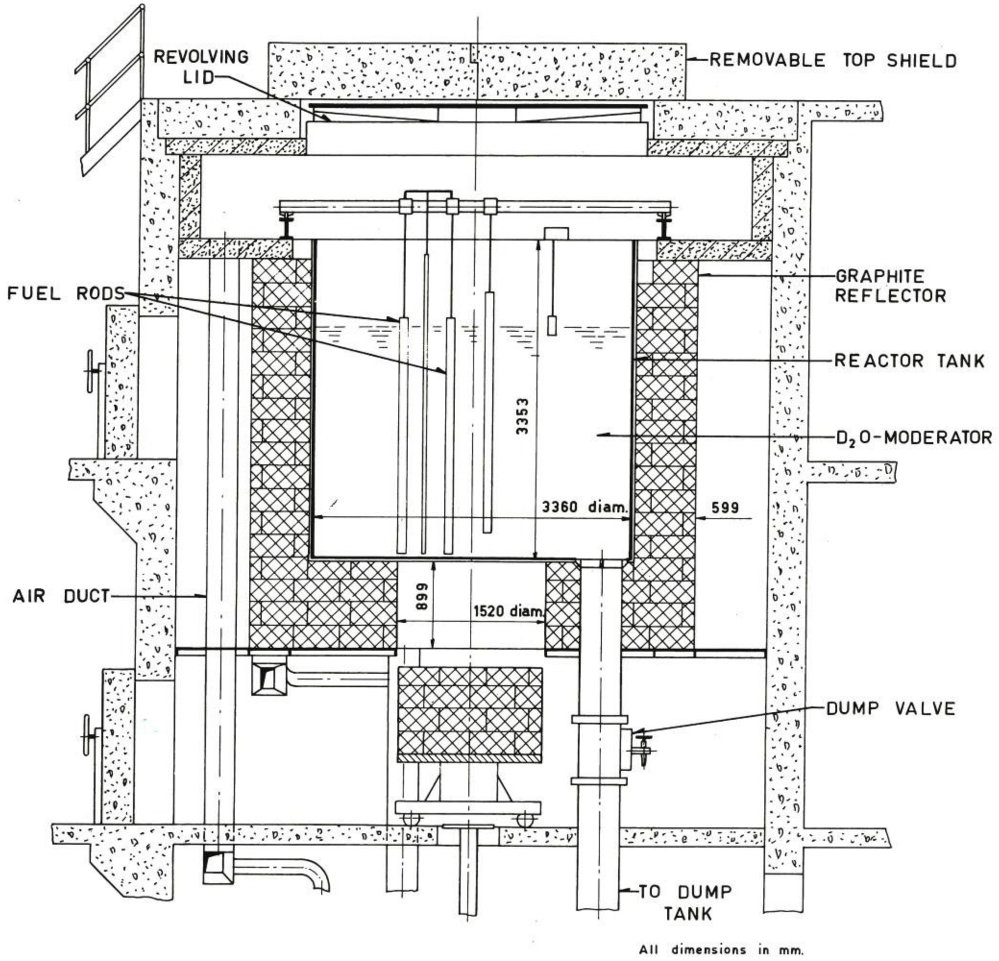
Questions?

Luke Yaraskavitch
Applied Physics Branch
Canadian Nuclear Laboratories
luke.yaraskavitch@cnl.ca

Useful Websites

http://www.cnl.ca	Canadian Nuclear Laboratories
http://www.cns-snc.ca	Canadian Nuclear Society
http://www.nuclearfaq.ca	Canadian Nuclear FAQ
http://canteach.candu.org	CANDU Owners Group Inc. (COG) CANTEACH Project
http://inis.iaea.org	IAEA International Nuclear Information System
http://www.nuceng.ca/candu/	The Essential CANDU textbook
http://www.nuclearheritage.ca	The Society for the Preservation of Canada's Nuclear Heritage
https://www.osti.gov/	U.S. DOE Office of Scientific and Technical Information
https://www.oecd-nea.org/science/wprs/irphe/handbook.html	OECD NEA Reactor Physics Benchmark Handbook
https://www.oecd-nea.org/science/wpncs/icsbep/handbook.html	OECD NEA Criticality Safety Benchmark Handbook

Facility Cross-Section



ZED-2 Capabilities

Value Proposition

In summary, ZED-2 measures critical configurations using its

- Large test region
  - Flexible fuel geometry
  - Flexible fuel type
- Zero power – negligible activation and fast turnaround

Practically, this lets us

- Measure reactor physics phenomena (e.g. fuel temperature coefficient of reactivity, absorber worth, kinetics parameters)
- Validate reactor physics codes
- Validate nuclear data

Limitations

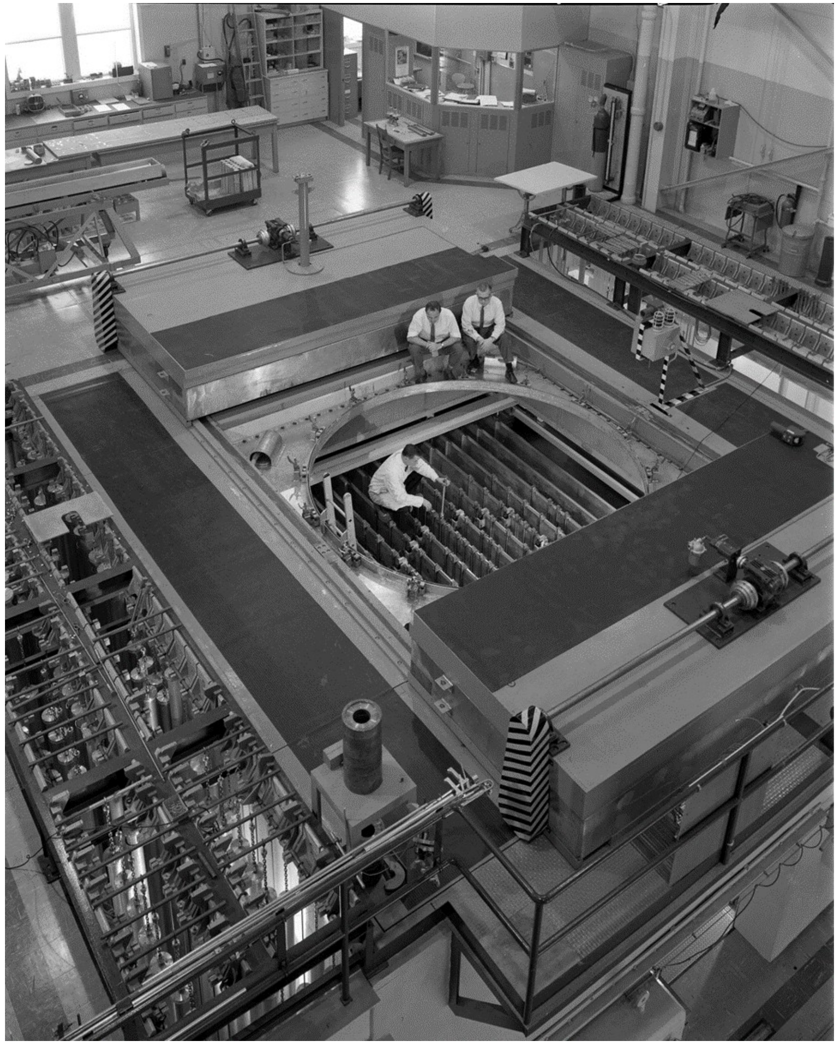
ZED-2 is not currently equipped to conduct:

- Irradiation/burnup experiments, or experiments with irradiated fuel (but fuel at a simulated level of burnup can be fabricated for ZED-2 by adding simulated fission products)
- Materials activation
- Neutron beam experiments
- Isotope production

Review and/or revisiting of the safety case may be required!

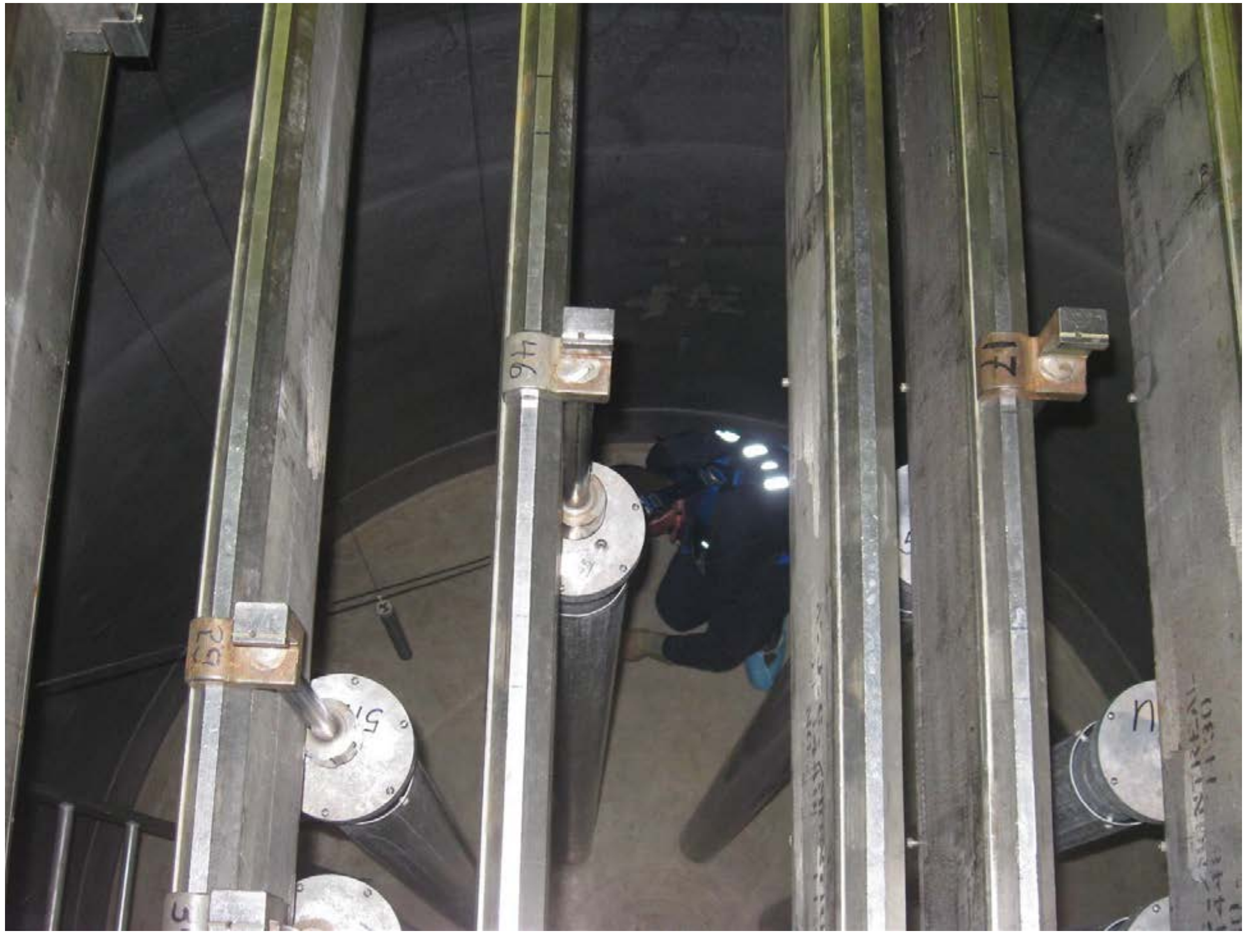
As designs pass through various stages of pre- and licensing review, needs will become clearer

Operations, 1970s

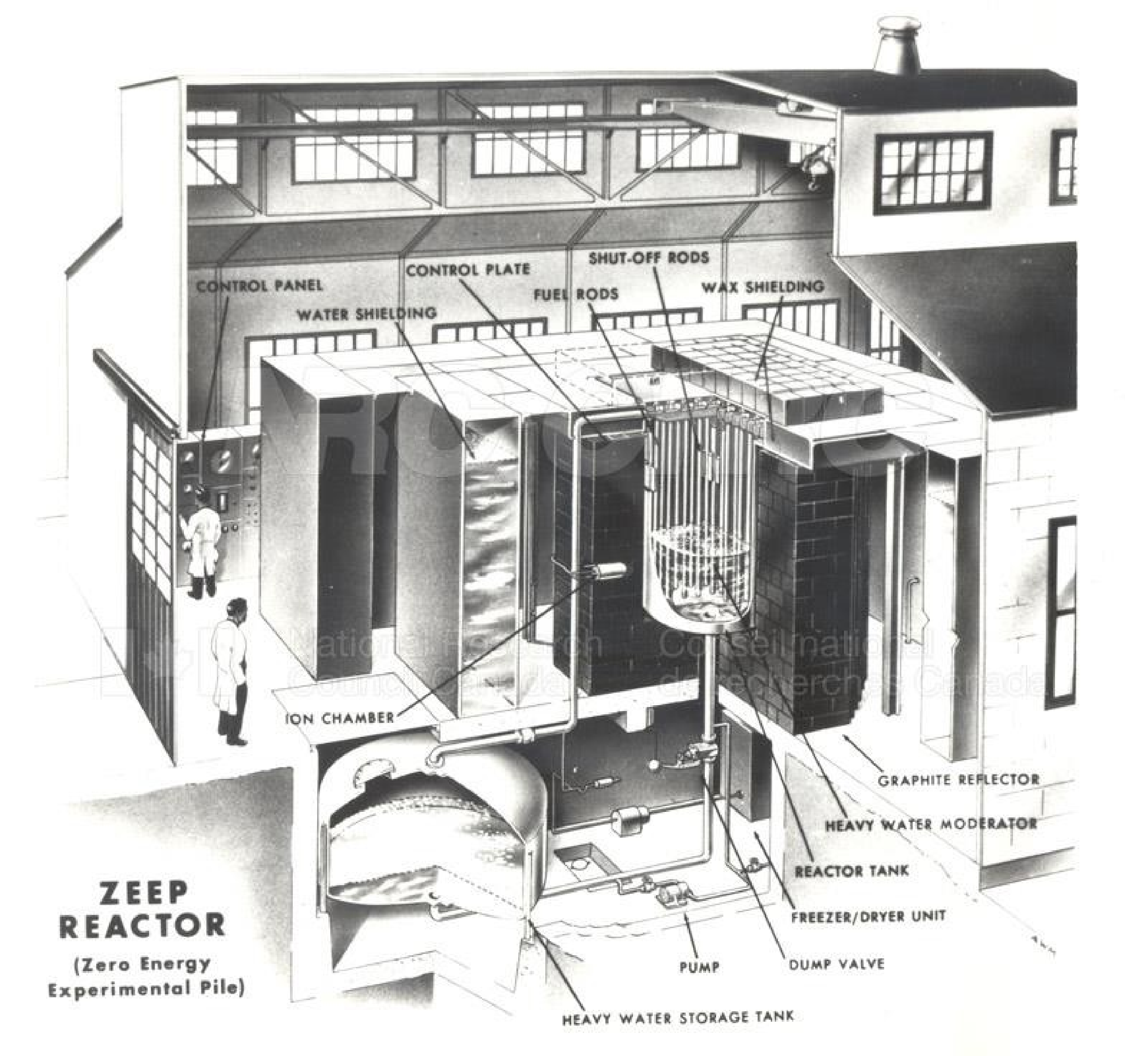


2014 ZED-2 Operations

View In Calandria

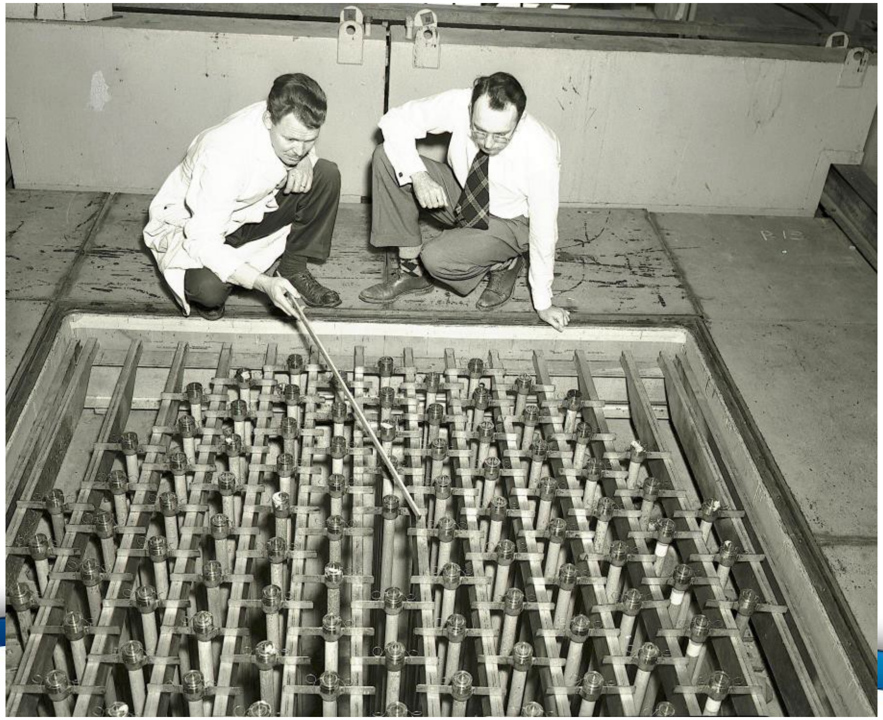


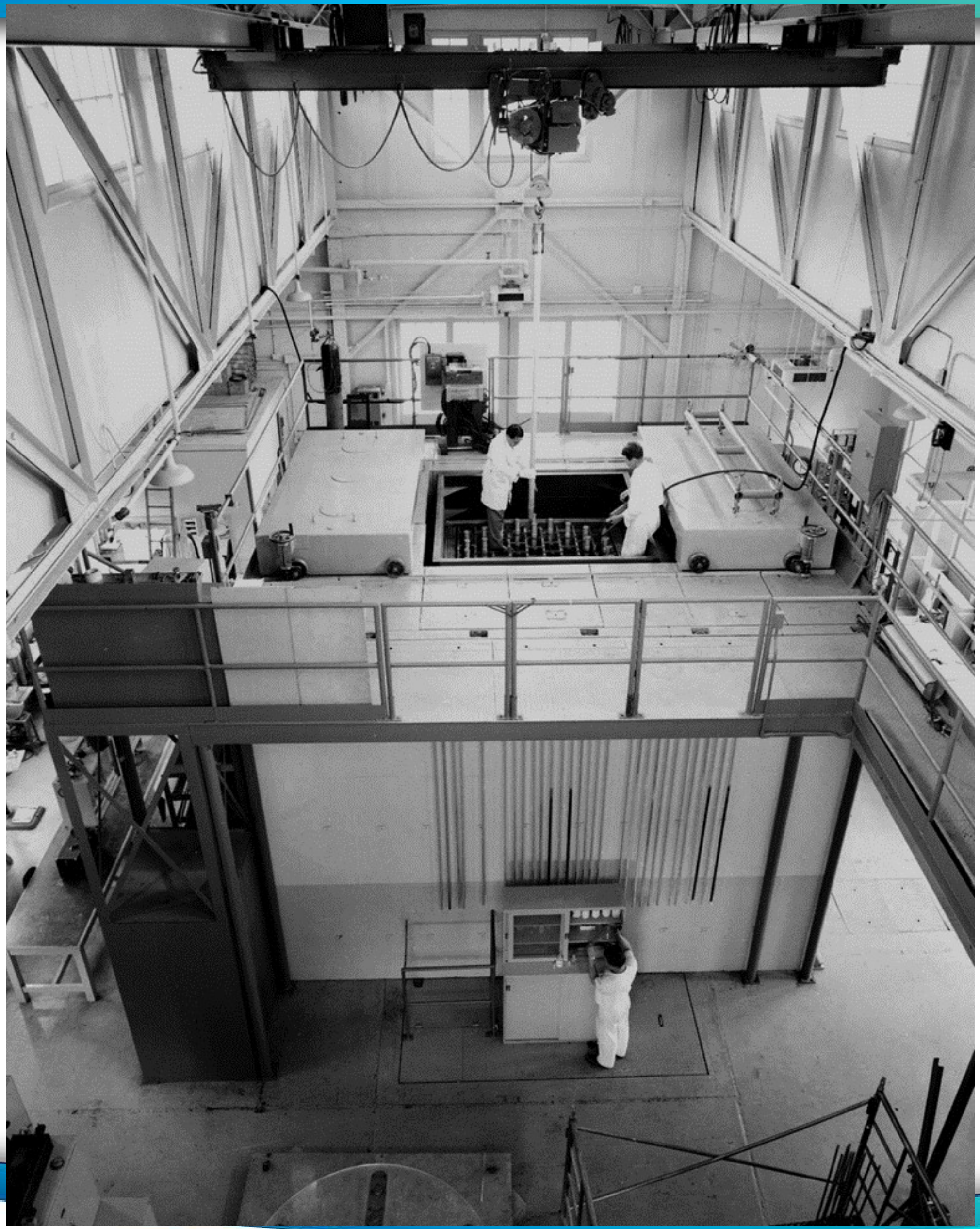
- Natural U-metal fuel (to start), heavy water moderator
- 5 September 1945, first critical!
- 1st reactor outside of U.S.
- 3.5 W, 30 W for short periods



Pictures: NRC National Science Library

- Flurry of activity to support NRX, after which D_2O was used to start NRX
- Tested NRU rod design
- Lattice physics for power reactors
- 25 years of service



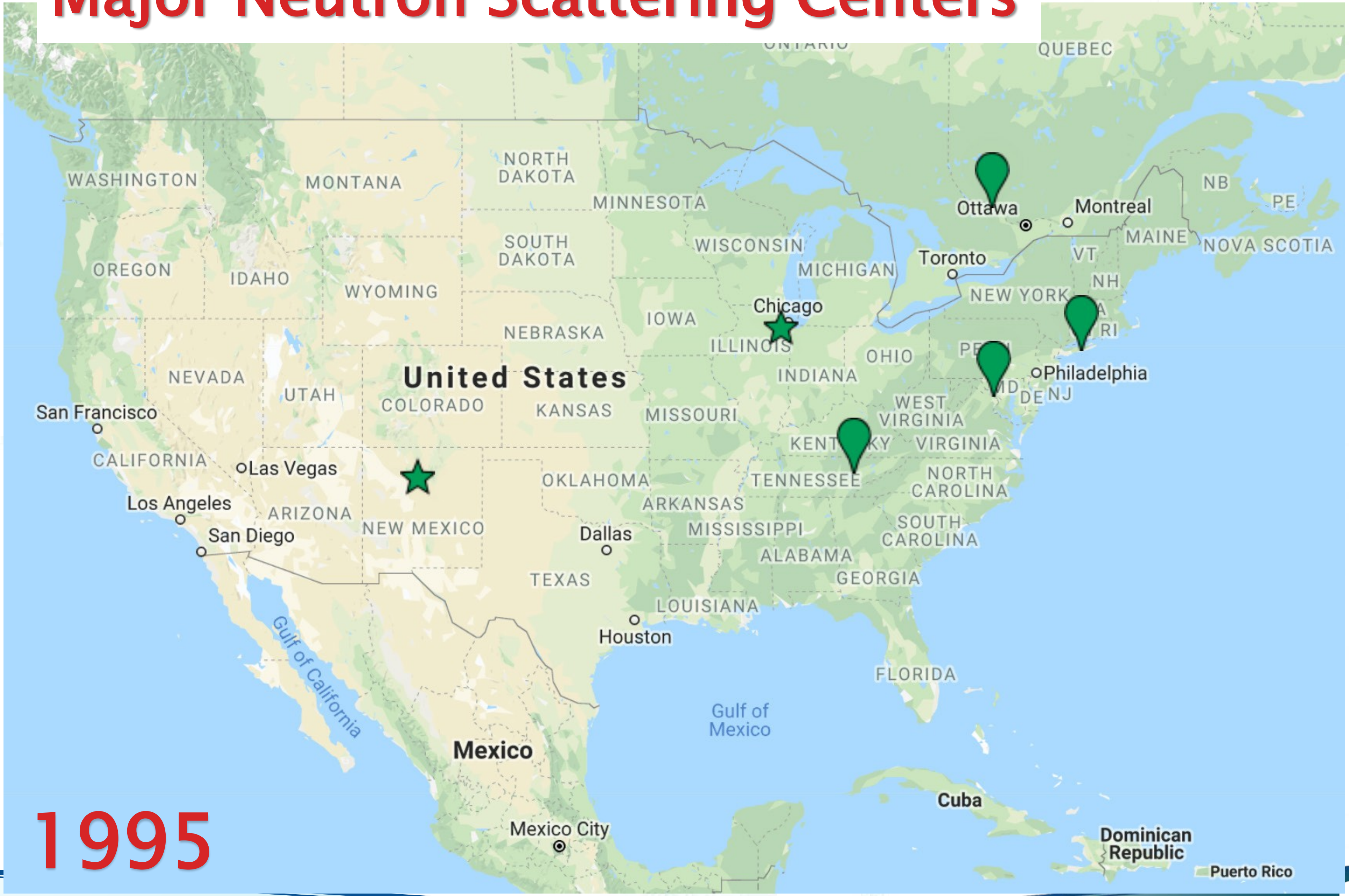


The Future of Neutron Science at the NCNR

Tom Newton, Danyal Turkoglu

TRTR Annual Meeting
September 22-26, 2019
Idaho Falls, Idaho

Major Neutron Scattering Centers



Major Neutron Scattering Centers

2019

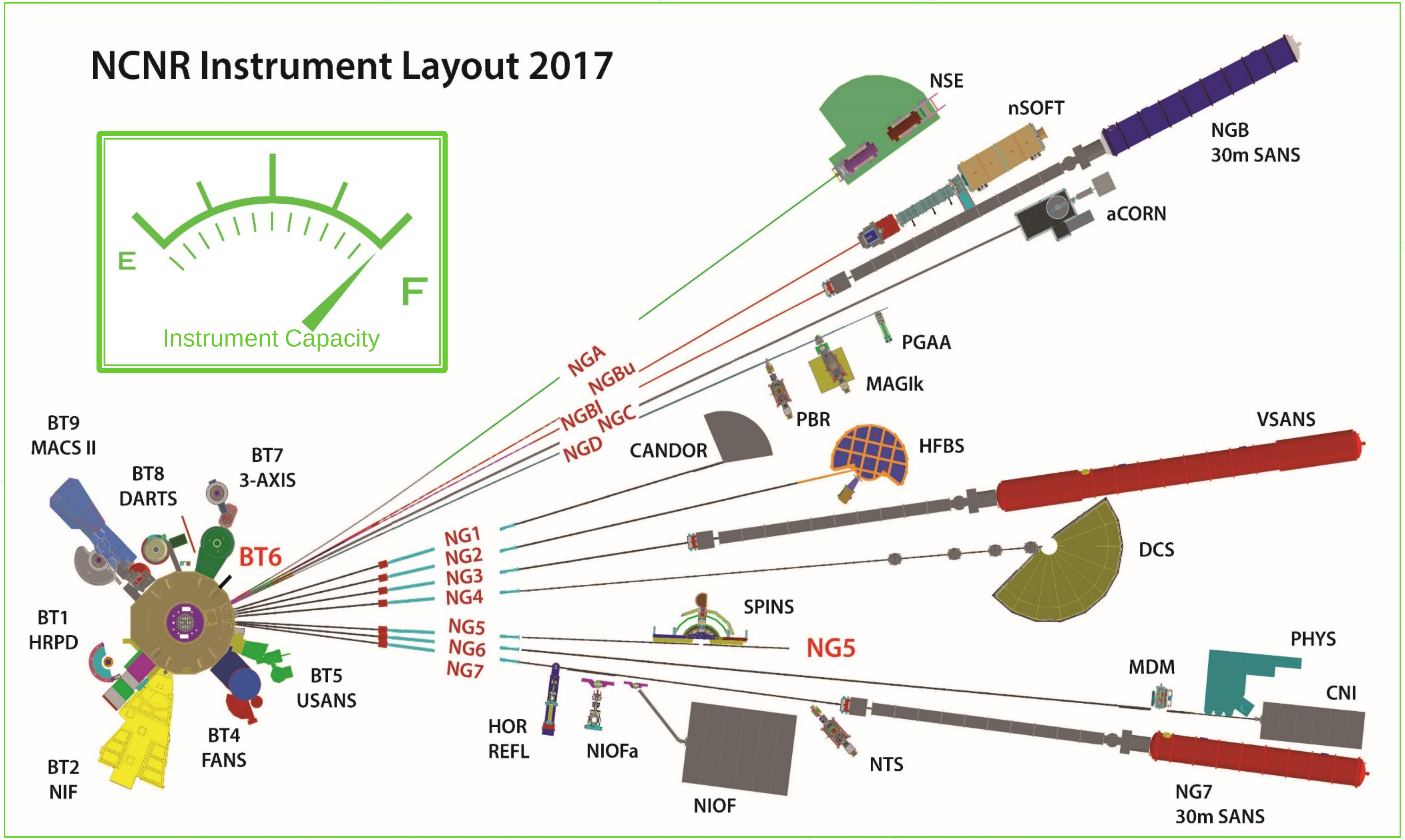
U.S. Neutron Scattering

- Three major facilities at two laboratories (number of instruments):
  - NIST: NCNR (29)
  - ORNL: HFIR (14)
    SNS (20)
  - All are oversubscribed
- CNBC (NRU reactor) closed in 2018, leaving US labs only major neutron scattering centers in western hemisphere
- North America has 1/3 capacity of Europe
  - Addition of ESS, a 5 MW long-pulse spallation neutron source, will put N.A. further behind

NCNR

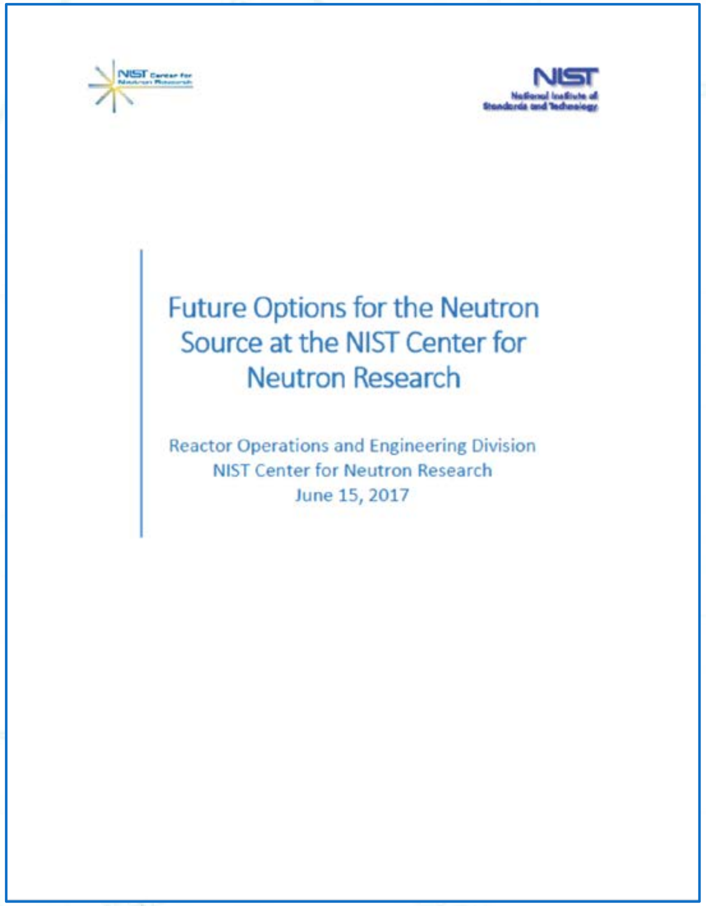
HFIR

SNS


NCNR Instrument Layout 2017
E
F
Instrument Capacity
NSE
nSOFT
NGB
30m SANS
aCORN
PGAA
MAGIk
PBR
HFBS
VSANS
CANDOR
NGA
NGBu
NGBI
NGC
NGD
BT9
MACS II
BT8
DARTS
BT7
3-AXIS
BT6
NG1
NG2
NG3
NG4
NG5
NG6
NG7
DCS
SPINS
NG5
PHYS
MDM
CNI
BT1
HRPD
BT5
USANS
BT4
FANS
BT2
NIF
HOR
REFL
NIOFa
NIOF
NTS
NG7
30m SANS

2017 study: An Exploration of Future Options for the NCNR Neutron Source

1. Maintain NBSR in current configuration.
2. Major upgrade to the NBSR to enhance flux.
   - In conjunction with conversion to LEU fuel
3. Replace the NBSR with a new reactor.
   - Two large cold sources feeding two guide halls


NIST Center for Neutron Research

NIST National Institute of Standards and Technology

Future Options for the Neutron Source at the NIST Center for Neutron Research

Reactor Operations and Engineering Division
NIST Center for Neutron Research
June 15, 2017

Options for neutron source at NCNR

Maintain NBSR	Refurbish NBSR	Replacement Reactor

Maintain NBSR

Pros

- Cheapest
- Status quo until major problem

Cons

- Riskiest in terms of long-term availability and reliability
- No room for new scientific instruments

Refurbish NBSR

Pros

- Eliminates potential existential issues (thermal shield and vessel)
- Allows some optimization

Cons

- Long downtime (> 2 y)
- No continuity
- Expensive (> $0.5B)
- Risky unknowns

Replacement Reactor

Pros

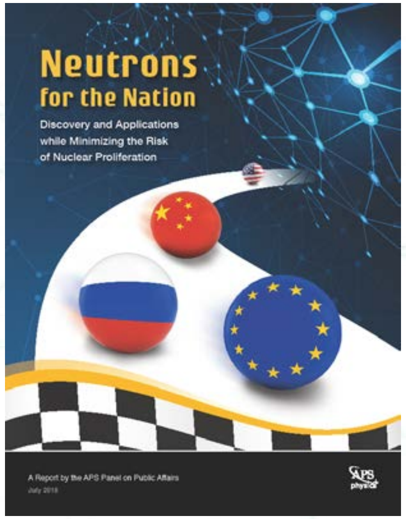
- Establishes NCNR as a world leader in neutron science capability
- Allows continuity of science via NCNR

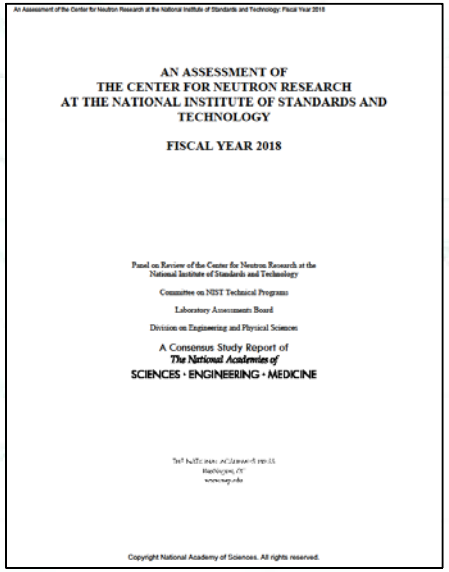
Cons

- Most expensive (≈$1.0B)
- Long lead time (8-15 y)

Future neutron source

- APS recommendation (2018):
  - "The United States should initiate an effort to competitively design and build a new generation of LEU-fueled high-performance research reactors …"
- NAS recommendation (2018):
  - "The reactor is 50 years old. Loss of this facility would have a strongly negative impact on neutron science within the United States and the scientific disciplines that NCNR serves."
  - "The NCNR should commission a detailed assessment of the current facility and begin the conceptual design of a new reactor."




An Assessment of the Center for Neutron Research at the National Institute of Standards and Technology: Fiscal Year 2018

AN ASSESSMENT OF
THE CENTER FOR NEUTRON RESEARCH
AT THE NATIONAL INSTITUTE OF STANDARDS AND TECHNOLOGY

FISCAL YEAR 2018

Panel on Review of the Center for Neutron Research at the National Institute of Standards and Technology

Committee on NIST Technical Programs

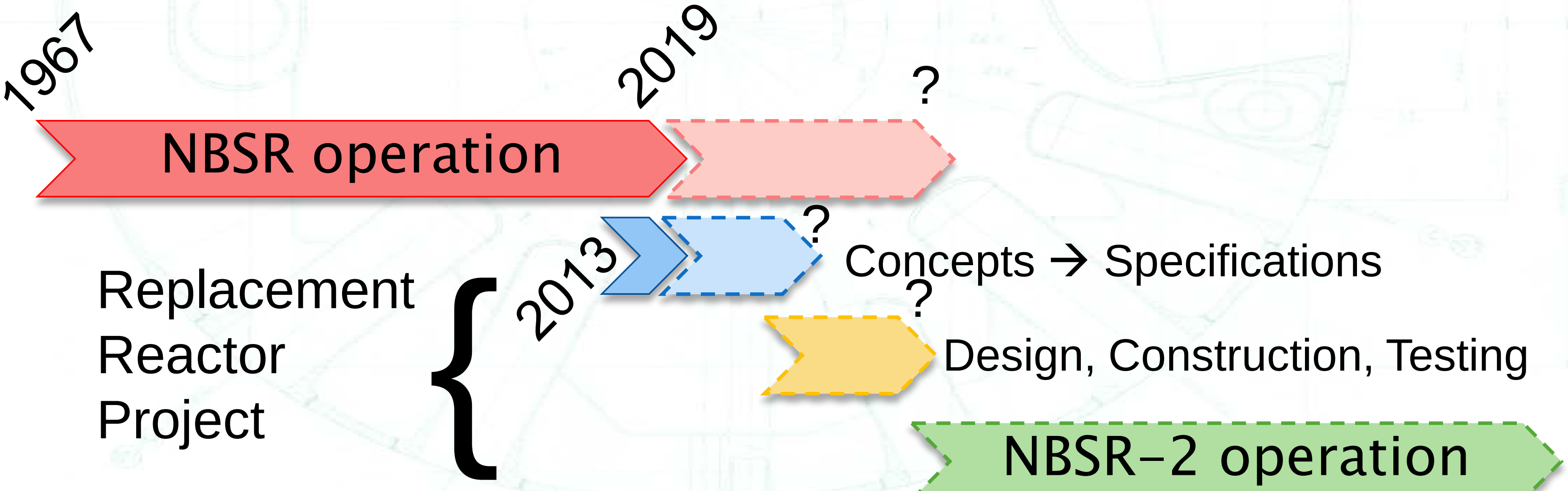
Laboratory Assessments Board

Division on Engineering and Physical Sciences

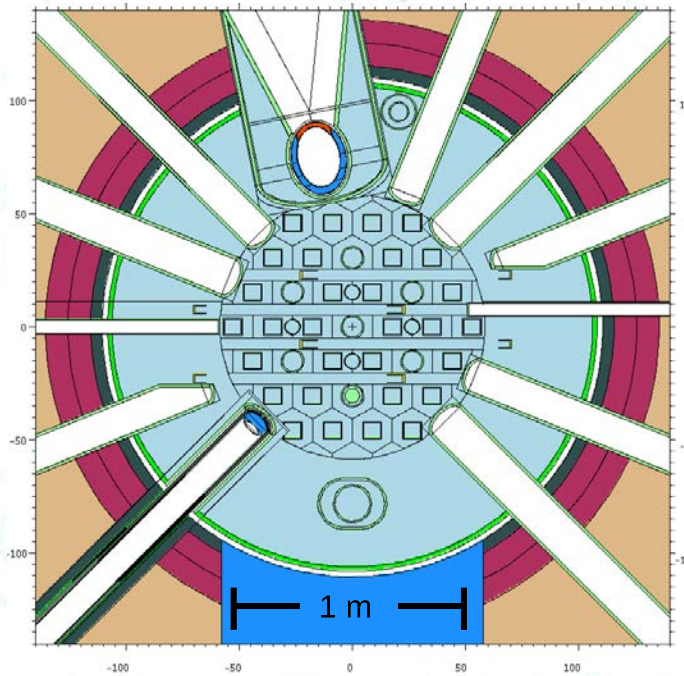
A Consensus Study Report of
The National Academies of
SCIENCES · ENGINEERING · MEDICINE

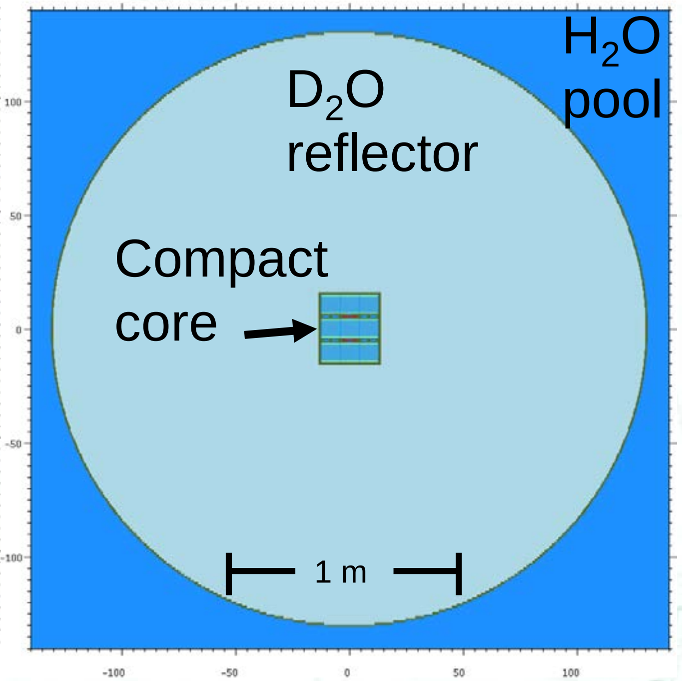
Pathway to a new source

- First began looking into a replacement reactor in 2013
- Several concepts have been investigated in an effort to optimize a reactor design for cold neutron science
- A succession plan that minimizes time between operation of NBSR and the replacement reactor is ideal



Current reactor concept





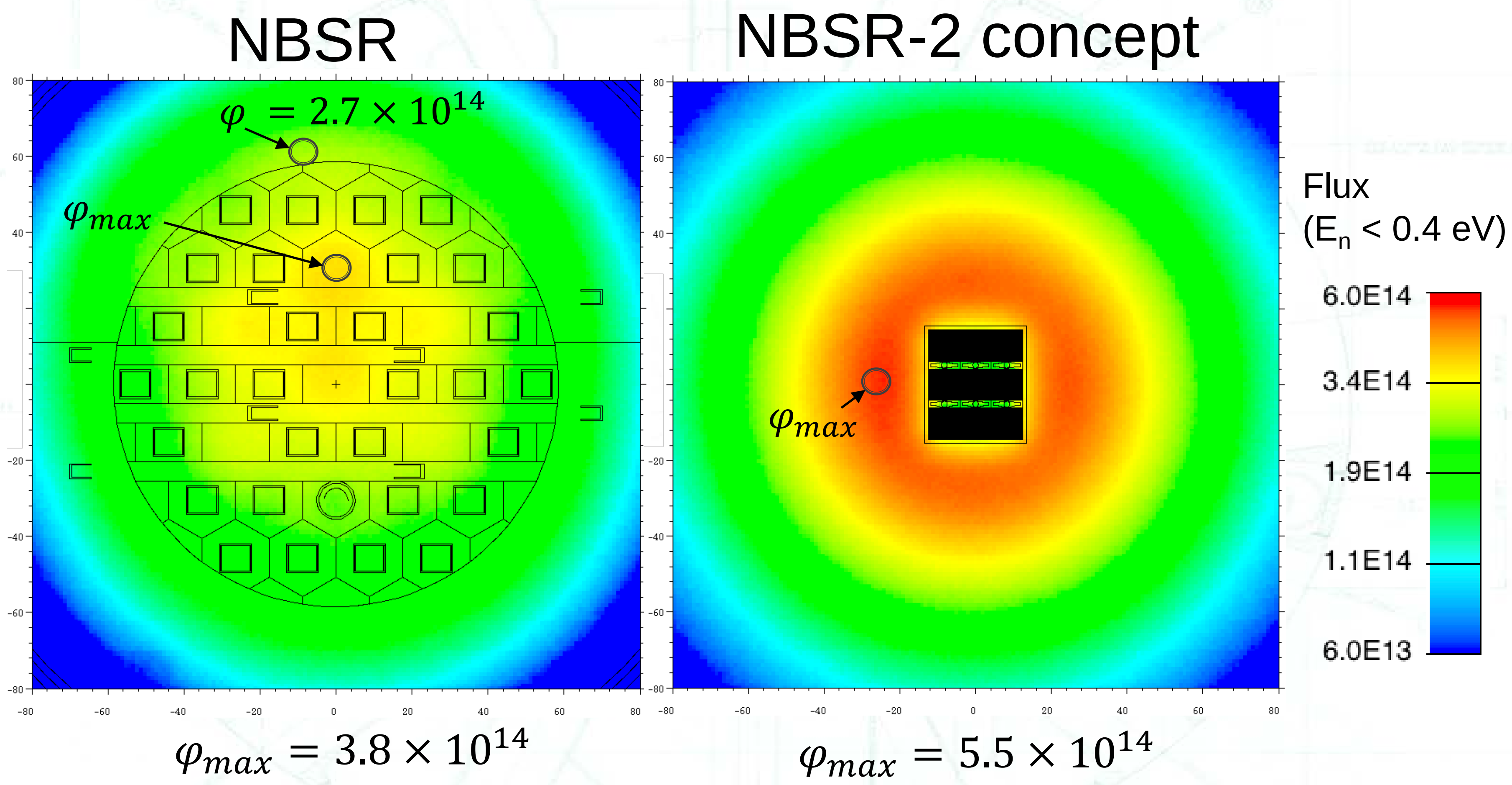
<u>NBSR</u>

20 MW
D_2O coolant
Closed vessel
HEU (U_3O_8/Al) fuel
30 fuel assemblies
38.5 d cycle

<u>NBSR-2 concept</u>

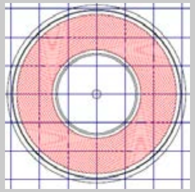
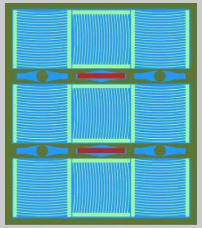
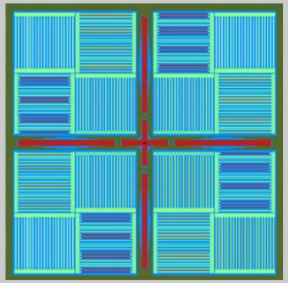
20 MW
H_2O coolant
Open pool
LEU (U10Mo) fuel
9 fuel assemblies
50 d cycle

Unperturbed neutron flux



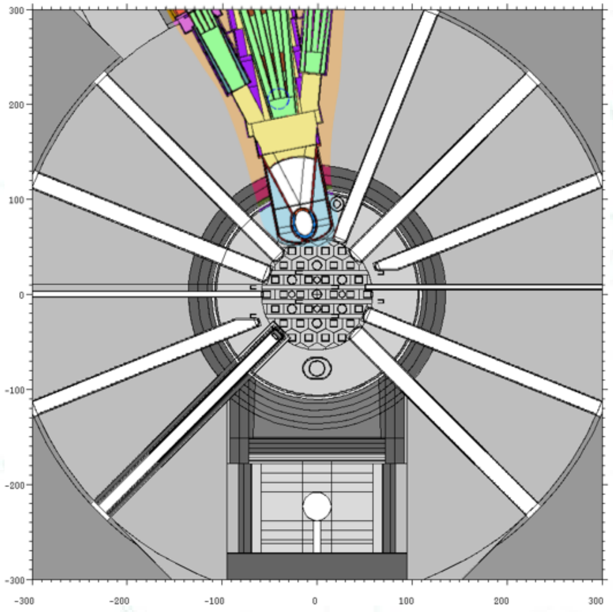
- A factor of >2 gain in flux at cold source locations

20 MW reactor comparison

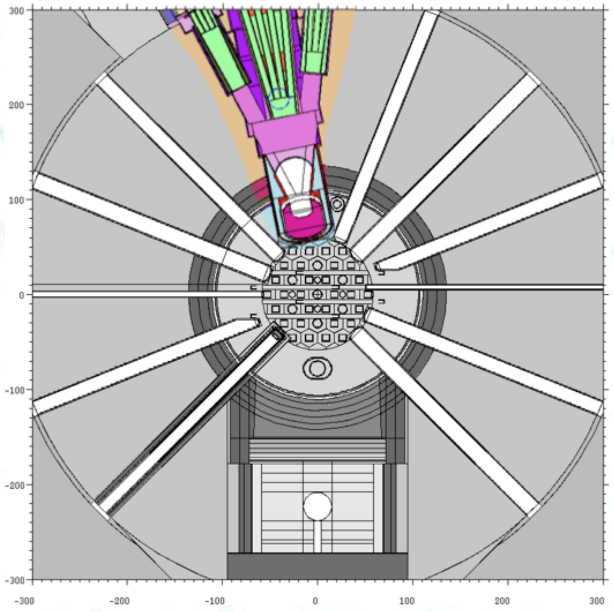
Reactor:	FRM-II	NBSR-2 concept	OPAL
Cross-sectional plan view of reactor core			
Fuel	HEU U_3Si_2/Al	LEU U10Mo*	LEU U_3Si_2/Al
First critical	2004	n/a	2007
Volume	28 L	41 L	69 L
Peak thermal neutron flux in reflector	8×10^{14} $cm^{-2}s^{-1}$	5.5×10^{14} $cm^{-2}s^{-1}$	4×10^{14} $cm^{-2}s^{-1}$

*Fuel is not yet qualified for use

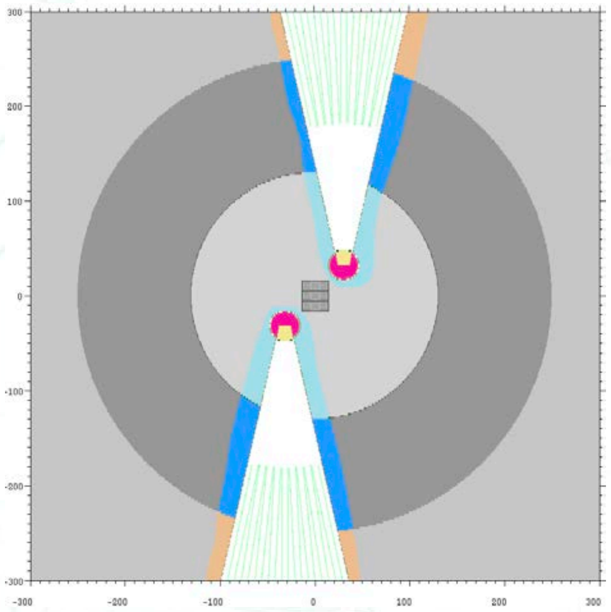
Cold neutron source (CNS) comparison



NBSR with LH_2 source



NBSR with LD_2 source (installation in 2023)

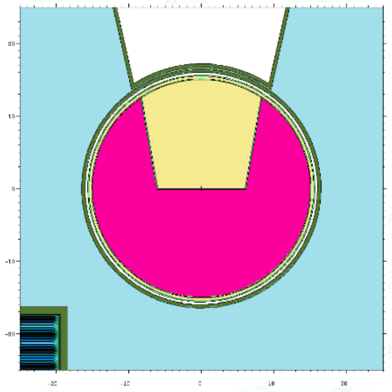


Concept reactor with two vertical LD_2 sources

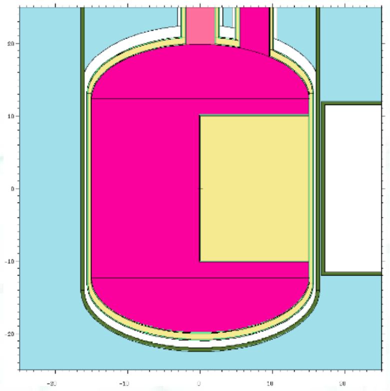
- MCNP model of NBSR is well benchmarked for CNS performance
- Primary design objective is to demonstrate substantially higher cold neutron beam intensities over NBSR capability

Vertical cold neutron sources

- Decouples guides and CNS
- 20 L volume
- Heat loads: <4 kW
  - Allows for thermosiphon natural circulation
- Not yet optimized

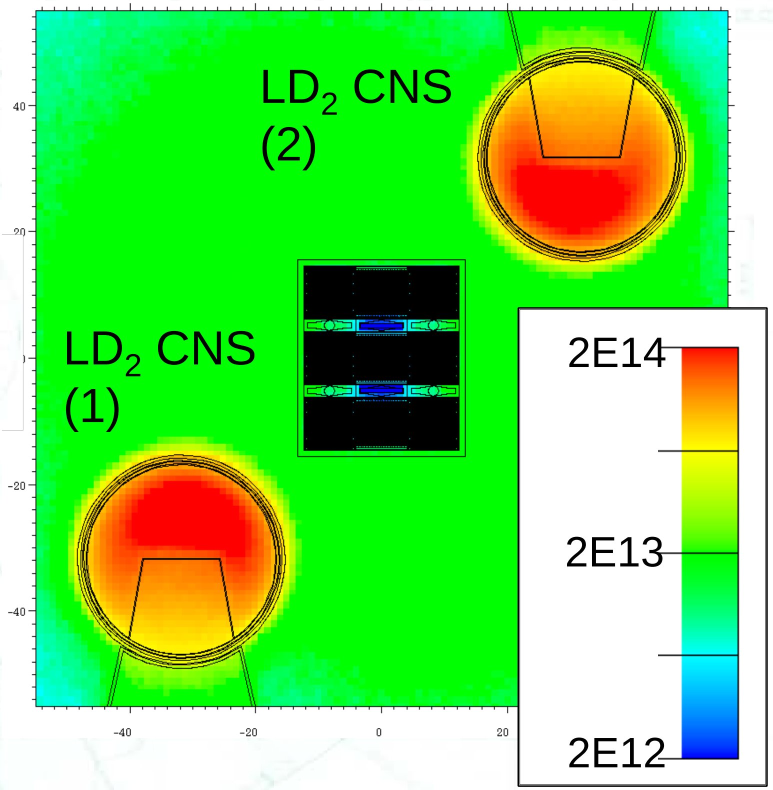


Plan view (*xy*)



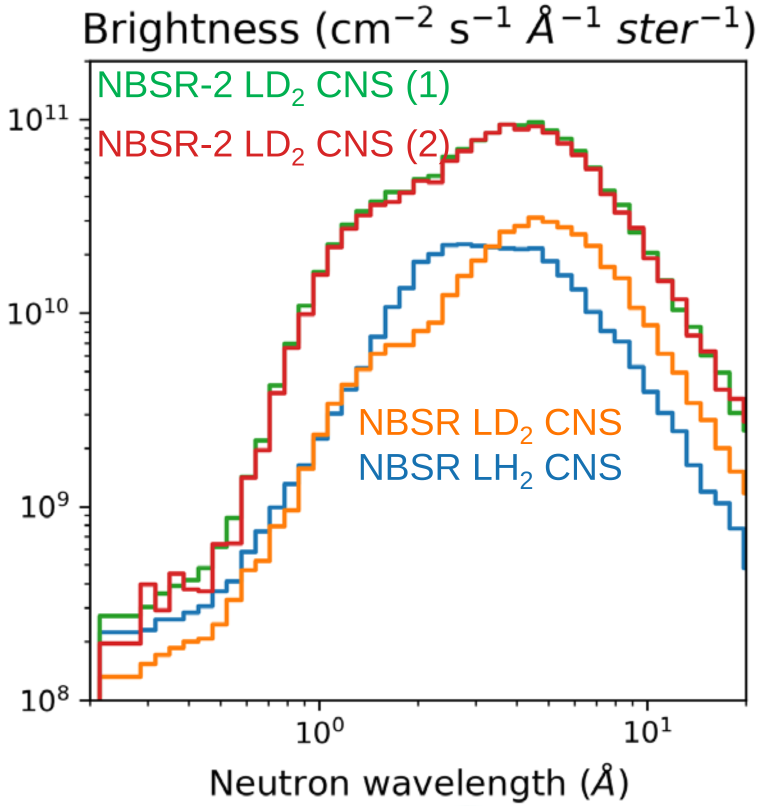
Elevation view (*yz*)

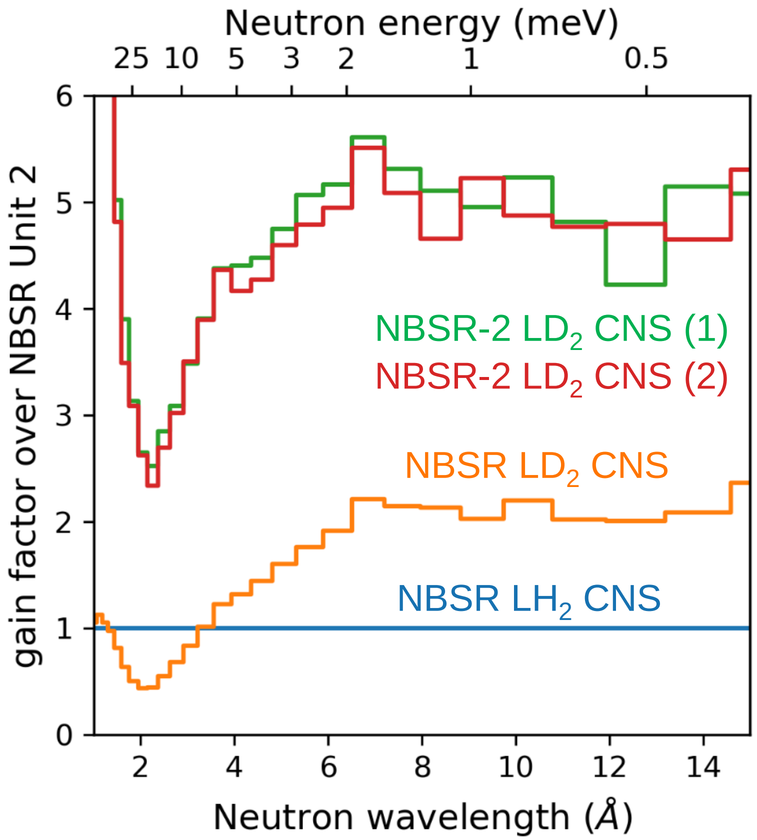
Cold flux (E_n < 10 meV)



Cold neutron source brightness

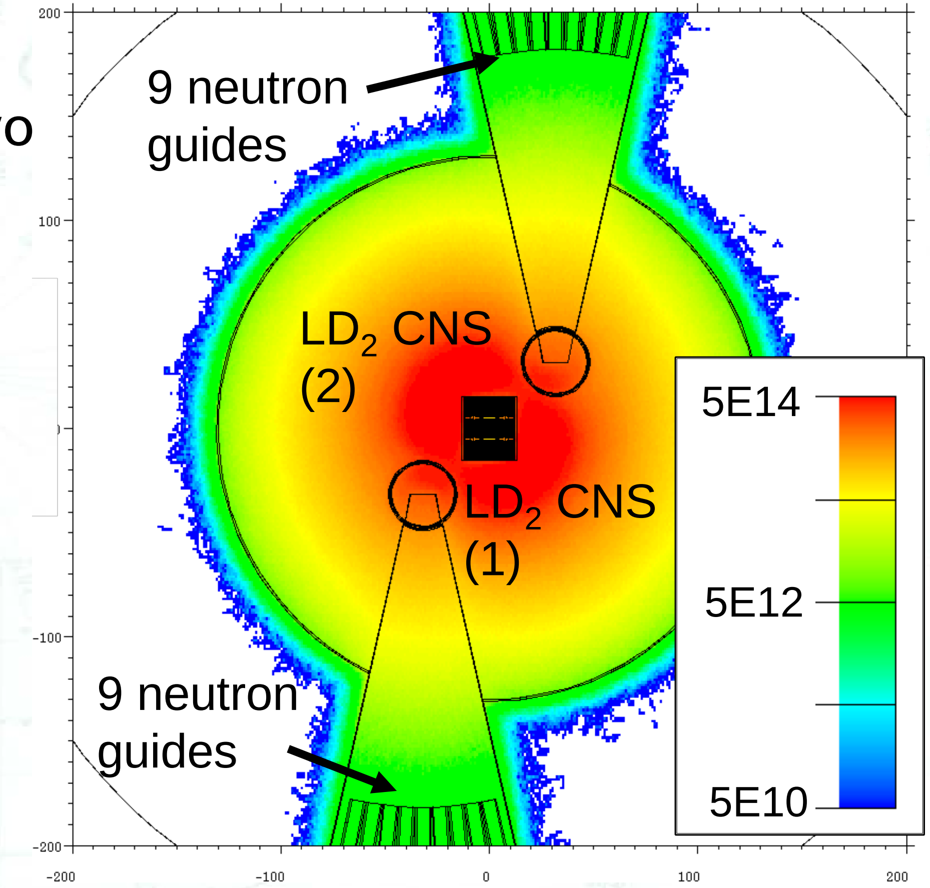
- Calculated with MCNP6 surface (F1) tally
- Represents the neutron intensity within an acceptance angle (2.9°) at neutron guide entrance per unit area (20 cm × 6 cm) per unit wavelength





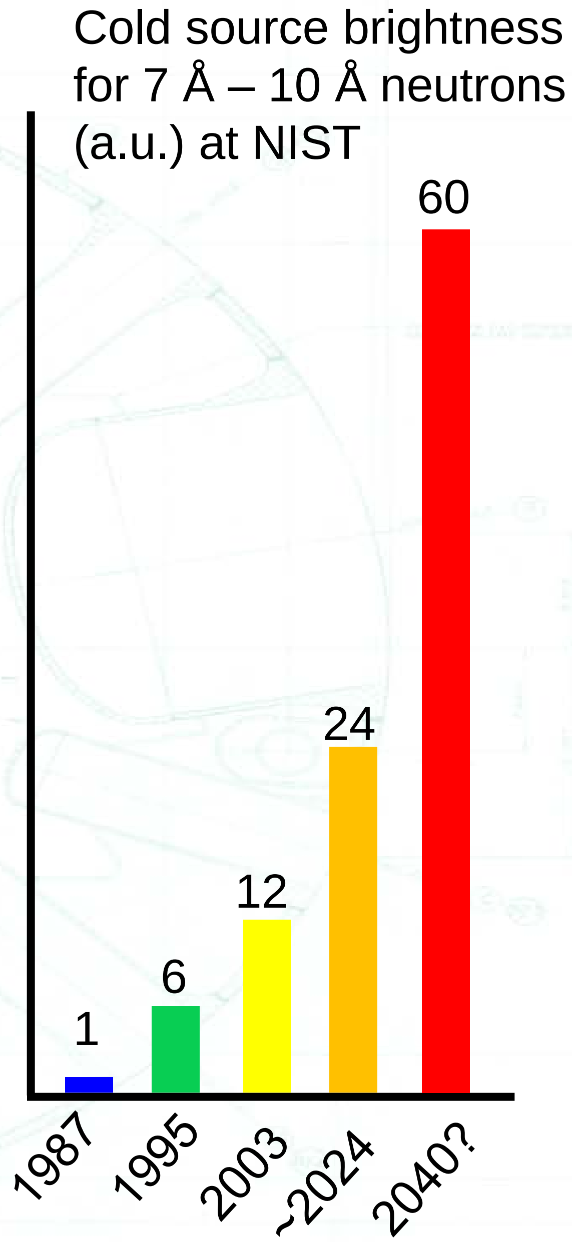
Expansion of neutron science capacity

- Substantial beam intensity increase
  - Reduces measurement times
  - Improves temporal resolution
- 20–30 neutron guides serving two guide halls
  - Reuse existing guide hall after NBSR operation ceases
  - Increases instrument capacity up to 60
- Ample space in reflector tank
  - Additional cold neutron sources?
  - Multiple thermal beam tubes
  - Multiple rabbit tubes
- What new capabilities can be unlocked?
  - Need to work with scientists and instrument designers



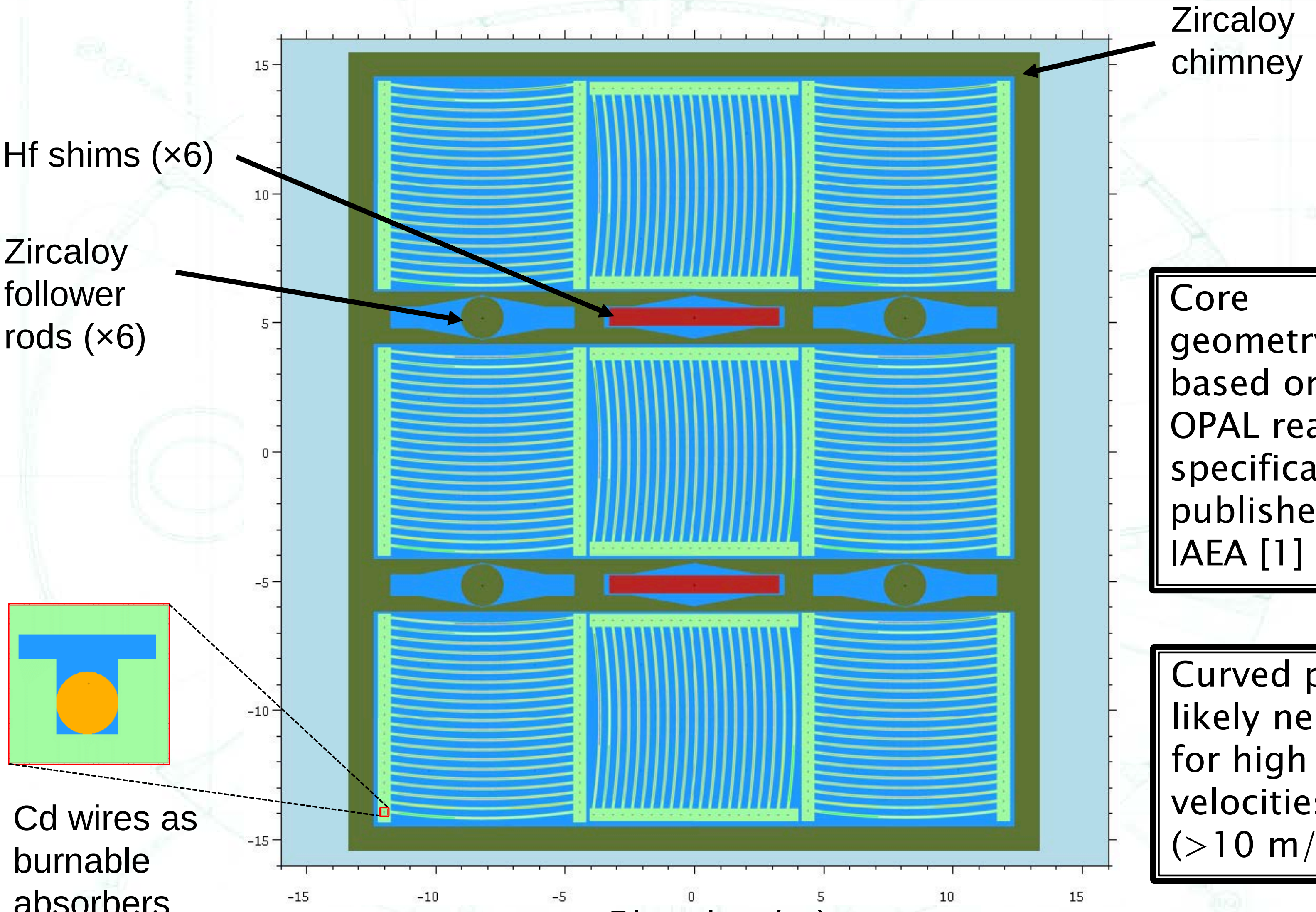
Summary

- NCNR strives to provide a world-class facility for neutron science
- Due to the NBSR age, a succession plan to create a new neutron source is needed to ensure continuity
- A reactor concept has been identified that could substantially expand neutron science capabilities at NIST for the 21st century



Backup slides

Core design



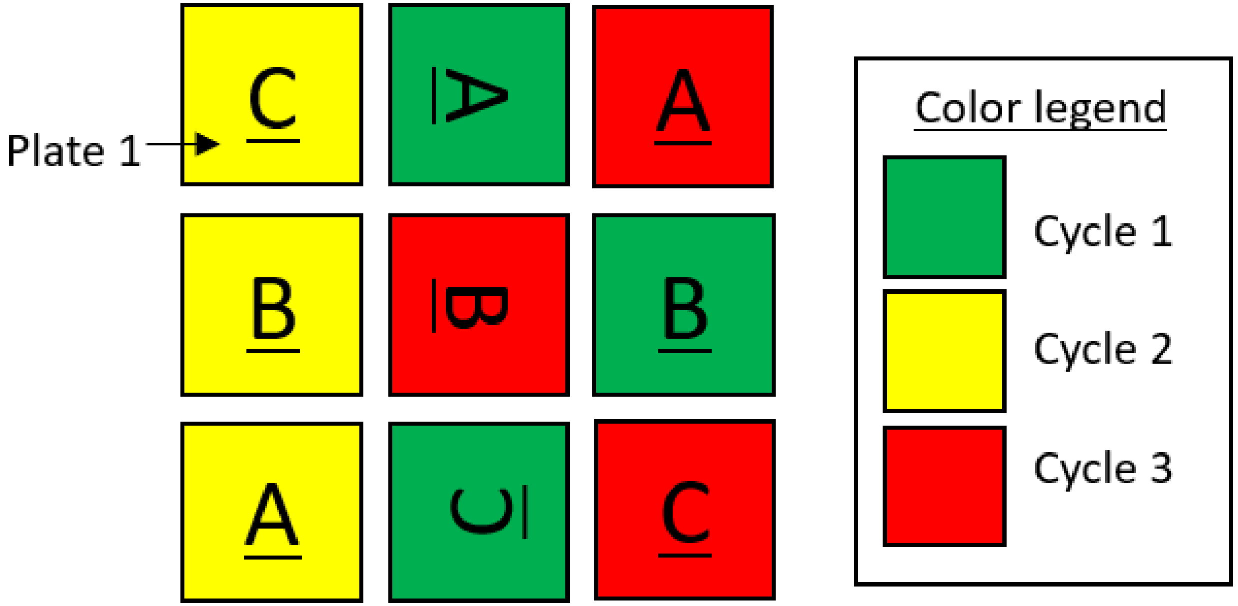
Plan view (*xy*)

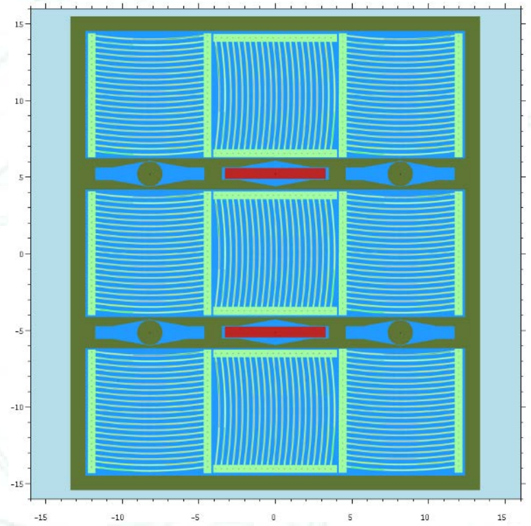
Core geometry based on OPAL reactor specifications published by IAEA [1]

Curved plates likely needed for high flow velocities (>10 m/s)

[1] IAEA, Technical Report Series No. 480 – Research Reactor Benchmarking Database: Facility Specification and Experimental Data, Vienna, (2015).

Fuel management scheme

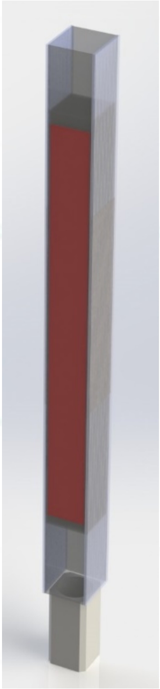


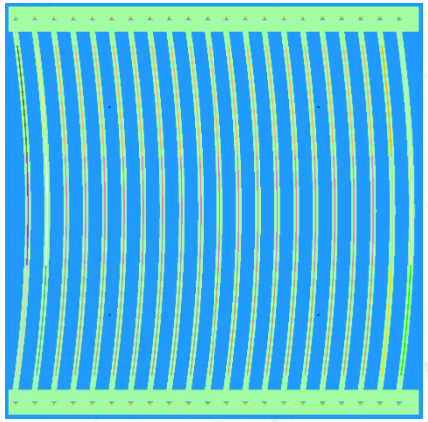


- 3 fresh fuel assemblies for a 50 d cycle
- Rotations during refueling
- Asymmetric power profile

LEU Fuel Assembly Design

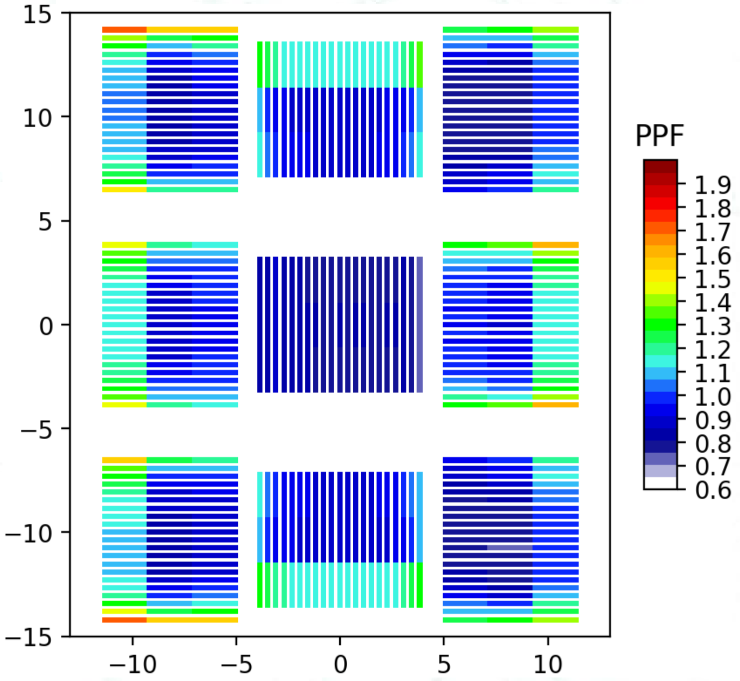
	NBSR	Concept Reactor
Foil thickness	0.0216 cm	0.0250 cm
Foil width	6.134 cm	6.5 cm
Foil height	27.94 cm	70 cm
Foils per FA	34 (17×2)	21
U-235 mass per FA	383 g	726 g
Fresh FAs per cycle	4	3
Cycle length	38.5 d	50 d

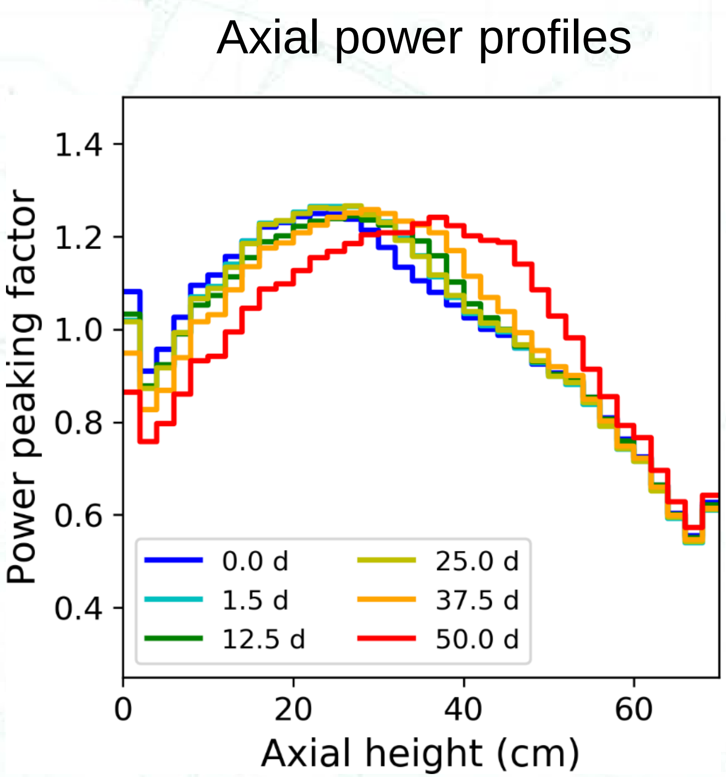




Square profile
(8.05 cm × 8.05 cm)
allows rotations
during refueling

Power distribution

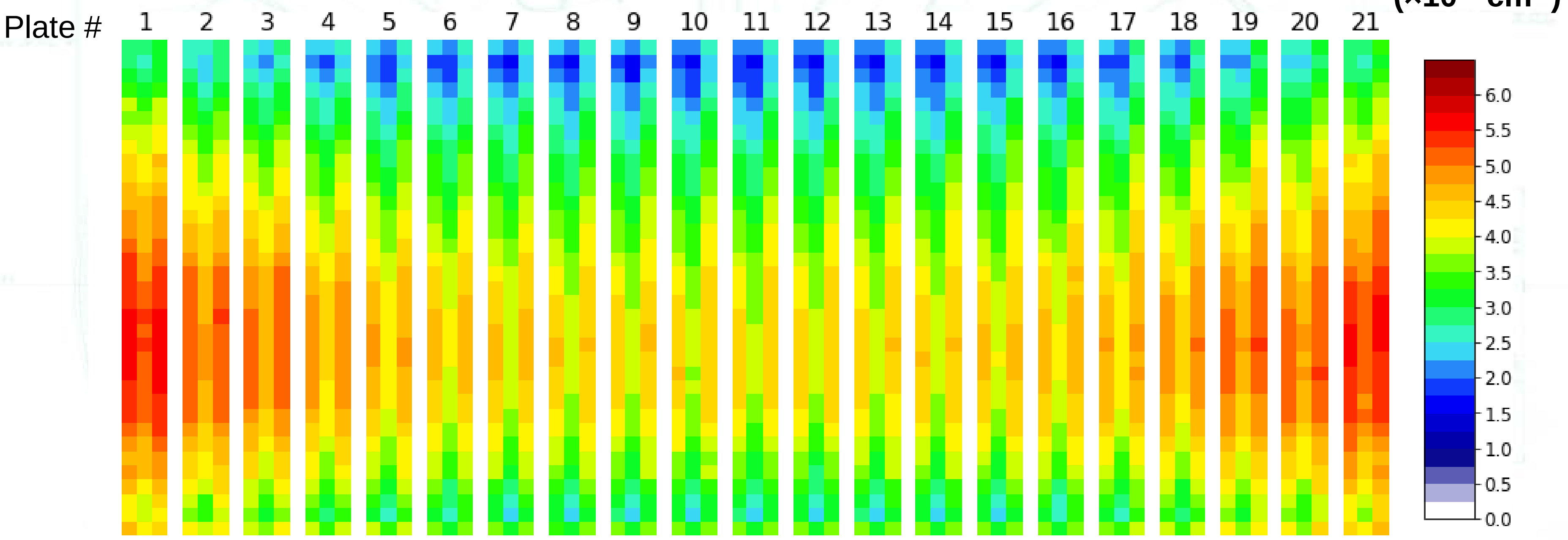




- Hot spot power peaking factor: 2.13
  - → Maximum power density: 9.3 kW/cm^3 × 2.13 = 19.8 kW/cm^3
  - → Maximum heat flux: 116 W/cm^2 × 2.13 = 247 W/cm^2
- Heat flux exceeds NUREG-1313 limit for U_3Si_2 fuel

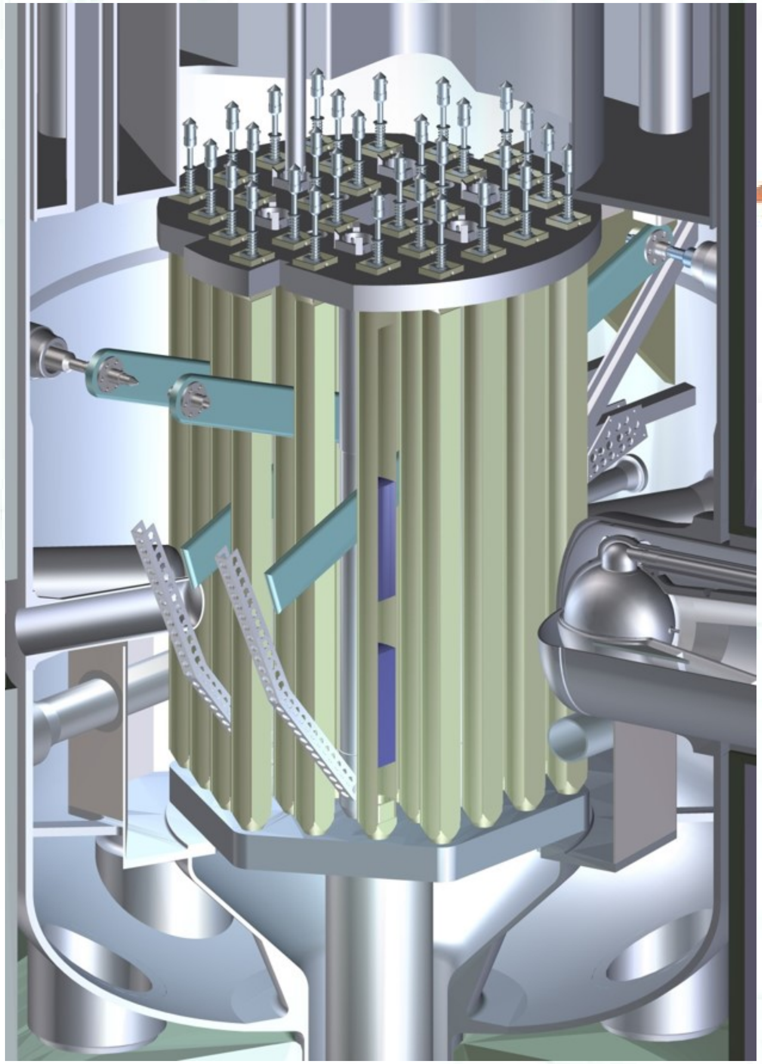
Fission density distribution

Fission density versus plate number for discharged FA with highest burnup



- Potential for high local fission densities: 6×10^{21} cm^{-3}
- Mean fission density of discharged FAs: 4.4×10^{21} cm^{-3}
  - Equivalent to 56% U-235 burnup

Annual Reactor Health Evaluation



Evaluation of 17 major reactor systems, based on 11 health indicators

- All systems are fail-safe, so no safety issues, but possible reliability impacts
- All systems now in good or fair condition
  - Thermal shield and vessel are inaccessible, but no evidence to suggest issues.

2019 Test, Research and Training Reactors Annual Conference

TRTR

MURR

Bringing quality nuclear research, education and service to a global community

INL
Idaho National Laboratory

Sponsorship Moment

Facility Overview

MURR is the highest-powered of the 24 University-operated / NRC-licensed Research Reactors in the U.S.

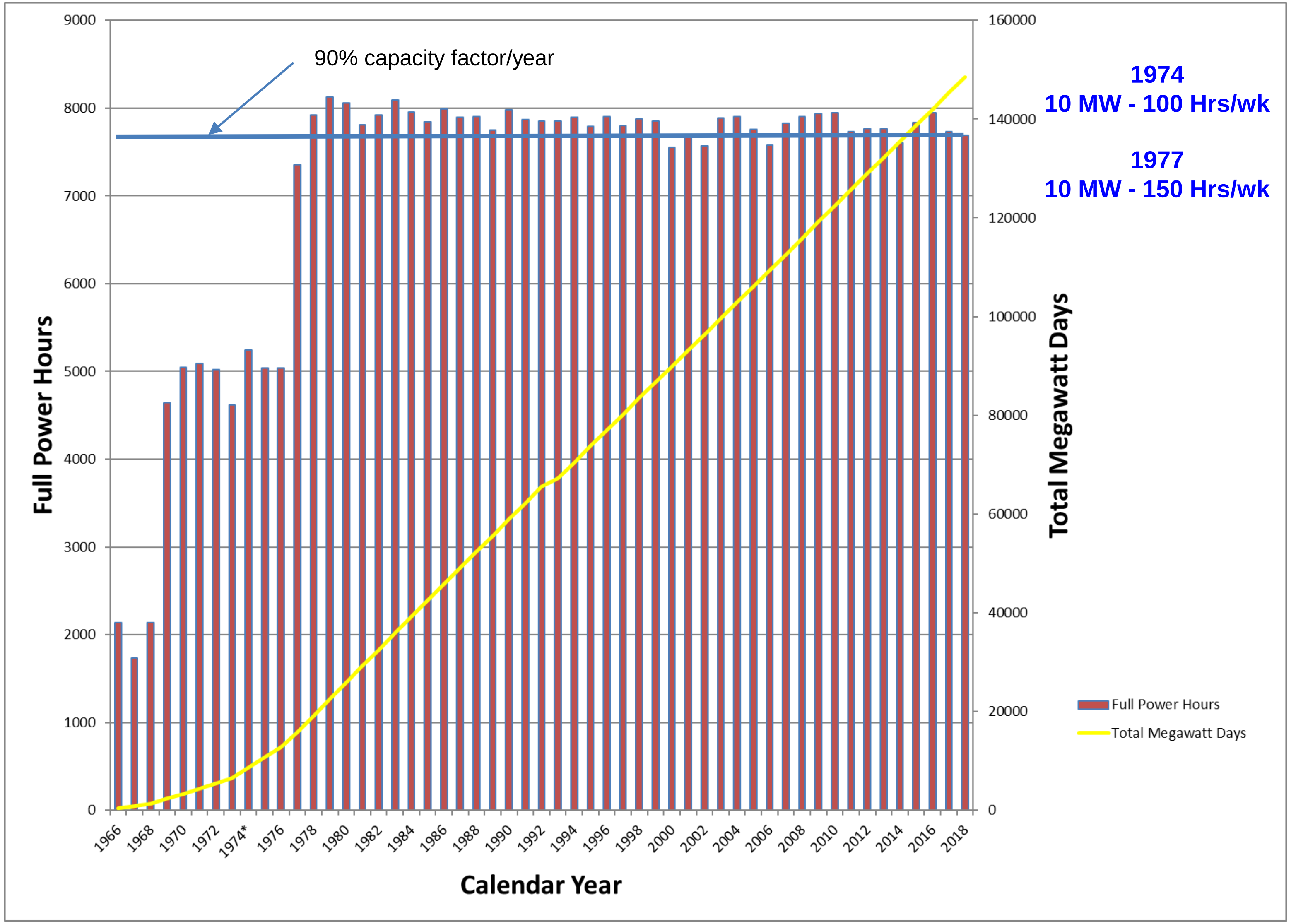
Facility	Power
University of Missouri-Columbia (MURR®)	10 MW
Massachusetts Institute of Technology	6 MW
University of California-Davis	2.3 MW
Oregon State University	1.1 MW
University of Texas, Austin	1.1 MW
Pennsylvania State University	1.1 MW
North Carolina State University	1 MW
Texas A&M University - TRIGA	1 MW
University of Massachusetts-Lowell	1 MW
University of Wisconsin	1 MW
Washington State University	1 MW
Ohio State University	500 kW

Facility	Power
Kansas State University	250 kW
Reed College	250 kW
University of California-Irvine	250 kW
University of Maryland	250 kW
Missouri University of Science and Technology (Rolla, MO)	200 kW
University of Florida	100 kW
University of Utah	100 kW
Purdue University	1 kW
Rensselaer Polytechnic Institute	100 W
Idaho State University	5 W
University Of New Mexico	5 W
Texas A&M University - AGN	5 W

Facility Overview

- Operate 24 hours a day, seven days a week, 52 weeks a year – ~90% of available time at 10 MW
- ~200 full-time employees
- In 2018, MURR shipped 34 different isotopes to 7 different countries via 688 shipments – Classified as Irradiations
- Also in 2018, MURR shipped 12 different isotopes to 4 different countries via 1,316 shipments – Classified as Products
- Each and every week MURR supplies the active ingredients for <u>three</u> FDA-approved drugs: Quadramet®, TheraSpheres® and Lutathera®
- Sole provider of I-131 and Mo-99 in North America

Facility Overview



1974
10 MW - 100 Hrs/wk

1977
10 MW - 150 Hrs/wk

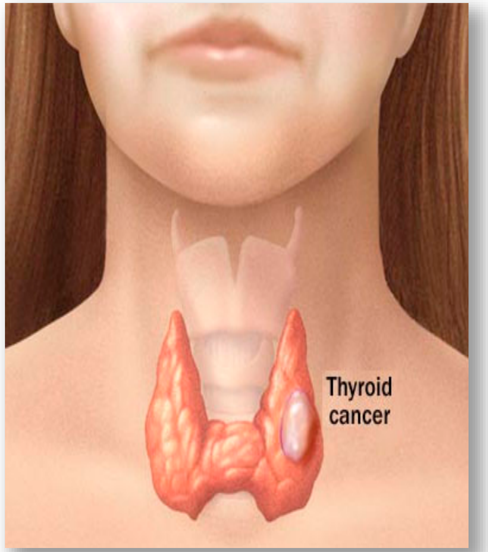
Facility Overview

Distinct Subcultures working together under the same roof
to improve the quality of life

Isotope Supply Activities

36 Different Isotopes Supplied by MURR		
Au-198	Ir-192	Sb-122
Au-199	Kr-79	Sb-124
Ba-131	I-131	Sc-46
Ca-45	Na-24	Se-75
Cd-115	P-32	Sm-153
Ce-141	Mo-99	Sn-117m
Co-60	Pd-109	Sr-89
Cr-51	Po-210	W-181
Cu-64	Rb-86	Y-90
Fe-59	Re-186	Yb-169
Lu-177	Ru-103	Zn-65
Hg-203	S-35	Zr-95

Iodine-131



- Iodine-131 is the 2nd most commonly used radiopharmaceutical and benefits millions of U.S. patients each year.
- Iodine-131 sodium iodide was the FIRST radiopharmaceutical to be FDA-approved (in 1951) and has been a MAINSTAY for thyroid cancer diagnostics & treatment ever since.
- There is NO U.S. supply...UNTIL NOW!

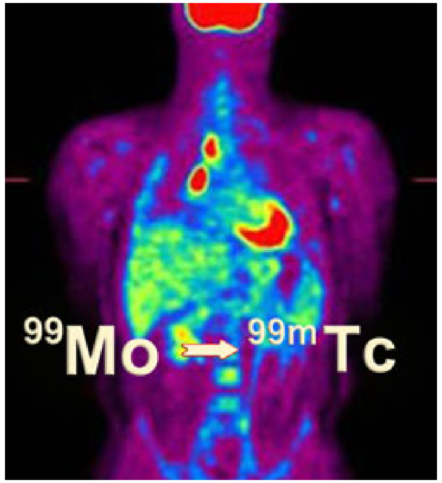
- Radioisotope decay gives Iodine -131 a short product shelf-life…like a melting ice cube.
- I-131 cannot be accumulated.

Molybedum-99

- Molybedum-99 is the number 1 used radiopharmaceutical, there is NO U.S. supply….UNTIL NOW!
  - ✓ Tc-99m is used in more than 30 radiopharmaceuticals ~35,000 times/day in the U.S. to diagnose disease and assess organ structure and function.
- Mo-99 (66 Hr half-life) is the parent for Tc-99m (6 Hr half-life).

  → Short half-life → short product shelf-life

  → Cannot accumulate a supply
- So what is the U.S. Answer?
  - ✓ MURR is actively working with 3 private industry players each with unique technology platforms.
  - ✓ MURR's eventual goal is to supply at least 50% of the U.S. weekly need.



Federal Drug Administration

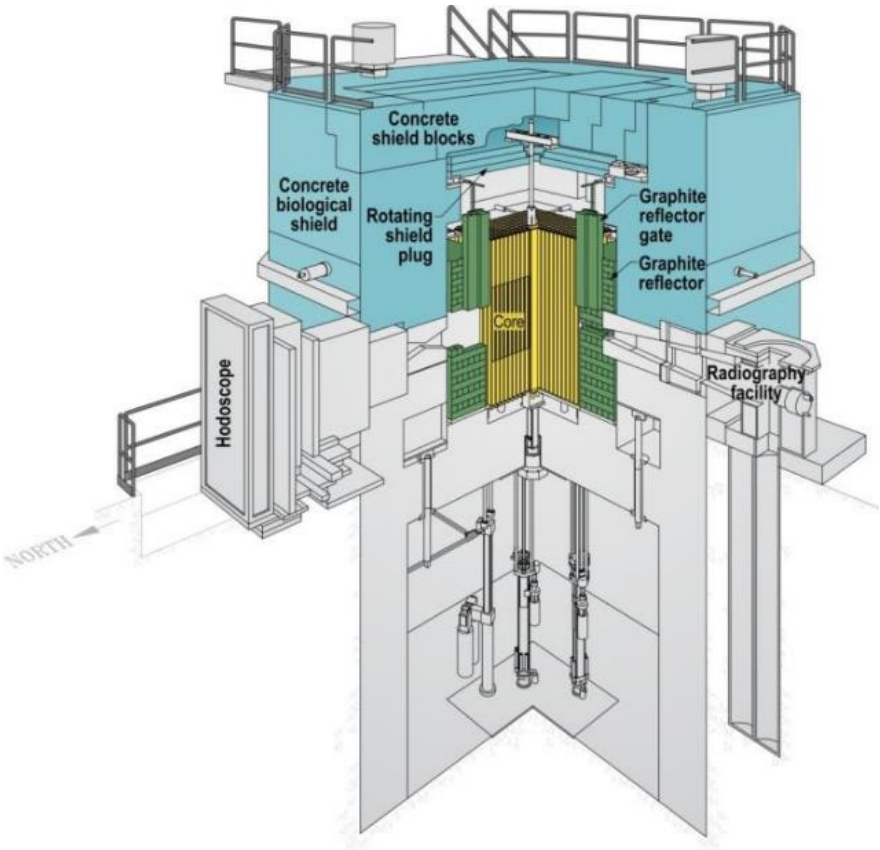
- FDA Quality Systems Compliance, being registered with the FDA as:
  - ✓ *API Manufacturer*
  - ✓ *Analysis Lab*
- Drug Master Files with FDA:
  - ✓ *MURR has filed multiple DMFs*
- Weekly supply of isotopes for:
  - ✓ *Existing treatments*
  - ✓ *New drug clinical trials*
  - ✓ *Global distribution*
- Partnering with Private Industry:
  - ✓ *Confidential R & D Contracts*
  - ✓ *Collaborative Projects*

An Overview of TREAT Operations Since Restart

A. A. Beasley

Introduction

- Transient Reactor Test (TREAT) operations support fuel safety testing
- Graphite-based air-cooled reactor
  - 120 kW steady state, 19 GW peak in pulse mode
  - Operated 1959 – 1994, 2017-present
  - Virtually any power history possible within 2500 MJ max core transient energy
- Experiment design
  - Reactor provides neutrons, experiment vehicle does the rest
  - Safety containment, specimen environment, and support instruments
  - Tests typically displace a few driver fuel assemblies (each 10cm square, 122cm L)
- 4 slots with view of core center, 2 in use
  - Fast neutron hodoscope, neutron radiography

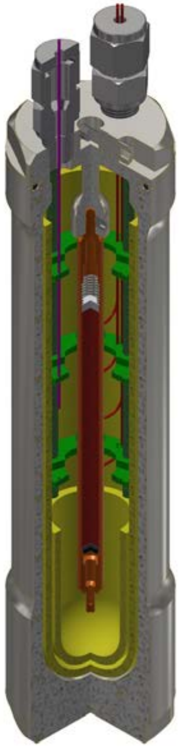


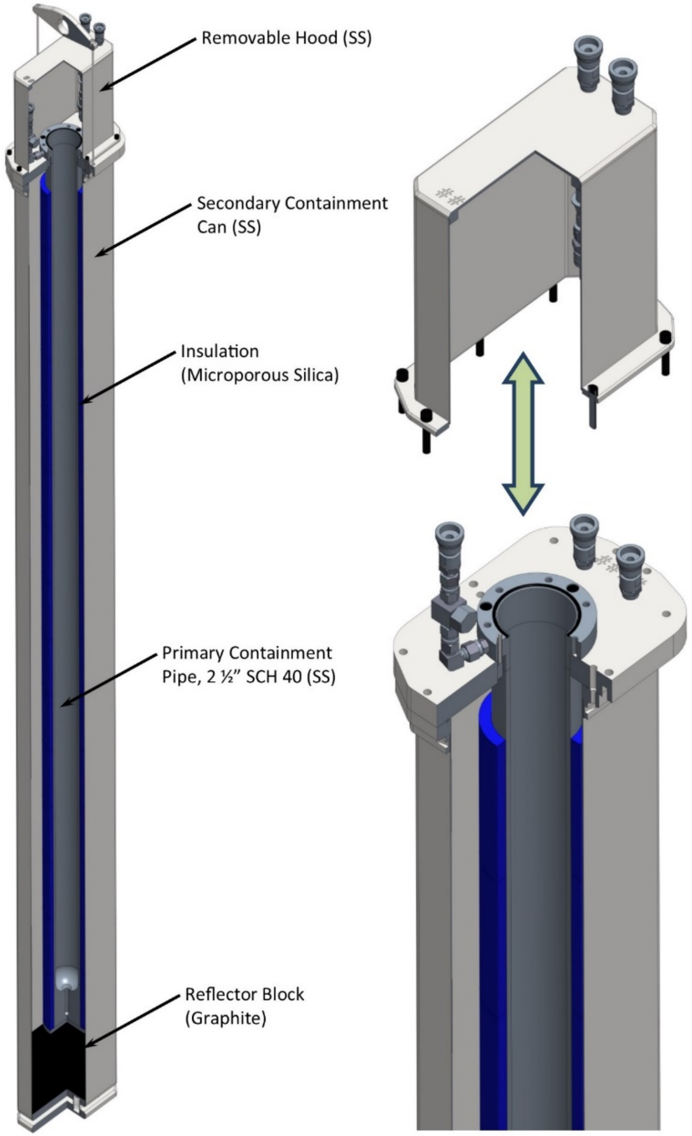
Restart Activities

- Operator qualification
  - Approximately 3 months after restart
- Core characterization
  - Demonstration that core performance was consistent with historical operation
  - Performed concurrently with initial operator qualification
- Transient development
  - Developed and performed transient prescriptions for typical testing operations
- Narrow pulse width
  - Transient development with a focus on minimum pulse width
  - 89 mS FWHM demonstrated
- MIT
  - Sensor test
- LOCA transients

Experiment operation

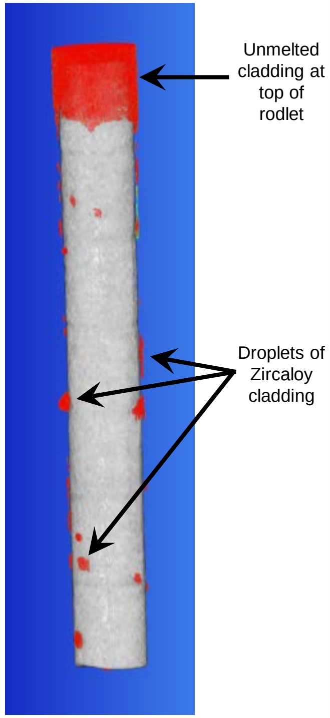
- MARCH
- ATF-SETH (UO_2)
- Aqua-SETH
- RUSL
- Sirius-Cal
- ATF-SETH (U_3Si_2)
- Sirius
- M-SERTTA





ATF-SETH (UO_2)

- UO2 zirconium clad fuel
- Initial qualification of SETH capsule
- Data to support interaction between dry capsules and TREAT reactor
- Verification of experiment to reactor calibration method

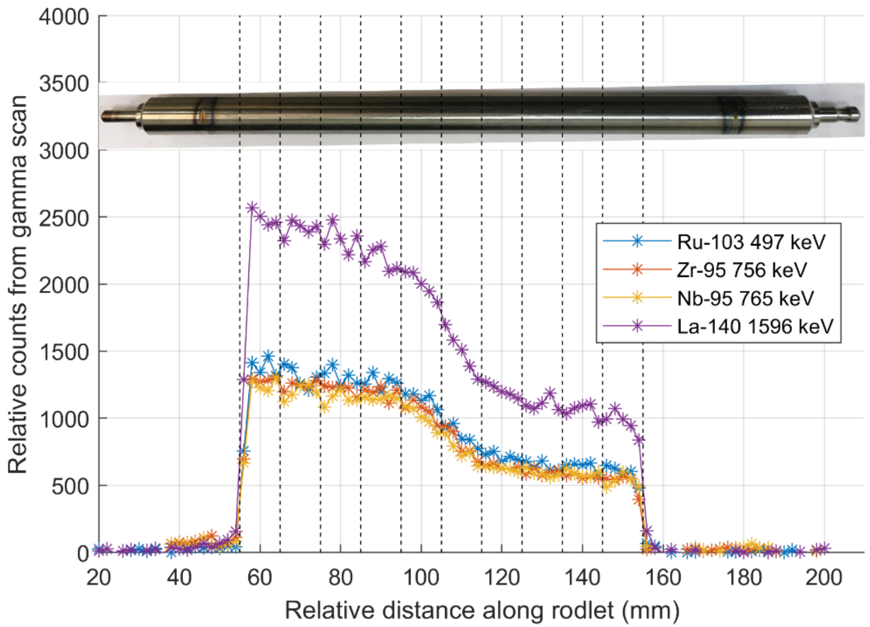


Segmented view showing zircaloy droplets down the sides of the fueled region in red.

Idaho National Laboratory

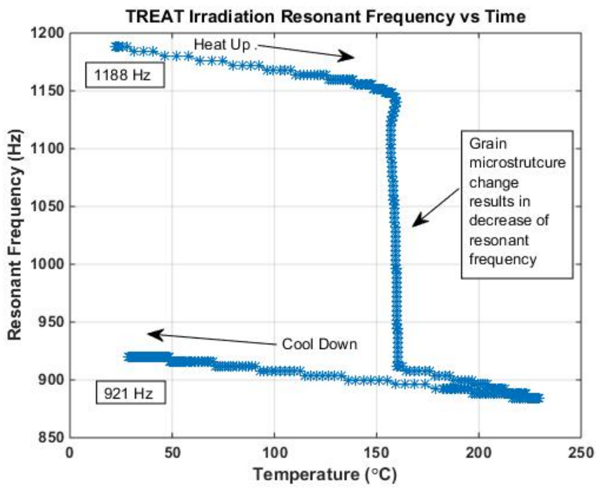
Aqua-SETH

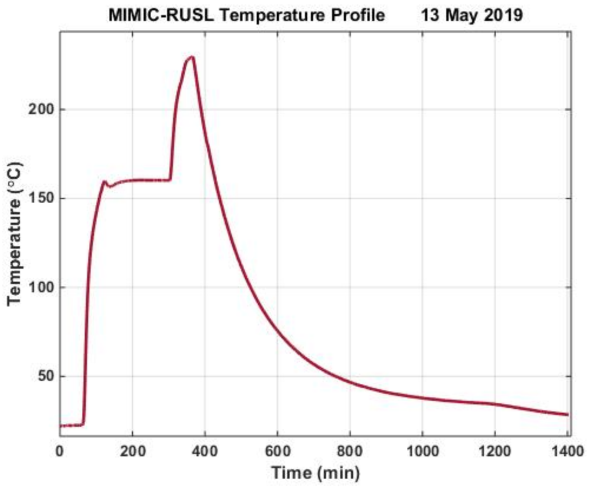
- Determine coupling factor in water filled environment vs. gas filled environment
- Same fuel pin design as SETH A-E with natural uranium pellets
- Water covered on lower 5 pellets helium gas fill for remaining 5 pellets
- Gamma spec to determine burnup
- Analysis of data and post irradiation radiographs demonstrate that actual fill level was higher than planned (6 pellets)
- This pin was reused to perform M-SERTTA-Cal



MIMIC-RUSL

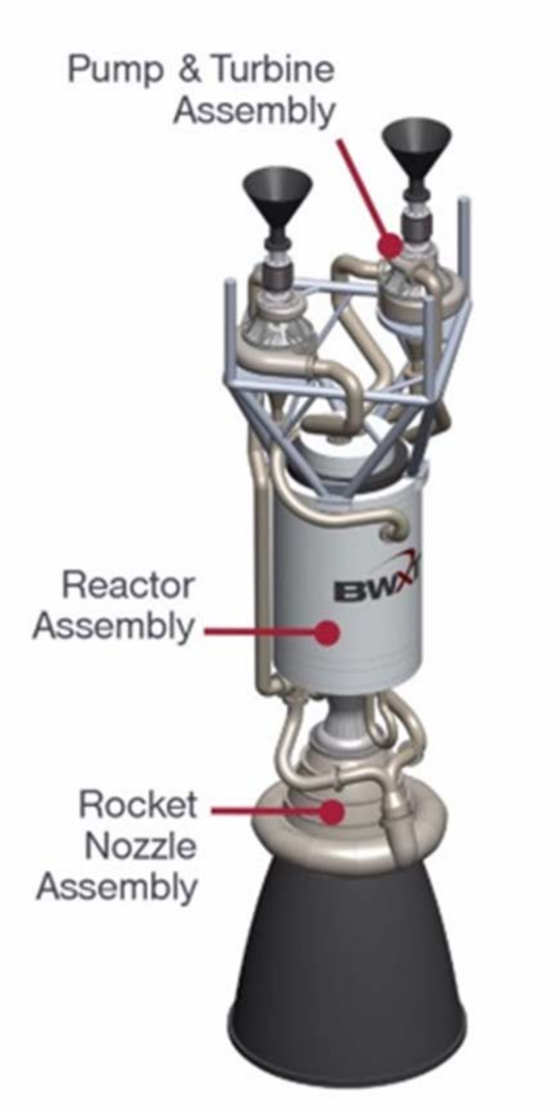
- MIMIC-RUSL (Resonant Ultrasonic Spectroscopy-Laser) INL-developed, advanced laser-based measurement
- Temperature ramping via MARCH's heater module created elastic property changes due to recrystallization in the specimen.
- Data compared to previously-performed out-of-pile tests
- Opens the door to advanced studies of material phase diagrams under irradiation and advanced in-pile measurements





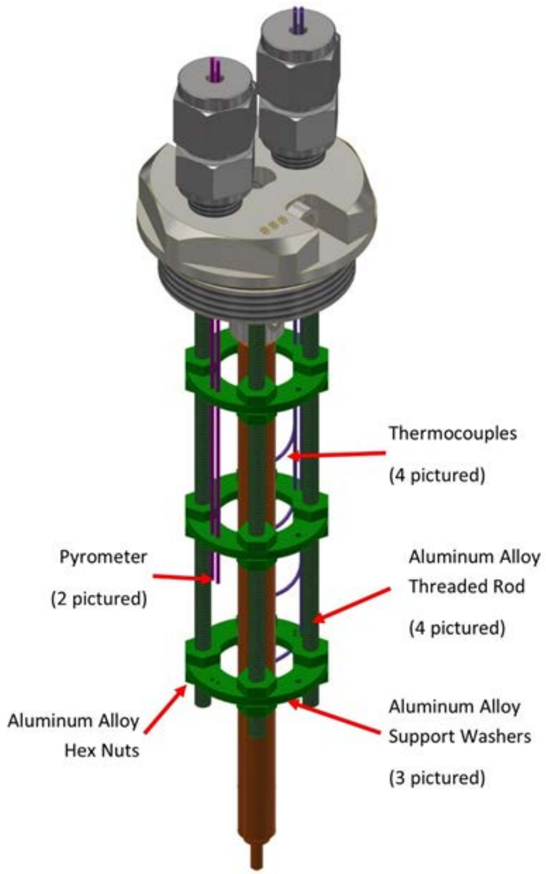
Sirius-Cal

- SETH based experiment to determine coupling factor in NASA CerMet-UN fuel.
- Supports NASA Sirius series of tests on nuclear thermal propulsion fuel



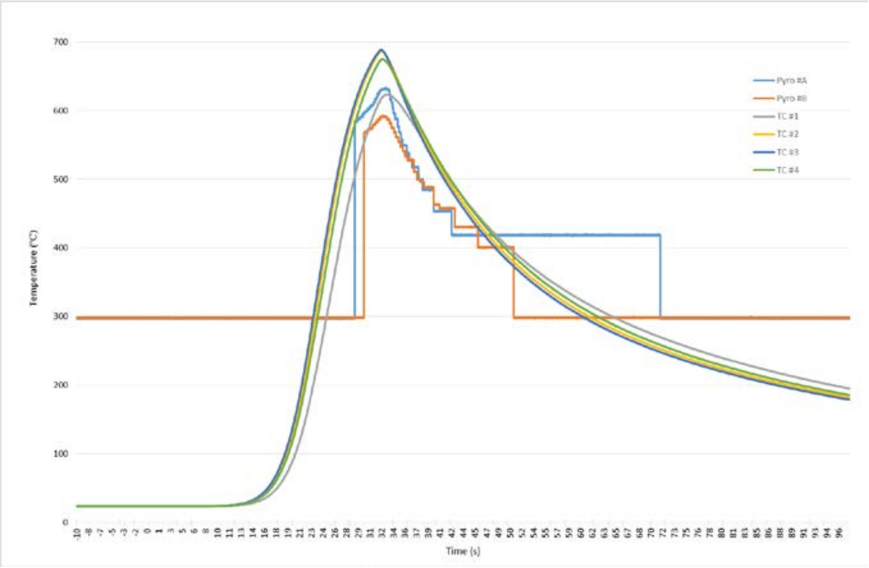
ATF-SETH (U_3Si_2)

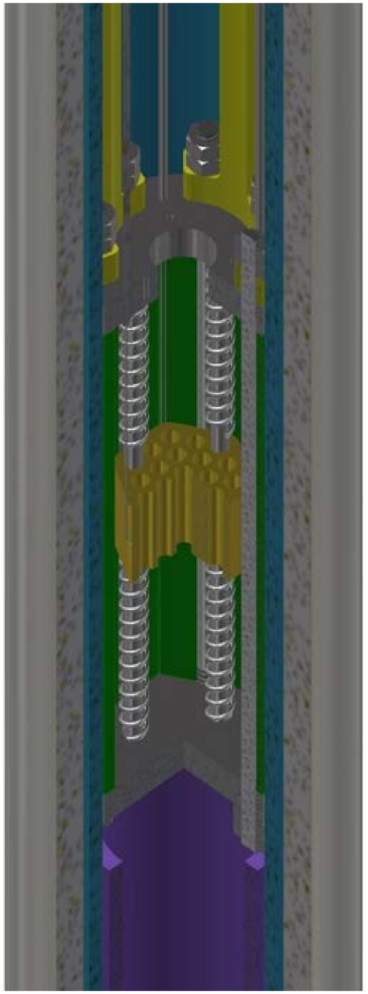
- Separate Effects Testing on Uranium Silicide fuel type
- First two tests completed in September
- U_3Si_2 fuel clad in Zr
- This is the first accident tolerant fuel tested in TREAT
- Test series to continue in 2020



Sirius

- NASA Nuclear Thermal Propulsion Fuel
- Tungsten-Rhenium CerMet with UN fuel
- High Heat-up rate ~95 ^{o}C/S
- High operating temperature 2850 K





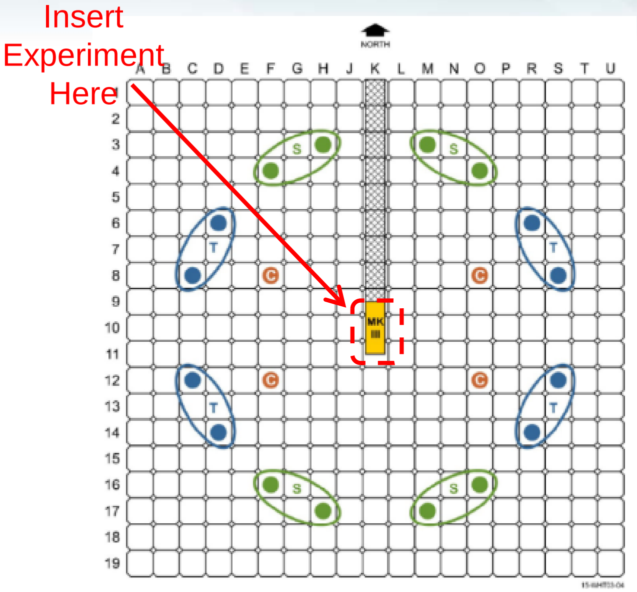
MARCH-SERTTA

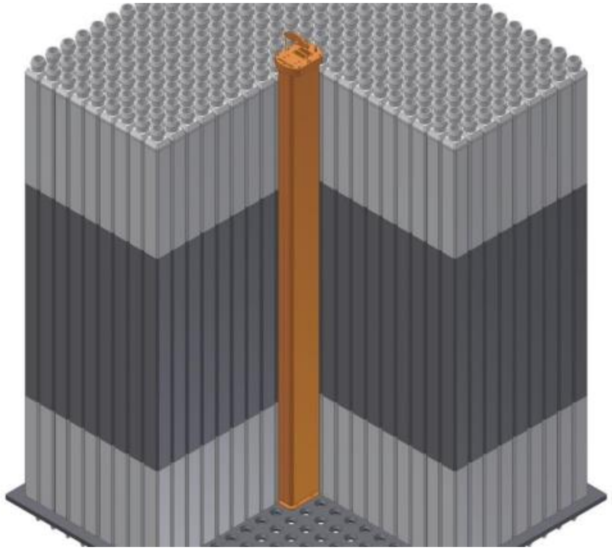
- M-SERTTA-Cal will complete in September (running today)
- Commissioning test for the first water based capsule
- Rodlet from Aqua-SETH used for calibration test
- Low power testing to demonstrate coupling between test and reactor

Future testing

- All 2020 testing will all be performed in MARCH vehicles
- SETH
  - ATF-SETH F and J-U_3Si_2 fuel in SiC-SiC cladding (ATF-SETH cont.)-2 fueled transients
  - Transformational Challenge Reactor (TCR)-investigation of thermo-mechanical limits of advanced manufacturing fuel-2 fueled transients
- M-SERTTA
  - ATF UO_2 with Zr clad-6 fueled transients
  - Critical Heat Flux (CHF)-borated steel provides heat to explore CHF under transient conditions-5 to 10 fueled transients
  - ATF-RIA-test PCMI with pre-hydrided Zr clad fuel-4 fueled transients
- CINDI
  - Steady state irradiations to investigate onset of cracking in U-Zr and Pu-U-Zr fuel-2 irradiations
- SIRIUS
  - Two additional fuel types for NASA NTP will be tested-14 fueled transients
- THOR
  - Sodium based test of advanced fast reactor fuel (U-Zr and Pu-U-Zr)-stretch for operation in 2020-3 transients
- Automatic Reactor Control System (ARCS) replacement-3 month outage

Questions?





SERTTA shown in TREAT core ¾ section view
Secondary containment "can" visible

ATR Path to Operational Excellence

IRRADIATION TESTING FACILITY OF CHOICE

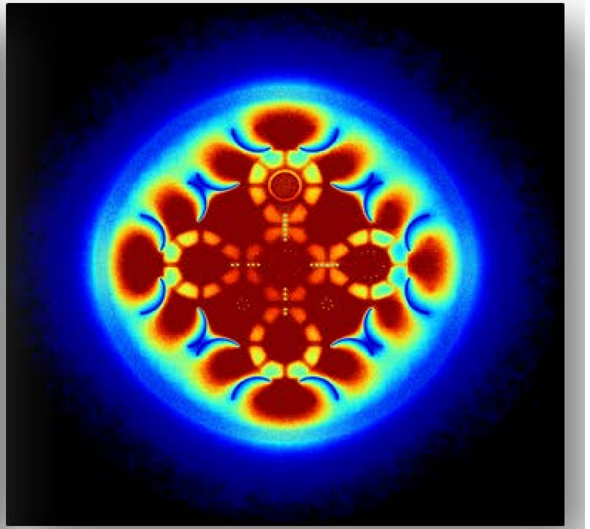
Kelly R. Estes
Director, ATR Business Affairs Division

September 2019

www.inl.gov

Idaho National Laboratory

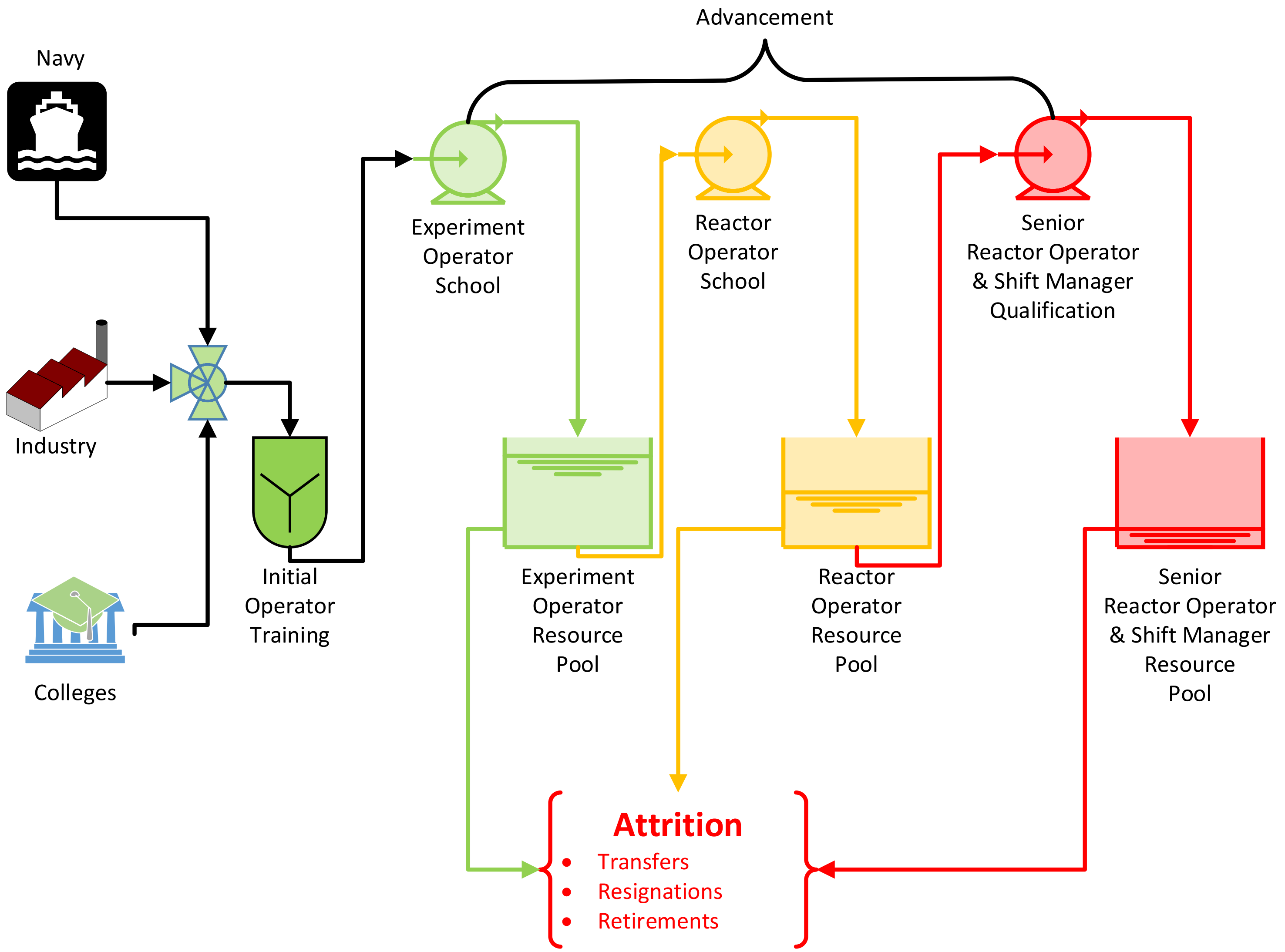
Research Using The Advanced Test Reactor

Research Using The Advanced Test Reactor

University Environment vs. DOE Environment

Advanced Test Reactor - Operations Staffing Model



Conduct of Operations – What is it?

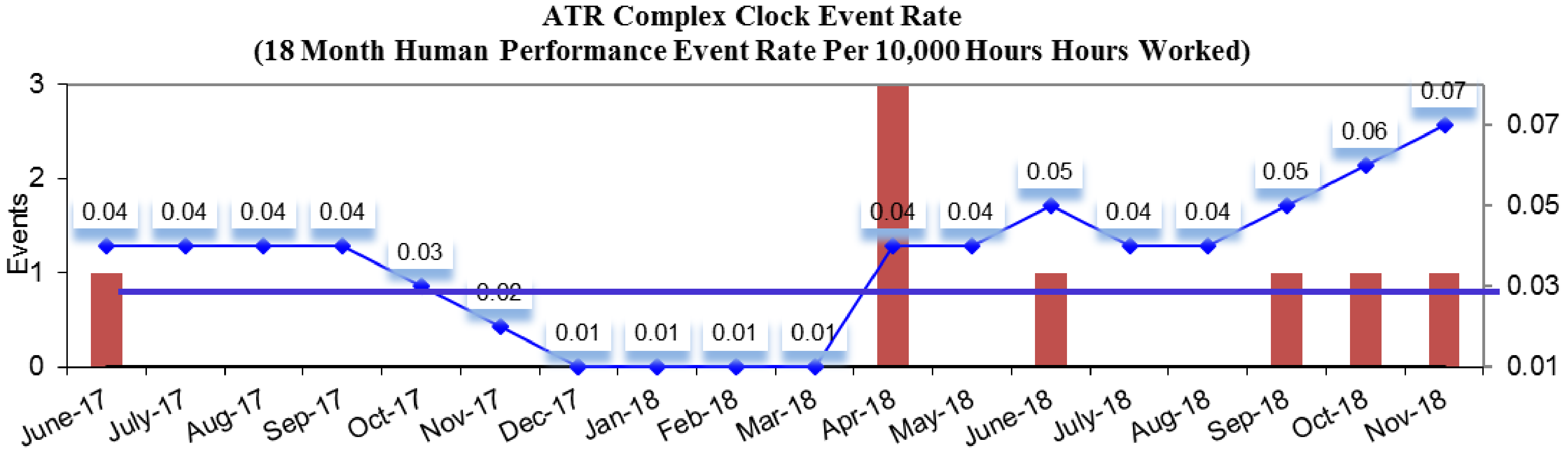
OPERATIONS MANAGEMENT AND LEADERSHIP

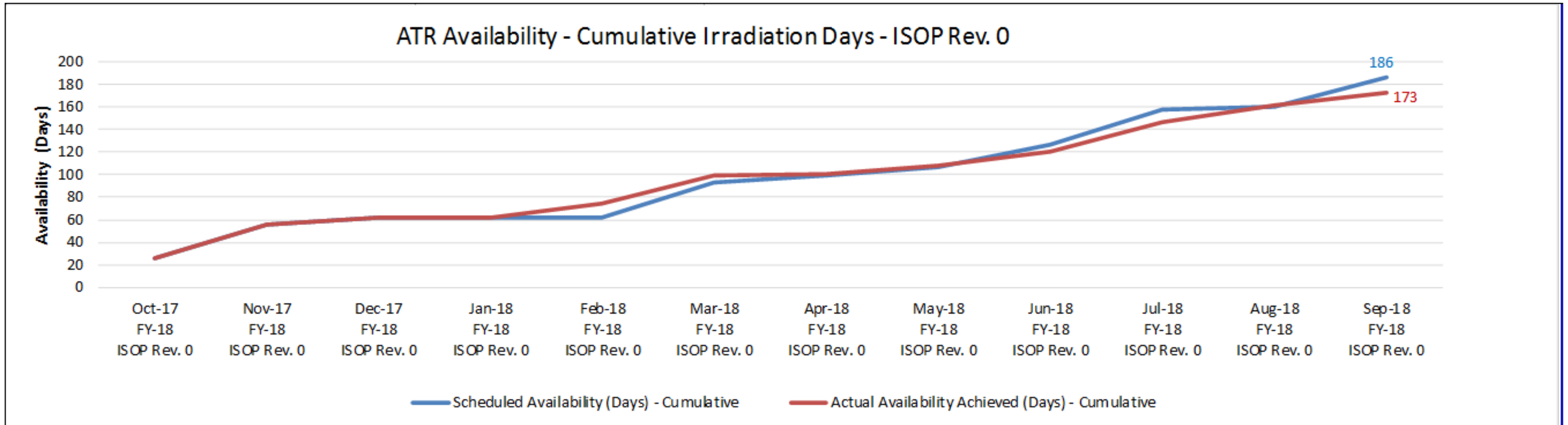
CONTROL ROOM ACTIVITIES

ADMINISTRATIVE CONTROLS

OPERATIONS STAFFING AND PIPELINE MANAGEMENT

ATR Measures of Success





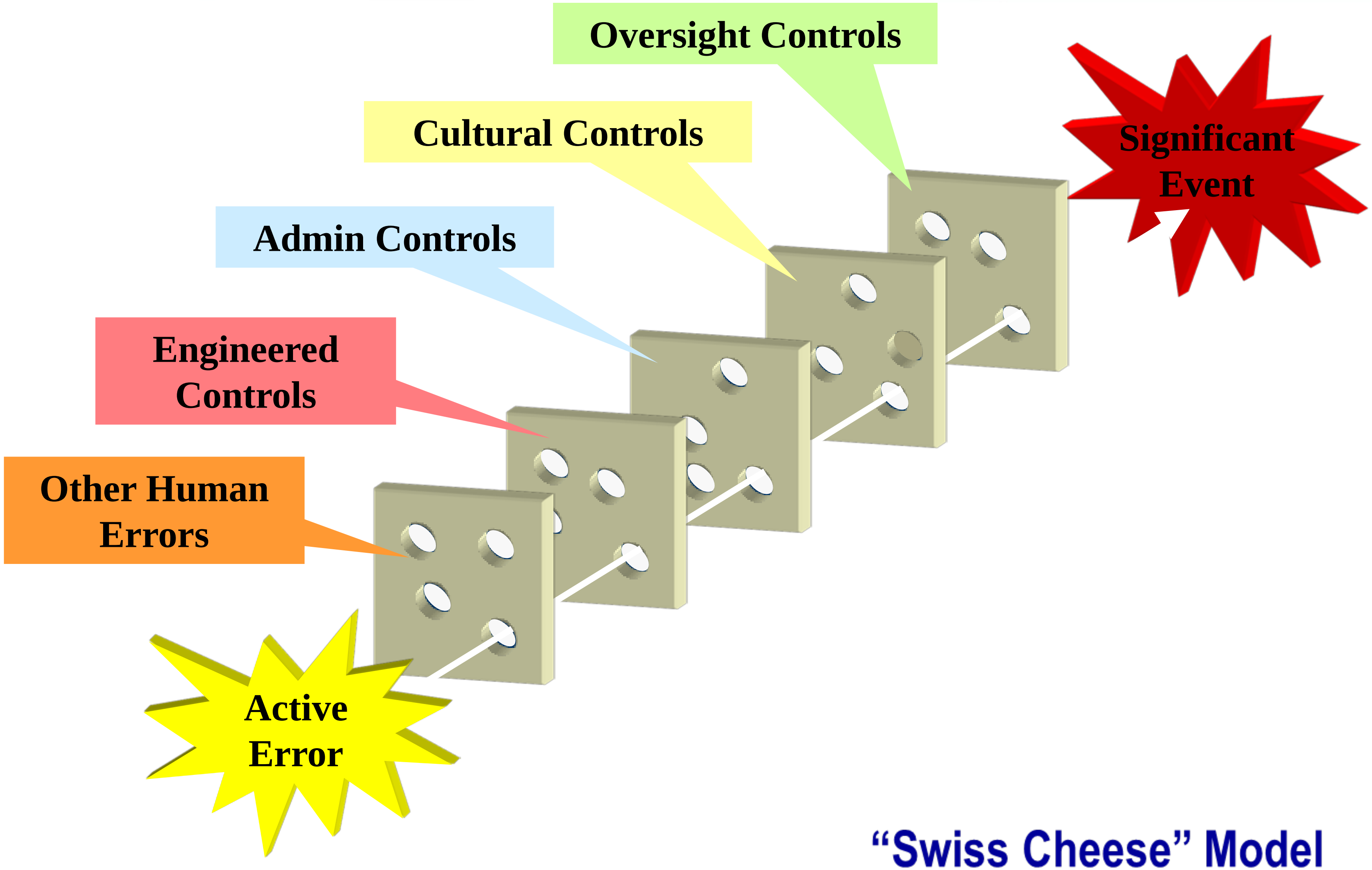
	Oct-17 FY-18 ISOP Rev. 0	Nov-17 FY-18 ISOP Rev. 0	Dec-17 FY-18 ISOP Rev. 0	Jan-18 FY-18 ISOP Rev. 0	Feb-18 FY-18 ISOP Rev. 0	Mar-18 FY-18 ISOP Rev. 0	Apr-18 FY-18 ISOP Rev. 0	May-18 FY-18 ISOP Rev. 0	Jun-18 FY-18 ISOP Rev. 0	Jul-18 FY-18 ISOP Rev. 0	Aug-18 FY-18 ISOP Rev. 0	Sep-18 FY-18 ISOP Rev. 0
Scheduled Availability (Days) - Cumulative	25	55	62	62	62	93	100	107	126	157	161	186
Actual Availability Achieved (Days) - Cumulative	25	55	62	62	74	100	101	108	120	147	162	173
Availability Achieved (%) - Cumulative	100%	100%	100%	100%	119%	107%	101%	101%	95%	93%	101%	93%

Lessons Learned from Game of Thrones

"*I'm sure cutting off heads is very satisfying, but that's not the way to get people to work together*."

Sensa Stark

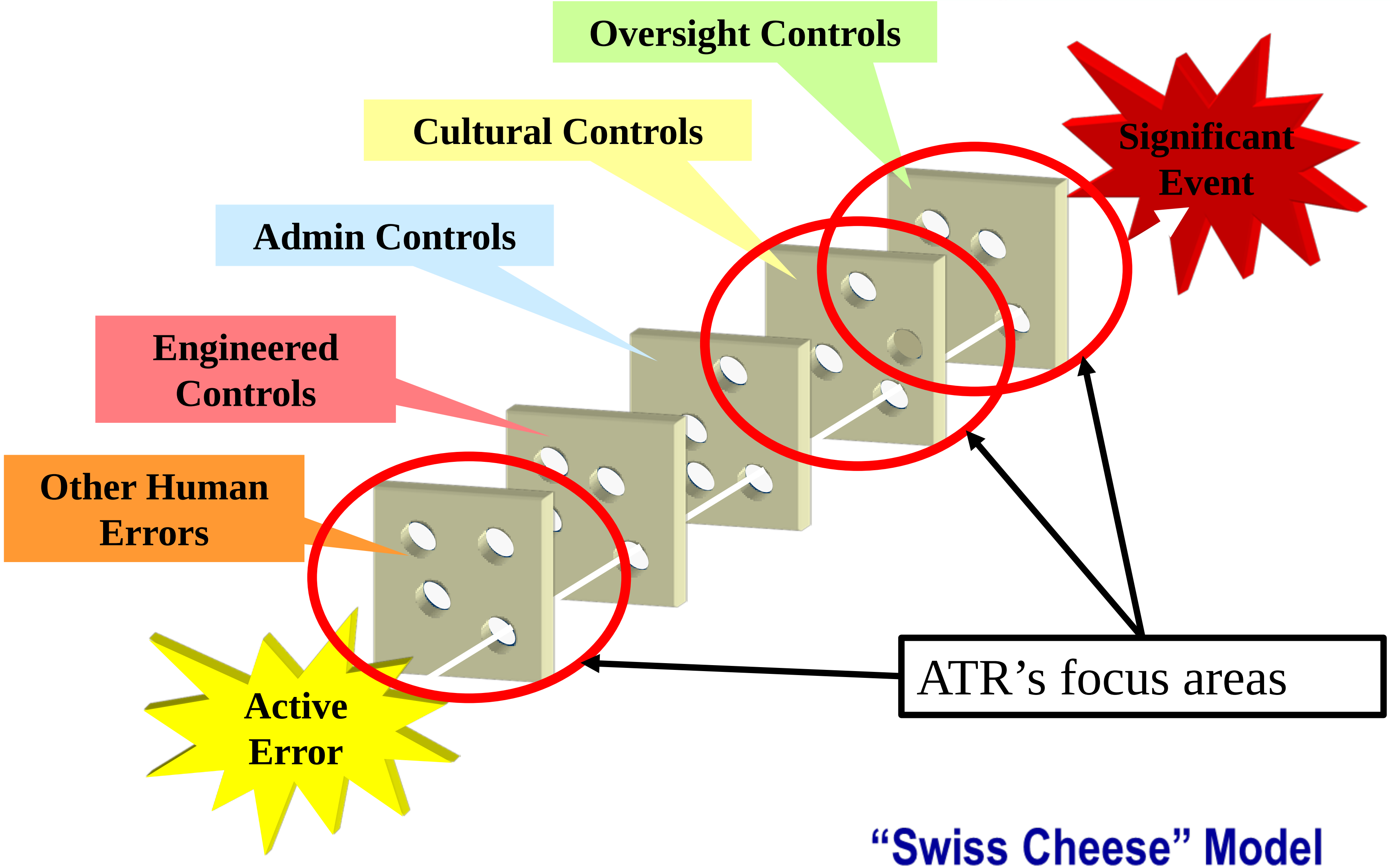
Flawed Defenses - ▲Severity



"Swiss Cheese" Model

James Reason, *Human Error*, 1990.

Flawed Defenses - ▲Severity



"Swiss Cheese" Model

James Reason, *Human Error*, 1990.

Resources ATR Uses for Conduct of Operations Processes

BATTELLE

BushCo HPI.com

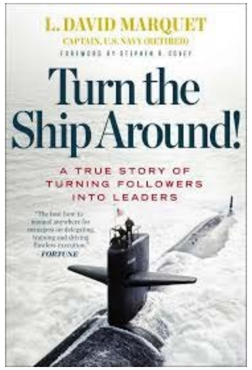
U.S. DEPARTMENT OF ENERGY

EFCOG

DUPONT

GOODNIGHT CONSULTING

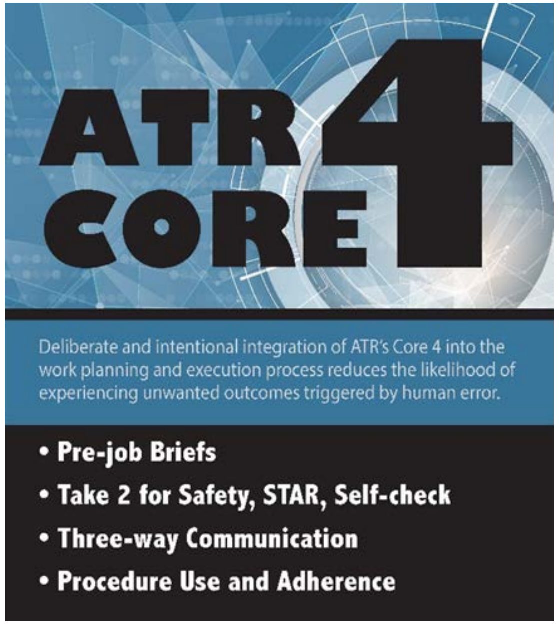
INPO



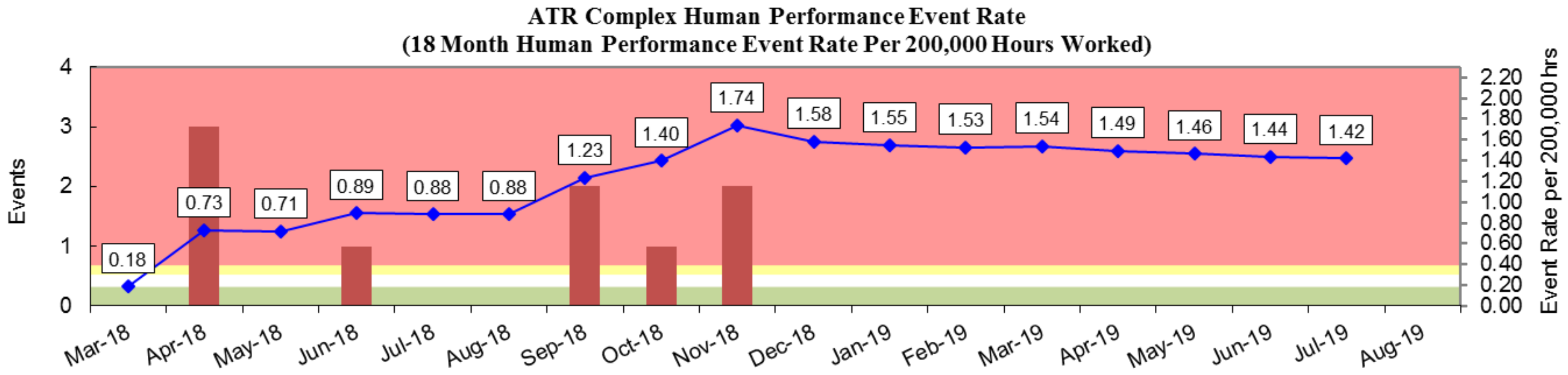
ppa Pinnacle Performance Associates

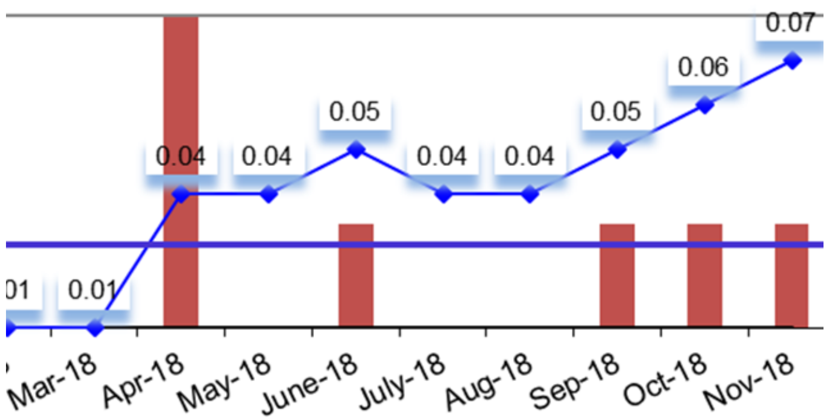
Marathon Consulting Group

ATR Human Performance Improvement

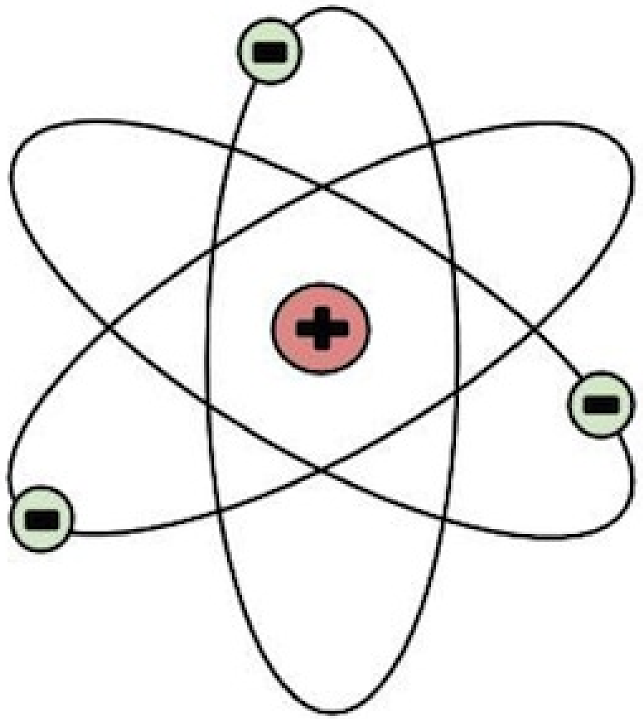


ATR Complex Event Rate – After





Q&A



Answers

Questions

INL

Idaho National Laboratory

How ATR Uses These Resources

- Battelle – Assessments & Guidance Committees
- BushCo – Human Performance Improvement
- DOE- Assessments & Guidance
- DOE EFCOG – Working Groups
- DuPont – The Risk Factor, Human Performance
- Goodnight Consulting – Staffing Capacity Assessment
- INPO – Conduct of Operations Guidance & Training
- Marathon Consulting – Assessments, Cause Analysis, & Coaching, Safety Culture Deep Dive
- Pinnacle Performance Associates - Consulting
- Tarpinian Consulting – Cause Analysis
- Turn the Ship Around – Intent Based Leadership

Operational Overview and Capabilities of the Transient Reactor Test Facility (TREAT)

S. H. Giegel, B. M. Chase, D. T. Willcox

www.inl.gov

Idaho National Laboratory

Outline

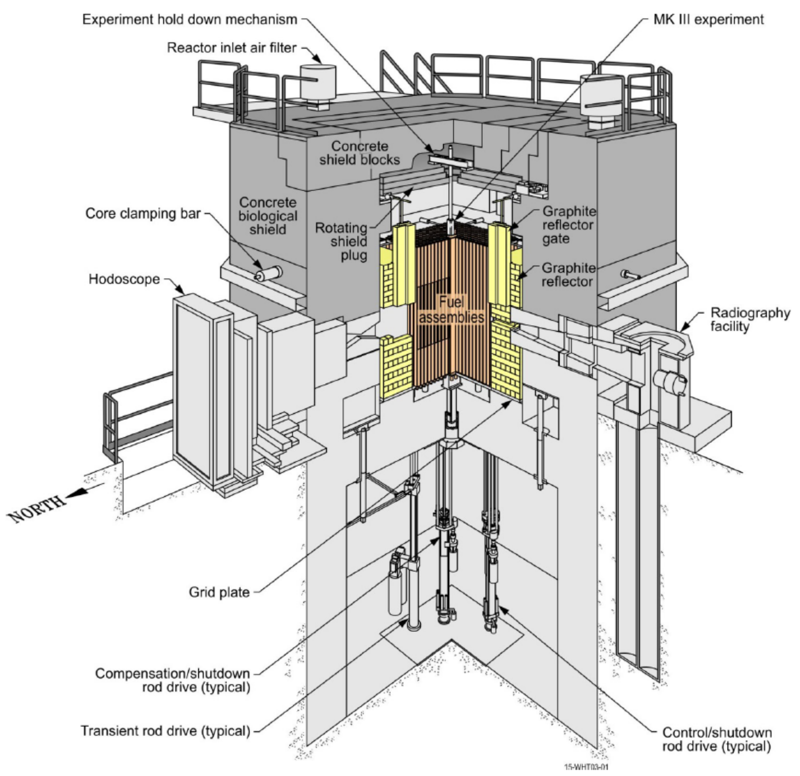
- History of TREAT
- TREAT Reactor Description
- TREAT Experiment Process
- Transient Capabilities

History of TREAT

- Constructed in 1958 and achieved first criticality in 1959.
- The primary mission of TREAT was to support the Fast Reactor Safety Program by providing accident type events in a controlled setting.
- After over 30 years of operation, TREAT was placed in standby mode in 1994 due to reductions in Fast Reactor programs.
- The Accident Tolerant Fuels (ATF) program as well as a renewed interest in generation IV reactor systems development sparked the decision to commence a restart of TREAT which was completed in 2017.

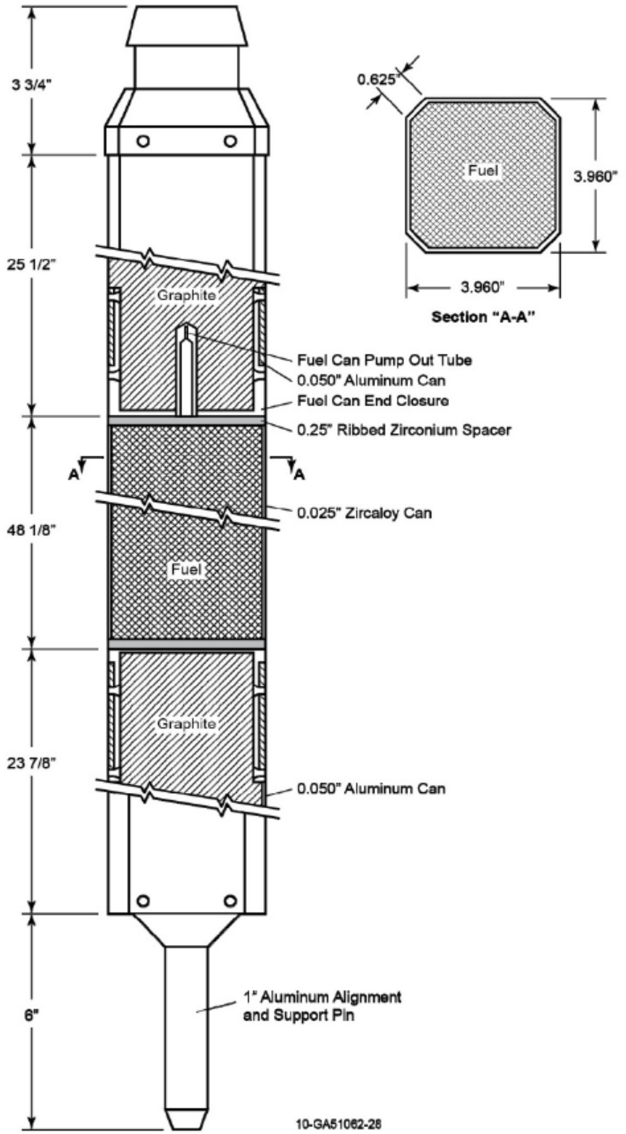
TREAT Reactor Description

- Core Layout
  - 19 x19 array of fuel and reflector assemblies ~ 4" x 4" x 9'
  - Permanent graphite reflector
  - Concrete biological shielding
- Coolant
  - Air
- Control Rods (B_4C)
  - Control/Shutdown rods
  - Compensation/Shutdown rods
  - Transient rods
- Two modes of operation
  - Steady-state mode (120 KW)
  - Transient mode (~20 GW)



TREAT Reactor Description

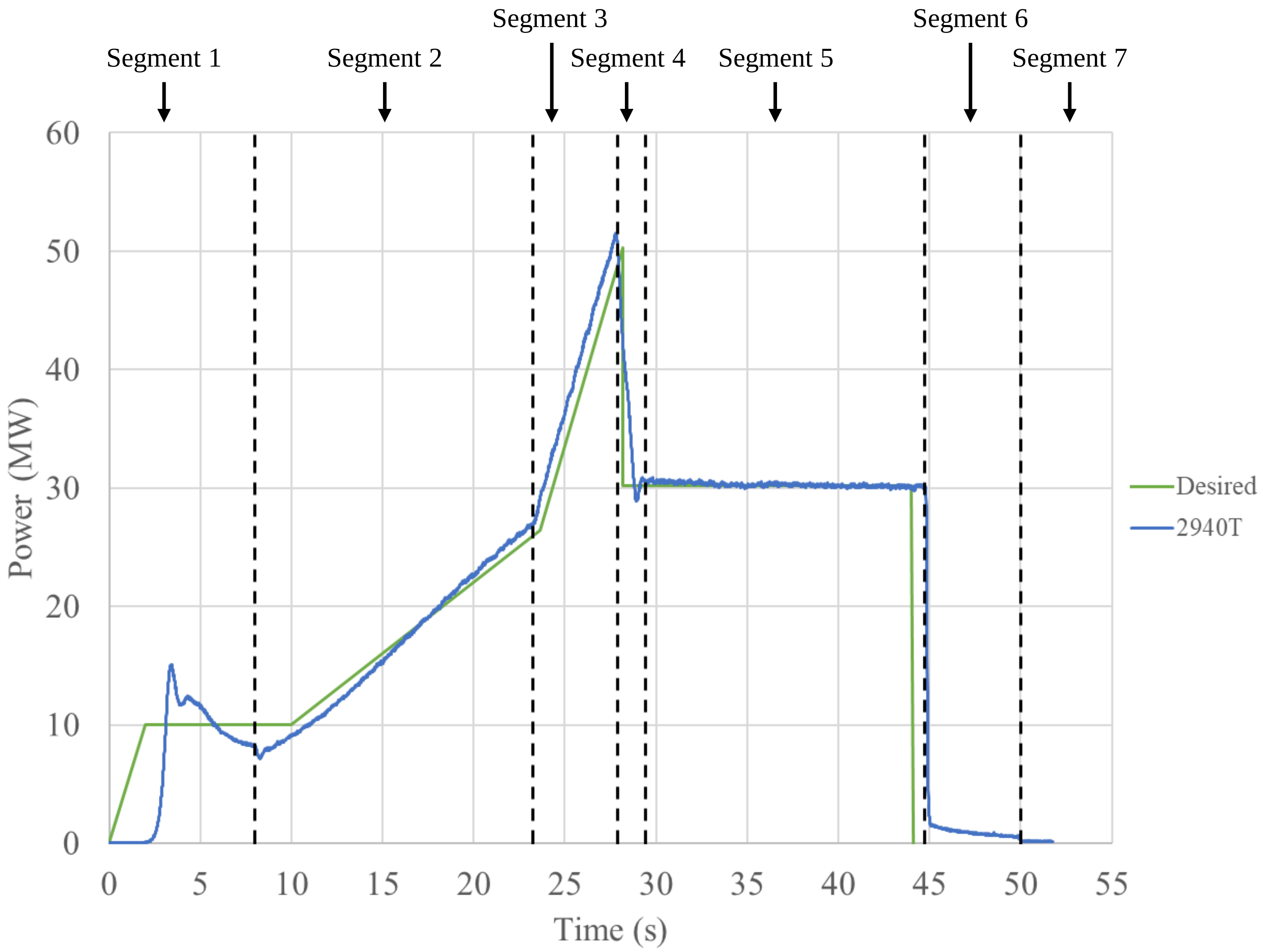
- Fuel design
  - Highly enriched UO_2 dispersed in graphite moderator
  - Carbon-to-uranium (^{235}U) atom ratio is 10,000:1
  - Zircaloy-3 cladding surrounding fueled section
    - Peak transient temperature < 820 ^oC; Safety Limit
    - Peak transient temperature < 600 ^oC; Limiting Control Setting
- Negative temperature coefficient of reactivity
    - Decreased neutron absorption upon heating of ^{235}U/graphite mixture.
    - Shift in neutron energy spectrum due to increase in moderator molecular energy



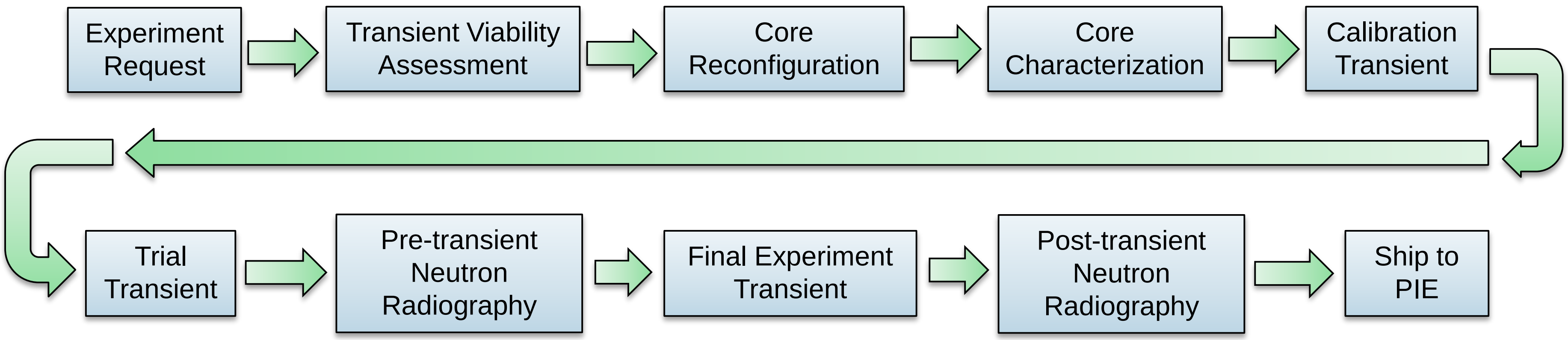
Transient Control Rods and ARCS

- Transient Rods
  - Hydraulically driven: 140 inches per second
  - Rapid step withdrawal initiates transients
  - Controlled by Automatic Reactor Control System (ARCS) during transient operations
- ARCS Developed and installed in the 80's
  - Control algorithm is used to generate rod demand signal sent to transient rod drives
  - Transient profiles are not limited to pulses
  - Transient power profile (prescription) composed of segments
    - Rise or fall on a period, linear power ramp, steady power, etc.
  - Segments terminate based on reaching prescribed parameters
    - Power level, energy deposition, time, etc.

ARCS Prescription Example

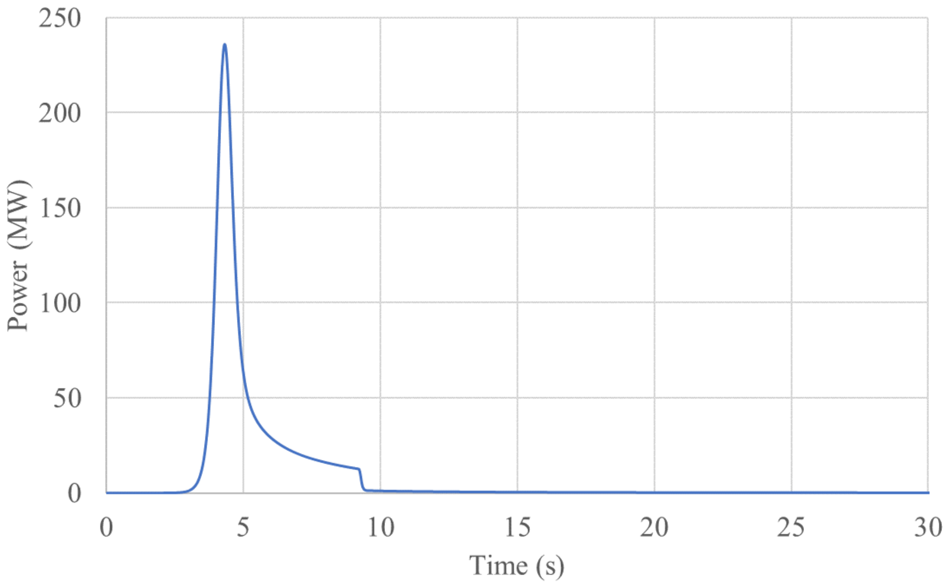


TREAT Experiment Process

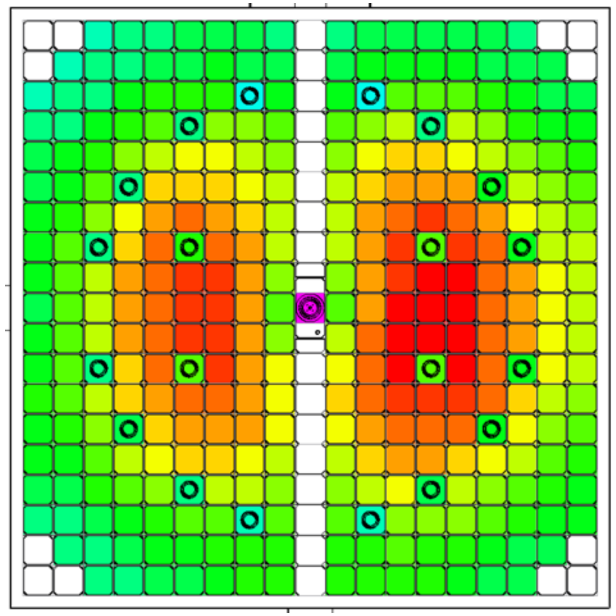


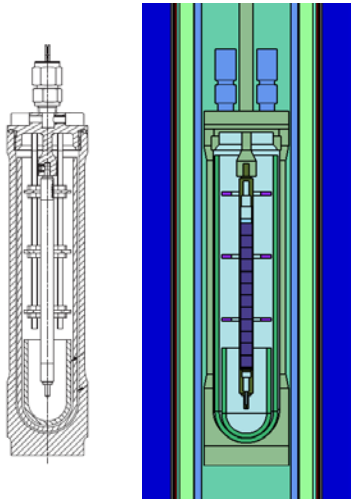
Transient Viability

- Can the current core configuration meet the needs of the requested experiment?
  - Excess reactivity requirements
  - Core temperature profile
  - Core safety limits



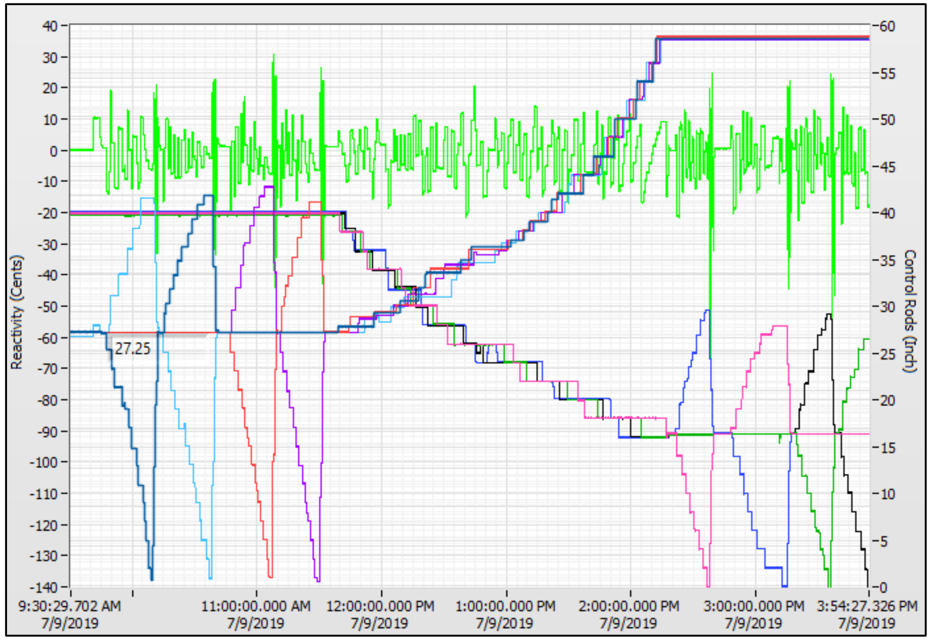
Max Temp (C)	127.9523
Total Energy (MJ)	298.8173
Peak Power (MW)	235.9727





Core Characterization

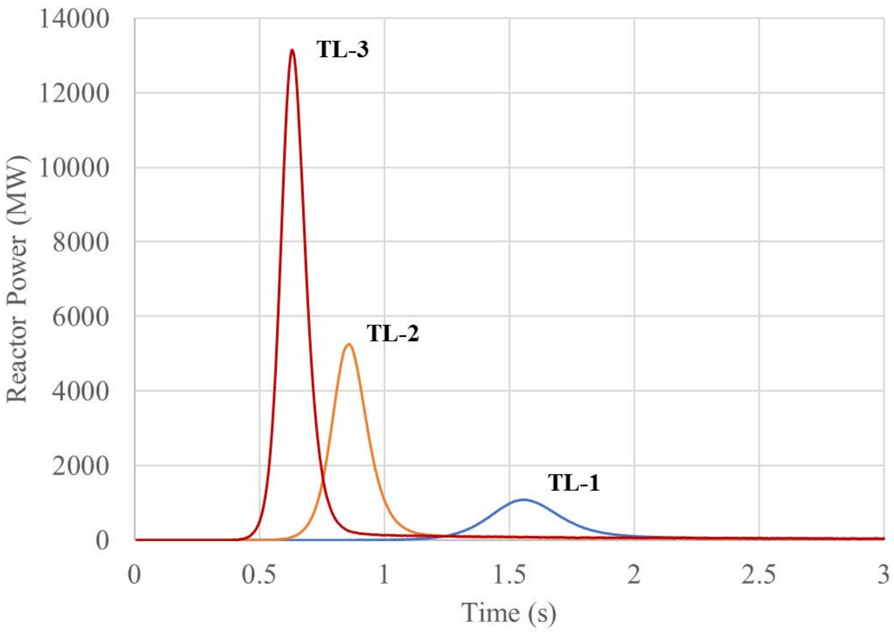
- Core reconfiguration
- Heat balance
- Rod worth measurements
- Temperature-limited transients



Core Characterization Cont.

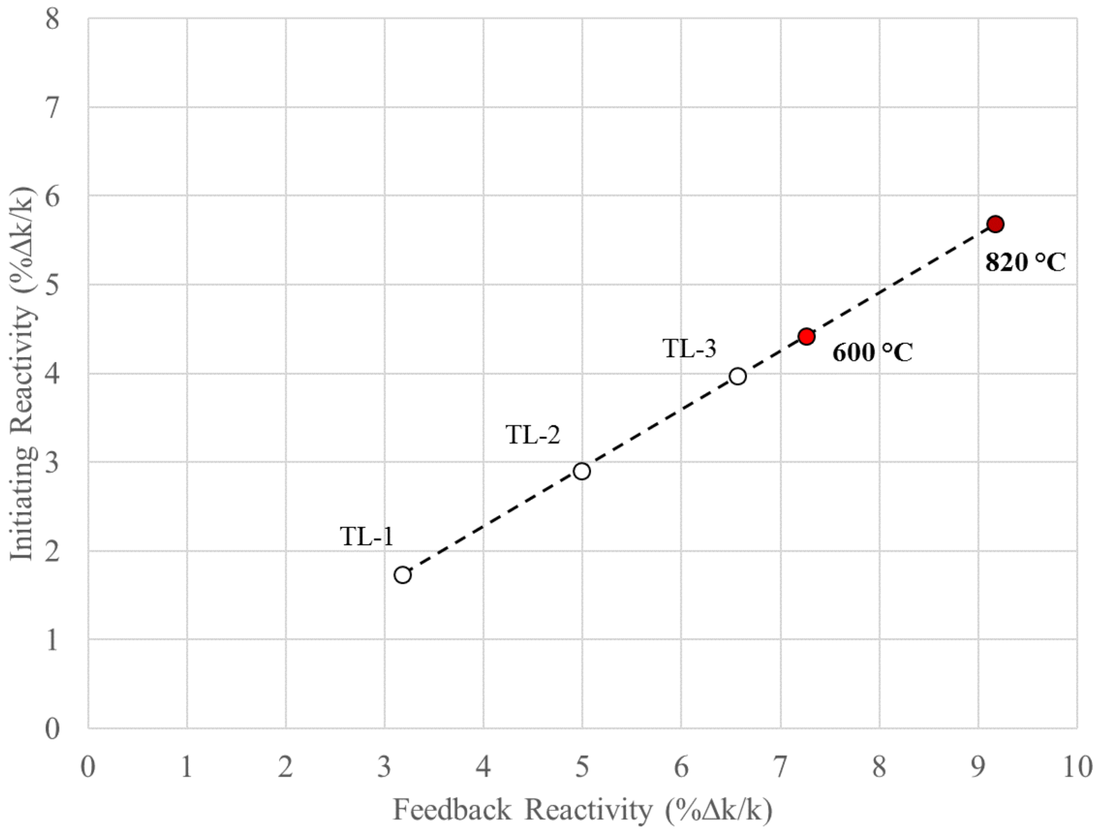
- Temperature-limited transients
  - TL-1: ~1.8 %Δk/k
  - TL-2: ~3.0 %Δk/k
  - TL-3: ~4.0 %Δk/k

	LCS	SL
Initiating rho (%Δk/k)	4.0	5.2
Period (sec)	0.0265	0.0190
Power (MW)	13,000	29,000
Energy (MJ)	1,800	2,7000



Transient 2936T			
Period (sec)	0.0265	Reactivity (%)	3.97
Peak Temperature (°C)		526.4	
Energy (MJ)		Peak Power (MW)	
RTS A	2201	RTS A	13010.6
RTS B	2215	RTS B	13147.6
RTS C	2106	RTS C	12897.8
RTS Average	2174	RTS Average	13018.7
ARCS	2028	ARCS	9993.8

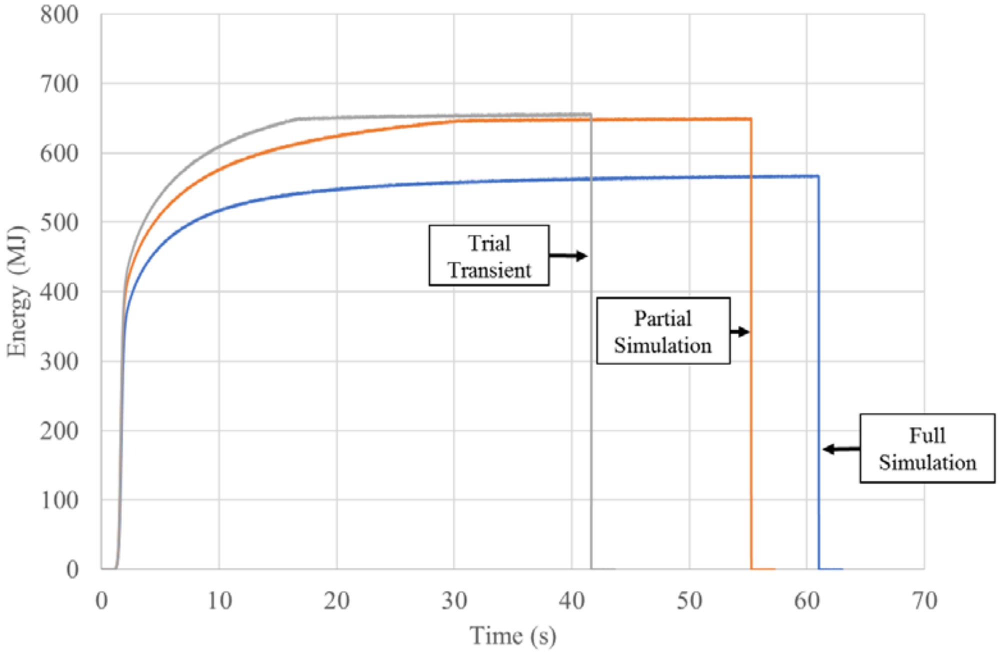
Core Characterization Cont.

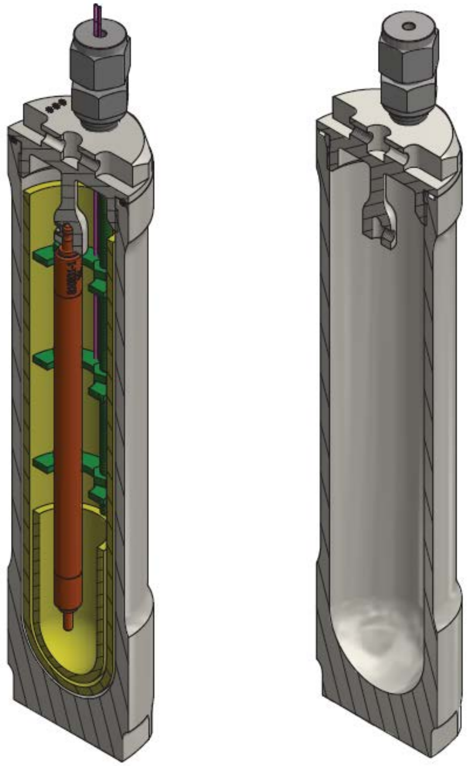


	LCS	SL
Initiating rho (%Δk/k)	4.41	5.67
Period (sec)	0.0232	0.0171
Power (MW)	18,150	38,866
Energy (MJ)	2,614	4,060

Final Experiment Transient

- Full Simulation
  - ARCS
- Partial Simulation
  - ARCS + Transient Rod Motion
- Trial Transient
  - Neutronically Equivalent Dummy (NED)



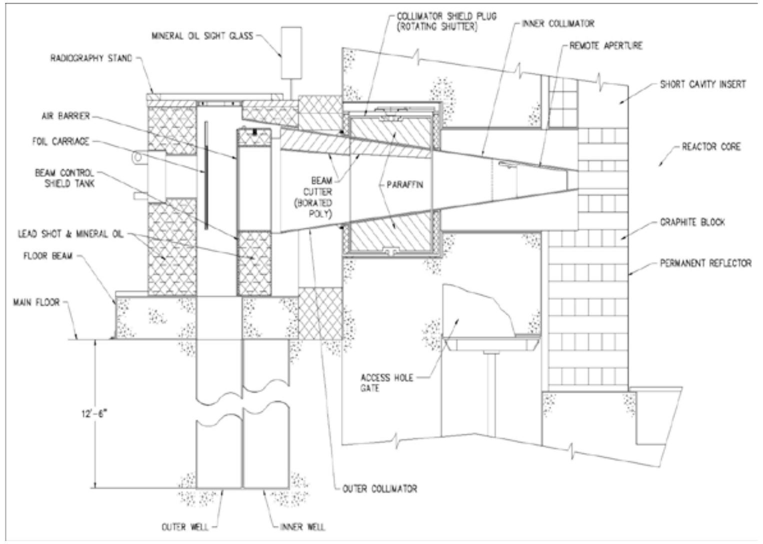


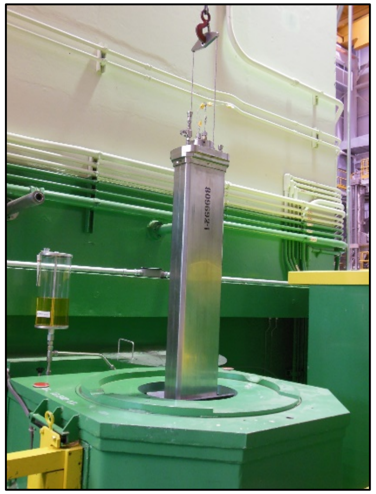
Fueled SETH

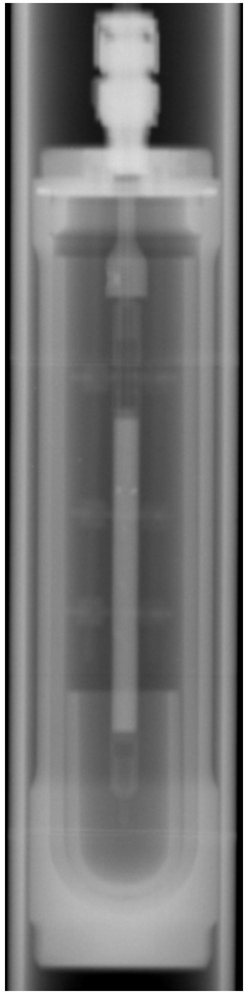
Dummy SETH

Pre- and Post-Transient Neutron Radiography

- Capable of imaging specimens 10 cm x 20 cm and up to 4 m long
- High resolution images can be obtained in 2.5 hours
  - Lower resolution images within 30 minutes





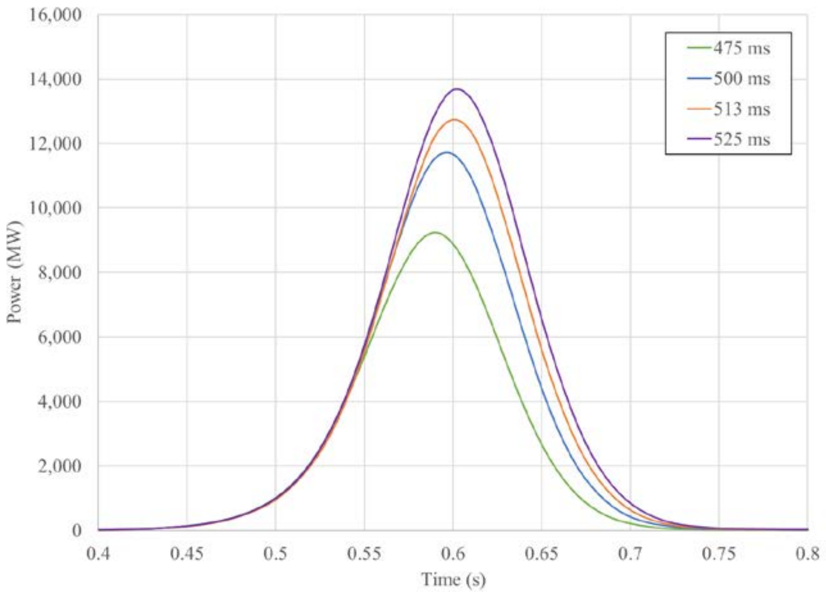

Pre-Transient


Post-Transient

**TREAT neutron radiography of the Separate Effects Test Holder (SETH)-D experiment*

Transient Capabilities

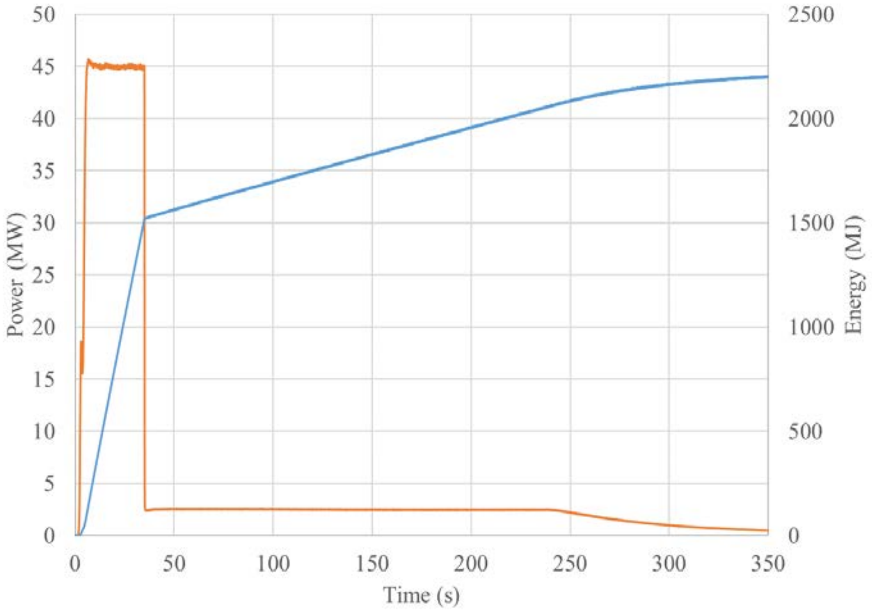
- Narrow Pulse Width Transients
  - Rod Clipping
  - PIRANA
    - Borated poison assemblies to reduce neutron lifetime
    - Testing projected to occur within one year
  - Helium Injection System
    - Rapid ejection of poison He gas with subsequent injection
    - Conceptual design phase

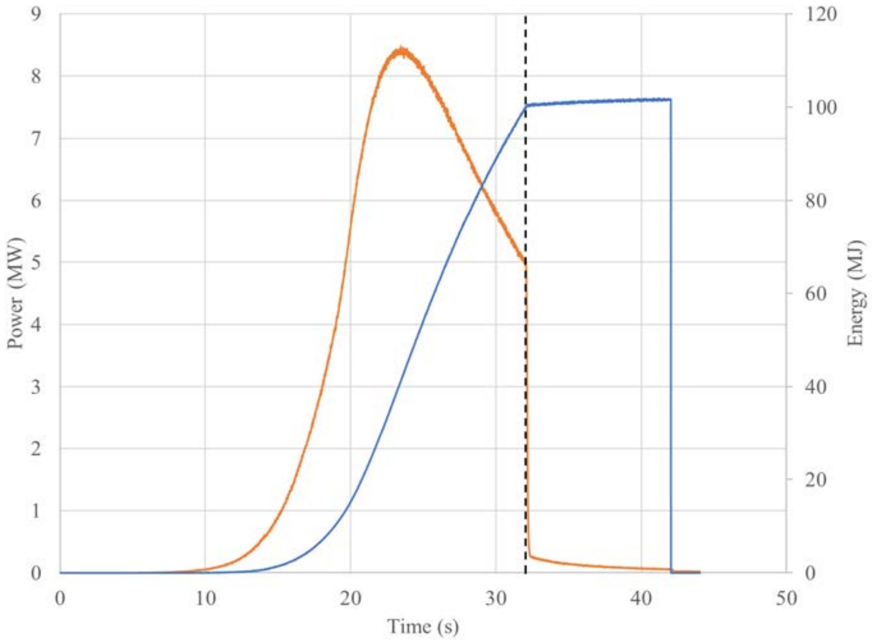


Transient	Rod Clip Time (ms)	FWHM (ms)
2904T	475	91.3
2905T	500	91.9
2906T	513	91.9
2907T	525	93.2

Transient Capabilities

- Shaped Transients
  - LOCA
    - Able to perform LOCA type transients similar to the Halden Reactor
  - SIRIUS-1
    - Testing NASA rocket fuel





SEPT 2019

The University of Texas at Austin

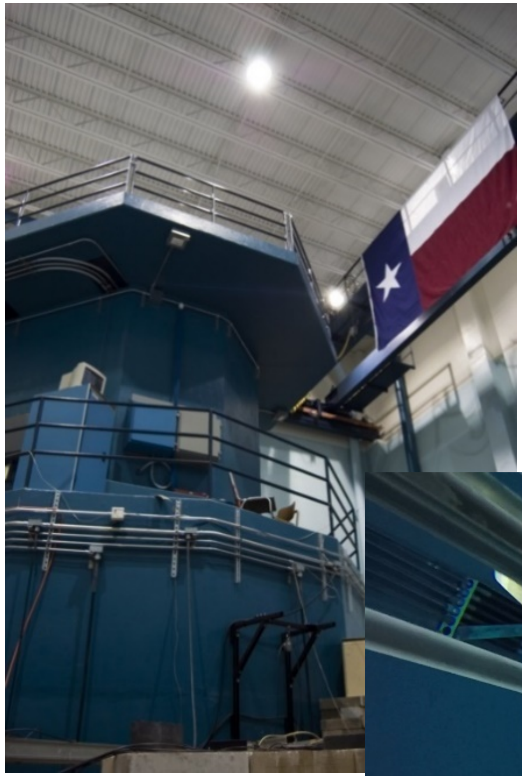
UNIVERSITY OF TEXAS SPENT FUEL INSPECTION

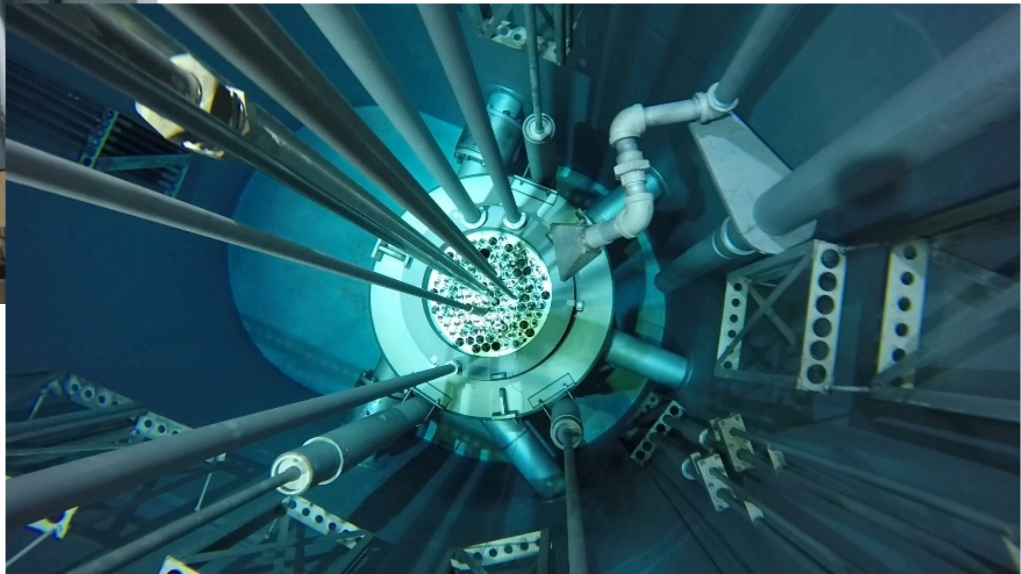
Overview Presentation for TRTR 2019

LARRY HALL AND JIM TERRY
(Nuclear Engineering Teaching Laboratory Reactor Manager and Electronics Technician)

Nuclear Engineering Teaching Lab at UT-Austin

- Radiation Sources
  - 1.1 MW TRIGA Nuclear Reactor
  - Thermo MP320 14-MeV Neutron Generator ($1x10^8$ n/s with a pulse rate up to 20 kHz)
  - Pu(Be) Sources
  - α, β and γ Radiation Sources





Nuclear Engineering Teaching Lab at UT-Austin

- Why do spent Fuel inspection now?
  - Senior member of inspection team eligible to retire
    - Allowed training of newer members
    - Provided video recording of the inspection for historical value for future move
  - Removed a step in the transport of spent fuel from facility to INTEC CPP-603 IFSF

- Who performed inspection?
  - Idaho National Lab (INL) for future storage at INTEC CPP-603 IFSF
    - Mr. Alan Robb
    - Mr. Eric Crapo
    - Mr. Matt Hunt
    - Mr. Mark Argyle
  - NETL-UT staff supported

Nuclear Engineering Teaching Lab at UT-Austin

- Inspection 12-18 August 2018
- 70 elements were inspected
  - 2 Series 2000 Aluminum LEU Rods
  - 19 Series 2000 StainlessSteel LEU Rods
  - 11 Series 3000 Stainless Steel LEU Rods
  - 16 Series 4000 Stainless Steel LEU Rods
  - 20 Series 5000 Stainless Steel LEU Rods
  - 1 Series 6000 Stainless Steel LEU Rod
  - 1 Series 10000 Stainless Steel LEU Rod

- Inspection conducted in accordance with:
  - PLN-218 “Examination of Training Research Isotope General Atomics (TRIGA) Fuel”
  - Engineering Design File, EDF-6293 “Inspection of TRIGA Fuels”

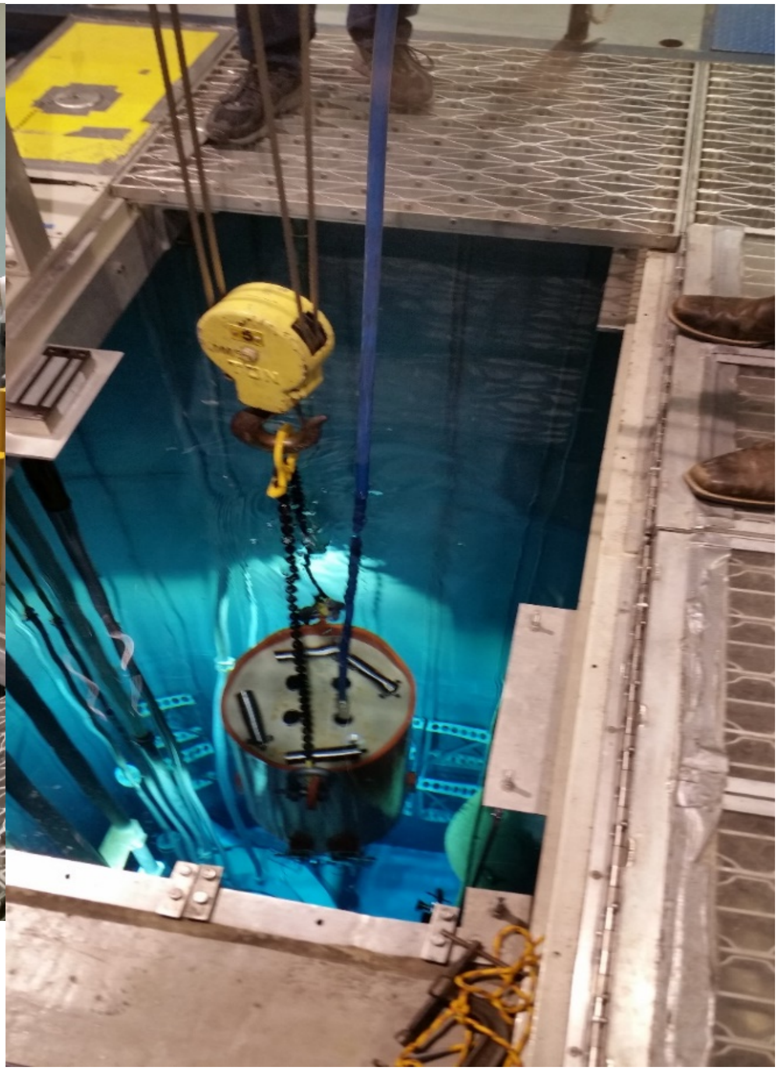
- 2 of the 70 elements were considered failed and stored in dry well. After examination they were placed in sealed failed fuel cans CAN-GSF-130-47-436 and E-cup Tamper Indication Devices installed and returned to dry well until transfer to INL.

Nuclear Engineering Teaching Lab at UT-Austin

- Prepare
  - Spent fuel was stored in storage wells in floor of reactor bay.
  - Fuel was moved in custom cask that holds four fuel elements at a time.
  - The cask was lifted by UT overhead 5 ton crane.

Nuclear Engineering Teaching Lab at UT-Austin

- Prepare
  - The cask moved the fuel from the storage wells to the reactor pool.
  - The fuel was removed from cask under water with fuel movement tool.
  - Dose rates in occupied areas remained at background throughout the event.



Nuclear Engineering Teaching Lab at UT-Austin

- Setup
  - INL shipped their equipment to UT prior to scheduled inspection.
  - UT staged equipment at top of reactor prior to INL arrival.
  - A custom fuel holding rack was developed by INL so they could leave it behind after inspection to prevent need to ship potentially contaminated equipment.

Nuclear Engineering Teaching Lab at UT-Austin

- Procedure (360° inspection performed)
  - Multiple persons viewed each element.
  - Slowly lowered camera down side of fuel element while it was in holder with measurement values down the side. Three passes conducted to ensure 100% coverage.
  - Video recorded inspection to include recording each elements serial number.

Nuclear Engineering Teaching Lab at UT-Austin

- Procedure (360° inspection performed) cont.
  - Recorded voice of each person discussing inspection.
  - Marks (pits, scratches discoloration) annotated on fuel examination data sheets.
  - Radiation readings taken for each fuel element 4 inches away. (Range from 0.0 R/hr to 451.0 R/hr)

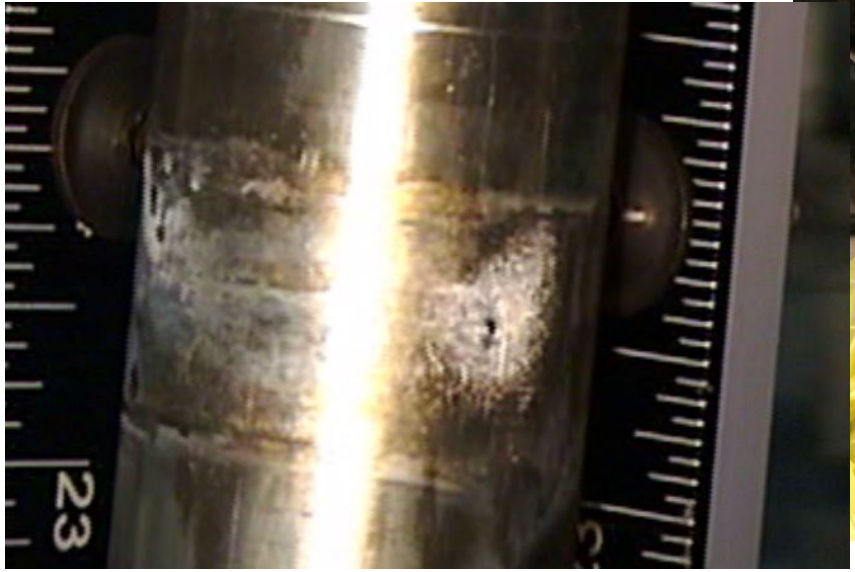
Nuclear Engineering Teaching Lab at UT-Austin

- Results
  - 7 elements considered to have some type of issue requiring further examination prior to shipment.
  - Noticed galvanic corrosion on top and bottom of elements due to being stored in an aluminum element rack submerged in water.
  - Various minor nicks, pits and scratches.
  - Dark color on stainless steel in fuel region.

Nuclear Engineering Teaching Lab at UT-Austin

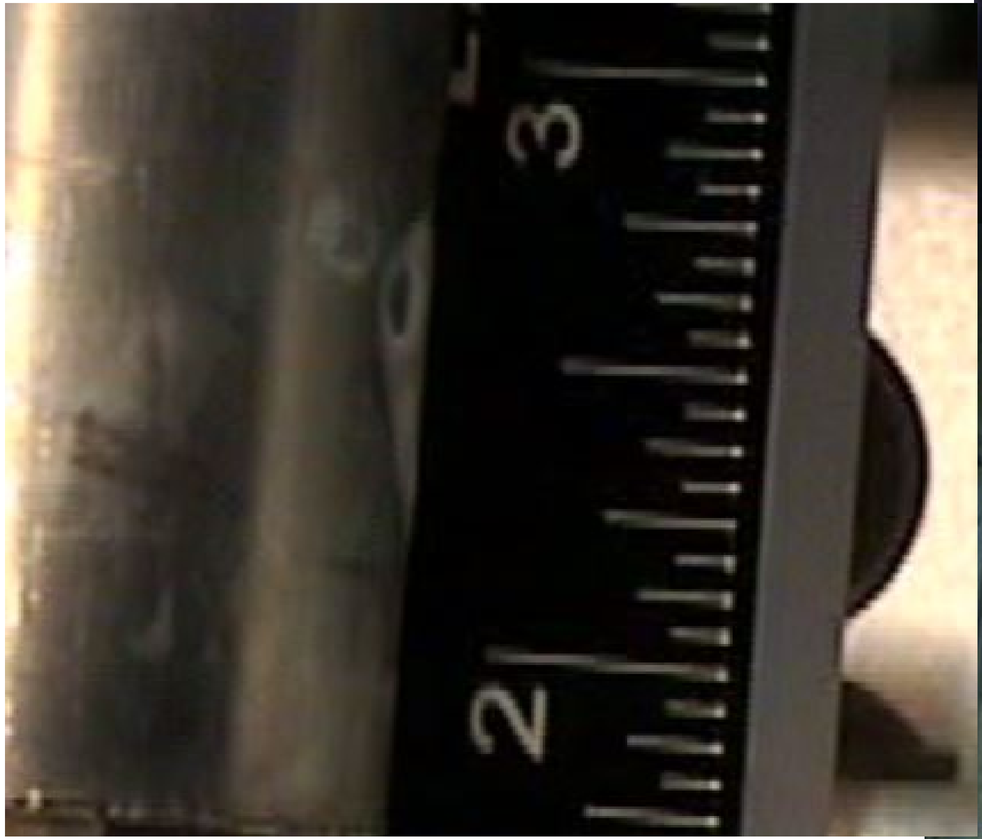
Region where fuel touches geometric storage rack

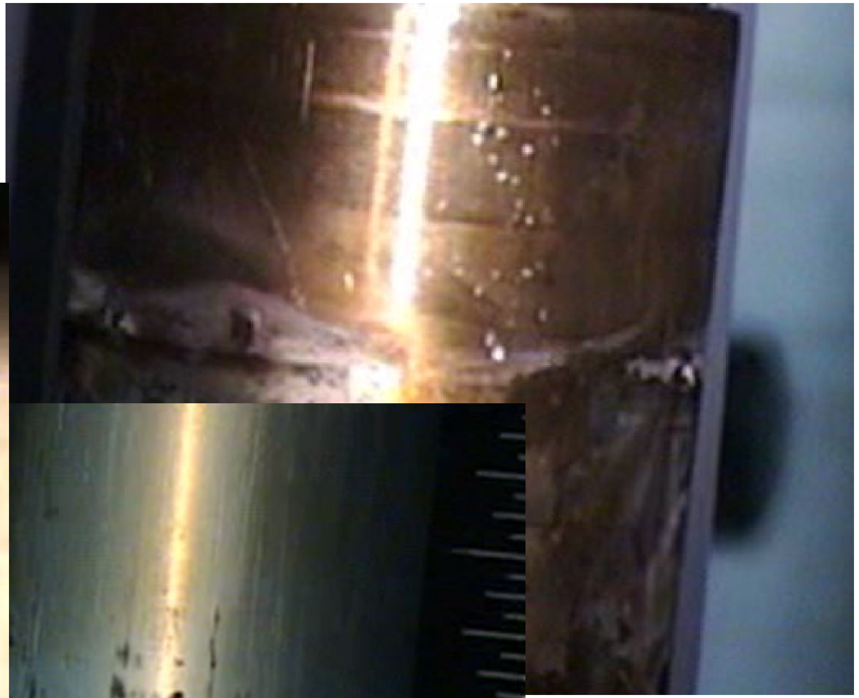
Fuel stored in wells filled with water causing galvanic corrosion.



Nuclear Engineering Teaching Lab at UT-Austin

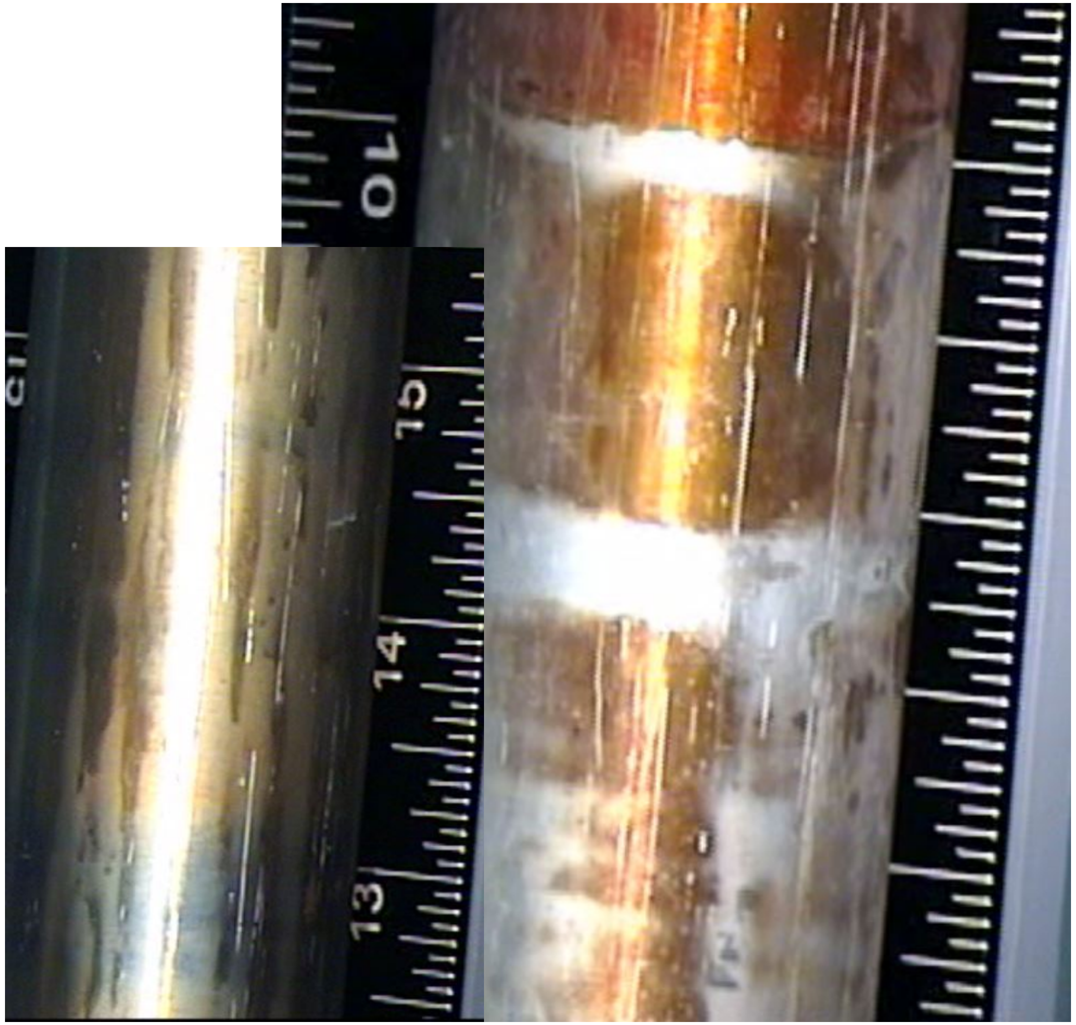
- Pitting-severity determined by shine off pitted region.





Nuclear Engineering Teaching Lab at UT-Austin

- Scratches and fuel color region.

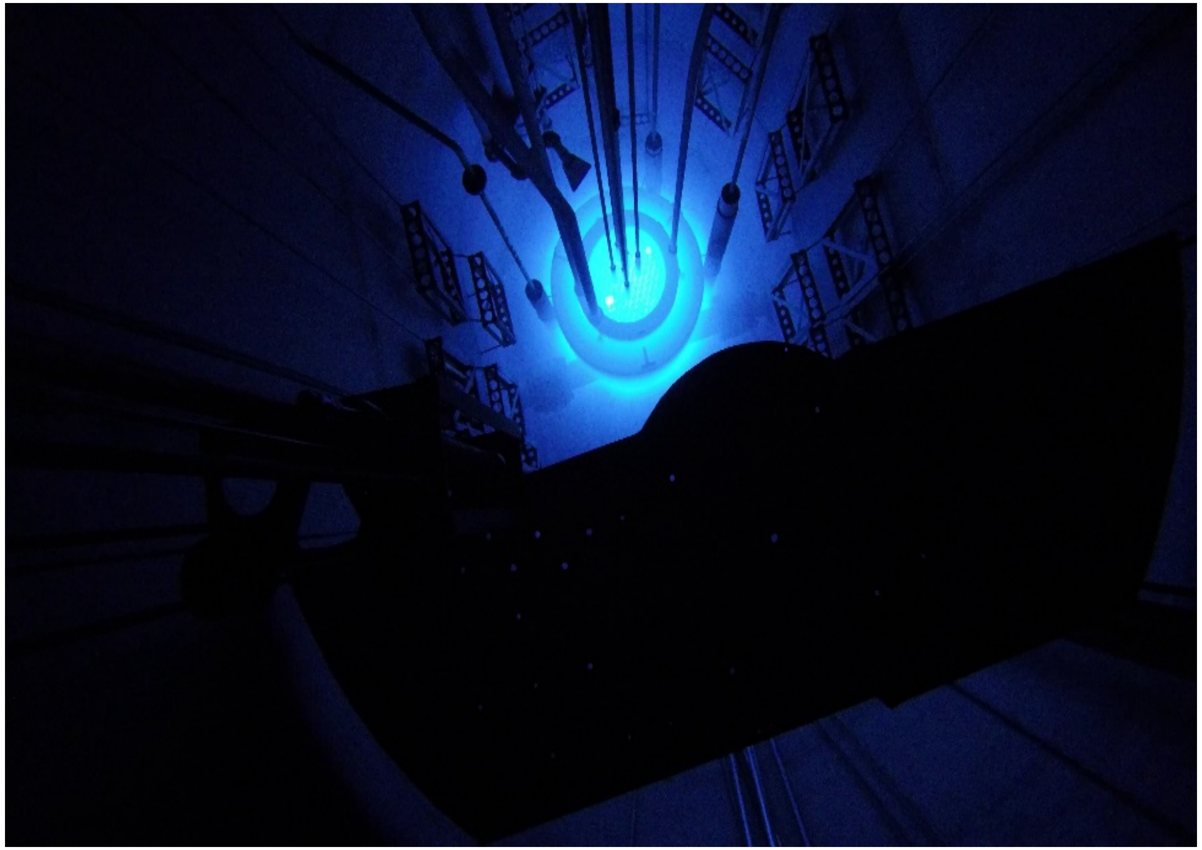


Nuclear Engineering Teaching Lab at UT-Austin

- Wins
  - Have a complete library of fuel inspection videos.
  - Inspections completed for when future shipment authorization is awarded for transport to Idaho.
  - 2 failed elements already placed in sealed failed fuel cans for shipment.
  - Facility flexed ability to move fuel from wells to pool without incident.
  - Inspection was a smooth operation with no incidents.

Nuclear Engineering Teaching Lab at UT-Austin

- Lessons learned
  - Hazards of storing spent fuel in an aluminum fuel rack in a wet well.
  - Nothing quick about fuel inspection, but it is necessary.



Nuclear Engineering Teaching Lab at UT-Austin

- Next Step
  - Solve issue with fuel shipments to the State of Idaho.
  - If shipments cannot be cleared, determine way to store additional old fuel locally.
  - Determine what requirements must be met to store spent fuel in dry wells.

Questions?