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Transient Testing Reactor Physics Workshop – May 2016

May 2016

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ABSTRACT

This Workshop was conducted in conjunction with the PHYSOR Conference being held in Sun Valley, Idaho, USA from May 1st -5th, 2016. The intent was to conduct a technical exchange on transient test reactor physics and modeling techniques used to support the Idaho National Laboratory TREAT facility and CABRI (CEA/IRSN) of France. Currently, several parallel programs are independently supporting critical aspects of TREAT physics analysis. Successful implementation will require active collaboration across programmatic boundaries to ensure the transfer of critical information as well as coordination and prioritization of activities.

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Transient Testing Reactor Physics Workshop – May 2016

1. INTRODUCTION

The Workshop was conducted in conjunction with the PHYSOR Conference being held in Sun Valley, Idaho, USA from May 1st -5th, 2016. The Transient Testing Physics Workshop had two primary objectives. The first was to open a dialogue between U.S. and French technical experts related to reactor physics at the Idaho National Laboratory (INL) Transient Testing Reactor (TREAT) facility and the French CABRI Research Reactor project (CEA/IRSN). This dialogue was to inform the development of action plans within both the DOE/CEA and INL/IRSN bilateral cooperative agreements. The second objective was to help integrate the diverse domestic efforts in this area related to the TREAT project. Currently, several parallel programs are independently supporting critical aspects of TREAT physics analysis. Successful implementation will require active collaboration across programmatic boundaries to ensure the transfer of critical information as well as coordination and prioritization of activities.

1.1 Basic Topical Areas

The workshop consisted of four basic topical areas (as shown in the attached agenda) including:

- **Transient testing facility overviews** (TREAT and CABRI descriptions and programs);
- **Reactor startup requirements** (relationship between operating requirement, reactor analysis codes, and physics testing);
- **Advanced modeling and simulation tool development** (code development and benchmark cases for validation), and
- **Fuel motion monitoring system recovery**.

This workshop was constrained to these areas in order to focus discussion. However, the scope of follow-up workshops will expand to other critical areas outlined in the bilateral agreements.

2. TOPIC 1 – TRANSIENT TESTING FACILITY OVERVIEWS

Presentations were provided on both TREAT and CABRI. Nick Woolstenhulme presented the history of transient testing at INL (emphasizing TREAT). A joint presentation was provided by Jean-Pascal Hudelot (CEA) and Bruno Biard (IRSN) on CABRI. Both presentations are included in the appendix.

3. TOPIC 2 – REACTOR STARTUP REQUIREMENTS

Presentations were provided by Jim Parry (TREAT Chief Reactor Scientist) on the TREAT startup plan and Jean-Pascal Hudelot (CEA) on the CABRI commissioning progress and plan. Both presentations are included in the appendix.

Parry summarized some key aspects of the TREAT physics plan. TREAT operational limits are based on experimentally obtained reactivity insertion limits. Physics codes are used to design sub-maximal transients of interest to experimenters and do not serve any safety function. As such, conservative point-kinetics codes have traditionally been used for first-order core and transient design for experiments. Detailed reactor operation parameters were then refined through operational testing (e.g. calibration tests).

The approach to physics testing for new core configurations was reviewed. Fundamentally, reactor engineering will characterize a new core design by first conducting rod worth tests, heat balance tests (to calibrate power instruments), and, finally, a series of sub-maximal temperature-limited transients to estimate the limiting reactivity insertion. Subsequent testing is then conducted by the experiment program to determine coupling between the experiment and the reactor for the specific test device being used.

While computational analysis is used to inform these tests, the data required for experiments is supplied by empirical results collected during these calibration tests.

The existing instrumentation was reviewed (power measurements are based on instruments located in biological shielding, far from the experiment). It is also important to note that the thermocouples in many of the existing instrumented fuel elements have failed during service and only a limited number are still functional. New devices will need to be developed and qualified to measure local power. Potential areas for insertion of new instruments were described and include the ‘coolant channels’ between elements, unused control rod drive positions, and specially designed replacement elements. The use of more sophisticated codes to reliably design experiments without calibration runs may require improved material property data for TREAT driver fuel and the availability of advanced instrumentation in order to validate the codes.

TREAT reactor engineering is working to maintain a suite of codes that can be validated (benchmark studies or future physics testing) and made available to experimenters. These codes will include both steady-state and transient codes. The steady-state codes are used to estimate the power coupling factor (PCF) between the reactor and experiment.

Steady-state models are being developed on multiple platforms by various TREAT stakeholders and include

- SCALE (TREAT Reactor Engineering –INL),
- MCNP (TREAT Experiments – INL, Hodoscope modeling – Texas A&M University/Idaho State University, and LEU Conversion - ANL),
- SERPENT (Advanced TREAT M&S - INL), and
- PARCS (TREAT Benchmark studies – University of Michigan).

Transient codes include the

- ARCS simulator (TREAT Reactor Engineering),
- S-TREK (a code being adapted by TREAT Reactor Engineering from the ARCS simulator for use on a more universal platform),
- a RELAP-5 model (being developed by the TREAT Experiments program to support experiment design),
- TREKIN (being updated by ANL for LEU Conversion core design), and
- MAMMOTH/RATLESNAKE (NEAMS-TREAT)

Jean-Pascal Hudelot provided an overview of the CABRI physics modeling and commissioning effort. CABRI steady-state core neutronics are described using TRIPOLI4 (also possible with MCNP). These calculations were initially validated against historic critical tests that were conducted while configured with the sodium loop. Further validation for the water-loop configuration is based on updates and improvement to representation of the core configuration. This includes both the core geometry and the material compositions within the core.

In addition to the core neutronics, tools to describe test pins are also required. The calculations performed at commercial power plants to estimate fuel pin composition are not always accurate enough to support determination of the power coupling factor during transient testing. CEA and IRSN both maintain a suite of codes to perform this function (CEA – APOLLO2/REL2005, APOLLO3, CESAR5.3 and IRSN- MORET4, MCNP, and VESTA).

CABRI transient behavior is treated using the point kinetics code DULCINEE. This code uses a pseudo steady-state method where the feedback parameters are calculated after each time step using preliminary TRIPOLI4 calculated results (this includes Doppler effect and delayed phenomena like clad expansion and coolant density). Other critical parameters (delayed neutron fraction and generation lifetime) are determined either computationally, using MCNP or the latest version of the TRIPOLI4 code, or measured during the commissioning tests. DULCINEE also embeds simplified thermal and thermal hydraulics models for processing single- or two-phase flows in natural or forced convection. Heat transfer in the fuel rods is modeled from the inside to the outside. Several types of regions are described (fuel/gap/clad). It allows the user to specify the control system operational parameters and predict the resulting transient or to input the desired transient and calculate the necessary control strategy to achieve it. Validation of DULCINEE is based on 19 past transient tests.

CABRI driver fuel performance during each transient is analyzed using SCANAIR to ensure compliance with safety criteria including fuel temperature, clad temperature, and clad strain. The use of multi-physics codes that couple phenomena to predict driver fuel performance is being pursued using the ALCYONE code (fuel performance) coupled with APOLLO3 (neutronics).

CABRI primary cooling system thermal hydraulics are being treated using CATHARE, TRIO-U and the commercial CFD code STAR-CCM+.

A variety of tests are being conducted during the commission phase of the reactor to provide critical inputs to these models. Examples include the following areas.

- Prior to nuclear operations - He-3 system depressurization rate as a function of valve positions and impacts of He-3 purity.
- Low power operations (*Commission #1*) – neutronic characterization of the core. The results were compared with uncertainty targets laid out in advance of the testing program. First criticality was achieved October 20, 2015.
- Full power operations (*Commission #7*) – heat balances for up to 25 MW steady-state power and neutron detector calibration for transient power from 100 kW to ~25 GW.

Discussion revealed a few common physics issues related to uncertainties in the use of point kinetics methods. Most significantly, the neutron spectrum is known to change during the transient. In the case of TREAT, the spectrum hardens during the transient due to the increase in graphite temperature. In CABRI the spectrum softens as the He-3 is removed from the reactor. In both TREAT and CABRI, the radial power distribution also moves during the transient. Advanced codes may allow for explicit treatment of these phenomena.

4. TOPIC 3 – ADVANCED MODELING AND SIMULATION TOOL DEVELOPMENT

The development and validation of new codes requires detailed comparison of calculated results with relevant measured parameters. Benchmark documentation of historic TREAT tests is being developed (by TREAT Reactor Engineering, TREAT Experiments, NE IRP projects, and NEUP projects) to enable the new MOOSE based codes (to be developed under NEAMS).

A presentation was prepared by John Bess (attached) to describe the objective, status, and early findings of the benchmark efforts. Dr. Bess is leading an effort within INL to develop the benchmark cases required to understand uncertainties in the TREAT core that could impact reactor operations (criticality, control rod worth, excess reactivity, shutdown margin) and to support validation of nuclear data utilized in the existing TREAT operation codes. Three core configurations are being assessed including the minimal critical core, a mid-size core to be determined, and the M8CAL configuration. At the end of this process, results will be packaged and documented for the International Reactor Physics Experiments Benchmark (IRPhEP) handbook. These cases will include the required core geometry, component compositions, and experimental results. A Baseline Assessment of TREAT for Modeling and Analysis Needs (BATMAN) report (INL/EXT-15-35372) outlining the core component geometry and material compositions is complete and is being widely used for model development. The experimental data for the selected transients is being collected but users are encountering difficulty finding all the desired data as well as defining the quality level/pedigree on the data that is located. It is anticipated that additional testing will be required during startup physics testing to 'fill in the blanks'.

However, a few dominant dimensional and material property unknowns have already been identified that substantially impact the uncertainty in reactor modeling. In particular, the actual position of the poison section of the control rods in the core (to be measured by TREAT Operations), boron content in the TREAT driver fuel, extent of graphitization in the TREAT fuel blocks, and the specific heat of the TREAT graphite. Strategies to measure or estimate these properties are being developed.

Additional benchmark cases are being developed under NE-4 sponsored projects to support complex experiment analysis. Tom Downar and Bill Martin (University of Michigan) are working within an NE IRP (led by Oregon State University) to develop benchmark cases for the minimum critical, M8CAL, and a third core to be determined. The UM team is using PARCS to assess sensitivity and uncertainty in the measured parameters. Ayman Hawari (North Carolina State University) is working within an NE NEUP to develop similar benchmark cases for the M2 and M3 experiments. The NCSU team is using SERPENT for sensitivity and uncertainty calculations.

Progress in the development of modeling tools to perform coupled kinetics calculations was presented by Mark DeHart. This team is working to use MAMMOTH to couple nuclear time-dependent neutron transport (RattleSnake) in the TREAT core with thermal mechanical behavior (BISON) of the fuel elements. This tool will allow for prediction of the full time dependent response of TREAT and its experiments to transient tests. A simplified model of TREAT was constructed from an infinite lattice of fuel elements to explore general physical response of the system. The analysis showed the appropriate qualitative pulse response as well as the fuel element temperature and flux distribution. However, difficulty was encountered due to the streaming effects of the coolant channels located at each elements corner. These openings allow for axial streaming of neutrons that required a modification to the computational methods used. This feature is expected to be even more significant in treatment of the hodoscope slot.

A MAMMOTH model of the minimum critical core was developed by Tony Alberti (Oregon State University). This was followed by 'small core' configuration that allowed for simulation of TREAT transient #15. Initial results suggest excellent agreement after analysis/reduction of the available historic measured reactor power data. MAMMOTH predicts as wide array of additional reactor parameters that were not historically measured that could be used for more comprehensive validation in the future.

M8CAL analysis is underway and poses some new challenges. Most notably, the presence of the hodoscope slot, multiple control rod types, complex geometry of the test section, and the use of dysprosium flux shaping collars in the test. Currently, the steady state calculations and measurements don't match and must be resolved prior to meaningful transient analysis.

Developers plan to follow analysis of these tests with work on the Multi-SERTTA device to be used in the ATF-3 campaign. Exchange of device descriptions and modeling needs was initiated between device designer (Nick Woolstehnulme) and MAMMOTH analysts (Javier Ortensi).

5. TOPIC 4 – FUEL MOTION MONITORING SYSTEM RECOVERY

The status of hodoscope recovery at both CABRI and TREAT was presented (see attached).

David Chichester described TREAT hodoscope recovery goals and specific activities completed to date. Although the TREAT hodoscope includes two detector banks (proton recoil scintillator detectors and proton recoil proportional counters), only one set is currently being addressed to support startup operations. The scintillators were selected for initial use due to their relative simplicity, to mitigate technical risks. All of the scintillator detectors were extracted from the hodocopse, an evaluation process was developed and implemented, and a refurbishment technique was developed. 99 of 327 detectors are considered candidates for refurbishment and 20 have been refurbished to date. The photomultiplier tubes attached to the scintillator were found to have substantially degraded and must be replaced. Candidates were tested and a preferred commercial product was selected. A data acquisition system (DAS) was designed and a prototype constructed. The 16-channel system will be replicated as many times as necessary to support deployment of a limited view system (64-96 channels) in the near term and eventually to support the full system. Advanced DAS capabilities may result in collection of neutron and gamma ray data. This could substantially enhance data gathered from the hodoscope during future test. Further analysis and testing is required.

The hodoscope was suggested as a tool that could be used to support physics testing. The device provides unique time and space-dependent fast neutron flux distribution data that is being predicted using the advanced codes. The device could be enhanced with additional detectors to simultaneously collect thermal neutron data.

Bruno Biard presented the status of the CABRI hodoscope. The device consists of 51 rows and 3 columns of collimated neutron impinging on 153 fission chambers (Np-237) and 153 proton recoil counters (methane). The system is capable of collecting data every 1 ms. The fission chambers are designed to provide measurements during full-power transient mode and the proton recoil counters provide measurements at low power. A few operating concerns were identified. Some electronic components may not be readily available and there are only a limited number of spares. Also, the system had to be moved during the facility refurbishment and confirmation of alignment is a critical first step in commissioning.

The process for calibrating and converting the hodoscope signal to a local mass distribution was described. The background neutron signal was processed during first critical tests and showed excellent agreement with the power profile measured via dosimetry.

A follow-up workshop on hodoscope technology to be hosted by IRSN in Cadarache in October was proposed.

6. CONCLUSION

6.1 Highlights and Action Items

- Significant synergy exists between the TREAT and CABRI programs. Formal collaboration in the future is empowered by bilateral agreements between DOE/CEA and DOE/IRSN.
- TREAT startup physics lead (Jim Parry) should engage CABRI commissioning lead (Jean-Pascal Hudelot) to discuss methodologies used and lessons learned during early physics testing at CABRI.
- TREAT reactor engineering should be formally tasked with reviewing all TREAT core models being developed, act as a hub for storing and distributing the models to users, and provide revision control functions to the validated versions of these models. Active participation of TREAT engineering in advanced code development is also recommended to accelerate implementation.
- Advanced codes should be used to explore time dependent reactor behavior (spectrum shifts in particular) that currently cannot be described using point kinetics.
- A strategy to reduce uncertainty in key TREAT material properties (boron content, graphitization, and specific heat) should be developed for consideration.

- Integration of the MAMMOTH team with the ATF-3 Experiments team should be expanded to provide ‘qualitative’ analysis of phenomena of interest that cannot be determined using existing codes (to later become ‘quantitative’ after validation of the codes during physics testing).
- Expanded technical exchange between CABRI and TREAT experts in the areas of experiment design, safety basis development, instrumentation, irradiation test device, modeling and simulation, and fuel motion monitoring would substantially improve the programs in both countries.

An expanded workshop to be hosted by CEA and IRSN is proposed for October 2016.

Appendix A

Workshop Agenda

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Transient Test Reactor Physics Workshop
Thursday, May 5, 2016

Objective: Conduct technical exchange on transient test reactor physics and modeling techniques used to support TREAT (INL) and CABRI (CEA/IRSN). The workshop will be conducted in conjunction with the PHYSOR conference being held in Sun Valley, Idaho, USA from May 1-5.

Attire: Business Casual

Host:	Dan Wachs, 526-6393	Revision Number 1
Meeting Coordinator:	Jeni Baker, 526-6624	April 25, 2016

Transient Test Reactor Physics Workshop
Thursday, May 5, 2016

Sun Valley Lodge, Larkspur Room, Sun Valley, Idaho, USA

08:00 Brief overview of TREAT and CABRI Reactors .... Dan Wachs/Bruno Biard

09:30 BREAK .... All

09:45 TREAT physics modeling and startup testing plan .... Jim Parry
TREAT Reactor Chief Scientist, (208) 533-8872

10:30 CABRI physics modeling and startup testing results and plans........ Jean-Pascal Hudelot
CEA

11:30 IRHP Benchmark process and results for TREAT. .... John Bess (Tom Downar)
Reactor Physics Design & Analysis, (208) 526-4375

12:00 Working Lunch: Discussion of dosimetry and radiation measurement techniques. .... All

1:00 Advanced modeling and simulation for TREAT. .... Mark DeHart
Reactor Physics Design & Analysis, (208) 526-1279

2:00 TREAT Hodoscope recovery and performance .... David Chichester
Global Security/International Safeguards, (208) 526-8920

2:45 BREAK. .... All

3:00 CABRI Hodoscope recovery and initial testing results.................................... Bruno Biard
IRSN

3:45 Discussion of development and validation of advanced analysis methods.................... All

5:00 Adjourn for Dinner. .... All

INL Facility tours (including TREAT) could be accommodated on Friday May 6 upon request.

The Past, Present, and Future of Transient Testing in Idaho

PHYSOR-2016
Lunchtime Talk, May 4, 2016

N.E. Woolstenhulme

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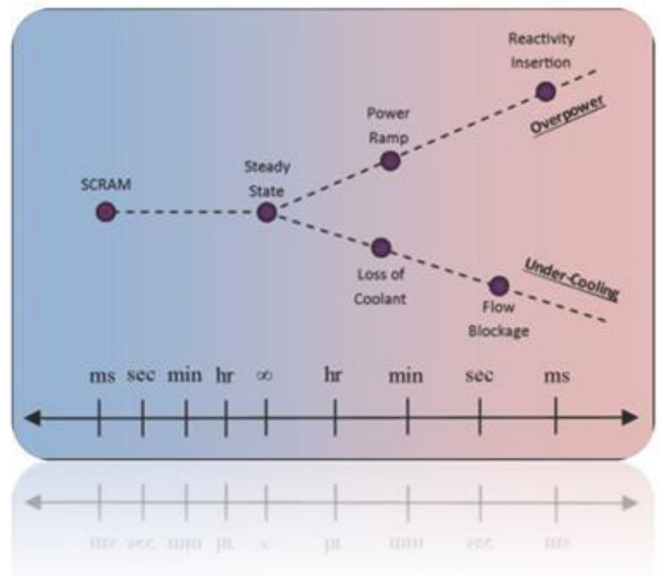
What is Transient Testing?

- Transient testing is like car crash testing for nuclear fuel
  - Demonstrate performance phenomena and limits for fuel development and reactor design
  - Show consequences of hypothetical accidents for licensing

- Transient testing is the study of fuel and fuel system behavior under <u>power-cooling mismatch</u> conditions
  - Slower event can be simulated out-of-pile or in steady state test reactors
  - Shorter events needed to be simulated with rapid nuclear heating



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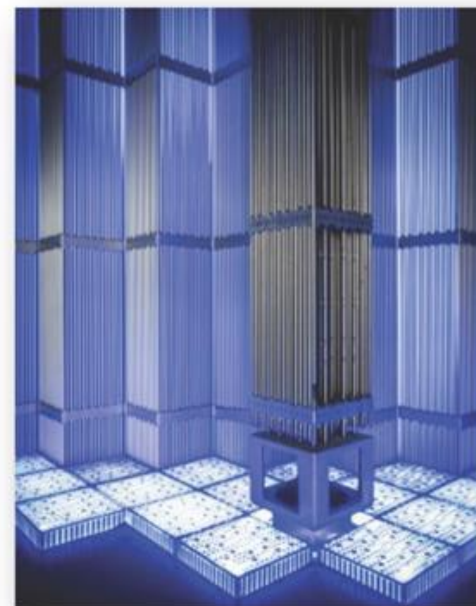
What is Nuclear Transient Testing?

- Nuclear transient testing is using fission heating, in part or in whole, to simulate power-cooling mismatch scenarios
- Electrically-heated non-nuclear transients tests are useful, but innately limited:
  - Lack of irradiation effects
  - Very difficult to achieve heating rates simulating rapid reactivity insertion accidents (RIA)
  - Heating "from the inside out" temperature profiles require fission heating
- Nuclear transient testing requires a transient test reactor



https://www.industryforum.co.uk/resources/articles/meeting-the-civil-nuclear-supply-chain-skills-challenge/

3

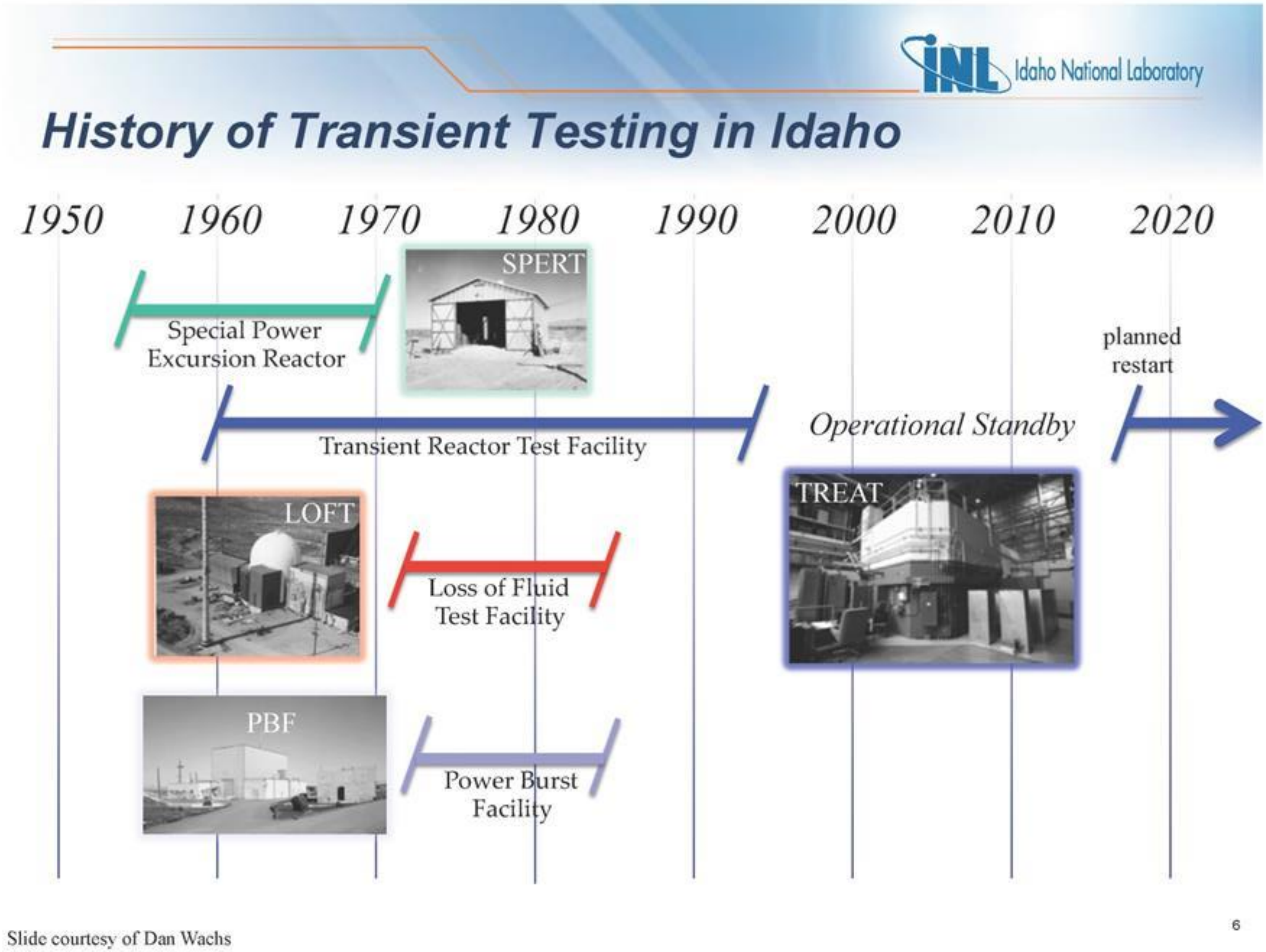
What is a Transient Test Reactor?

- Transient Test Reactors have special design features to enable accident simulation, for example:
  - Ability to safely insert large amount of reactivity:
    - Fast-acting transient control rods
    - Driver core tolerant of energy excursions
    - Strong negative temperature feedback (self limiting)
  - Ability to depressurize
    - Fast acting blowdown valves

Power Burst Facility

- Most of the time transient test reactors provide these conditions to an experiment position in the core
  - But sometimes the driver core itself was the subject of the test!
- The national reactor testing station (now INL) hub for nuclear transient testing

The Past



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History of Transient Testing in Idaho
1950
1960
1970
1980
1990
2000
2010
2020
SPERT
Special Power
Excursion Reactor
planned
restart
Transient Reactor Test Facility
Operational Standby
LOFT
TREAT
Loss of Fluid
Test Facility
PBF
Power Burst
Facility
Slide courtesy of Dan Wachs
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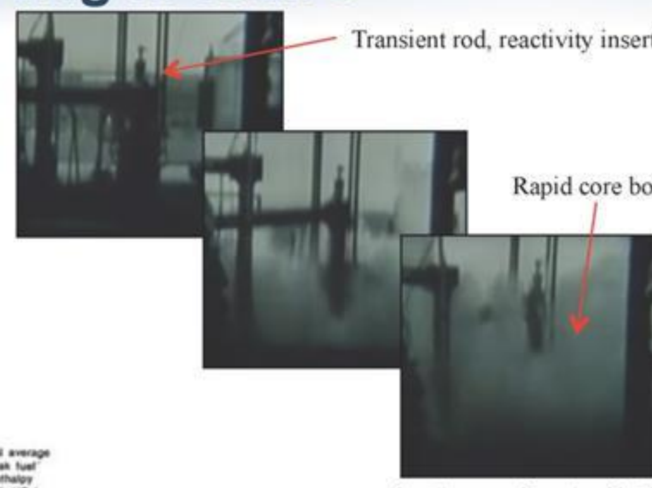
History of Transient Testing in Idaho

- The Special Power Excursion Reactor Test (SPERT) facilities was constructed in the 1950's
  - Actually several different core configurations, one of which was tested "destructively" under RIA conditions



- SPERT Capsule Driver Core: Tested Light Water Reactor (LWR) fuels in water-filled capsules under RIA pulse-type transients
  - Set the stage for future PWR transient testing, the crux is maintaining rod-like geometry
- SPERT now decommissioned

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History of Transient Testing in Idaho

- The Transient Reactor Test (TREAT) facility
  - Contemporary with SPERT, but primarily supported transient testing for sodium-cooled fast reactors
    - Also supported LWR and other reactor systems
  - Constructed in late 1950's, performed nearly 3000 transients
  - Placed in operational standby in 1994
  - More about TREAT later…

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History of Transient Testing in Idaho

- The Loss of Fluid Test (LOFT) facility was a small PWR designed to test plant system response to Loss Of Coolant Accident (LOCA)
  - Fast acting valves simulated break of primary piping
  - Instrumental in validating computational codes and PWR licensing process
  - Constructed in the 1970's, now decommissioned

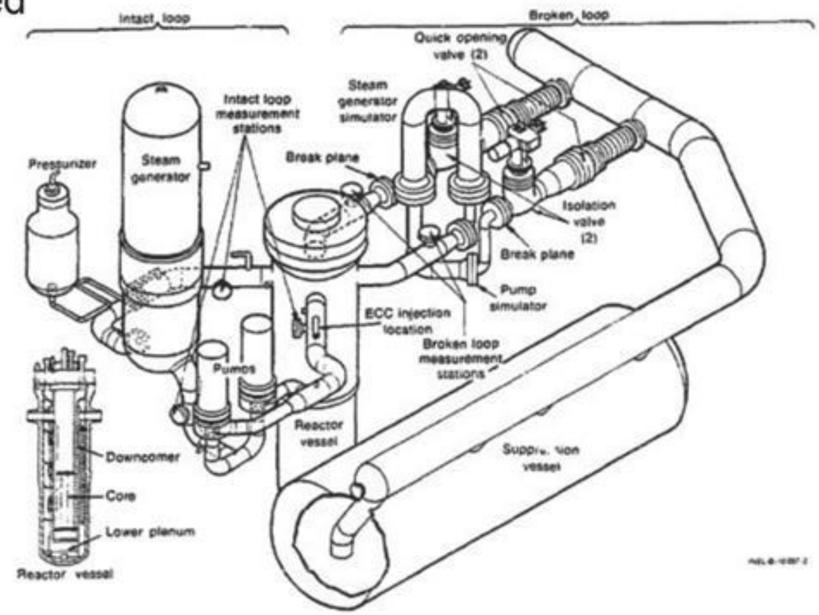

Fig. 1. - LOFT major components in cold leg break configuration

D.L. Reeder and V.T. Barta, "The Loss-of-Fluid (LOFT) Facility", EG&G Idaho, CONF-790803--13.

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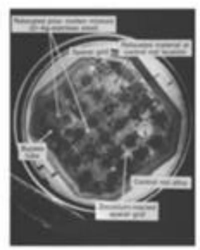
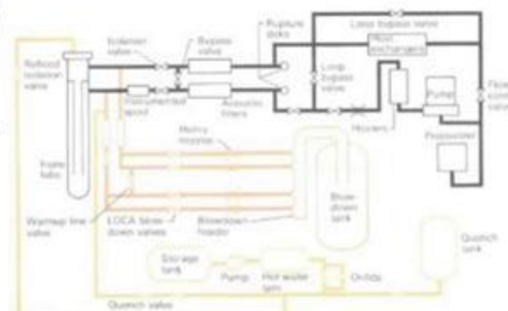
History of Transient Testing in Idaho

- The Power Burst Facility (PBF) facility was constructed in the 1970's
  - Contemporary to LOFT, but was designed to drive a central experiment position
  - Massive in-pile-tube, elaborate loop-system, and transient rods enabled sub-assembly testing of RIA and LOCA
  - Blowdown, reflood, and fission product transport measurements
- Post TMI: PBF tested 32-rod pre-irradiated PWR bundles in severe fuel damage tests
  - One of the most tremendous transient tests series in history
- Facility now decommissioned





Fun Trivia: PBF's driver fuel was actually tested in TREAT to demonstrate resilience to power pulses

W.A. Spencer, A.M. Jensen, R.K. McCardell, "Capabilities of the Power Burst Facility", EGG-M-06582, International Topical Meeting on Irradiation Technology, Sep 28 – Oct 1, 1982.
F.K. Clements and E. Feinauer, "Analysis of Safety Considerations for Transient Testing of PBF Prototype Fuel Rods in TREAT", IDO-17056, Apr 1964.
David A. Petti, Zoel R. Martinson, Richard R. Hobbins, and Daniel J. Osetek, "Results from the Power Burst Facility Sever Fuel Damage Test 1-4: A Simulated Severe Fuel Damage Accident with Irradiation Fuel Rods and Control Rods", September 12, 1990, Nuclear Reactor Safety.

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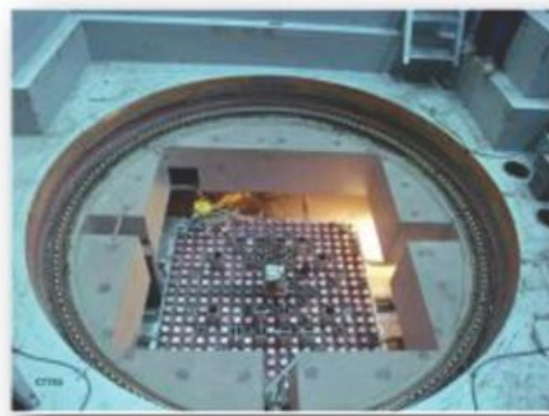
History of TREAT

- But TREAT outlived them all, why?
  - Contemporary to and collocated with the sodium-cooled fast reactor EBR-II
  - Water-free core design likely selected to simplify "what-if" scenarios for sodium-bearing tests
- Reduction in US fast reactor funding → TREAT went into operational standby in 1994
  - Dry and simple facility, little effort needed to maintain
  - So it sat hibernating for 20+ years
  - But more on that later….

Water-Sodium Reaction



Top view of TREAT core, rare picture with shield plug removed

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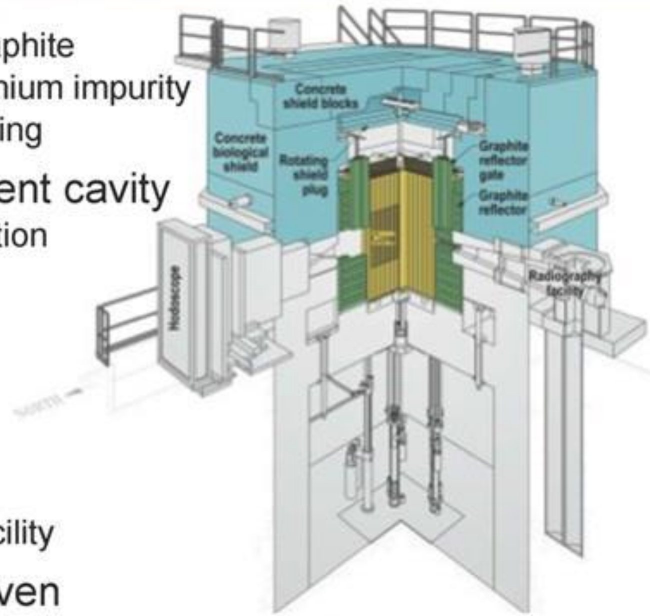
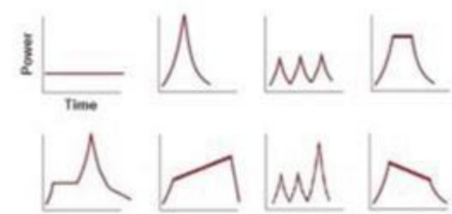
Introduction to TREAT

- TREAT core design:
  - Zircaloy-canned blocks of urania dispersed in graphite
  - Core is effectively a giant graphite block with uranium impurity
  - Strong negative temperature coefficient, self-limiting
- Displace fuel assemblies to create experiment cavity
  - Each fuel assembly is 10cm × 10cm in cross section
  - 1.2m of active core length
- Air cooling system
  - 100kW steady state
  - Not a required safety system
- 4 slots with view of core center, 2 in use
  - Fast neutron hodoscope, neutron radiography facility
- Fast-moving transient rods hydraulically driven
  - Allows for precise and flexible transient shaping
  - 2500MJ max core energy in prompt burst (<1 sec)
  - 2900MJ max core energy in shaped mode (up to ~5 min)
  - And practically anything in between





Example Transient Shapes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Arial View of TREAT
• 100 kW Steady-state power with 19 GW Peak Transient Power
• Core: ~1.2 m high x 2 m. dia.; surrounded by 0.25 m graphite reflector
• 19 x 19 array of 10 x 10-cm. fuel and reflector assemblies
• 12 steady-state and 8 transient control rods
• Immediate, large negative temperature coefficient
• 6604 reactor startups, 2884 transient irradiations
Fun Trivia: Some of TREAT's reflector graphite came from CP-1, the world's first nuclear reactor!
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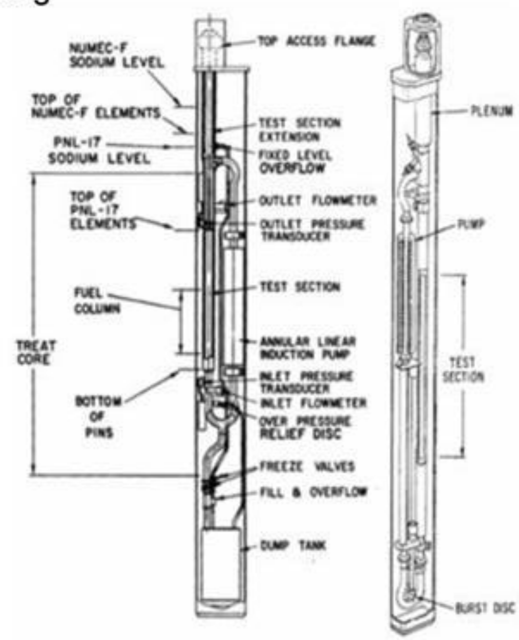
TREAT Experiment History

- TREAT is well suited to self-contained drop-in test devices
  - Installation, testing, and withdrawal in a matter of days
  - Enables support for different-environment test devices (e.g. water or sodium)
  - Assembly and disassembly in shielded hot cells
  - Device fits within shielded handling casks
    - Loop handling cask 25cm diameter X 387cm long

- TREAT's historic testing focused on sodium-cooled fast breeder reactor specimens
  - Highly successful with package-type loops and capsules
  - Robust piping primary containment, sheet metal leak-tight secondary enclosure
  - Pumps, heater, instrumentation, etc. all contained within enclosure
  - No contaminated coolant plumbing outside of shielding
  - Approach greatly facilitates testing of pre-irradiated fuel specimens



Historic Mk-series Sodium Loop

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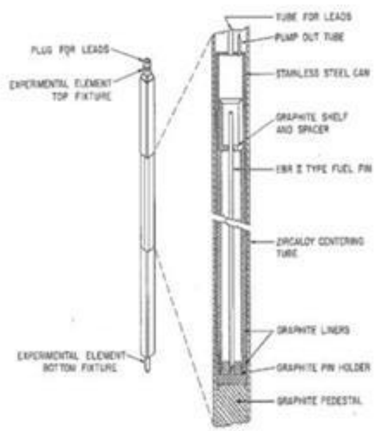
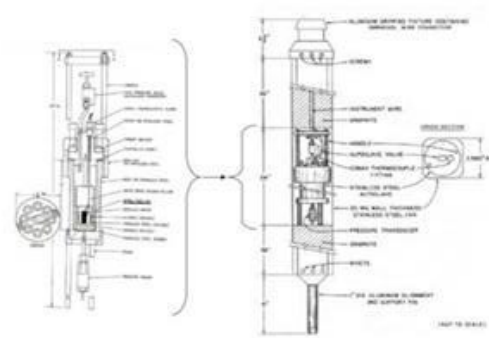
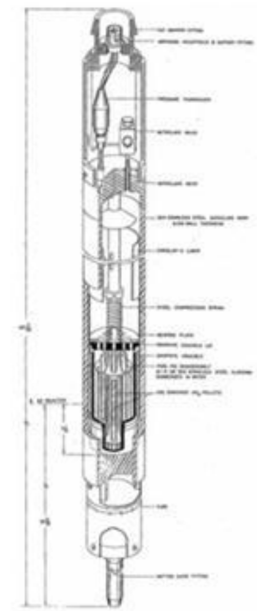
Static Capsule Testing

- TREAT has a rich history of transient testing in static capsules
  - Fast-reactor fuels, both dry and in sodium
  - PWR and research reactor fuels in water
  - Space nuclear propulsion fuels dry and in seawater







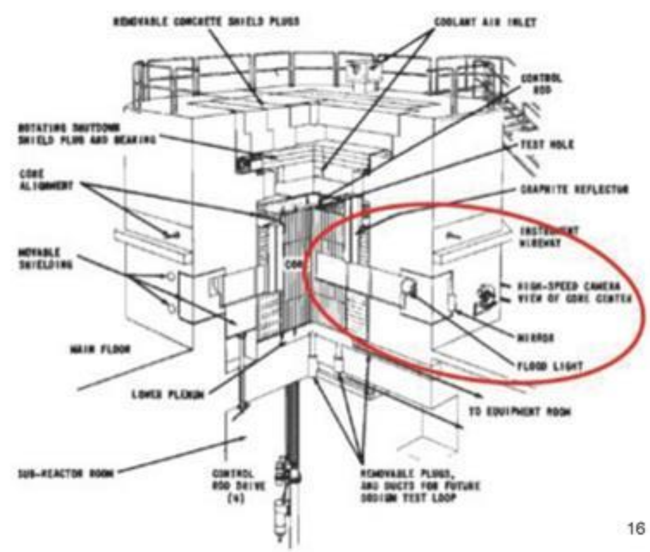
- Almost every geometry imaginable
  - Rods, pins, bare pellets, plates, extrusions, bundles, clusters
  - Fresh and pre-irradiated

N.E. Woolstenhulme and J.D. Wiest, "ATF Transient Testing Pre-Conceptual Design and Engineering Considerations Summary", August 2013, INL/EXT-13-29898

15

Fuel Visualization and Motion Monitoring

- TREAT's through-the-side access slots have been used to effectively watch the fuel in various ways
  - High-speed videography through transparent capsule with quartz windows (example videos on next slide)
    - Limited in providing pressurized water environments
    - Not terribly useful for testing in opaque sodium
    - But very useful in visualization basic phenomena
  - High-speed film-based camera (1960's)
  - Flood lamp and periscope
- Function later replaced by fast neutron hodoscope



Videos of Historic Transient Tests

Idaho National Laboratory

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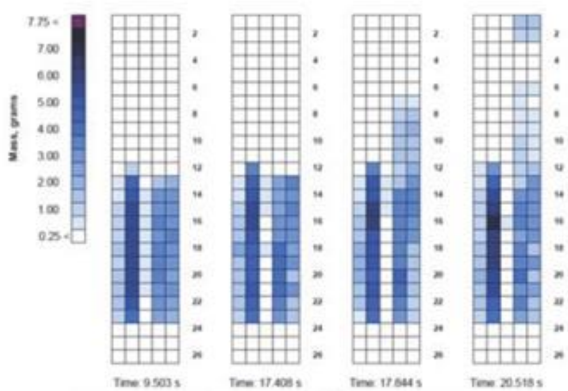
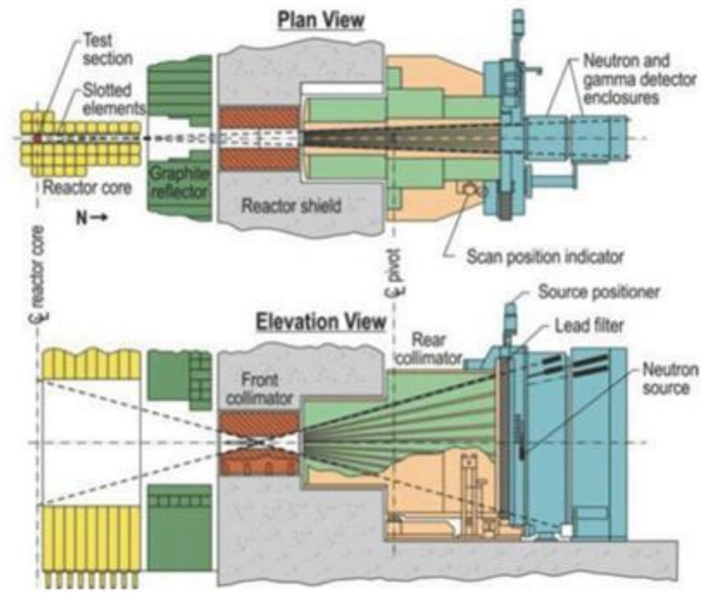
Fuel Visualization and Motion Monitoring

- Fast neutron hodoscope later became the key capability for monitoring fuel motion during the transient
- Fission-born fast neutrons emitted from specimen travel through vehicle's containment wall, through a collimator, and into detector array
- Provides pixelated view of fuel mass in each collimator slot



Snap-shot views of data from a Hodoscope experiment

This data shows the simultaneous response of two fuel pins to a transient. The pin on the right shows significant axial fuel relocation has occurred at 17.844 seconds. This observation establishes the failure point and the progression of fuel movement after the breach.



Slide Courtesy of David Chichester

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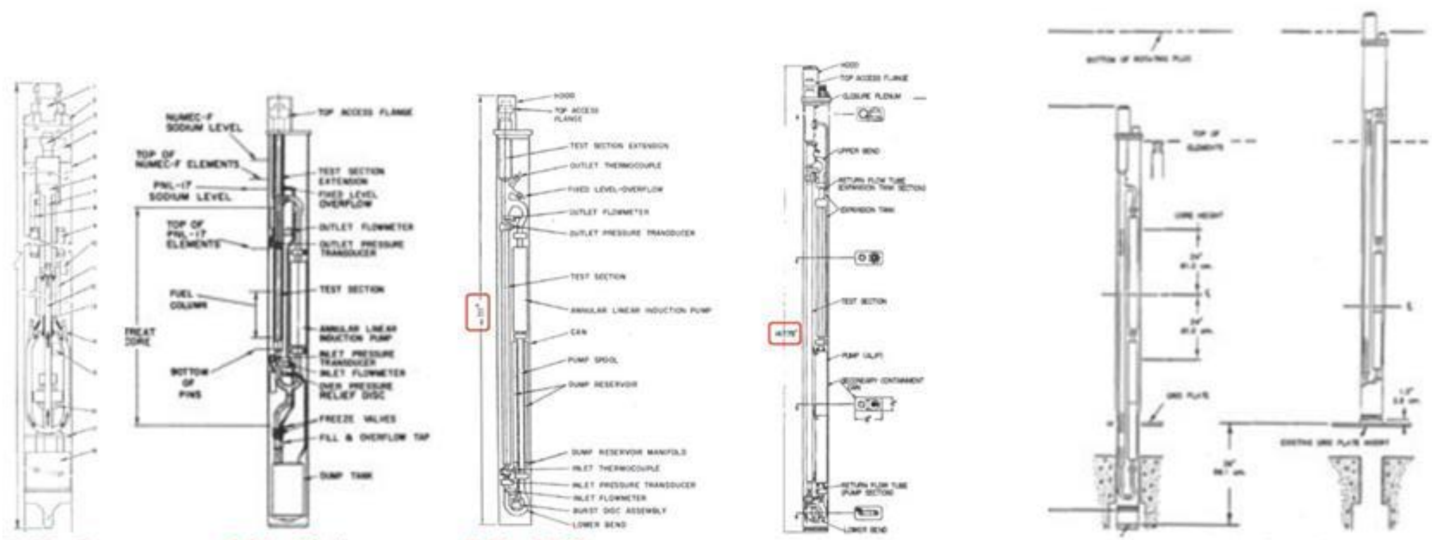
Mk-Series Loops

- Flagship fast reactor transient tests (1970's-1980's) occurred in Mk-series sodium loops
  - Very modular, could support test trains with 1, 2, 3, or 7 pins
  - 1 or 2 induction pumps depending on flowrate needed
  - Expansion tanks for additional pressure safety
  - Different axial configurations for upper or lower plenum pin designs



Mk-I Mk-IIA Mk-IIC Mk-III Axial Options

The Present
(from now up until TREAT restarts)

TREAT's Current Status

- DOE's accident tolerant fuels (ATF) program and other needs
  - Impetus for resuming transient testing in the US
  - TREAT selected, project is underway
- Other supporting infrastructure being revived
  - Hodoscope refurbishment
  - Hot cell equipment
  - Shielded handling casks

Resumption of Transient Testing Director John Bumgardner with TREAT in background, Published in Local Newspaper, Post Register "Bringing a nuclear test reactor back to life at INL", October 3, 2014.

Senator Ben Cardin and **Ernest Moniz** discuss the Transient Reactor Test Facility, commonly referred to as TREAT, with staff from the Energy Department's Office of Nuclear Energy at @COP21. | Energy Department photo by Matt Dozier.

Idaho National Laboratory's Transient Reactor Test Facility, commonly referred to as TREAT, is an air-cooled, thermal spectrum test facility designed to evaluate reactor fuels and structural materials. This 3D-printed model of the reactor shows a cutaway of its inner workings. | Energy Department photo by Matt Dozier.

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TREAT Restart Status

- Fuel Evaluations Demonstrate Acceptability for Continued Use
  - Fuel assemblies inspected, some by removal from core, some by boroscope in-situ
  - Completed installation of 16 poison assemblies allowing for subcritical operations with removal of all control rods.
- Control Rod Drives Acceptable for Continued Use
  - Successfully refurbished existing drives (i.e. gears, hydraulics, snubbers, etc.)
  - Completed functional and SCRAM testing of all rod drives

Images courtesy of Resumption of Transient Testing Team

Transient Rod Video

- Video of one transient rod pair moving, 8 total rods exist

Idaho National Laboratory

TREAT Restart Status

- Reactor control system testing to date indicates replacement not required
- Facility was left in remarkably good condition in 1994 and facility systems consistently maintained
- Current evaluations have affirmed functional plant system's conditions
- Updated Safety Basis To Current Requirements
  - Updated Safety Analysis Report (SAR) submitted to DOE for review
  - No issues anticipated with regulatory authorization to operate TREAT
- So what's left?
  - Primarily operator training
  - SAR review and approval
  - On schedule for operation in 2018, and maybe even sooner!

Images courtesy of Resumption of Transient Testing Team

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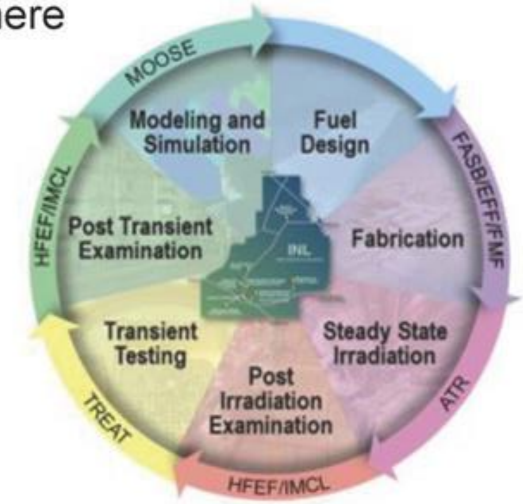
No Test Reactor is an Island

- TREAT might look like its in the middle of nowhere
- But its actually right by much of what it needs

Advanced Test Reactor



Fun Trivia: The Arco desert is actually green for a few weeks in spring

Materials and Fuels Complex
Hot Cells
Fuel Fabrication
Advanced Characterization

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Experiment Design Status

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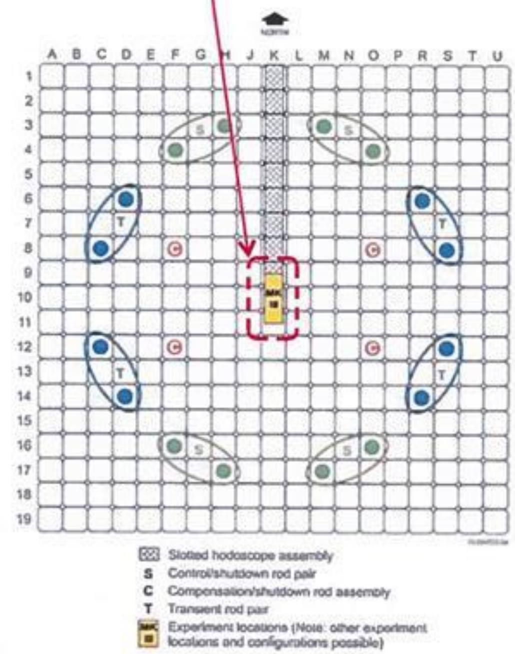
- TREAT is a brilliantly basic machine
  - But all it really does is provide neutrons
- The experiment vehicle (e.g. loop, capsule, etc.) does the other half of the work
  - Boundary conditions (heat transfer, coolant environment)
  - Instrumentation
- ATF transient tests likely to be the first transient tests
  - Support for congressional mandate to insert lead test rods in a commercial PWR
  - TREAT spent the last two decades of its prior operation (~1970-1990) largely supporting fast reactor tests
  - Transient testing experiment team developing pressurized water test capabilities for TREAT
- Revitalization of sodium-environment irradiation vehicles underway
- Development of vehicles for "science-based" specimens also underway

Insert Experiment Here
(or anywhere else really)



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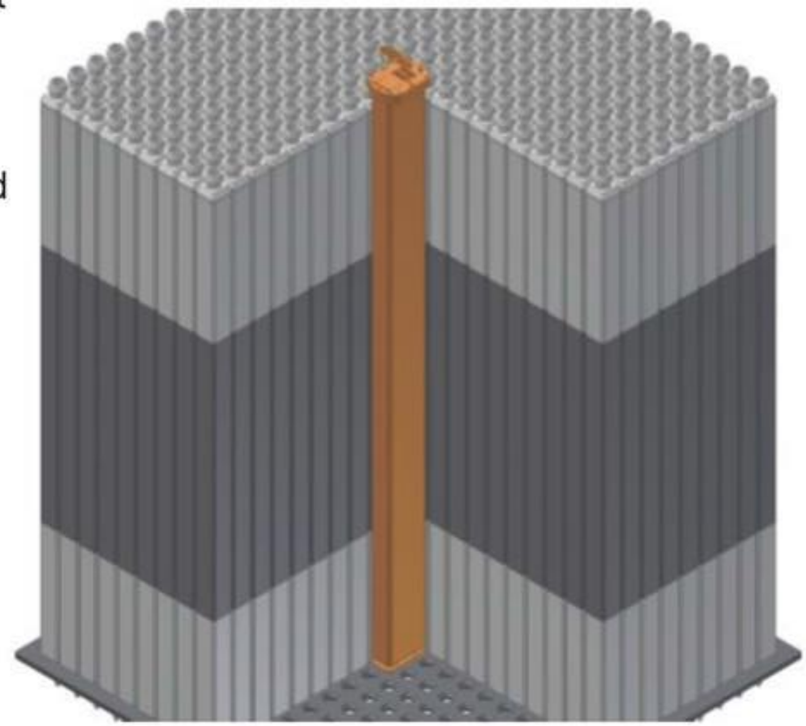
Static Environment Vehicles

Static Environment Rodlet Transient Test Apparatus (SERTTA)

- General purpose devices <u>without</u> forced convection
- Pre-pressurized and electrically heated
  - Liquid water up to PWR condition (320C 16 MPa)
  - Inert gas or steam
  - Liquid sodium
- Vessels designed with tremendous safety margin
  - Nickel-based superalloy UNS N07718 enables thin vessel wall to minimize neutron absorption
- Two SERTTA's under development
  - 4X capsule "Multi-SERTTA"
  - 1X capsule "Super-SERTTA"



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

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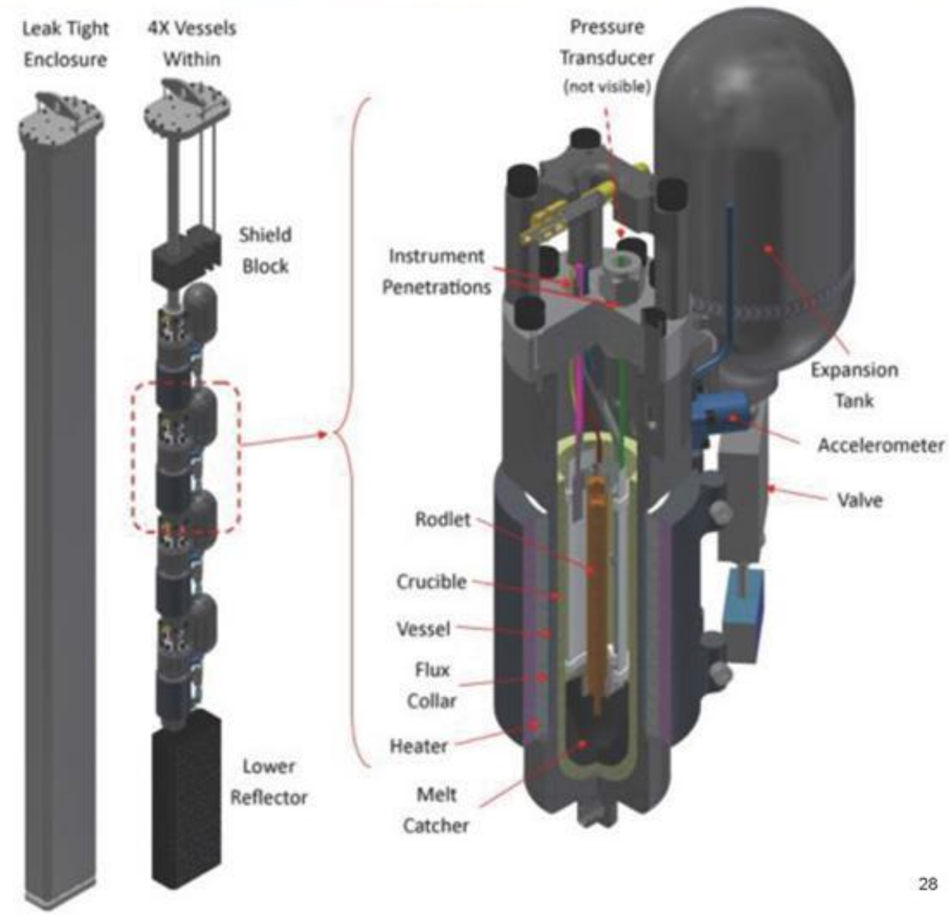
Multi-SERTTA

- Best for smaller scale specimens and four-for-one testing (concept screening)
- Planned to be the first "new" test to be used in restarted TREAT



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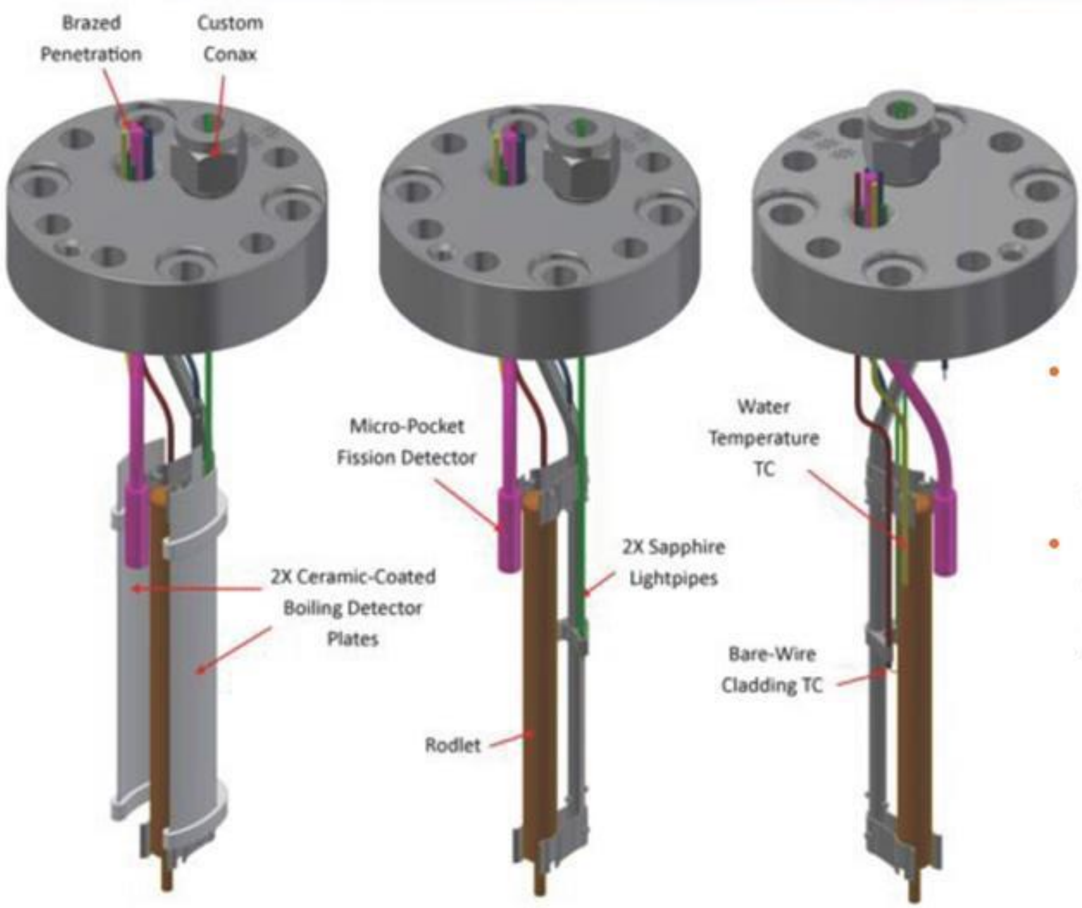
Multi-SERTTA

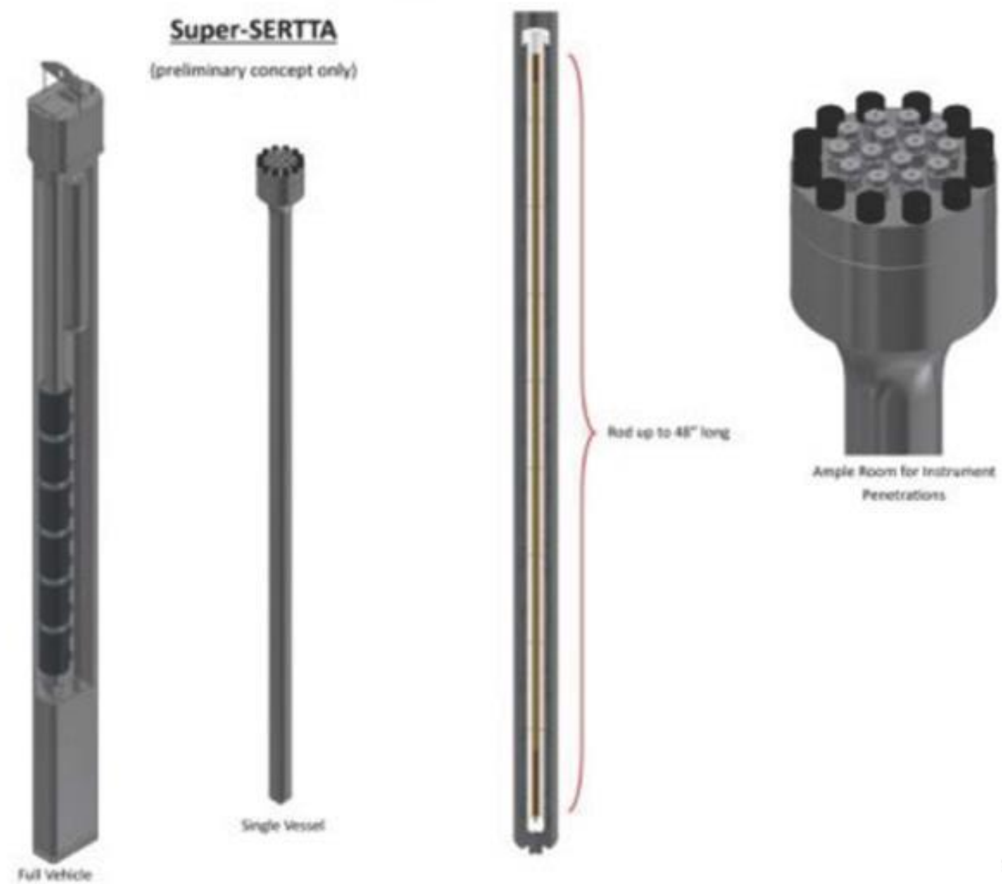


- Despite geometry limitations, has an impressive instrument array
- Modular, adaptable for other missions (version shown here is for PWR rodlets)

The Future

Super-SERTTA

- For larger specimens and/or bundles
- Higher energy capacity
- More geometry available for instrumentation



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Flowing-Water Loop

- But static water will only get you so far
- Forced convection need to simulate LWR conditions (boiling response, etc.)
- Developing the TREAT Water Environment Recirculating Loop (TWERL)
- Based on MK-series concept
- Test train is modular:
  - One rod in a flow tube for highly instrumented test trains
  - Up to three rods in individual flow tubes for concurrent testing
  - Four-rod bundle Test-specific instrument designs

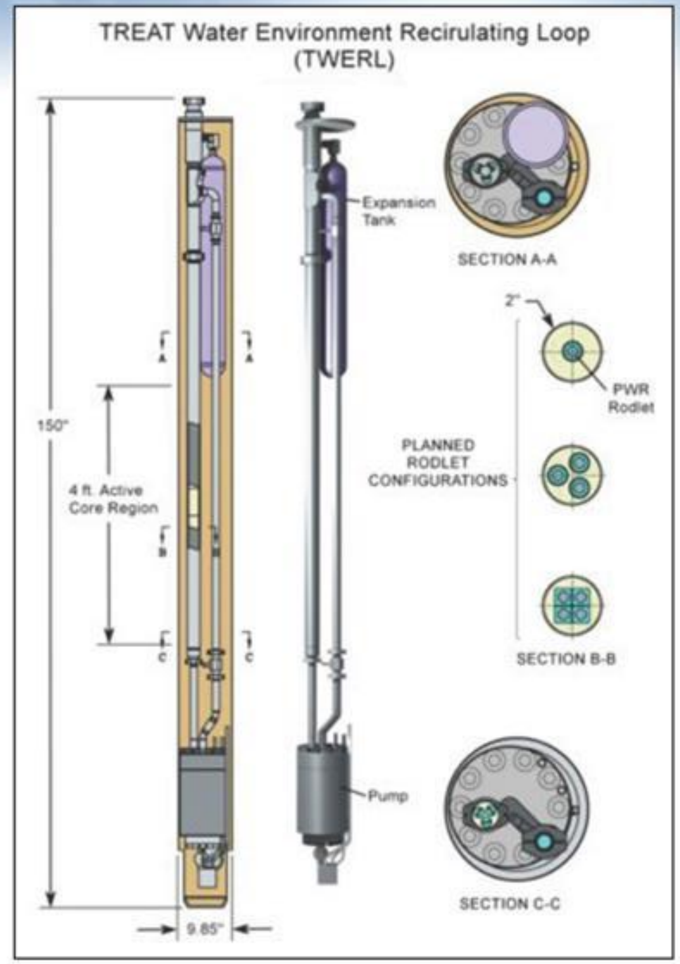

Image courtesy of Greg Housley

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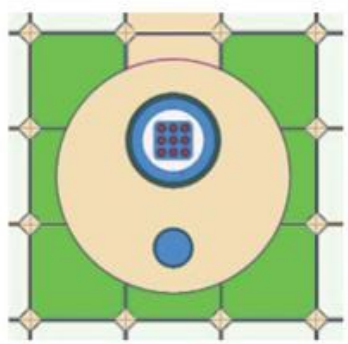
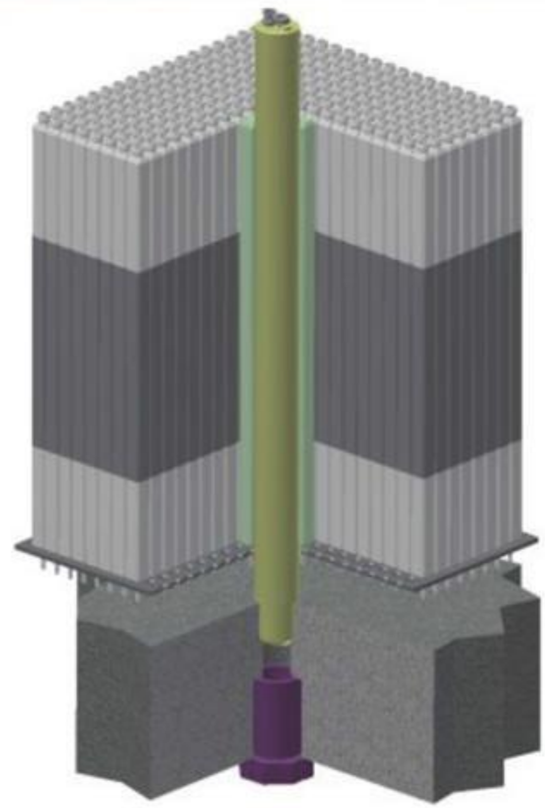
TWERL

- Larger cylindrical footprint in core
- Fits within existing shielded casks
- Further TWERL modules and evolutions envisioned
  - Blowdown valve and tank for LOCA simulation
  - 9-rod bundle "Super-TWERL" (nuclear analysis shows TREAT is capable)



MCNP Rendering of 9-rod "Super-TWERL"
Image courtesy of Connie Hill



TWERL shown in TREAT ¾ section view
Image courtesy of Greg Housley

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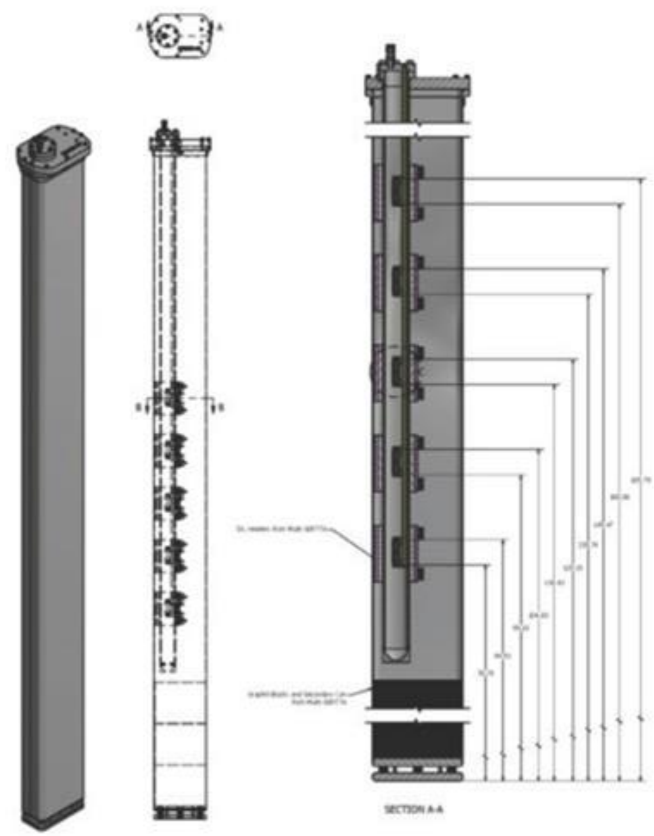
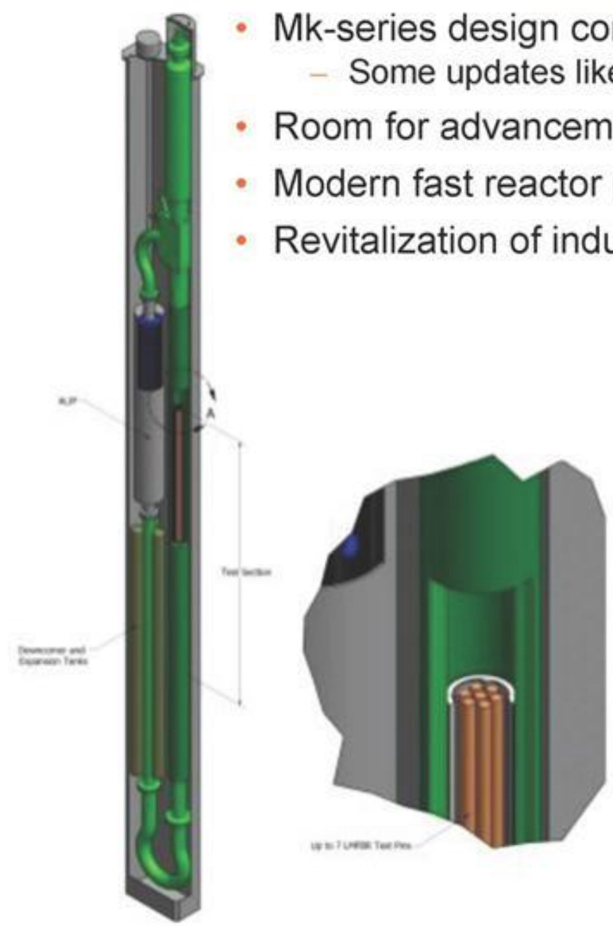
MARCH Vehicle

- Vehicle which enables small specimens to be irradiated at TREAT, extracted, and shipped for exams with little to no shielding
- Dubbed the Minimal Activation Retrievable Capsule Holder (MARCH)
  - Capability akin to hydraulic shuttle, (aka "rabbit"), but without the plumbing
  - Multiple small samples (fueled or unfueled) in low-activation capsules
  - Capsule-specific temperature control (heaters) and monitoring (thermocouples)
  - Small sample size greatly facilitates experiment safety analysis → the result is cheap and easy experiments
- Designed firstly for an LDRD that compares irradiation-induced microstructure changes to lower-length-scale performance models (MARMOT)
  - Many similar tests expected to follow
  - One could say it's designed to "Unify Theory and Experiments in the 21st Century"!



Mk-IV Sodium Loop

- Mk-series design concept is well-established
  - Some updates likely needed → “MK-IV” sodium loop
- Room for advancement – materials, instrumentation
- Modern fast reactor program needs to be incorporated
- Revitalization of induction pump capability



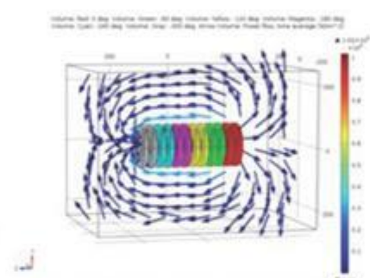
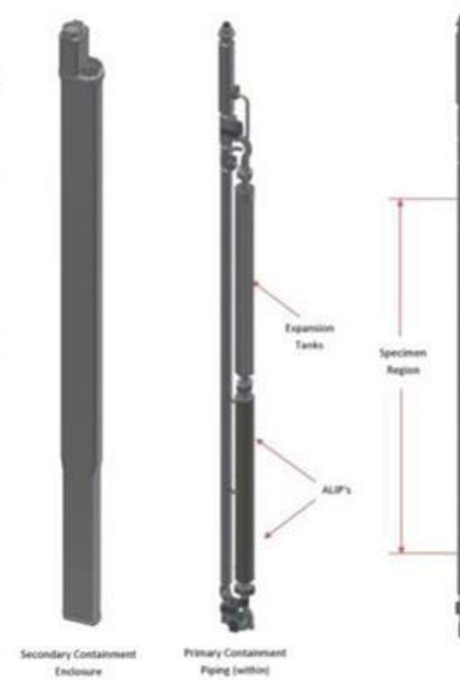


Figure 10. Vector simulation of 12 solenoids in a 2-pole electromagnetic pump.

Carlos O. Maidana and Juha E. Neiminen, “Multiphysics Analysis of Liquid Metal Annular Linear Induction Pumps: A Project Overview”, Proceedings of NETS2016 meeting.



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The Future

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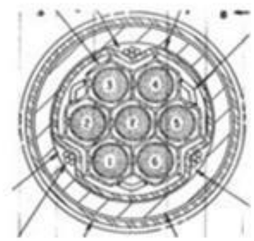
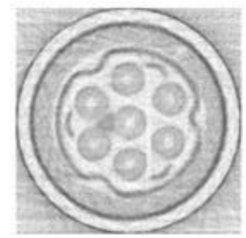
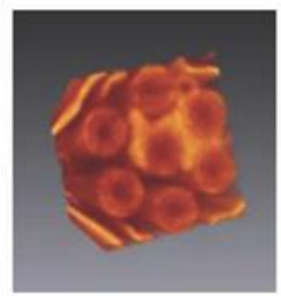
- 20 yrs of computational advances will set TREAT's future apart from its past:
  - Multi-physics modelling of experiments (reactor, fuel performance)
  - Advanced post-transient exams (3D computed tomography)
- The future of transient testing "in Idaho" will reach far beyond INL's border both domestic and abroad
  - Nuclear Science User Facilities (NSUF)
  - Industrial access through GAIN
  - Multiple university collaborations already, no doubt more to come
    - Instrument development, advanced hodoscope sensors, IRP led by UW Madison
    - Core/loop benchmarking, IRP led by OSU
  - Collaboration on in-pile advanced sensor development
  - International collaboration with other transient test reactors:
    - NSRR (Japan), CABRI (France), IGR (Kazakhstan)



Mk-series 7-pin test



Historic Neutron Tomograph



Modern 3D Reconstruction

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Thank you for your attention

DE LA RECHERCHE À L'INDUSTRIE

cea den

BRIEF OVERVIEW OF CABRI REACTOR

Jean-Pascal Hudelot
jean-pascal.hudelot@cea.fr
Nuclear Energy Division,
CEA Cadarache, France

The CABRI Past Programs

- **Common Objectives of the CABRI Programs**
  - Study of the fuel rod behavior under Reactivity Initiated Accident (RIA) Conditions

- **Fast Breeder Reactor Programs from 1978 to 2001**
  - 4 programs: CABRI1, CABRI2, FAST and RAFT
    - 59 experiments
    - Superphenix and Phenix Fuel Pin Types
    - in Sodium experimental loop

- **Pressurized Light Water Program: from 1993 to 2000**
  - REP-Na Program
    - 12 experiments
    - UO_2 and MOX Fuel Rods (Burn-up up to 65GWd/tU)
    - in Sodium experimental loop

The CABRI International Program (CIP)

Goal

- Study of PWR fuel rod behavior under Reactivity Initiated Accident
  - In prototypical irradiation conditions (pressurized water)
  - Post-DNB and post-failure fuel behavior
  - Filling gaps and enlarging the validation domain

Content

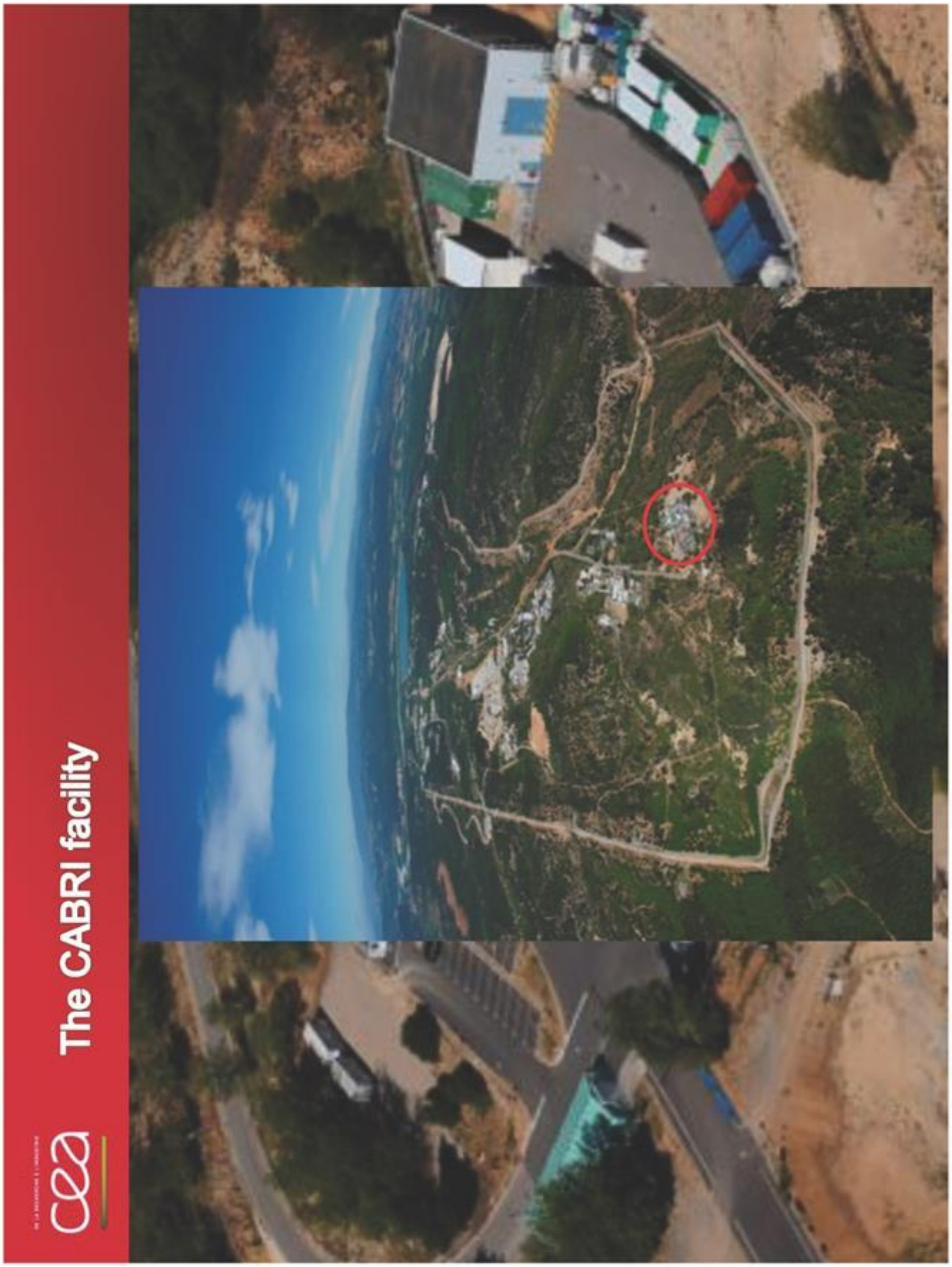
- UO_2 and MOX Fuel Rods (Burn-up up to $100GWd/t_U$) with:
  - New cladding Materials (M5, Zirlo, Duplex, …)
  - Present and New Fuel Types
- 12 experiments (with 2 in Na loop performed in 2002)

Program organization (OECD/NEA)

- Piloted by IRSN - Experiments performed by CEA
- Funded by IRSN, EDF, EPRI, USNRC, GRS, CEA, ...

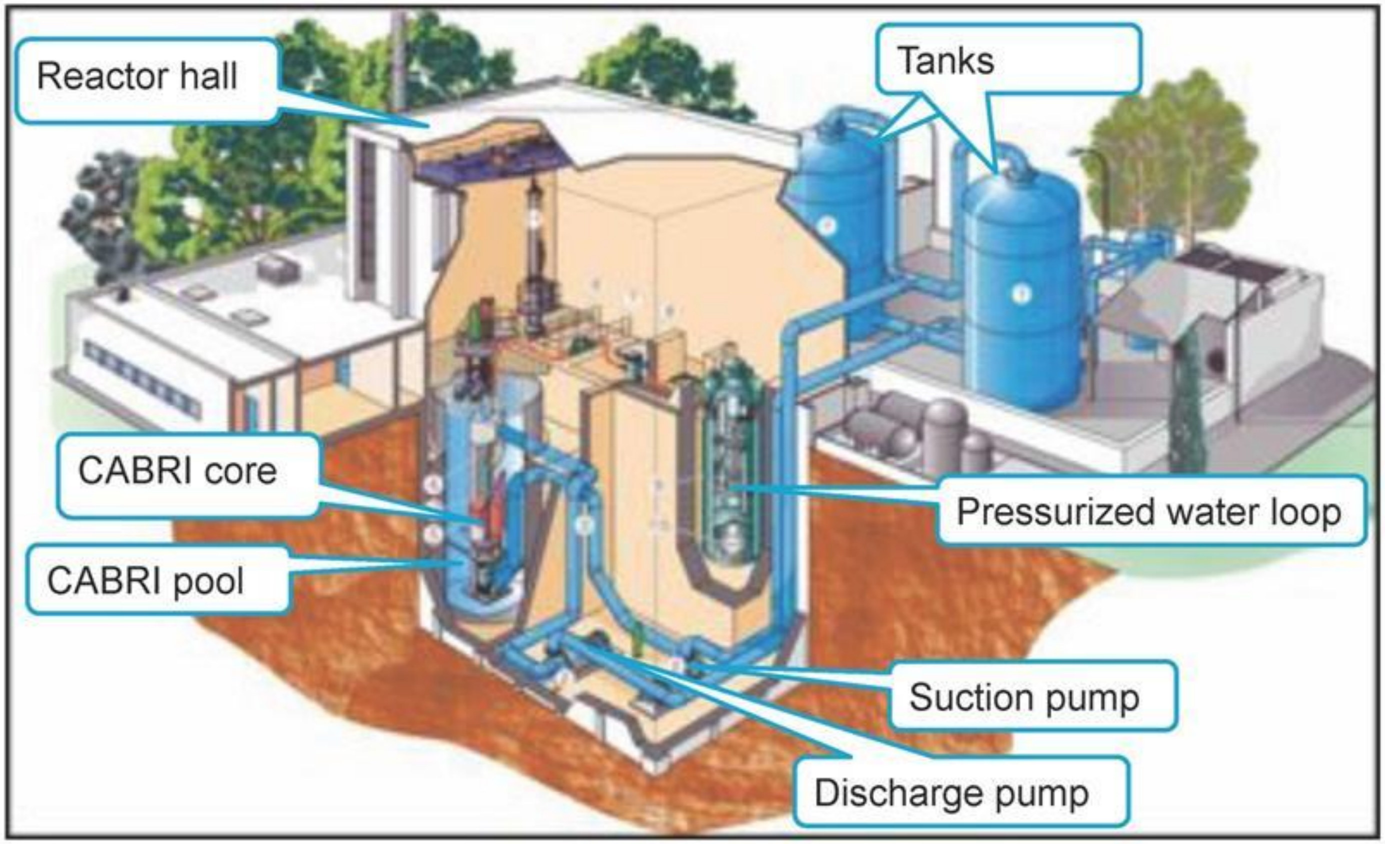
Main challenges for PWRs under RIA conditions

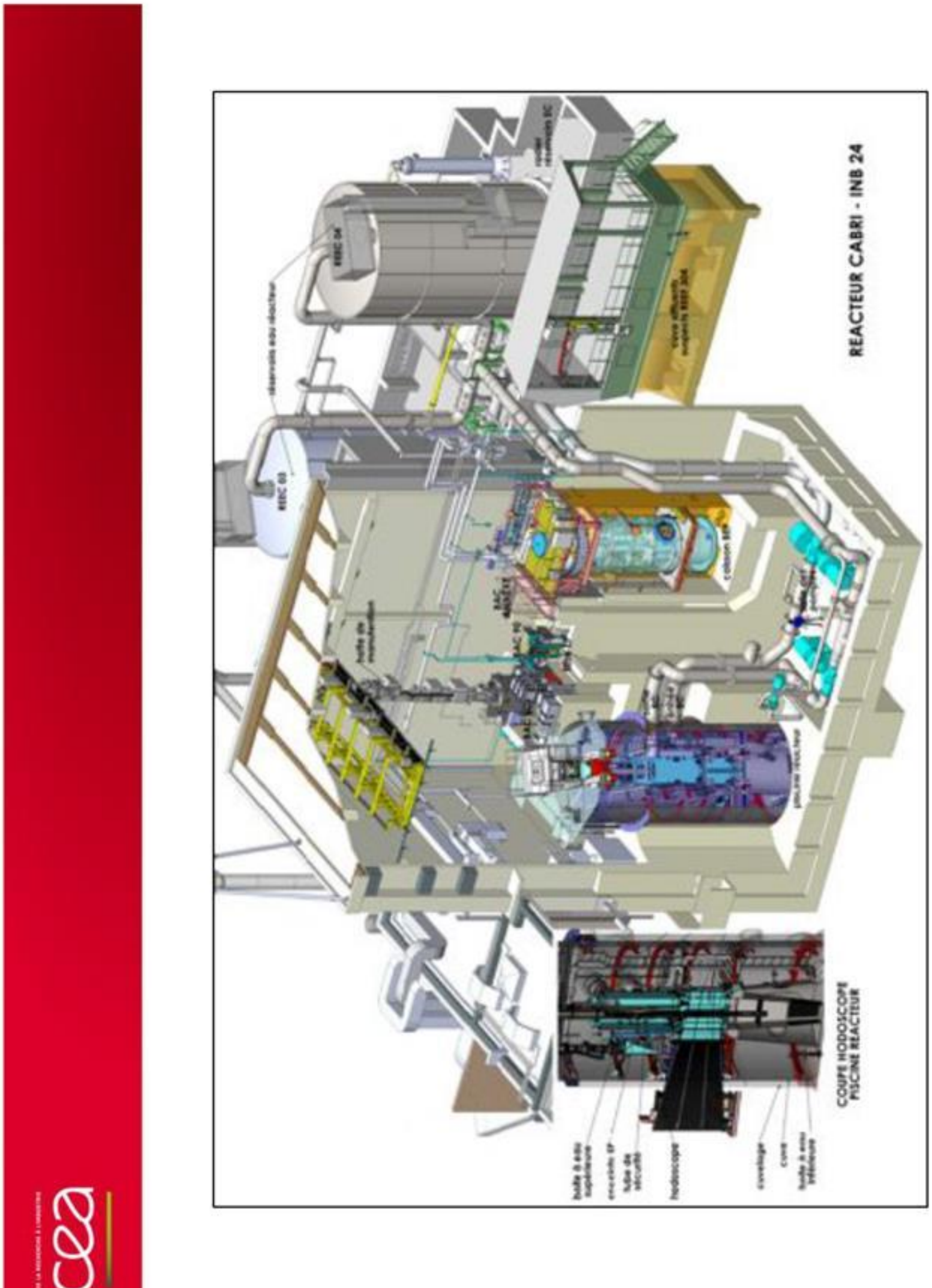
- Regulators
  - Evaluate safety criteria and margins
- Electricity Suppliers
  - Minimize the operation costs & optimize the operation flexibility
- Development & licensing of innovative fuels
  - Claddings: resistance to oxidation (M5/ZIRLO/M-MDA)
  - Microstructure of MOX: better control of fission gas release
- Validation of Multi-Physics calculation codes
  - Design and safety of fuel/reactors



The CABRI facility

Overall view of the CABRI facility




REACTEUR CABRI - INB 24
COUPE HODOSCOPE
PISCINE REACTEUR

The CABRI reactor

- **Reactor and Fuel Characteristics**
  - Pool Type Reactor
    - Core Size: 60x60x80cm
    - Power Max:
      - Steady State: 25MW
      - Pulse: 25GW
    - Forced convection Water Cooled
  - Fuel Rods
    - 1487 UO_2 (6% enriched)
    - Stainless Steel Cladding

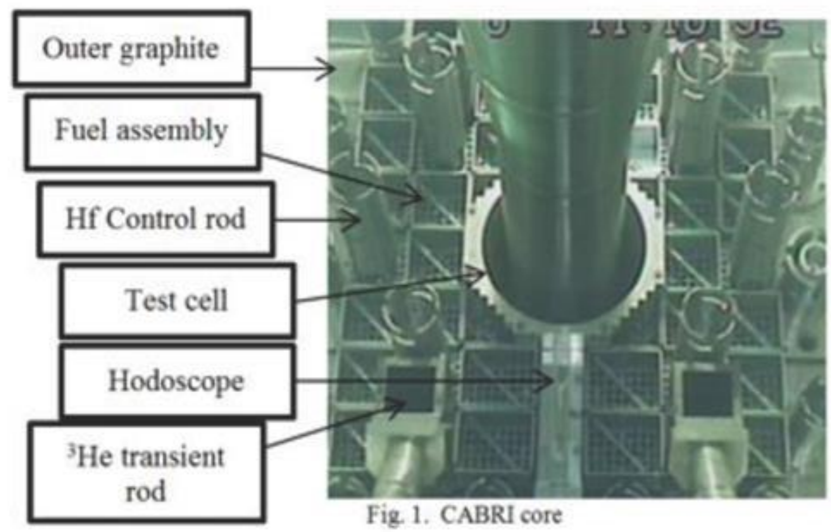
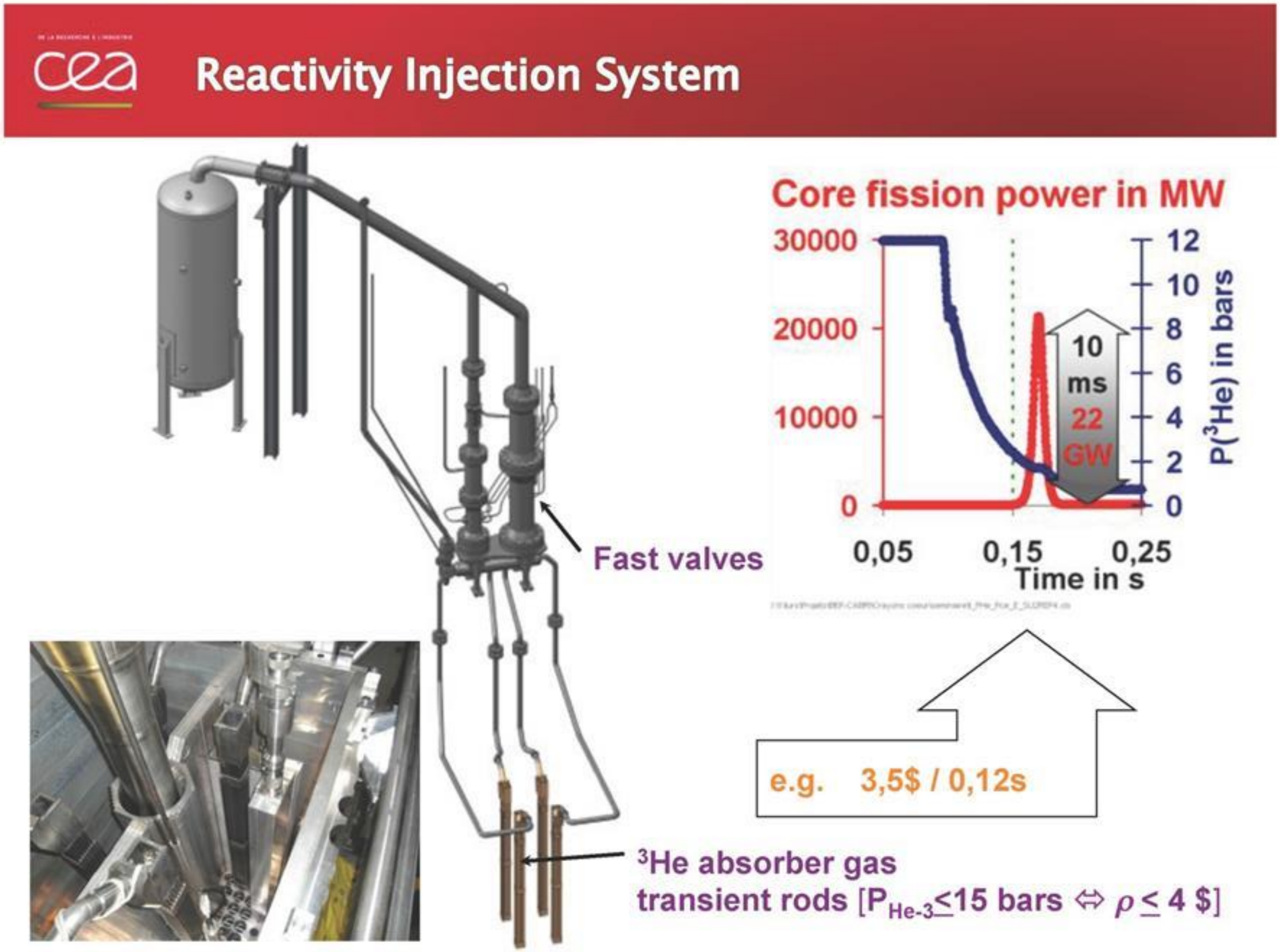
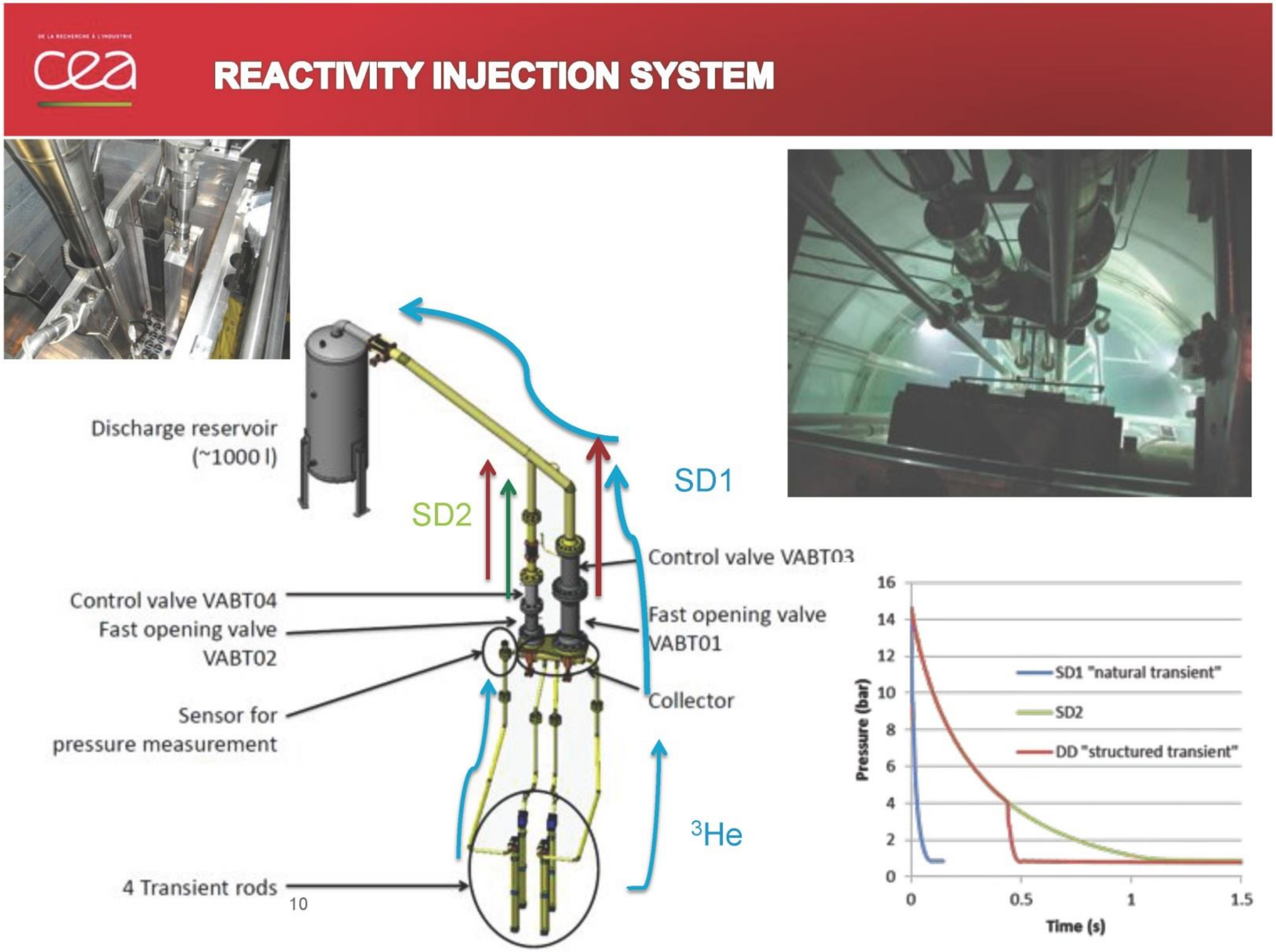


Fig. 1. CABRI core

- **Test device (1 to 3 rods max) is introduced in:**
  - Previous Test Cell
    - Na circulation up to 400°C
  - New Test Cell
    - Pressurized water circulation
    - P = 155 bar, T = 280°C, v = 4m/s

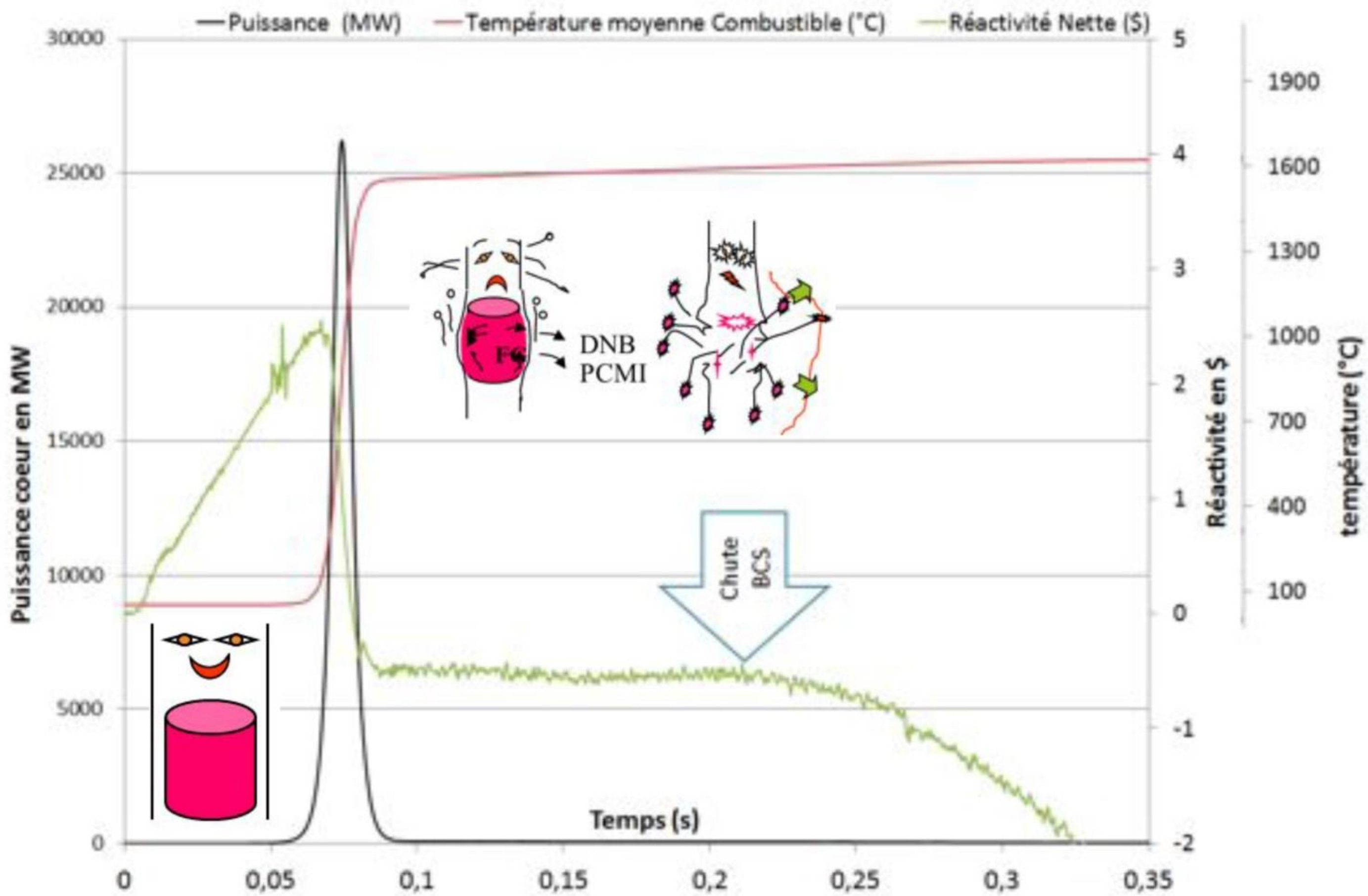
| PAGE 8


cea
Reactivity Injection System
Fast valves
Core fission power in MW
30000
20000
10000
0
0,05
0,15
0,25
Time in s
12
10
8
6
4
2
0
P(3He) in bars
10 ms
22 GW
e.g. 3,5$ / 0,12s
3He absorber gas
transient rods [$P_{He-3} \leq 15$ bars ⇔ $\rho \leq 4$ $]

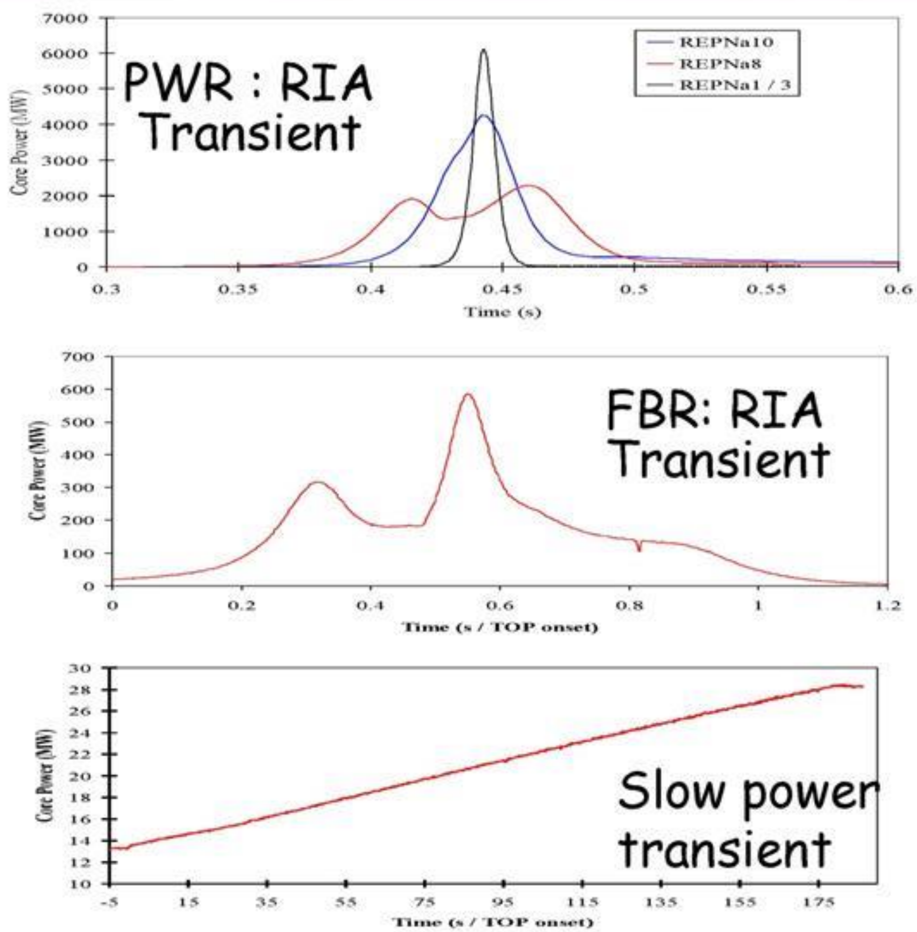


REACTIVITY INJECTION SYSTEM
Discharge reservoir
(~1000 l)
SD2
SD1
Control valve VABT03
Control valve VABT04
Fast opening valve
VABT02
Fast opening valve
VABT01
Sensor for
pressure measurement
Collector
3He
4 Transient rods
10
16
14
12
10
8
6
4
2
0
Pressure (bar)
0
0.5
1
1.5
Time (s)
SD1 "natural transient"
SD2
DD "structured transient"

Typical CIP RIA Sequence



CABRI POWER TRANSIENT EXAMPLES



Performances

- PWR UO2
  - ~100 cal/g
- PWR MOX
  - ~200 cal/g
- FBR RIA
  - ~300 cal/g
- FBR slow ramps
  - Transient:
    400 → 1000W/cm at 0,6% Pn/s

Upgrade of the CABRI Facility

- **For experimental needs**
  - Pressurized Water Loop
    - Dismantling of the sodium loop
    - Manufacturing of the water loop vessel
    - Manufacturing of a new test cell
  - Replacement of the handling cask and related equipment
  - Inspection and upgrade of the reactivity injection system
- **For regulatory needs**
  - Replacement of the core block
  - Upgrade of the overhead crane
  - Design and fabrication of high activity effluent circuits
  - Seismic reinforcements of equipment (reactor vessel, circuits, tanks, doors, chimney…)
  - Fire protection: division into fire areas
  - Creation of an ultimate emergency system
  - New public road network

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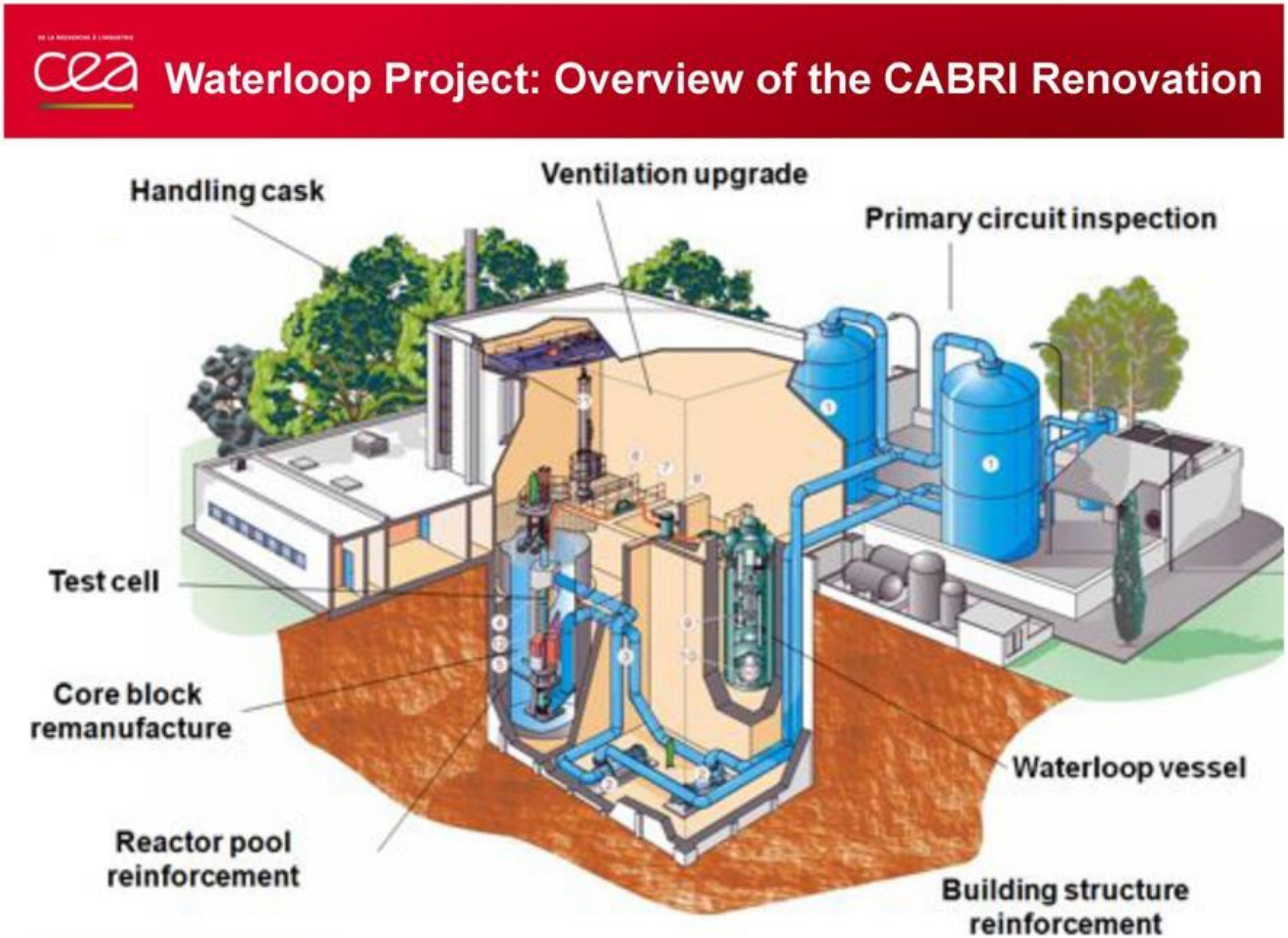
Refurbishment of the CABRI Facility

- Power and instrumentation & control system
- Reactor building roof
- Primary cooling system

- Ventilation

- …

| PAGE 14


Waterloop Project: Overview of the CABRI Renovation
Handling cask
Ventilation upgrade
Primary circuit inspection
Test cell
Core block remanufacture
Reactor pool reinforcement
Waterloop vessel
Building structure reinforcement

Safety Reassessment

Safety instruction

- The French Regulator organized 2 permanent groups in 2004 and 2008
  => recommendations and engagements

Safety documentation

- New CABRI safety report
- Update of the main rules of operation
- Update of the operation procedures

Safety authorizations

- For the first criticality and power commissioning tests: October 19th, 2015
- For CIP tests: 2nd semester of 2017

Organization of the Recommisioning

Commission 1	**Neutronics characterization of the core**
Commission 2	Ventilation
Commission 3	Reactor vessel
Commission 4	Handling
Commission 5A	Conventional circuits
Commission 5B	Circuits of the pressurized water loop
Commission 6	Instrumentation and control
Commission 7	**General operation during steady and transient power tests**

- Commission 1: 80% finished; end in May 2016
- Commission 5B: continued and finished in mid-2016
- Commission 7: end before the end of 2016

Conclusion

- All upgrade and refurbishment works are finished
- Important work achieved as for safety documentations for authorization of the French regulator
- First criticality of CABRI / Waterloop succeeded
- Neutronics tests will be ended in May 2016
- Steady and transient power tests: end of 2016
- CIP-Q test in 2nd semester of 2017
- CIP program during 5 years

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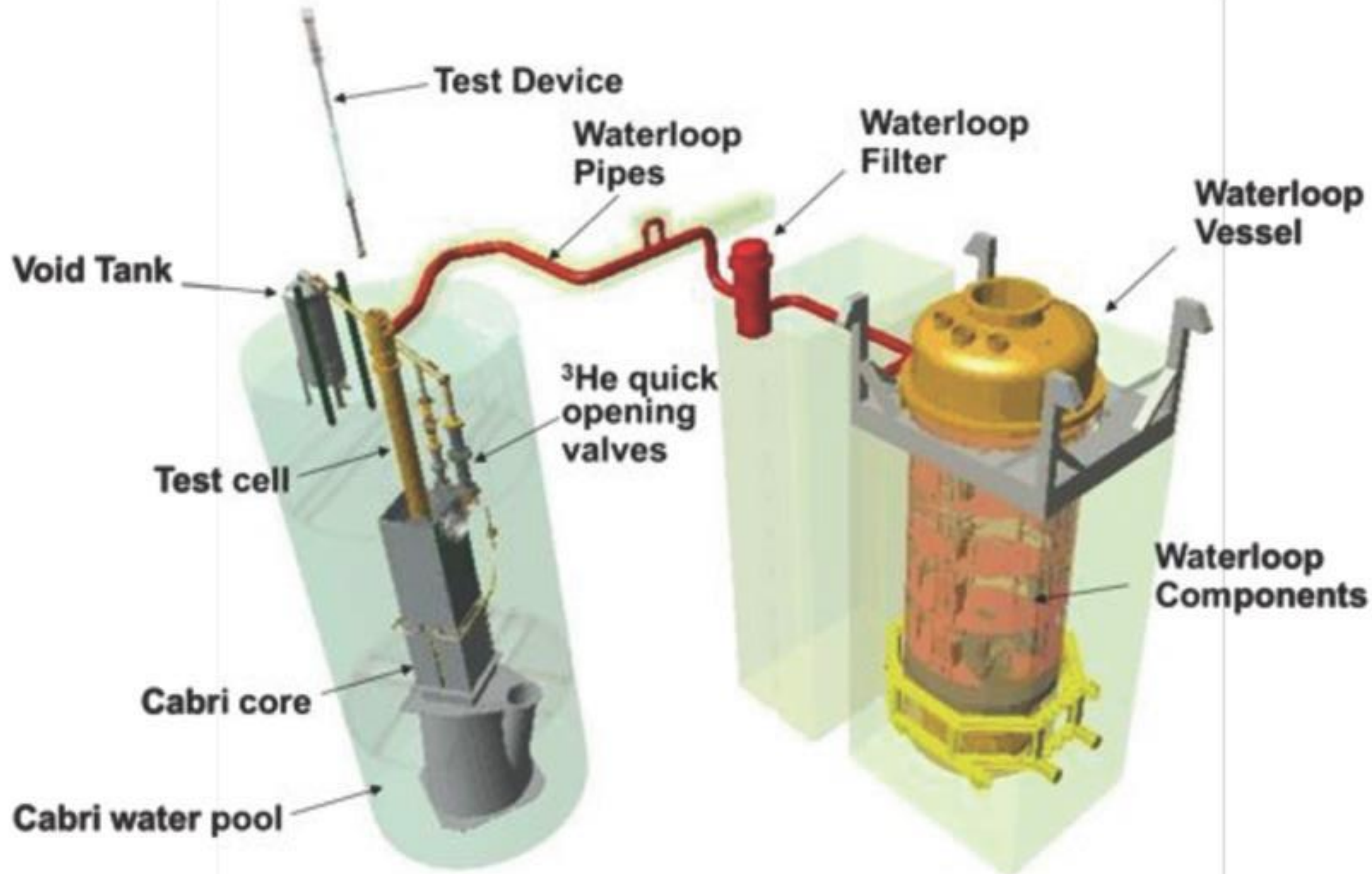
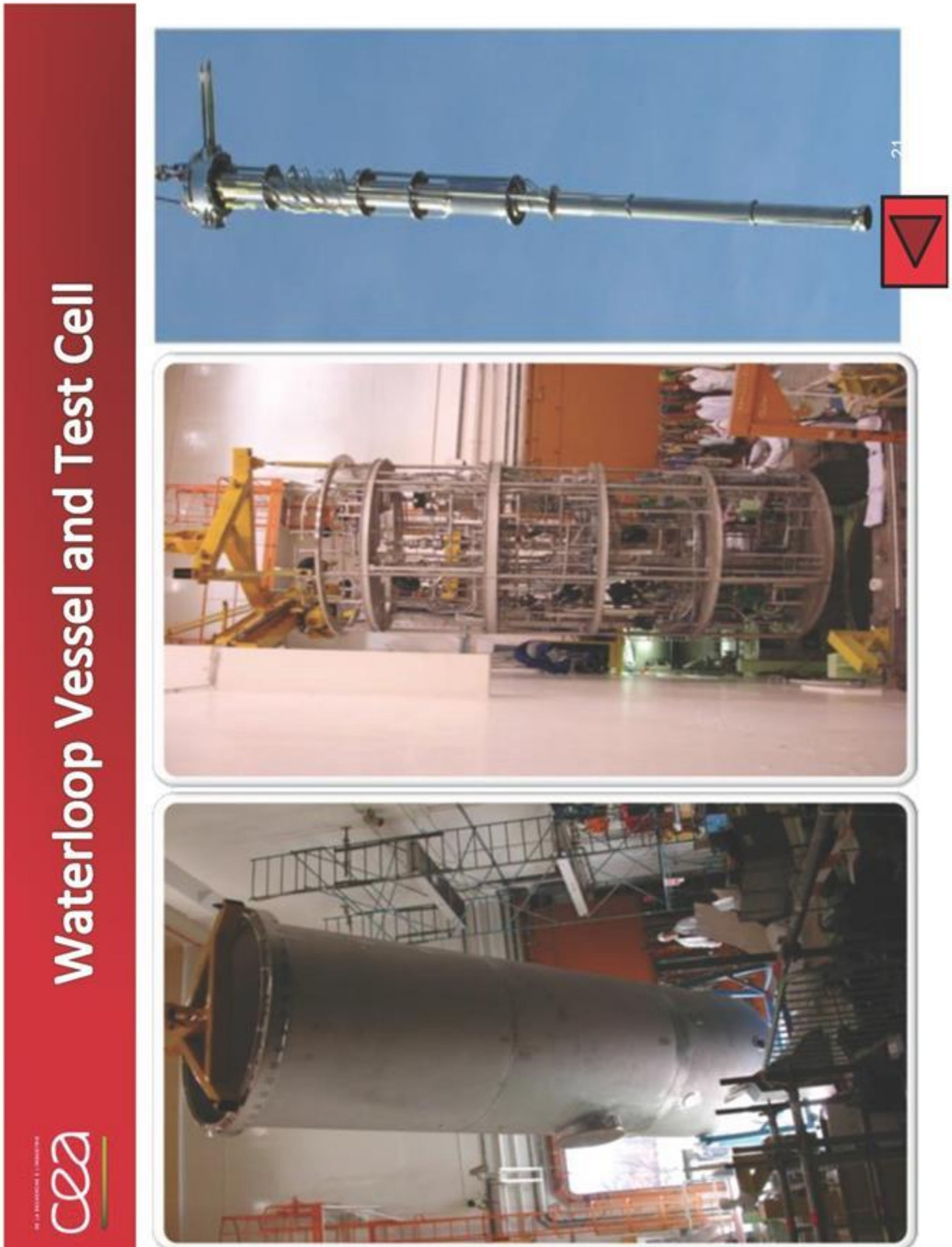
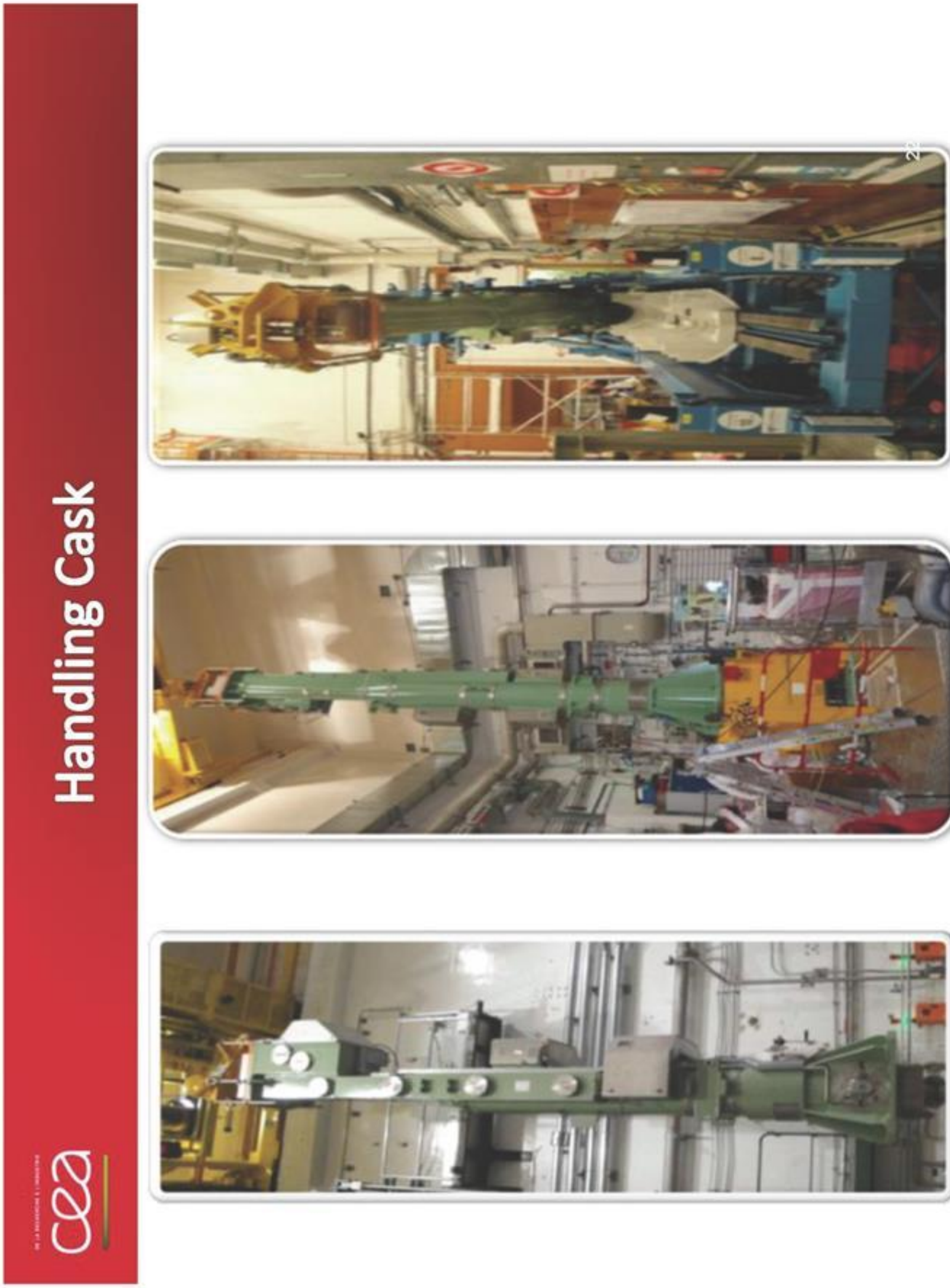
Thank you for your attention

The New CABRI Waterloop

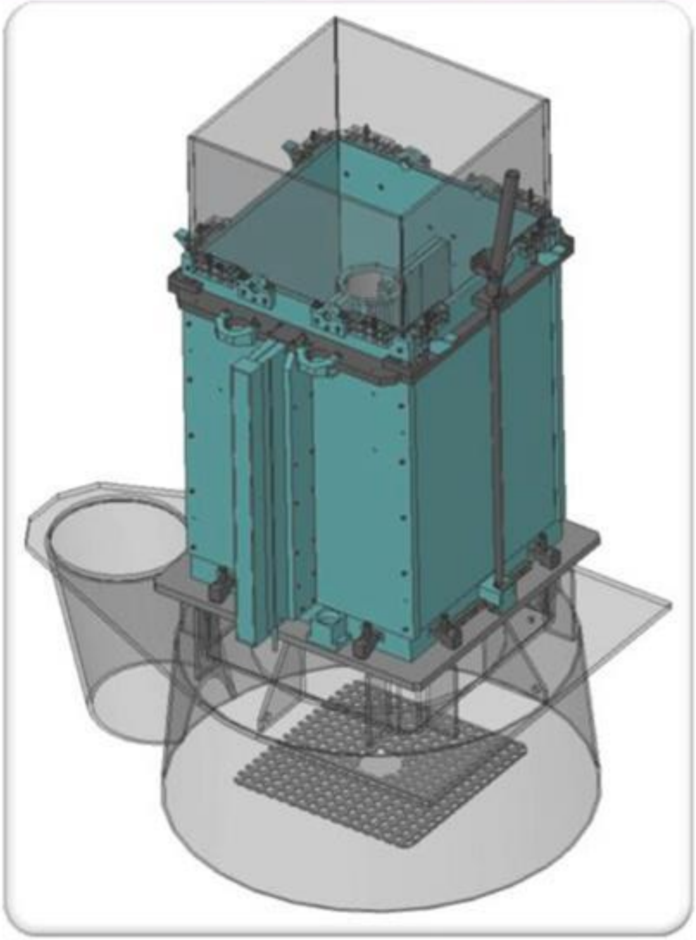


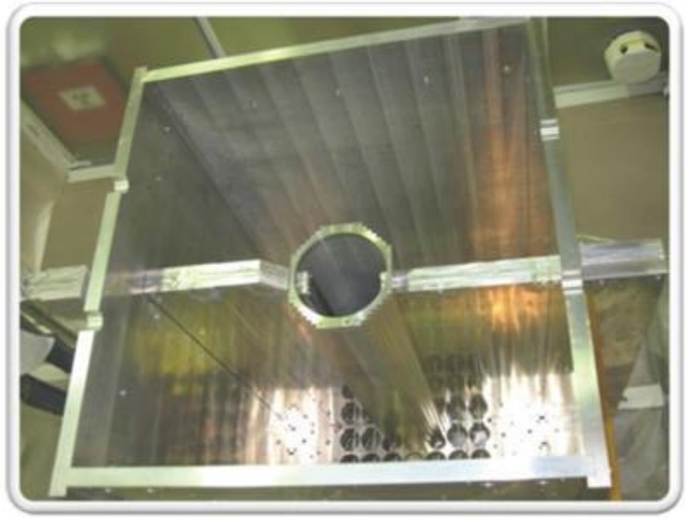

Waterloop Vessel and Test Cell
21


Handling Cask

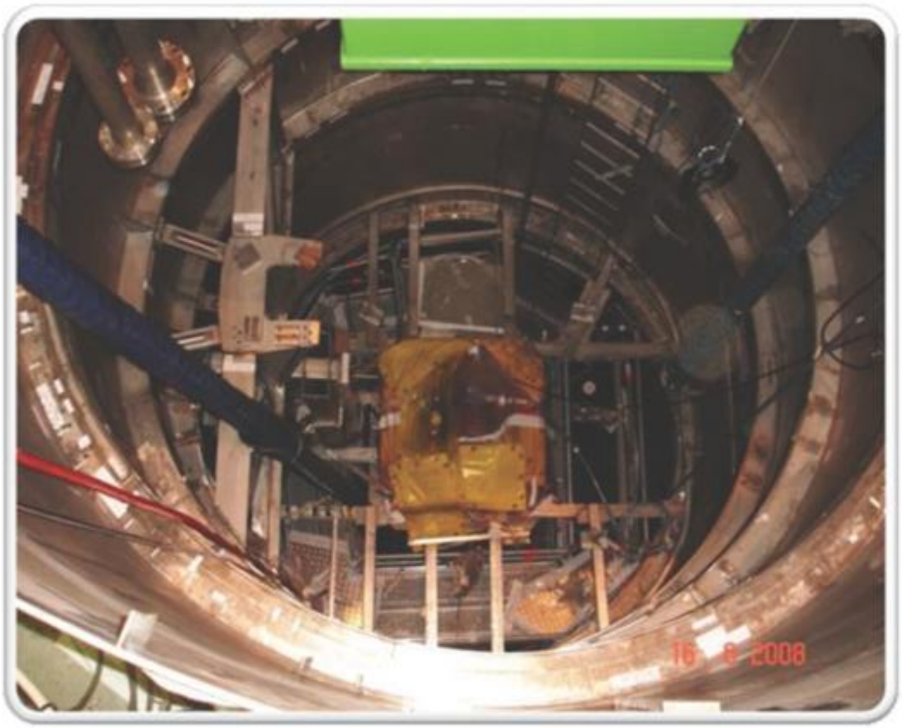
Trailer & Shielded Container

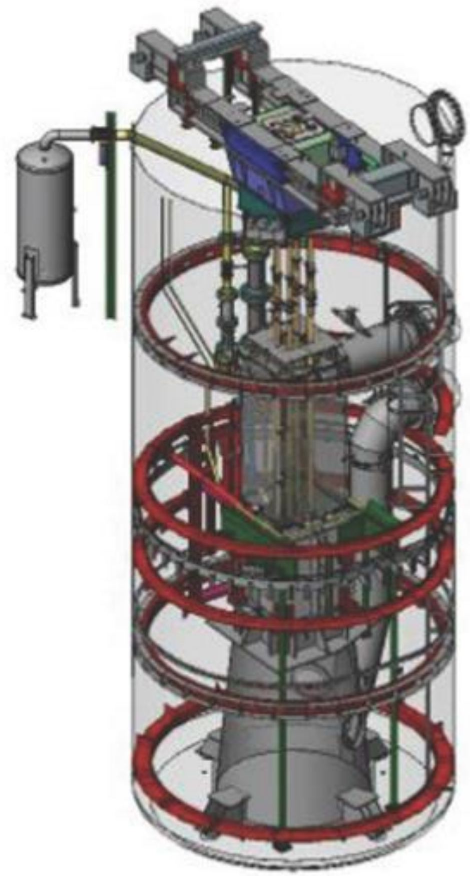
Core Block Remanufacture



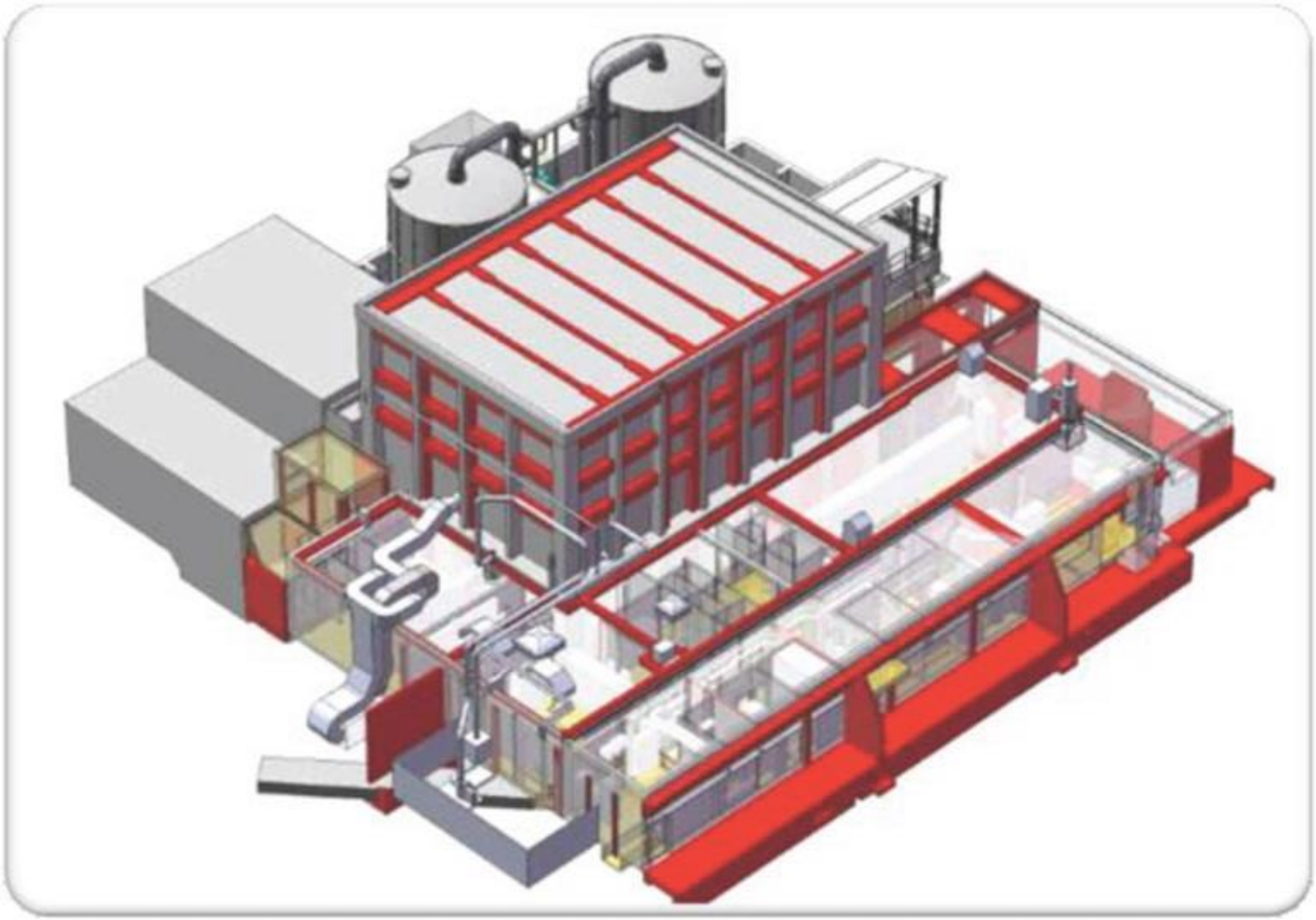


Reactor pool seismic reinforcement





Building structure seismic reinforcement



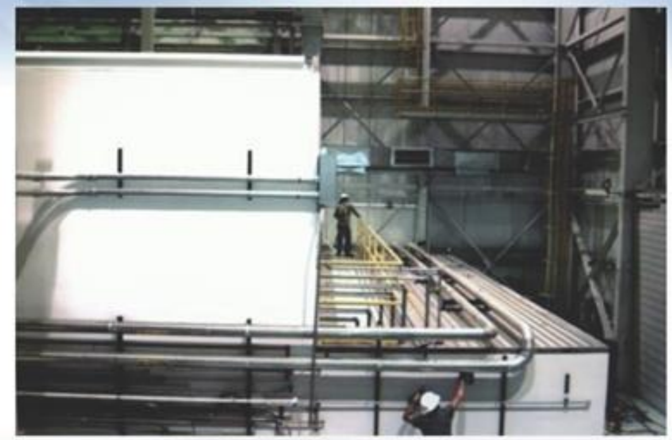
Building structure seismic reinforcement


TREAT Restart Activities
May 5, 2016
J. R. Parry
TREAT Chief Scientist
www.inl.gov
INL
Idaho National Laboratory
Experiment hold down mechanism
Reactor inlet air filter
MK III experiment
Concrete shield blocks
Rotating shield plug
Graphite reflector
Fuel assemblies
Radiography facility
Hodoscope
NORTH
Grid plate
Compensation/shutdown rod drive (typical)
Transient rod drive (typical)
Control/shutdown rod drive (typical)

Restart Progress

Idaho National Laboratory

- New roofs – Reactor building and Office building.
- Demolition of PHP building in the high bay.
- New water line.
- ARCS system operational.
- TREAT control rod drives will not require replacement or major refurbishment.
  - Completed initial functional and SCRAM testing of all rod drives.
  - 18 new shock absorbers have been received. Eight will be replaced over the next several weeks and eight plus the two refurbished will be placed in stock for future need.




Rod Testing
INL
Idaho National Laboratory
3

Idaho National Laboratory

Fuel Condition

- Fuel evaluations demonstrate fuel will be acceptable for continued use. A total of 109 fuel assemblies have been evaluated.
- Inspections found a variety of loose foreign material in the core cooling channels as well as melted yellow plastic on five fuel elements in the core.
  - All loose foreign material has been removed from the core and all melted plastic has been removed from affected fuel assemblies using a thermal removal technique.
  - Validation was completed to ensure that no melted plastic imprinting had occurred on elements that had been co-located with the affected elements.
  - Completed final visual and ultrasonic test (UT) examinations of the remediated assemblies that had melted yellow plastic removal to ensure no damage to the surface of the fuel clad had occurred.

4

TREAT Upgrades

- TREAT (MFC-720) Fire Protection Upgrades
  - Completed the MFC 720 Fire Protection System Modification milestone to complete construction, installation, testing, and partial project transfer from construction on January 21. Subsequently, completed successful acceptance testing of the clean agent gaseous fire suppression system and completed final acceptance test of the fire protection upgrade including fire alarm center notifications.
- Control Room (MFC-724) Refurbishment
  - Completed refurbishment activities allowed for turn over to TREAT, facilitating functional testing of poison assemblies, control rods and rod drives.

5

Core Poisoning

Idaho National Laboratory

- Completed installation of all 16 poison assemblies into the core November 2015.
- Poisoned core was confirmed April of 2016
  - Poisoning the core allows mock operations and supports qualification training of reactor personnel as well as reactor operational procedure validation.

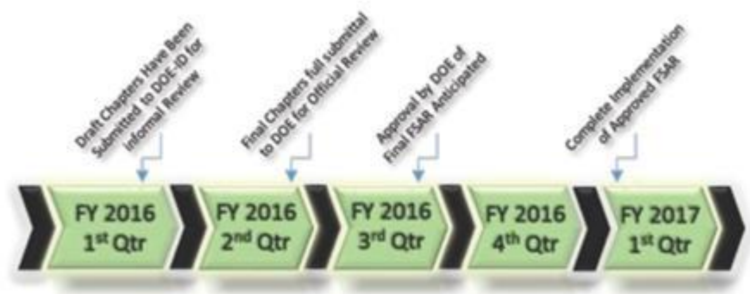
Look-ahead

- V&V of software for ARCS and RTS.
- Integrated operational testing of all equipment with poison core.

Upgraded SAR

- FSAR was reviewed on a chapter-by-chapter basis and updated to be compliant to accurately describe the facility, organization, and processes.
  - FSAR sections for reactor system modifications associated with the TREAT Upgrade Project were removed as this equipment was either designed but not supplied, or supplied but not installed.
  - FSAR update to current organization titles and responsibilities as well as updated process references, and accident analysis was updated to reflect current vs. upgraded core.
- Close coordination with DOE-ID nuclear safety has been maintained through routine meetings and DOE informal review of drafts.
  - All draft chapters were submitted to DOE-ID for informal review on 11/16/15.
  - The technical specifications (TS) were submitted for informal review on December 23, 2015.
  - All comments received from DOE-ID have been incorporated, the chapters have finished TORC and SORC review, and the final version was submitted to DOE-ID on March 29, 2016 for final review.

Final FSAR/DSA Timeline



7

Reactor Operations

- **New core configuration**
  - Measure control rod worth
  - Heat balance (to calibrate/position instruments)
  - Series of 3 temperature limited transients
    - Start with a low power transient with subsequent transients increasing in power
    - Linearly extrapolate to the reactivity required to achieve temperatures of 600° and 820° C

Startup Procedure

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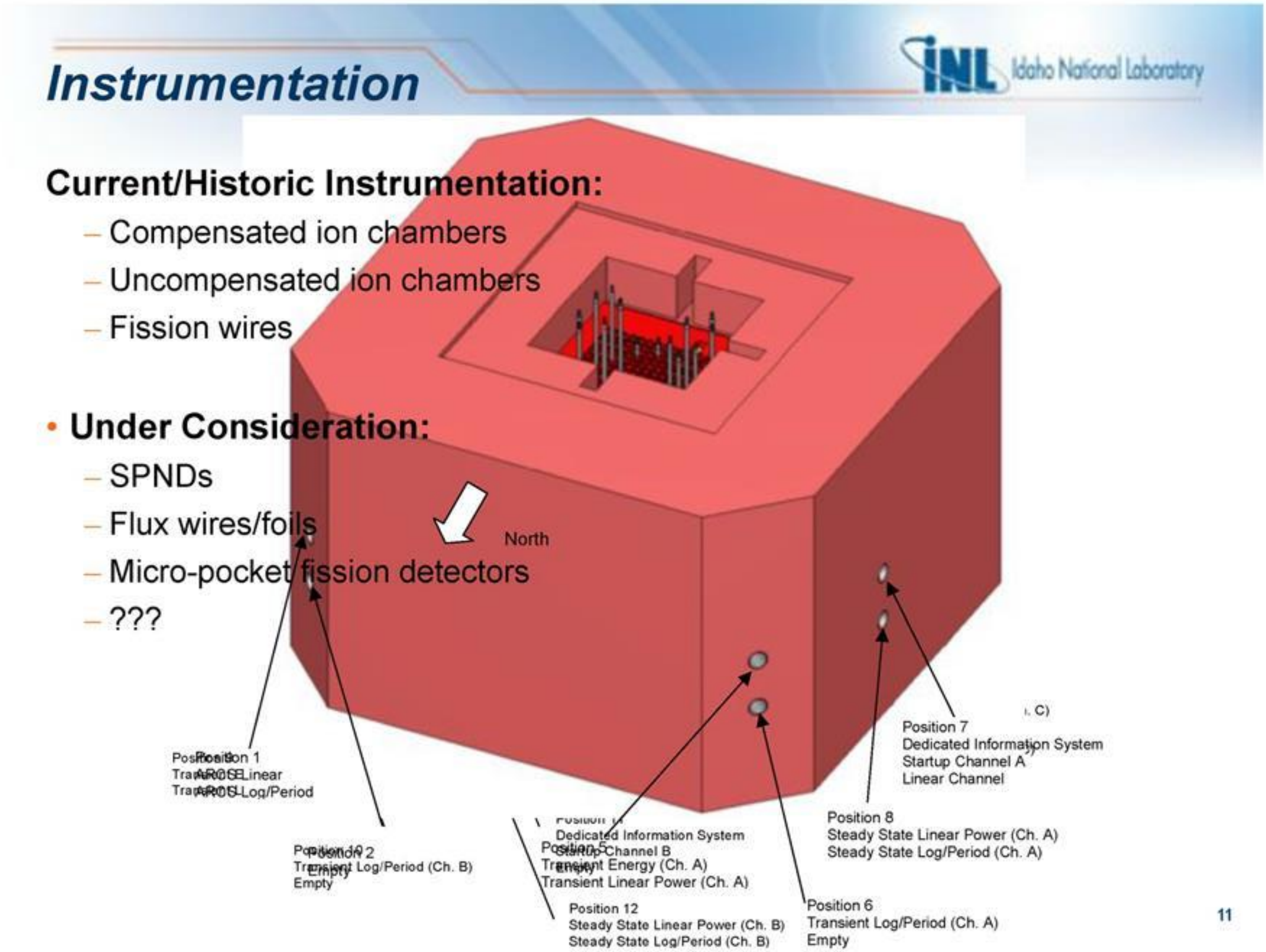
- **Completely withdraw all Compensation rods.**
- **Completely withdraw all Transient rods.**
- **Withdraw control rods to achieve critical.**
- **Increase power to 50 W.**
- **Bank control rods to same elevation at 50 W.**
- **Insert transient rods to a banked elevation determined to provide the transient reactivity needed for the prescribed transient**
  - Control rods are withdrawn to maintain criticality while Transient rods are inserted. (Control rods are banked at the same elevation)

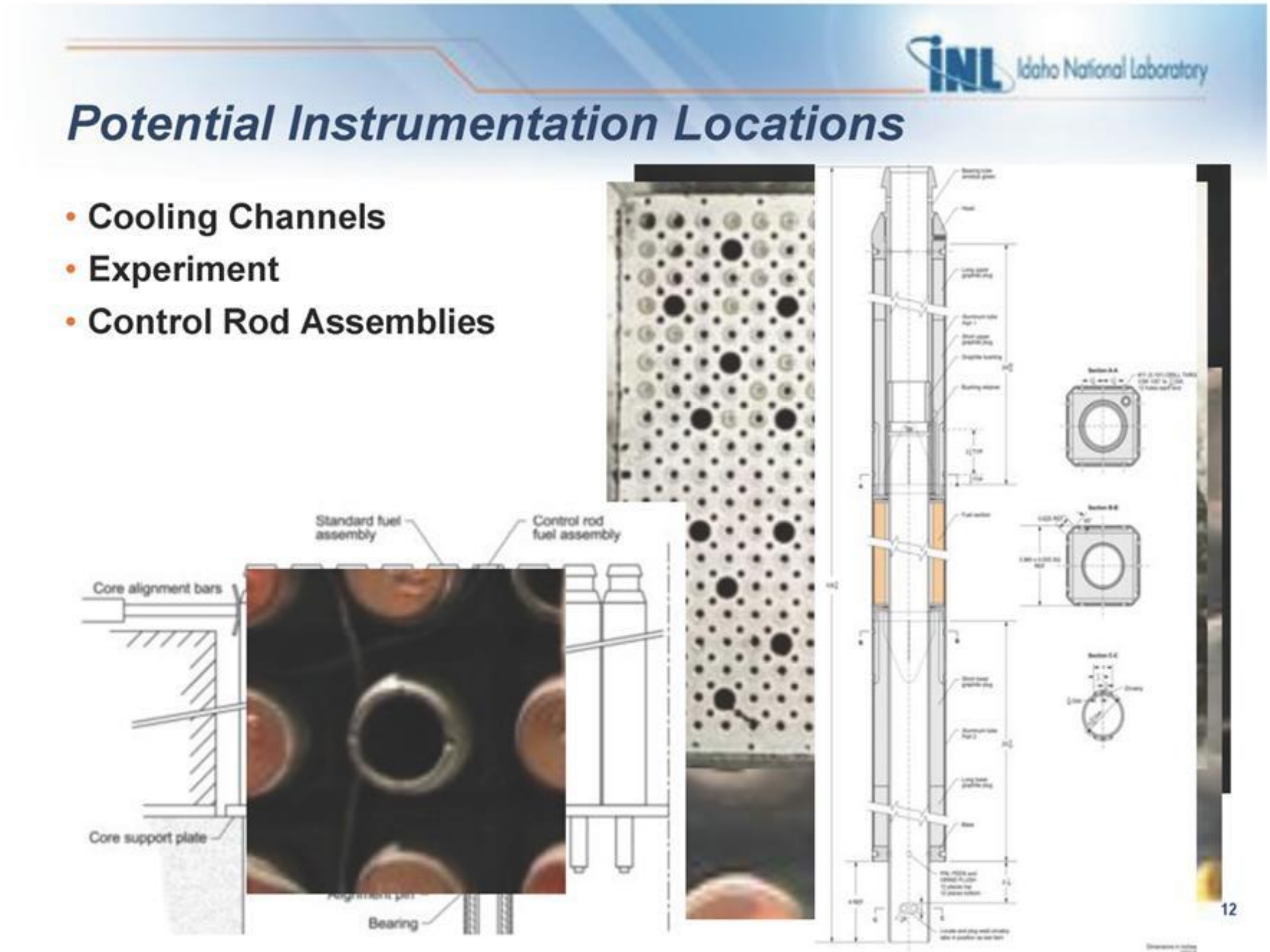
9

Startup Testing

- Planned
  - Critical rod position
  - Heat balance
  - Instrument calibration
  - Differential rod worth measurements
  - Trial transients to determine reactivity limits
  - Power coupling factor measurements

- Under Consideration:
  - Transfer function
  - Temperature coefficient
  - Flux mapping

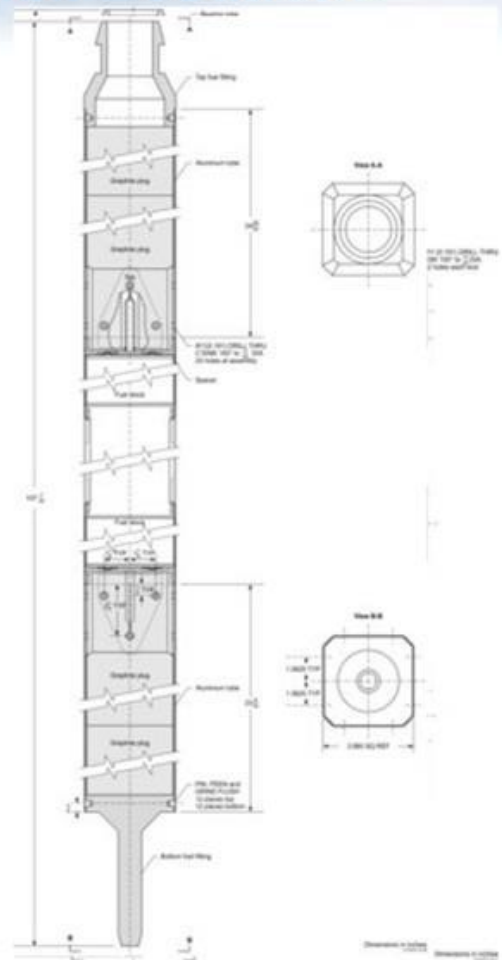

Instrumentation
Idaho National Laboratory
Current/Historic Instrumentation:
– Compensated ion chambers
– Uncompensated ion chambers
– Fission wires
• Under Consideration:
– SPNDs
– Flux wires/foils
– Micro-pocket fission detectors
– ???
North
Position 7
Dedicated Information System
Startup Channel A
Linear Channel
Position 8
Steady State Linear Power (Ch. A)
Steady State Log/Period (Ch. A)
Position 6
Transient Log/Period (Ch. A)
Empty
Position 12
Steady State Linear Power (Ch. B)
Steady State Log/Period (Ch. B)
Dedicated Information System
Startup Channel B
Transient Energy (Ch. A)
Transient Linear Power (Ch. A)
Transient Log/Period (Ch. B)
Empty
11


Idaho National Laboratory
Potential Instrumentation Locations
• Cooling Channels
• Experiment
• Control Rod Assemblies
Standard fuel assembly
Control rod fuel assembly
Core alignment bars
Core support plate
Bearing
12

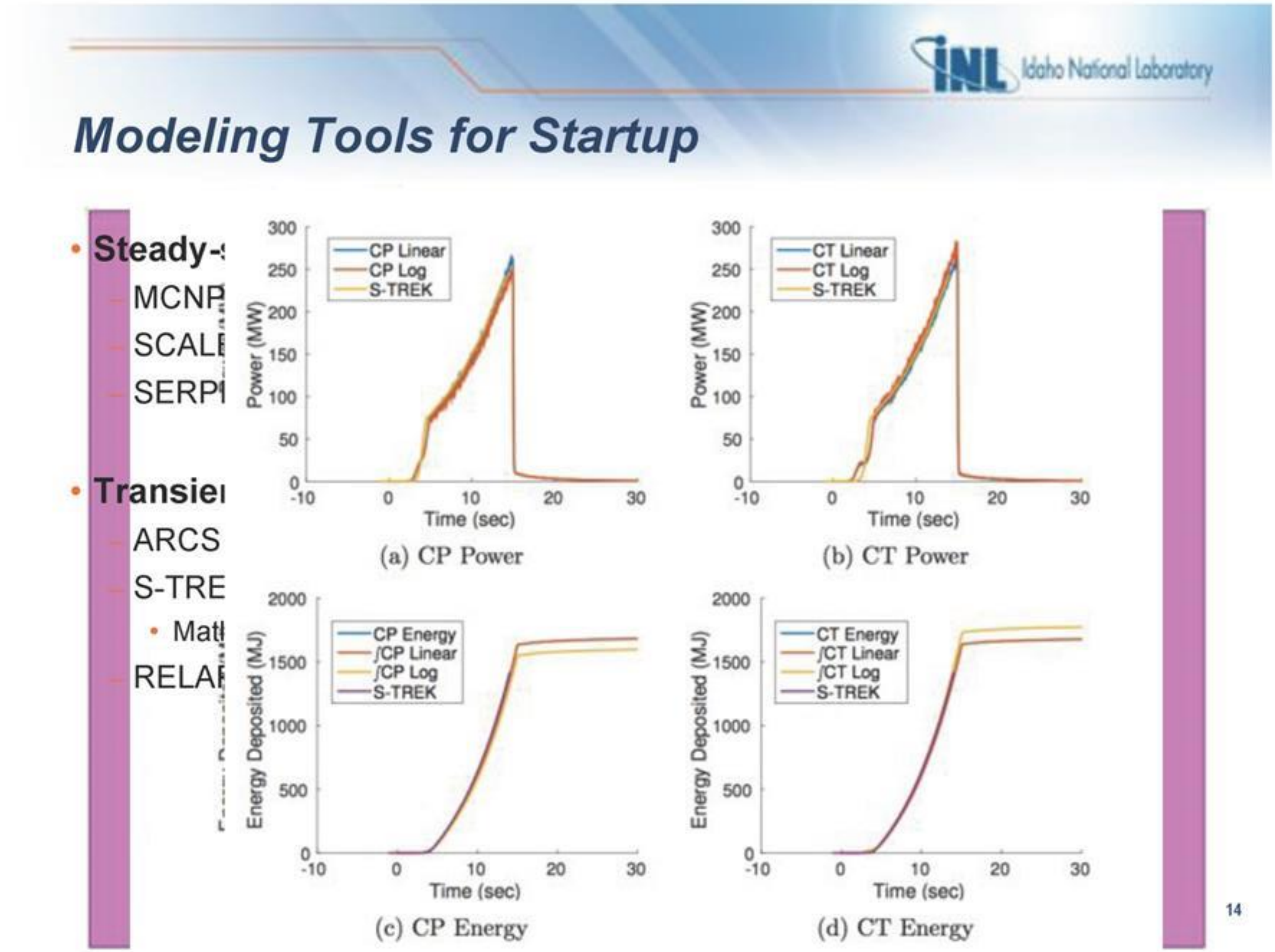
Idaho National Laboratory

More Possible Instrumentation Locations

- Hodoscope slot
- Vertical access hole
- Custom instrumentation assemblies



13


Idaho National Laboratory
Modeling Tools for Startup
Steady-
MCNP
SCALE
SERP
Transie
ARCS
S-TRE
Mat
RELA
CP Linear
CP Log
S-TREK
Power (MW)
Time (sec)
(a) CP Power
CT Linear
CT Log
S-TREK
(b) CT Power
CP Energy
∫CP Linear
∫CP Log
S-TREK
Energy Deposited (MJ)
(c) CP Energy
CT Energy
∫CT Linear
∫CT Log
S-TREK
(d) CT Energy
14

Questions?

DE LA RECHERCHE À L'INDUSTRIE

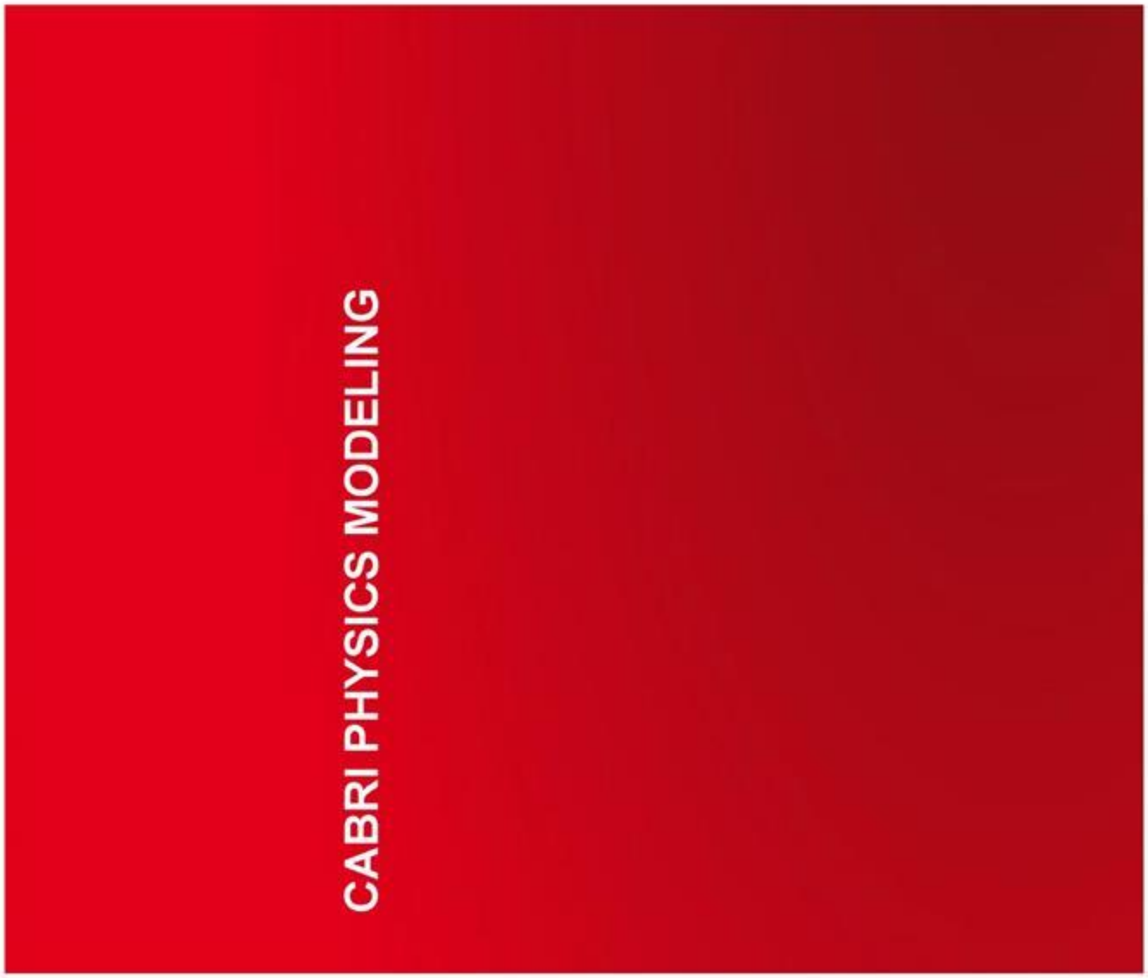
cea den

CABRI PHYSICS MODELING STARTUP TESTING RESULTS AND PLANS

Jean-Pascal Hudelot
jean-pascal.hudelot@cea.fr
Nuclear Energy Division,
CEA Cadarache, France

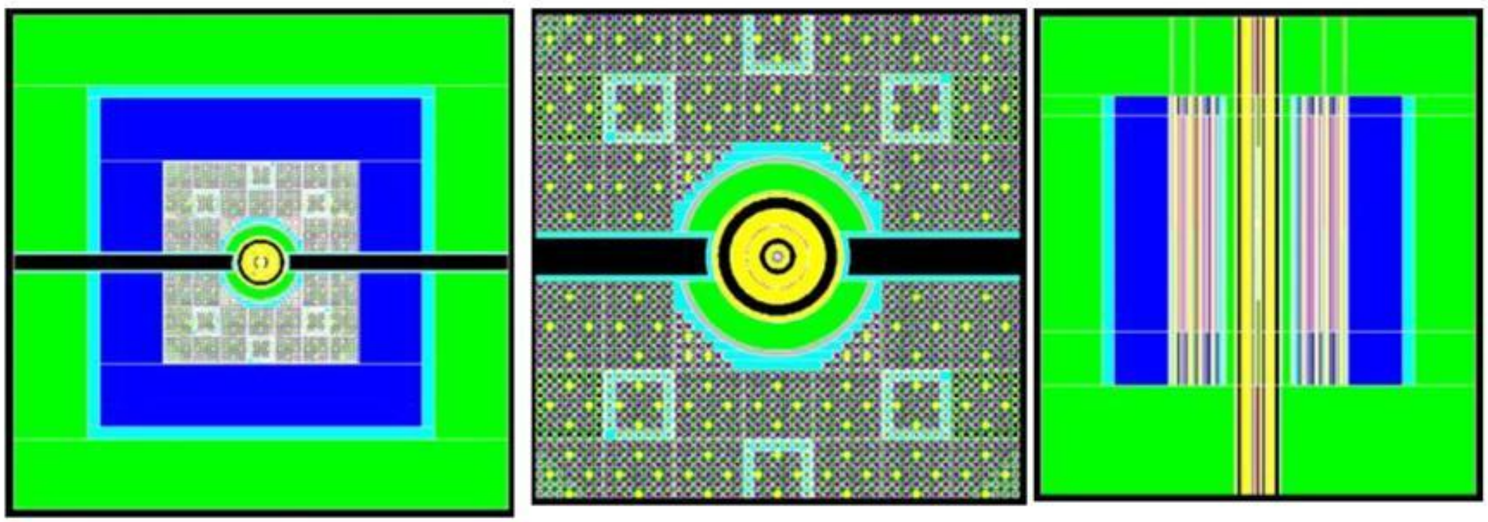
OUTLINE

- CABRI physics modeling
- Startup testing preliminary results and plans


CABRI PHYSICS MODELING

CORE NEUTRONICS

- TRIPOLI4 and MCNP core calculation schemes
  - ➢ Real geometry description of the core
  - ➢ JEFF3.1.1 nuclear data library
  - ➢ Validated on critical states of past tests in Na-loop



STATUS AND EFFORTS FOR THE VALIDATION OF NEUTRONICS CORE CALCULATION SCHEME

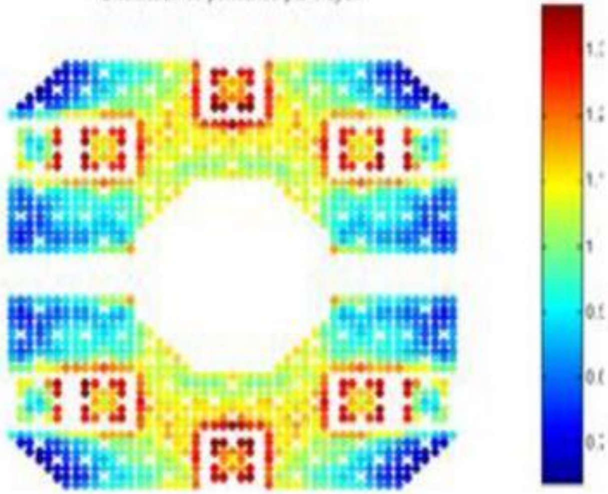
- Status: validated on past Na-loop critical states

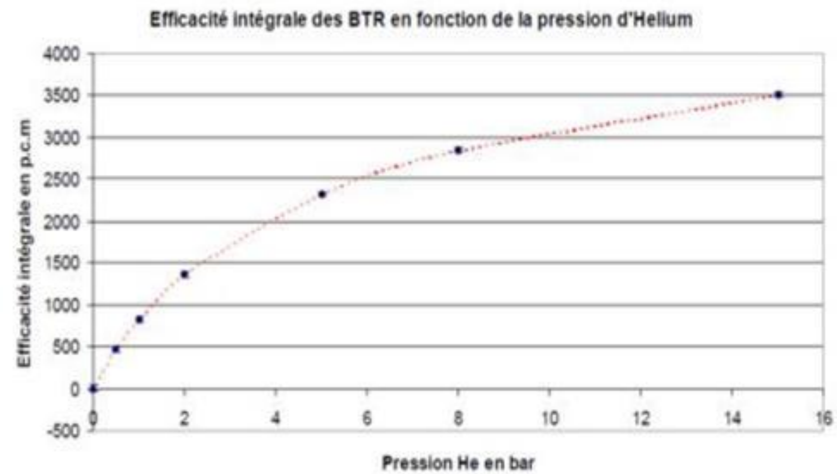
- Current efforts
  - ➢ Update and improvement of the core characterization in terms of material balance and geometry
  - ➢ Validation against other intergral and local parameters, taking benefit of current neutronics commssioning tests
    - Control rods integral and differential efficiencies in several operation conditions
    - Axial and radial flux profiles
    - Isothermal temperature coefficient
    - Kinetic parameters (β, Λ)
    - Reactivity worth of the control rods
    - Void effects in the test cell
    - Gamma-heating

- Other main use: coupling factor, doppler coefficient, local data assessment

EXAMPLE OF NEUTRONICS CALCULATIONS

Coeur CABRI 3D (grilles) - Boucle sodium (Ni 8mm) + CIP02
BCS à 595 mm - BTR 0.6 b He3
Distribution de puissance par crayon





Ref. 1: JP. Hudelot, AC. Colombier, C. D'Aletto, L. Gaubert, O. Guéton, O. Leray, P. Siréta, C. Vaglio-Gaudard, M. Valentini, "Development, validation and qualification of neutronics calculation tools for small reactors application to the JHR, CABRI and OSIRIS reactors", RRFM2012 conference, 18-23 march 2012, Prague

OTHER NEUTRONICS TOOLS

- Calculation of the test fuel inventory
  - ➢ APOLLO2/REL2005 reference scheme: Precise calculation but the fuel inventory is not complete
  - ➢ APOLLO3 code is the future alternative

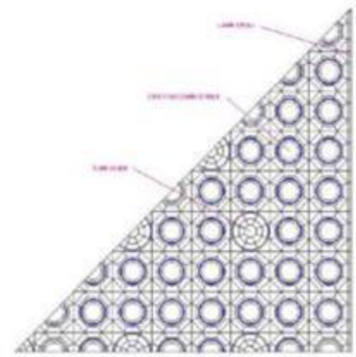


Figure 4: Modèle géométrique pour le calcul MOC

  - ➢ CESAR5.3 depletion calculations → complete fuel inventory
    - ➢ Numerical validation by comparison with the French reference tools DARWIN2.3

- IRSN calculation route
  - ➢ MORET4 and MCNP5 stochastic codes
  - ➢ VESTA depletion code

| PAGE 7

DULCINEE: CABRI TRANSIENT CALCULATION CODE

- DULCINEE (Ref. 2) point-kinetics code developed in the 1970s
  - characterizes the behavior of a nuclear core with low pressure coolant (water, sodium, etc.) during several types of transients (RIA, LOF, ramps, LOCA)
  - embeds simplified thermal and thermal hydraulics models.
    - can process single or double phase flows in natural or forced convection
    - Heat transfer in the fuel rods is modeled from the inside to the outside. Several types of regions are described (fuel/gap/clad).
    - The physical properties of each region are tabulated as a function of the temperature.
  - Also takes into account clad expansion or coolant density

- The CABRI core is modeled with 2 regions:
  - 1 hot channel
  - 1 average channel

Ref. 2: Dulcinee. Beyond neutron kinetics, a powerful analysis software, G. Ritter et al., RRFM IGORR 2012, Prague, Czech Republic, March 18-22, 2012

MAIN INPUT DATA FOR THE DULCINEE CODE

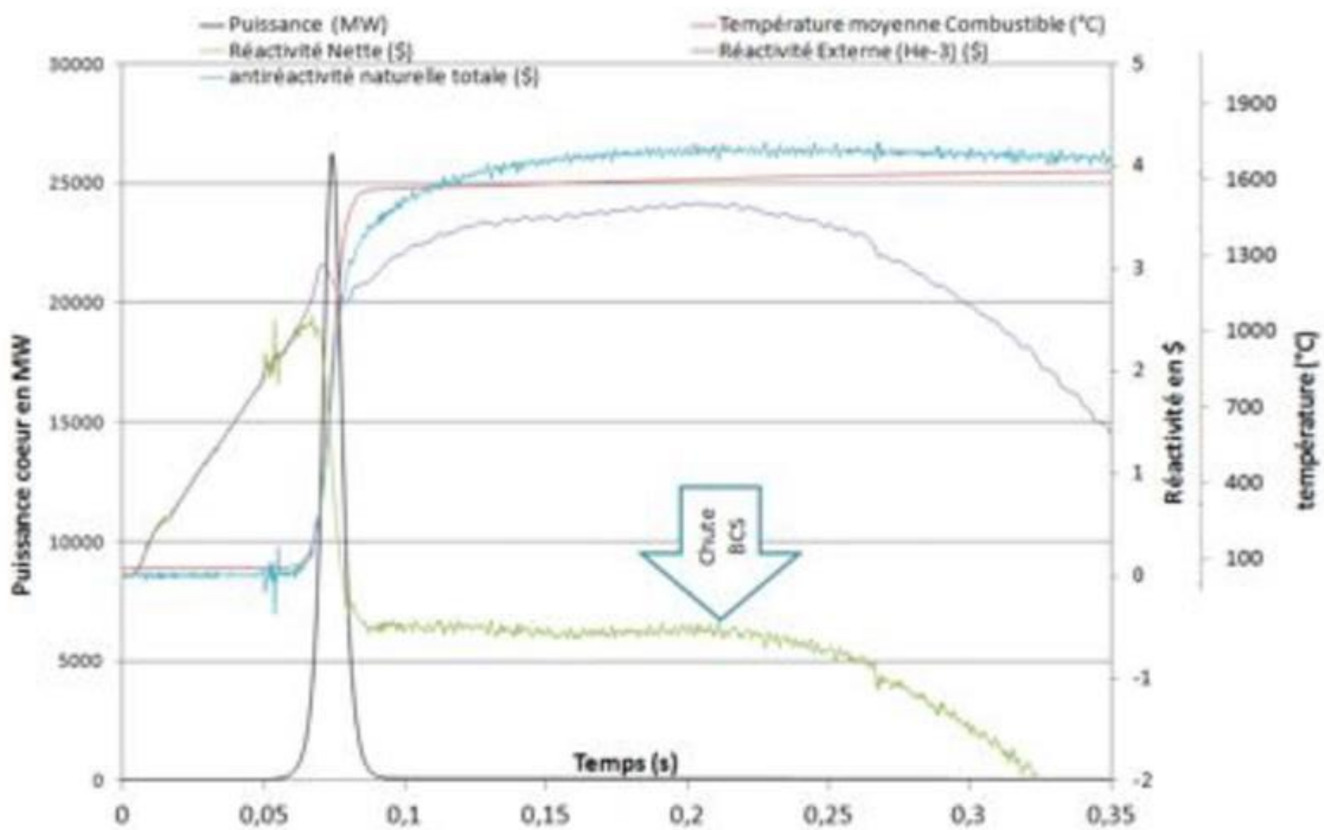
- Feedback coefficient from TRIPOLI4 code
  - Doppler coefficient
  - Void coefficient

- Other input parameters from measurements or TRIPOLI4 code
  - Kinetic parameters (β,Λ)
  - Axial and radial power profiles
  - Reactivity worth of helium-3 vs. pressure

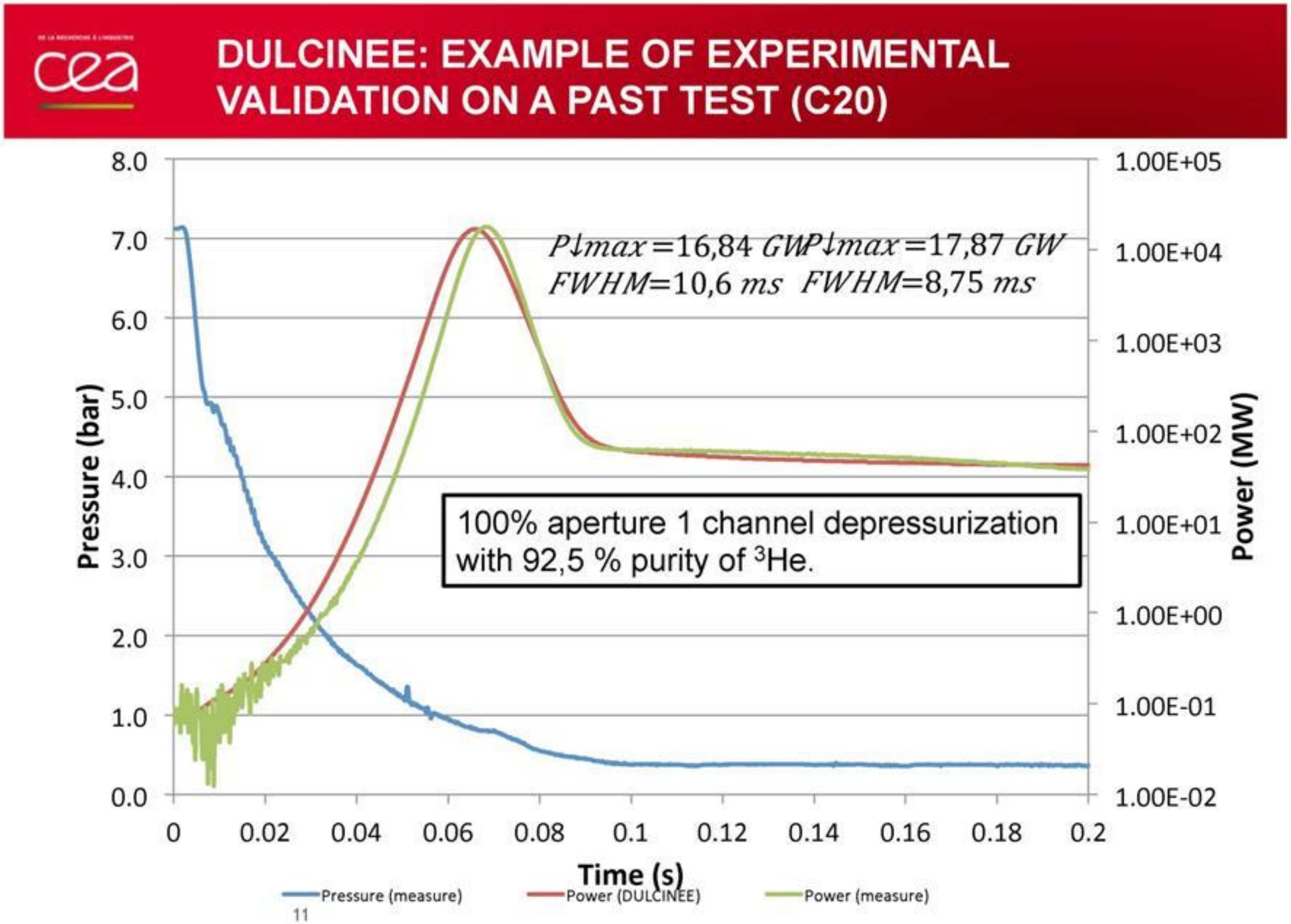
- After every temperature step, the neutron feedbacks can be re-assessed in order to provide the overall system net reactivity as an input of the point kinetics equations computed in DULCINEE

- DULCINEE is validated on 19 past transient tests

TYPICAL DULCINEE CALCULATIONAL RESULTS

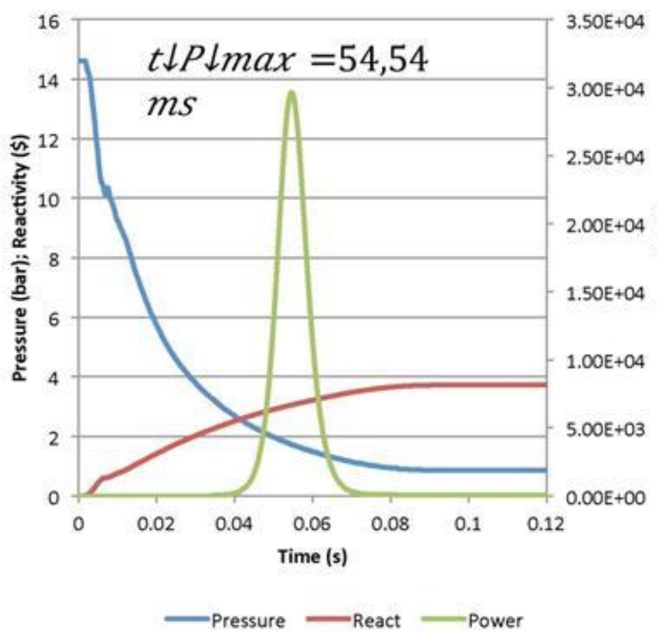


| PAGE 10


DULCINEE: EXAMPLE OF EXPERIMENTAL VALIDATION ON A PAST TEST (C20)
Pressure (bar)
8.0
7.0
6.0
5.0
4.0
3.0
2.0
1.0
0.0
Power (MW)
1.00E+05
1.00E+04
1.00E+03
1.00E+02
1.00E+01
1.00E+00
1.00E-01
1.00E-02
0
0.02
0.04
0.06
0.08
0.1
0.12
0.14
0.16
0.18
0.2
Time (s)
P↓max=16,84 GWP↓max=17,87 GW
FWHM=10,6 ms FWHM=8,75 ms
100% aperture 1 channel depressurization with 92,5 % purity of ³He.
Pressure (measure)
Power (DULCINEE)
Power (measure)
11

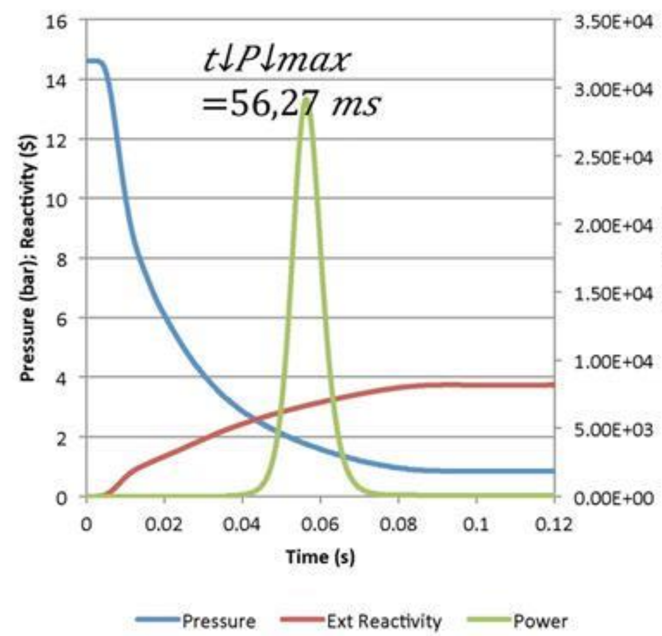
DULCINEE: EFFECT OF THE PRESSURE MEASURE LOCATION ON THE POWER PULSE

DULCINEE calculation of a power transient based on the pressure at the sensor coming from STAR-CCM+



$P{\downarrow}max$=29,64 GW
$FWHM$=8,651 ms

DULCINEE calculation of a power transient based on the pressure in the rods coming from STAR-CCM+



$P{\downarrow}max$=29,14 GW
$FWHM$=8,727 ms

12

THERMOMECHANICS CALCULATION TOOLS (1/2)

- SCANAIR is a "1.5-D code" of IRSN
  - Models a single rod surrounded by a coolant channel
  - Made of a set of three main modules:
    - thermal dynamics (including thermal-hydraulics in the coolant channel),
    - structural mechanics
    - gas behavior
  - The initial rod state is given by FRAPTRAN

- Used for safety assessment of the CABRI test and for interpretation
  - Safety Criteria are
    - Fuel temperature
    - Clad temperature
    - Clad strain

[Ref. 3] A. Moal, V. Georgenthum, O. Marchand, "SCANAIR: A transient fuel performance code Part One: General modeling description," Nuclear Engineering and Design 280 (2014) 150–171

[Ref. 4] V. Georgenthum, A. Moal, O. Marchand, "SCANAIR a transient fuel performance code Part two: Assessment of modeling capabilities," Nuclear Engineering and Design 280 (2014) 172–180

THERMOMECHANICS CALCULATION TOOLS (2/2)

- ALCYONE is a multi-dimensional application [see Ref. 5]
  - The main phenomena considered in ALCYONE are the following:
    - Fuel properties (Thermal conductivity, Thermal expansion coefficient, Elasticity coefficients, Creep, Cracking, Densification, Radial power profiles, Gaseous swelling),
    - Clad properties (Thermal conductivity, Thermal expansion coefficient, Elasticity coefficients, Inelastic clad behavior)
    - Pellet–clad interface (Pellet–clad gap thermal heat transfer and Friction between pellet and cladding)
  - Embeds simplified thermal-hydraulics and neutronics models

- ALCYONE can be coupled to the APOLLO3 neutronics tool [see Ref. 6]

- Ref. 5: J. Sercombe, B. Michel, G. Thouvenin, B. Petitprez, R. Chatelet, D. Leboulch, C. Nonon, "Multi-dimensional modeling of PCMI during base irradiation and ramp testing with ALCYONE V1.1," Proc. Top Fuel Conference, Paris, France (2009)
- Ref. 6: JC. Le Pallec et al., "Neutronics/fuel thermomechanics coupling in the framework of a REA transient scenario calculation, "PHYSOR 2016 – Unifying Theory and Experiments in the 21st Century Sun Valley Resort, Sun Valley, Idaho, USA, May 1 – 5, 2016

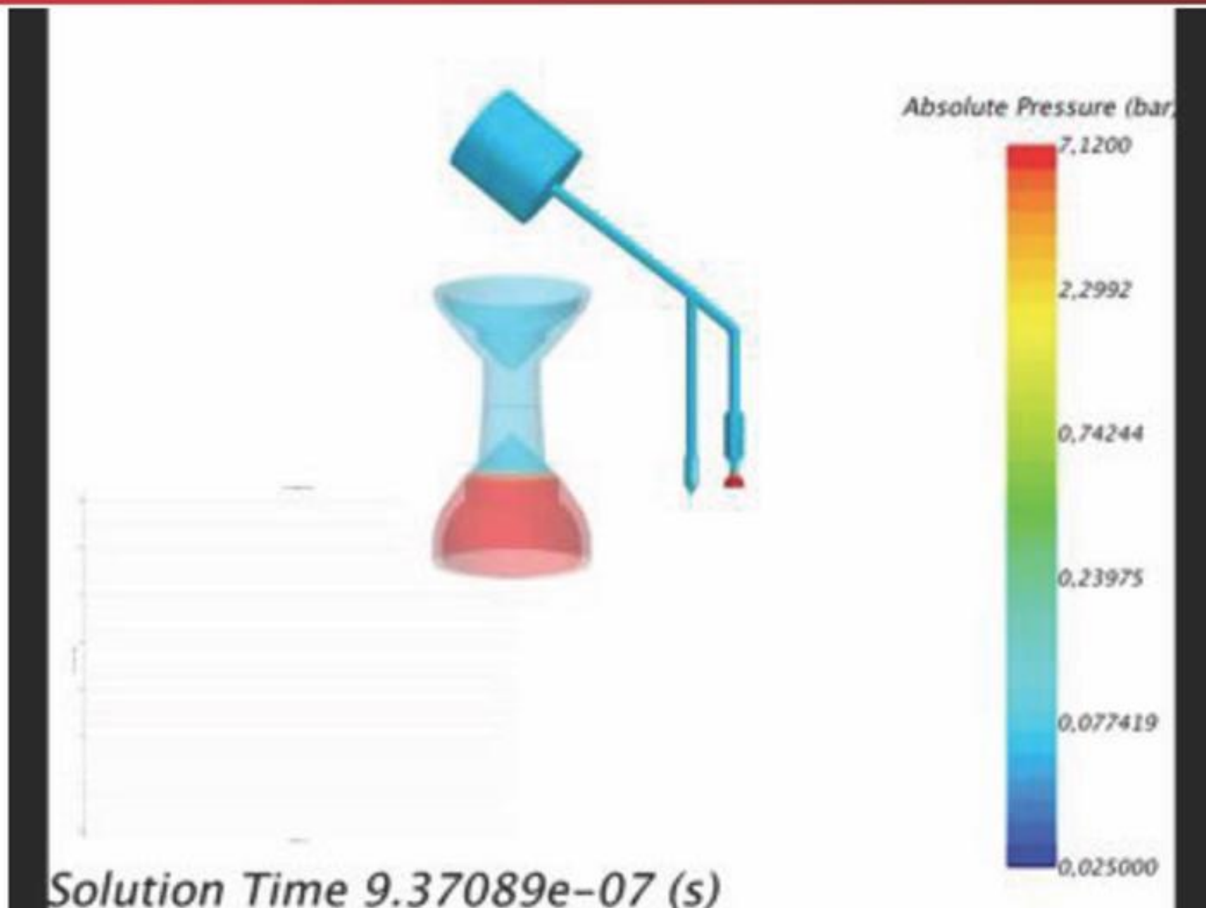
THERMAL-HYDRAULICS CODES

- CFD models (STAR-CCM+ or TRIO-U)
  - Transient rods depressurization (see Ref.7)
  - Core thermalhydraulics
    - Siphon-breaking in case of rupture of primary piping
    - Flow exchange between the core and the pool
  - Primary cooling system
    - Imbalance of the primary cooling system study (see Ref. 8)

- CATHARE modeling of the primary cooling system
  - Primary pump shutdown
  - Siphon-breaking in case of rupture of primary piping

- Ref. 7, O. Clamens, J. Lecerf, T. Cadiou, B. Duc, B. Biard, JP. Hudelot, "Assessment of the CABRI transients power shape by using CFD and point kinetics codes", PHYSOR 2016 – Unifying Theory and Experiments in the 21st Century Sun Valley Resort, Sun Valley, Idaho, USA, May 1 – 5, 2016, on CD-ROM (2016)
- Ref. 8, JP. Hudelot, Y. Garnier, J. Lecerf, M. Fournier, S. Magnetto, E. Gohier, "CABRI reactor commissioning: results and analysis of the tests on the primary cooling system and on the control and safety rods", IGORR 2014 conference, 17-21 November 2014, Bariloche, Argentina

EXAMPLE OF DEPRESSURIZATION FROM 7BAR, WITH OPENING OF THE LARGE FAST-VALVE



PRIMARY COOLING SYSTEM MODELING WITH STAR-CCM+

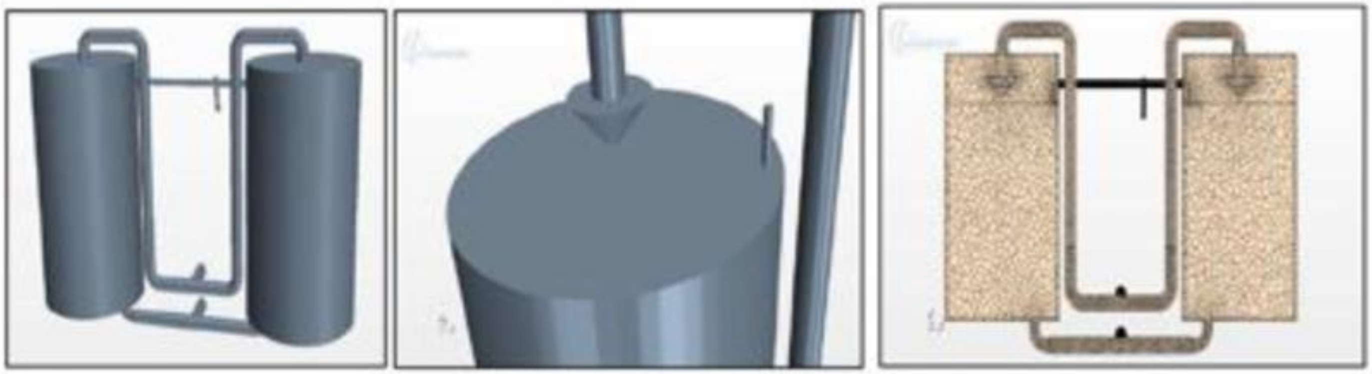
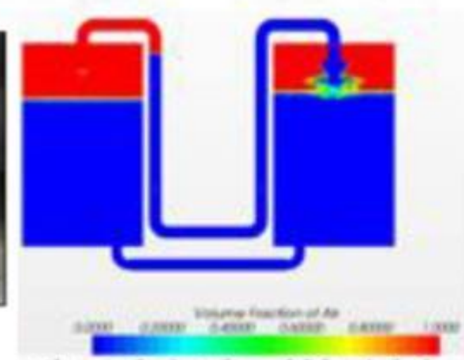


Figure 12: STAR-CCM+ geometries: outside view (left); discharge cones (center); polyhedralmeshing (right)

Figure 7: Outside view of the water tanks the primary cooling system of CABRI (left) – view of the discharge "T" pipe (center) - discharge cone of and holed cylindrical plate (right)



ıction of air after 300s (air in ı

CFD MODELING OF THE IN CORE PRIMARY CIRCUIT

Figure 16 : Représentation du maillage des parties internes du réacteur

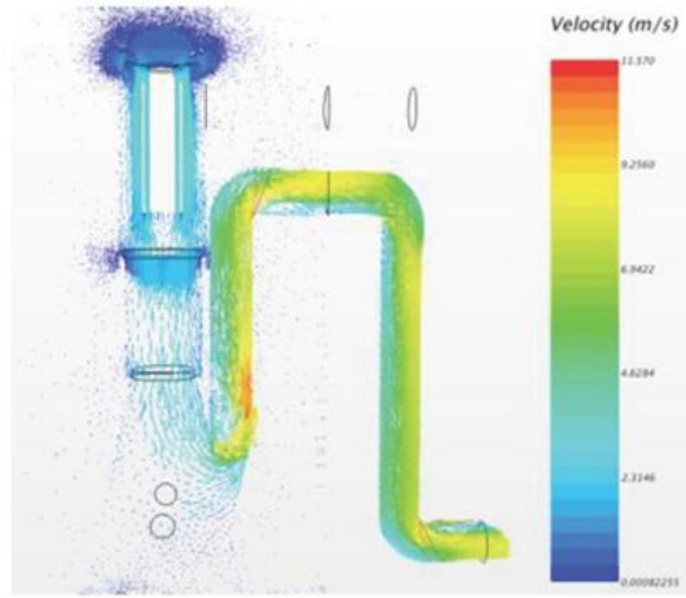


Figure 20 : Vecteur vitesse dans le « circuit primaire »

CFD CALCULATION OF FLOWRATE INSIDE THE CORE

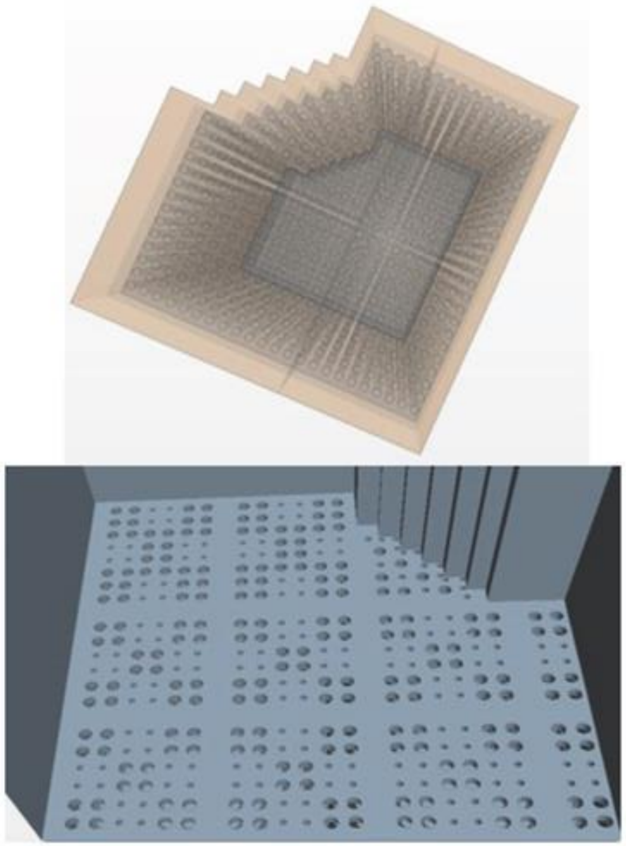


Figure 25 : Plaque de mélange modélisée avec STARCCM+

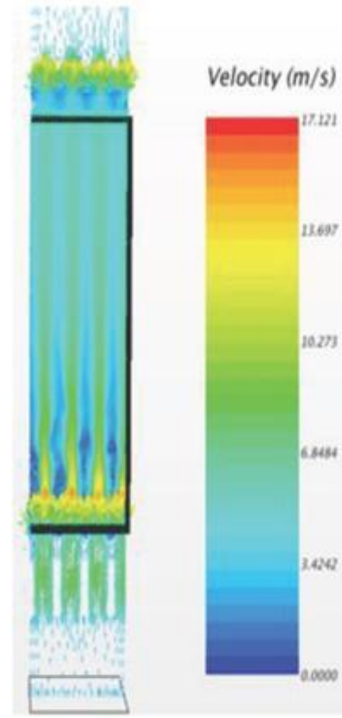
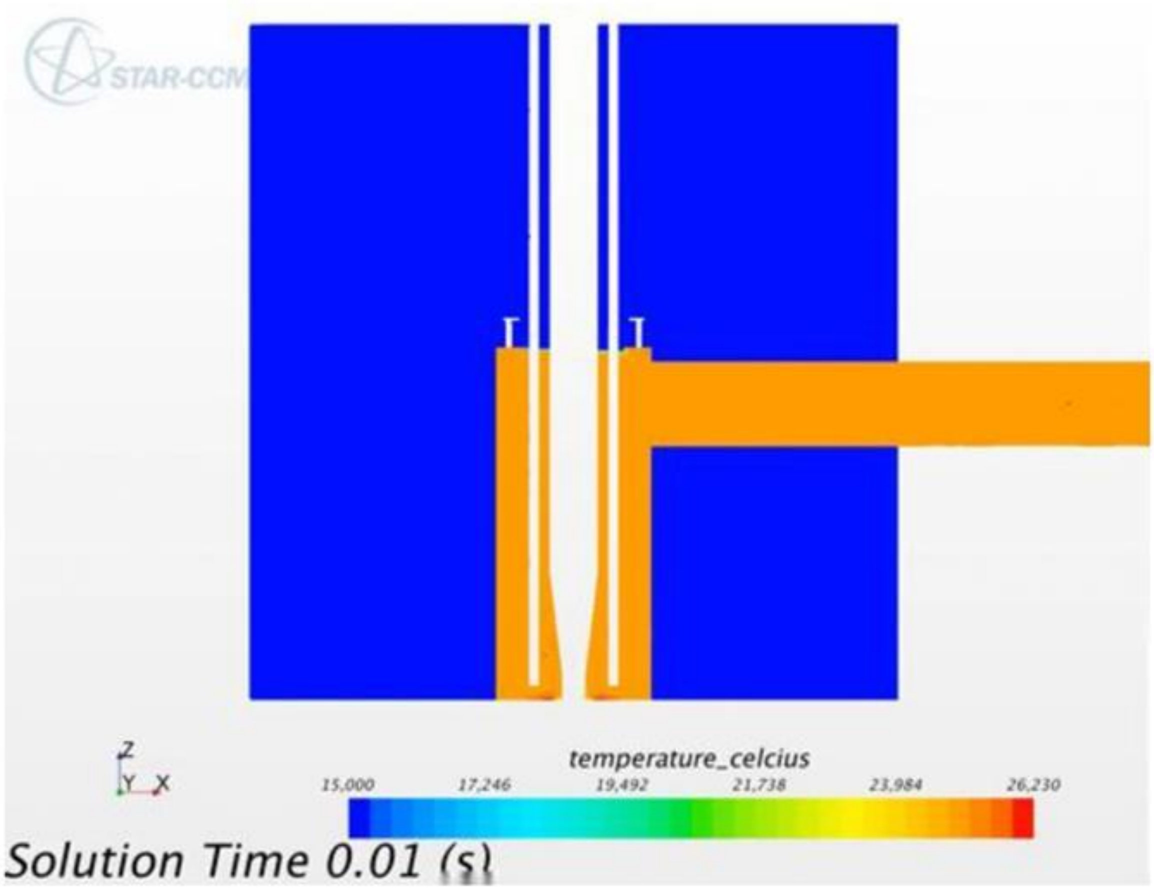


Figure 26 : Représentation des vecteurs vitesses

FLOW EXCHANGE BETWEEN THE CORE AND THE POOL



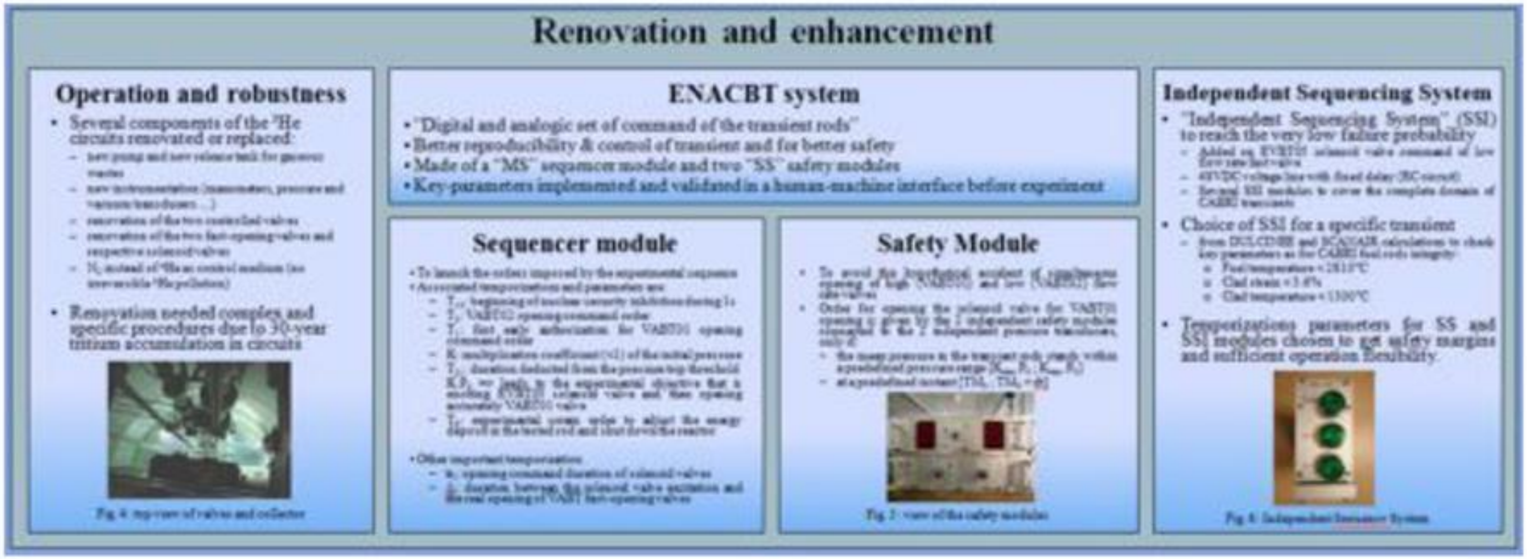
START UP TESTS

Organization of the Recommisioning

Commission 1	**Neutronics characterization of the core**
Commission 2	Ventilation
Commission 3	Reactor vessel
Commission 4	Handling
Commission 5A	Conventional circuits
Commission 5B	Circuits of the pressurized water loop
Commission 6	Instrumentation and control
Commission 7	**General operation during steady and transient power tests**

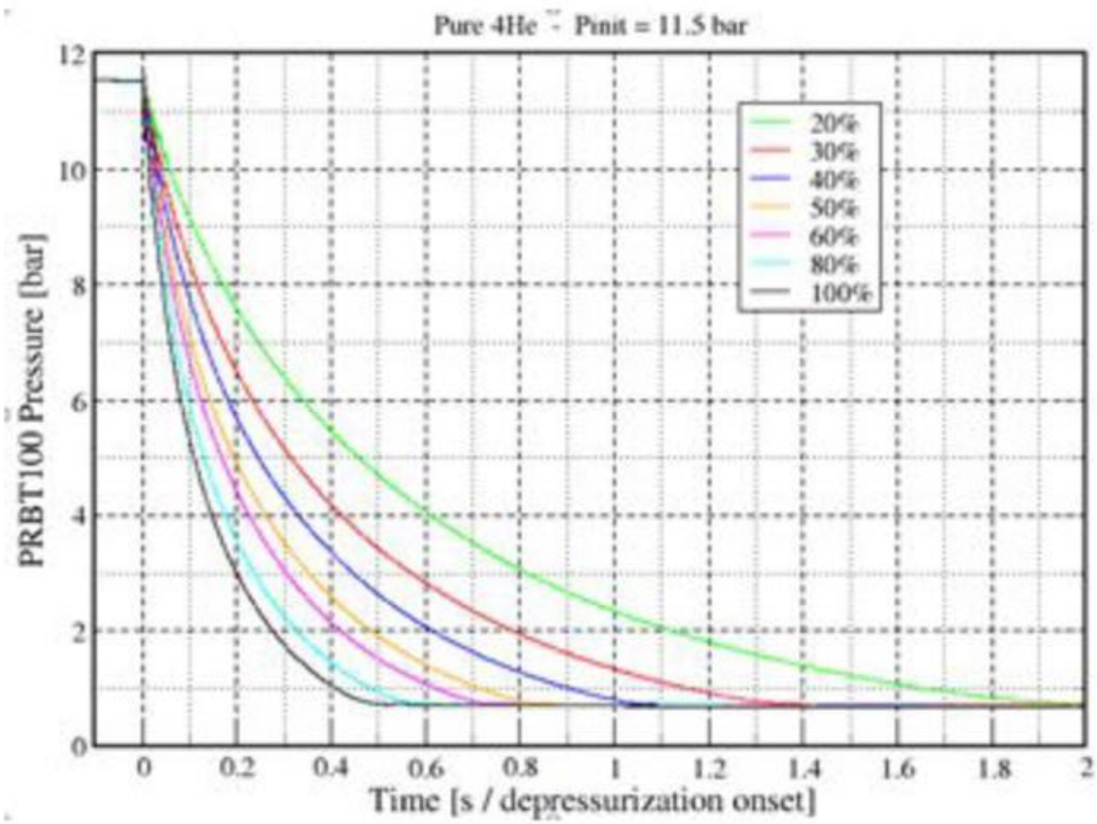
- Commission 1: 80% finished; end in May 2016
- Commission 5B: continued and finished in mid-2016
- Commission 7: end before the end of 2016

RENOVATION AND ENHANCEMENT OF THE TRANSIENT ROD SYSTEM

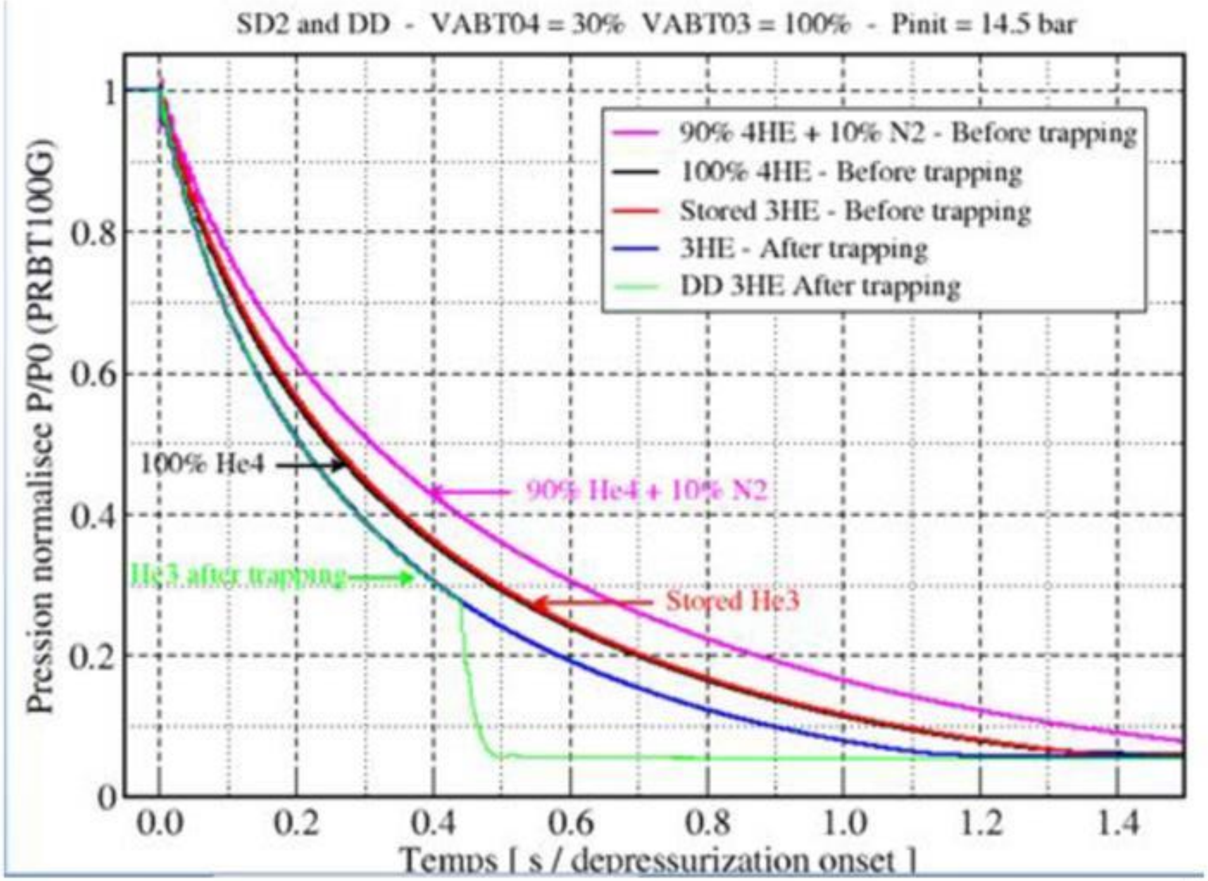


Ref. 9, B. Duc, B. Biard, P. Debias, L. Pantera, JP. Hudelot, F. Rodiac, "Renovation, improvement and experimental validation of the Helium-3 transient rods system for the reactivity injection in the CABRI reactor" ", IGORR 2014 conference, 17-21 November 2014, Bariloche, Argentina

CONTROL OF THE DEPRESSURIZATION SEQUENCE



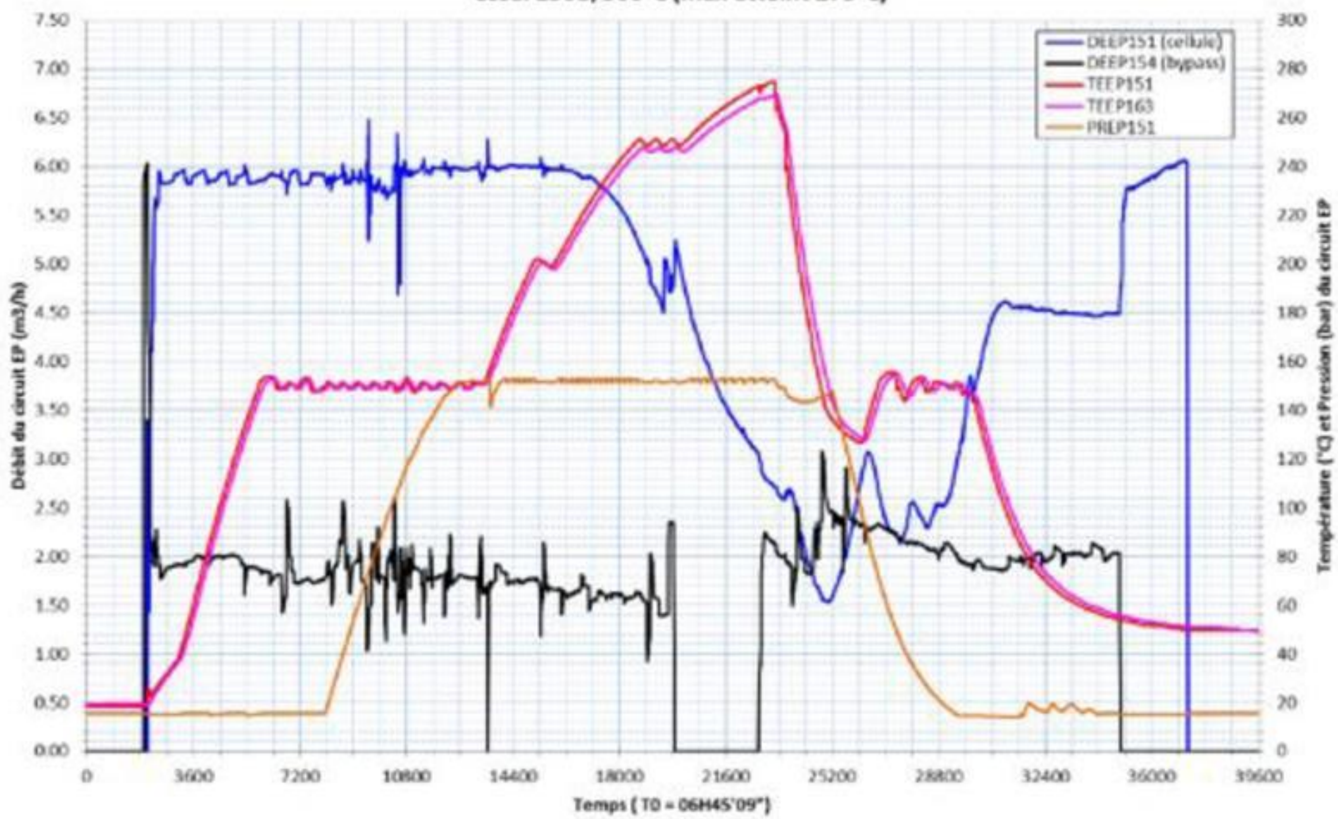
OPTIMIZATION OF THE REACTIVITY INJECTION

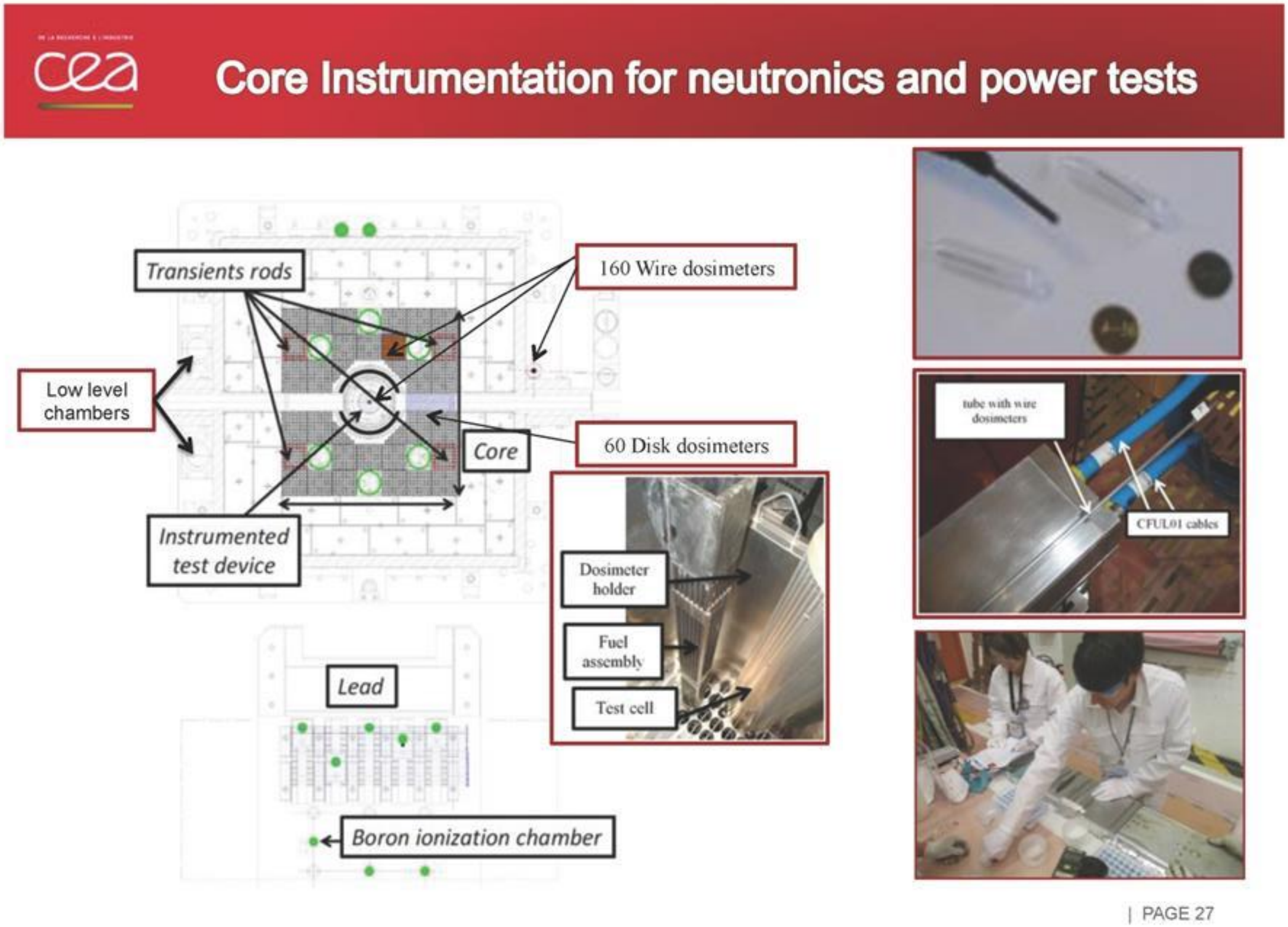


| PAGE 25

WATERLOOP VALIDATION TESTS (MAY 2014)

Fonctionnement de la POEP151 du 13/05/2014
essai 150b/300°C (max atteint 275°C)




cea
Core Instrumentation for neutronics and power tests
Transients rods
Low level chambers
Instrumented test device
160 Wire dosimeters
60 Disk dosimeters
Core
Lead
Boron ionization chamber
Dosimeter holder
Fuel assembly
Test cell
tube with wire dosimeters
CFUL01 cables
| PAGE 27

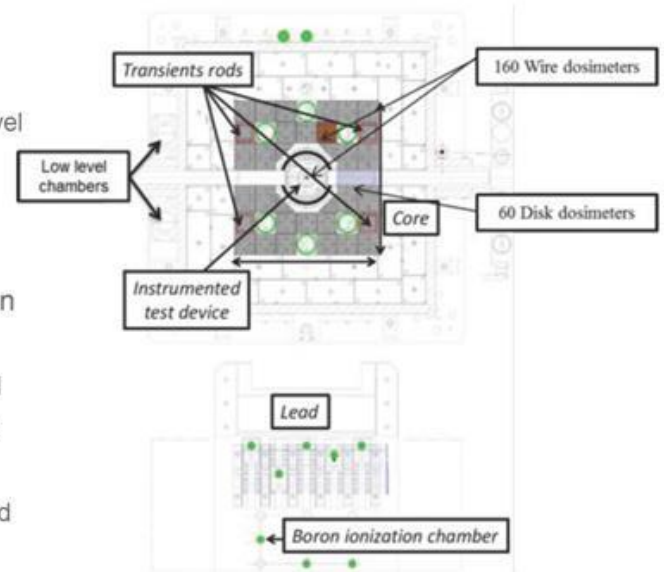
Commission 1: Neutronics Characterization of the Core

- Precise characterization of the neutronics parameters of the core
  - At low power (< 100 kW) at ambient temperature
  - Ability to achieve target performances towards testing and safety margins
  - To validate the preliminary design and safety neutronics calculations

Neutronics parameters	Measurement technique	Target uncertainty (1σ)
Critical height	Critical state	± 1mm
Integral reactivity worth of control and safety rods	MSA/MSM & « *rod drop* »	± 4%
Differential reactivity worth of control and safety rods	Doubling time	± 1%
Core power distribution	Dosimetry	± 2%
Isothermal temperature coefficient	Critical state	± 1 pcm/°C
Reactivity effects inside the water loop	Critical state	± 5%
Axial flux profile	Dosimetry	± 2%
Core stacking reactivity worth	Critical state	± 5%
Effective fraction of delayed neutrons	Rossi and Feynman-α methods	± 3%
Effective prompt neutron lifetime		± 3%
Axial distribution and integral of fission rate in the core	Dosimetry	± 2%
Reactivity worth of the ^{3}He transient rods	Critical state	± 5%
Integral reactivity worth of control and safety rods	MSA/MSM & « *rod drop* »	± 4%
Differential reactivity worth of control and safety rods	Doubling time	± 1%

Commission 7: Core Power and Energy Deposit Determination

- Core power by heat balance method
  - Measurement of the primary coolant heating for an accurate monitored flowrate during steady states of power
  - Current delivered by boron neutron detectors is calibrated vs. the power measured by heat balance
  - only possible up to a 25MW power (maximum deliverable power level for a steady state operation of CABRI)
  - uncertainty around 4% (2σ) is expected.



- Linearity of experimental neutron detectors is good between 25MW and 25GW
  - Transient power (from 100kW to ~25GW) is recorded via 5 identical boron neutron detectors are positioned at 5 different distances from the core
  - Comparison of the activities of ^{59}Co and ^{197}Au dosimeters generated during heat balance measurements and during start up power transients, targeting a same injected energy in the driver core (270MJ) for both measurements

→ minimize the experimental uncertainty on absolute power measurements

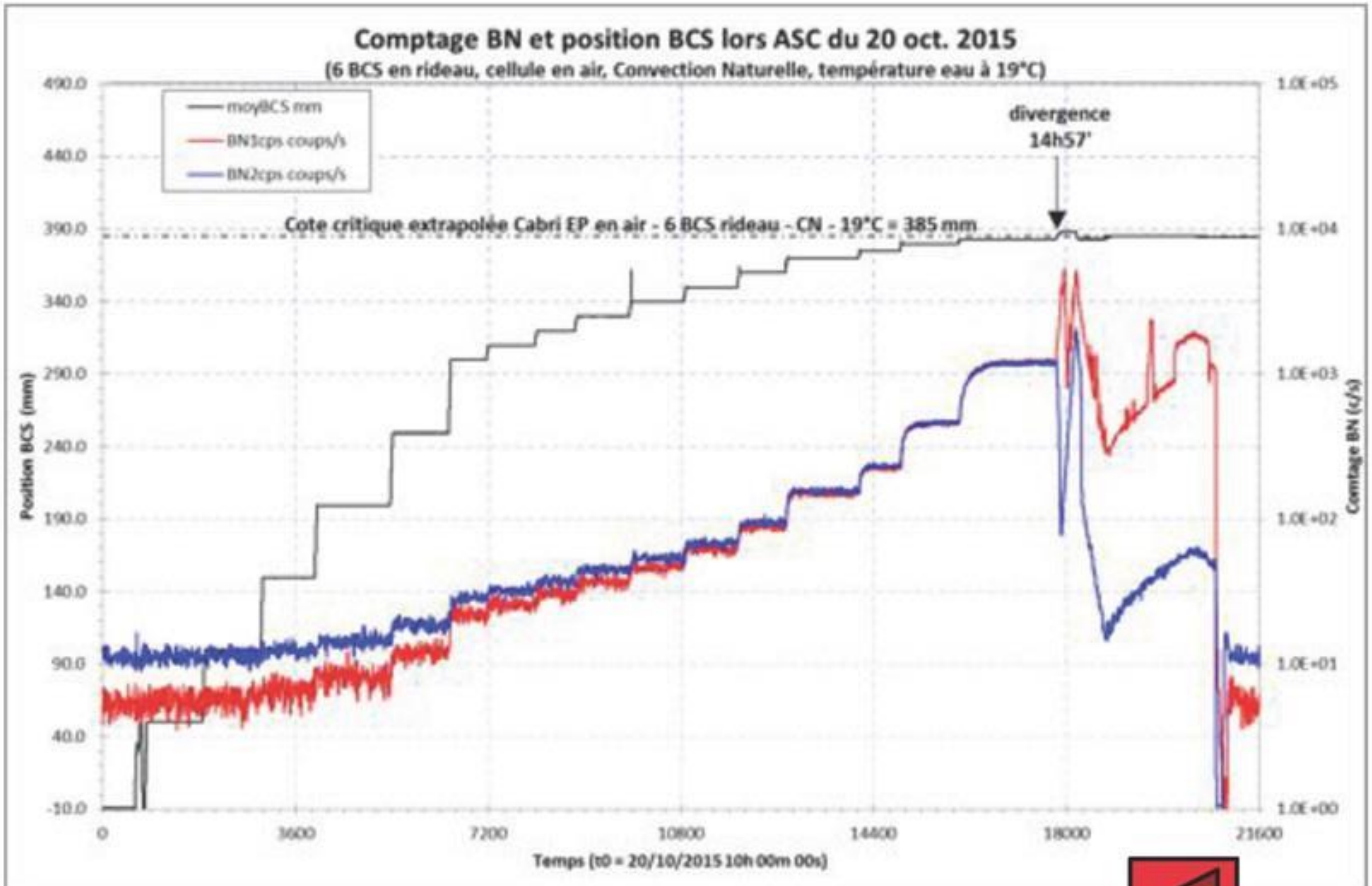
→ strengthen the estimation of the uncertainty on the energy deposit during transients.

Conclusion

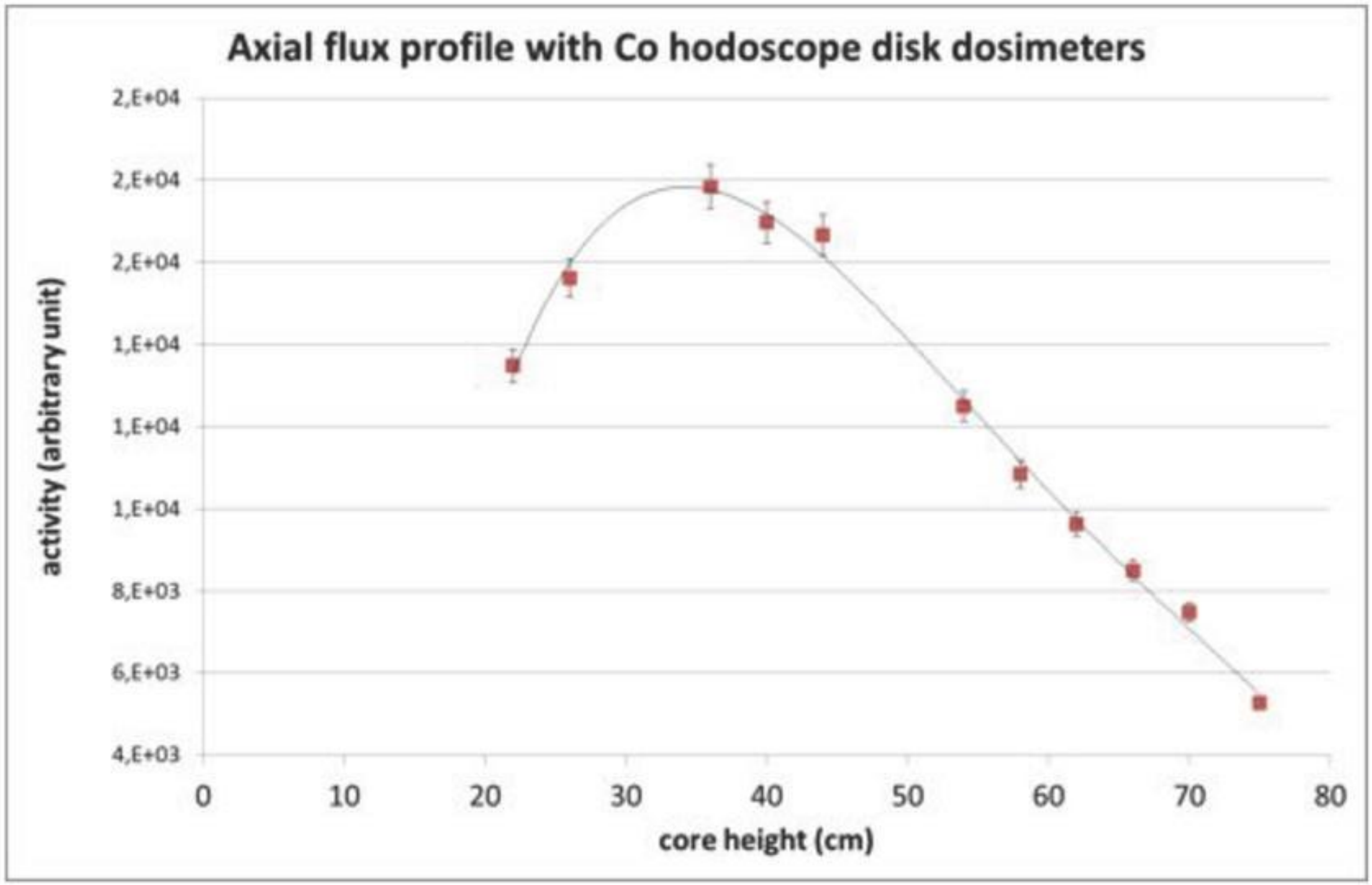
- First criticality of CABRI / Waterloop succeeded
- Neutronics tests will be ended in May 2016
- Steady and transient power tests: end of 2016
- CIP-Q test in 2^{nd} semester of 2017
- CIP program during 5 years

Thank you for your attention

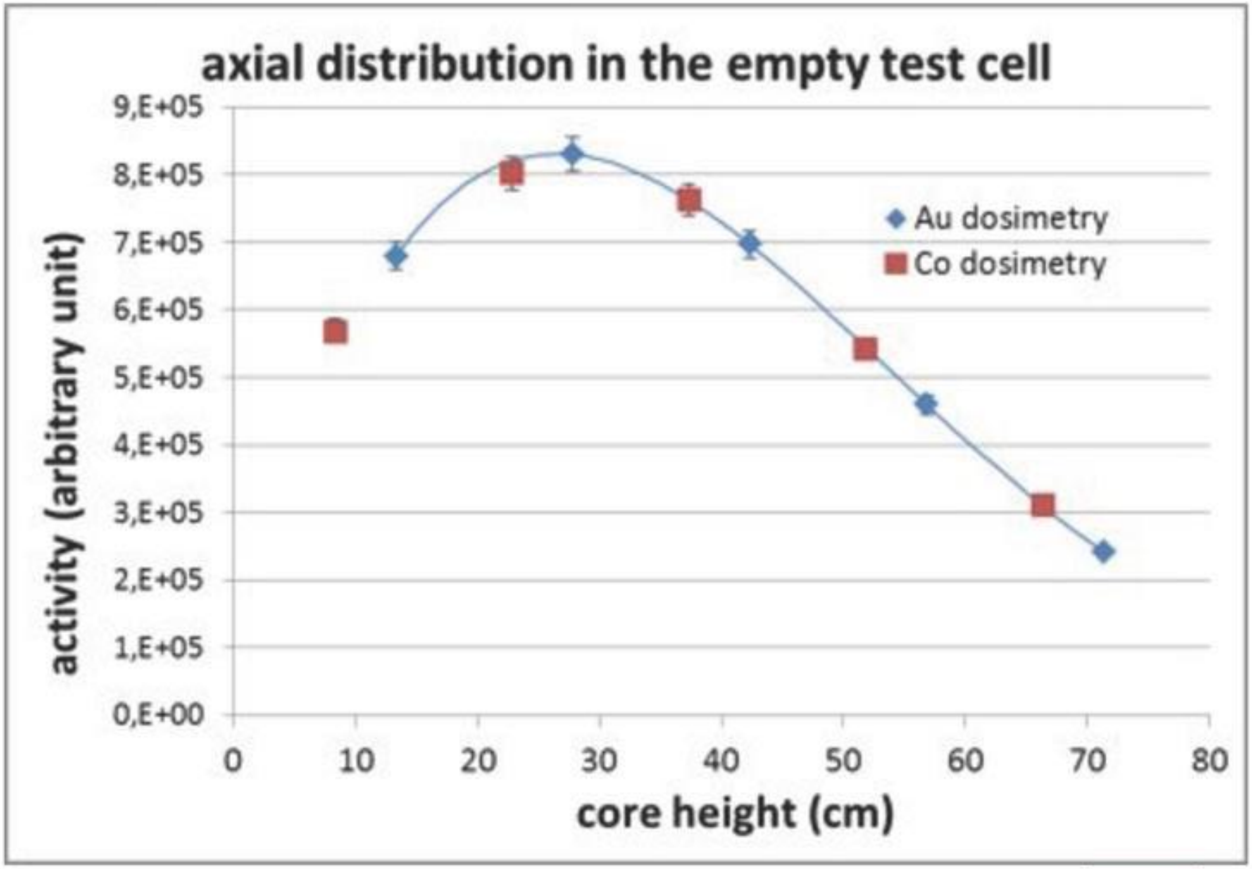
1st criticality of CABRI: October 20th, 2015



Axial flux distribution close to the hodoscope



Axial flux distribution in the test cell



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Appendix B

Presentations Slides

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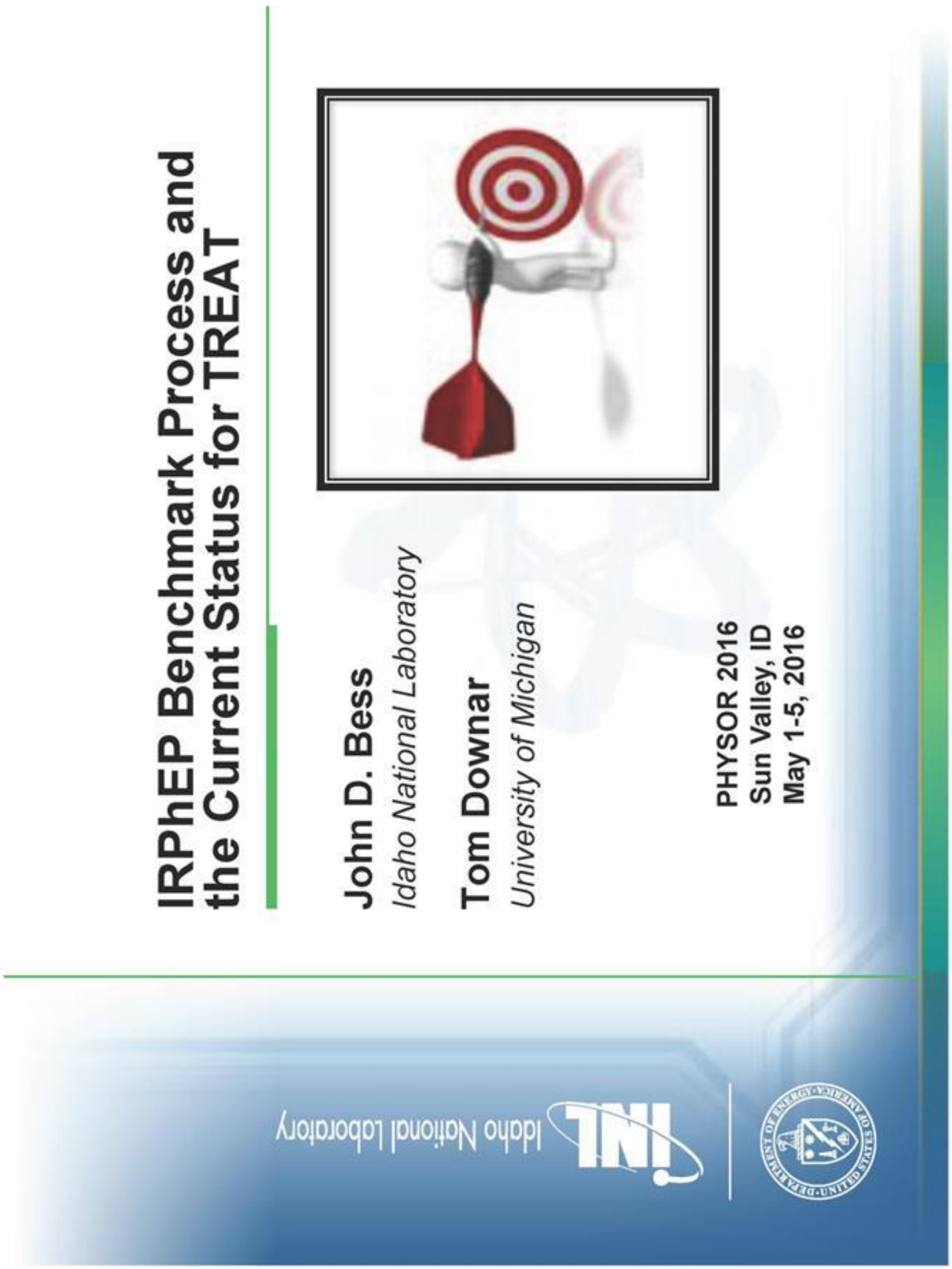
IRPhEP Benchmark Process and the Current Status for TREAT
John D. Bess
Idaho National Laboratory
Tom Downar
University of Michigan
PHYSOR 2016
Sun Valley, ID
May 1-5, 2016
Idaho National Laboratory

Outline

- US Benchmarking Efforts
- INL Benchmarking Efforts
- Minimum Critical Mass Core
- M8CAL Core
- Publication in IRPhEP
- Where Do We Stand

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US Benchmarking Efforts

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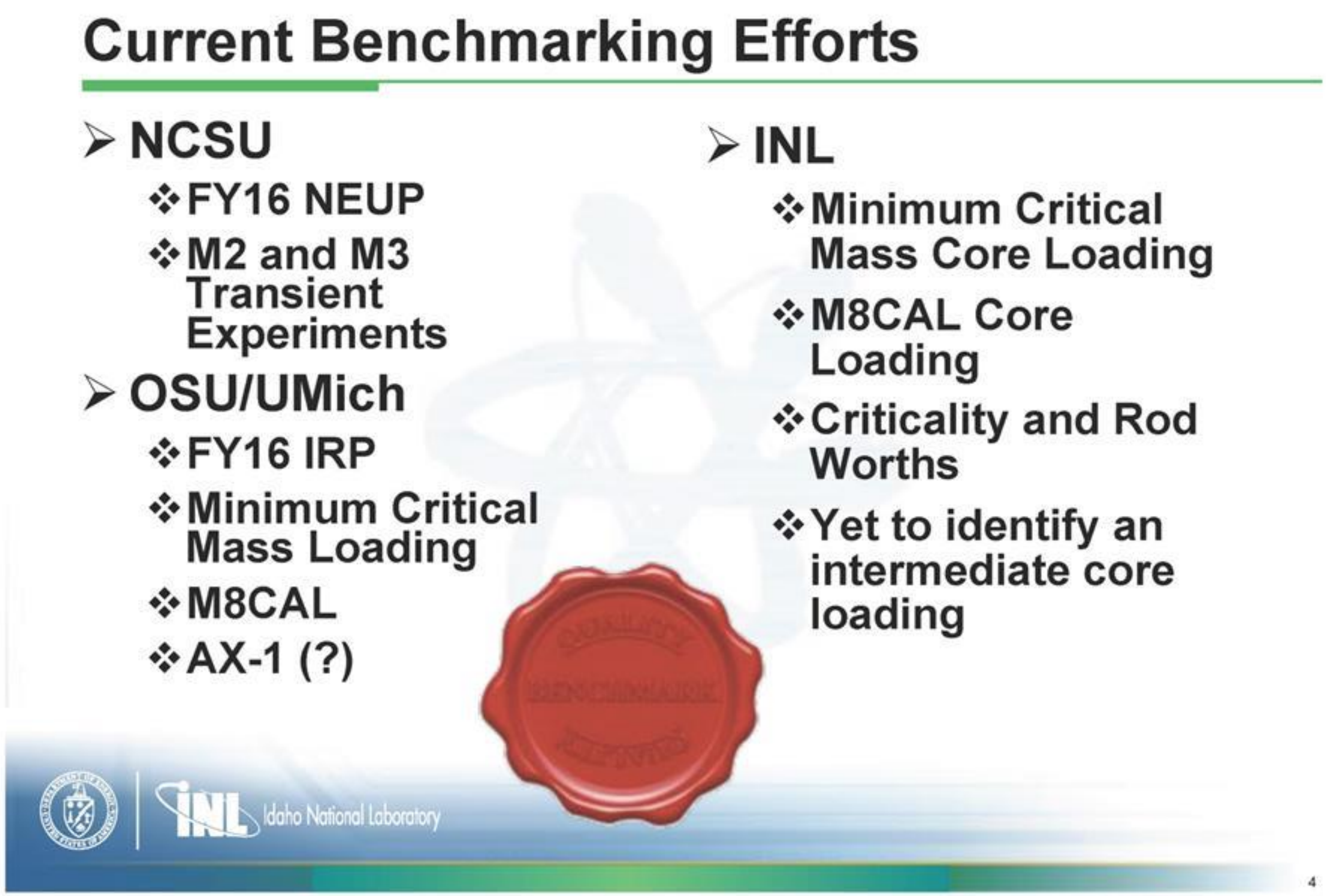
Current Benchmarking Efforts
NCSU
FY16 NEUP
M2 and M3 Transient Experiments
OSU/UMich
FY16 IRP
Minimum Critical Mass Loading
M8CAL
AX-1 (?)
INL
Minimum Critical Mass Core Loading
M8CAL Core Loading
Criticality and Rod Worths
Yet to identify an intermediate core loading
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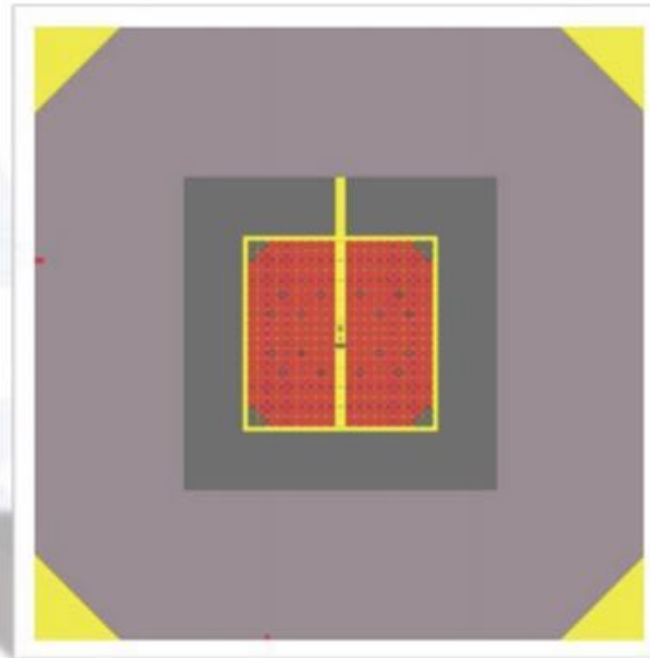
NCSU – February 2016 Status Update

- Collecting M2/3 calibration test data
- Calculating graphite thermal neutron scattering cross sections
- Serpent Monte Carlo model of M2 calibration test



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UMich – February 2016 Status Update

- Serpent model of Minimum Critical Mass core loading
  - Used parameters from BATMAN report
- Performing sensitivity analyses
  - Boron in fuel
  - Graphitization of fuel
  - # Zr-clad dummy fuel
  - B/Fe in reflector
  - Cross sections effect
- Developing deterministic models of TREAT
  - PARCS
  - PROTEUS
- Developing Monte Carlo transient capability in OPENMC

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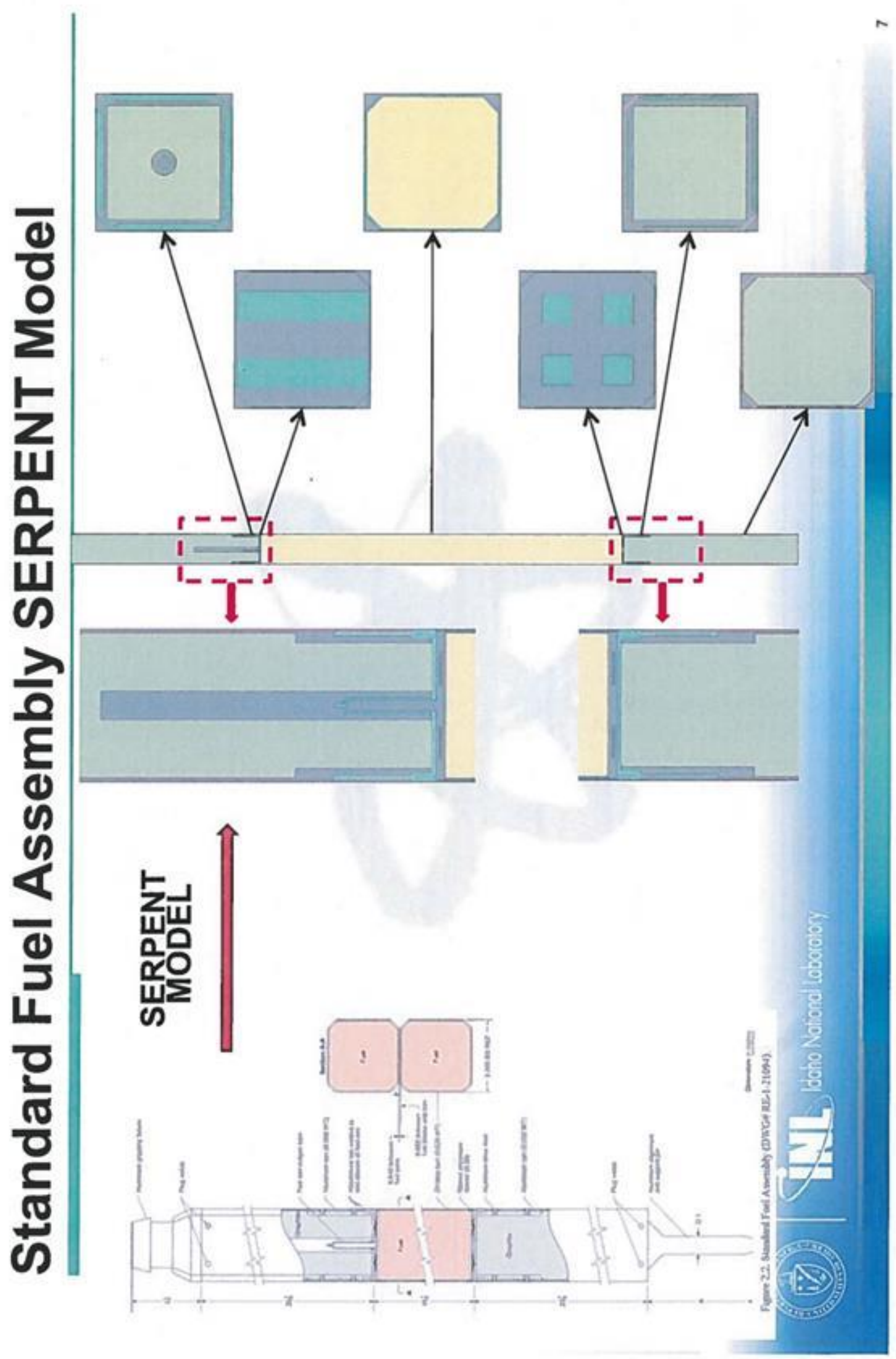
Standard Fuel Assembly SERPENT Model
SERPENT MODEL
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Parametrics with the SERPENT Core Model

Case	5.9ppm Boron	Diff (pcm)
Base: 7.6ppm Boron** 600ppm Fe *** 59%Graphitization ENDF/B-7.1 16 Zr Assembly	1.00130 ±19pcm	-
5.9ppm Boron 267ppm Fe	**1.01846 ±23pcm***	1716
5.9ppm Boron 267ppm Fe 100%Graphitization	1.00394 ±23pcm	1452
5.9ppm Boron 267ppm Fe 0 Zr Assembly	1.01639±21pcm	207

* Used as basis for PARCS Model
** Was used in all previous analyses
*** Argonne work, (Brittan's memo)

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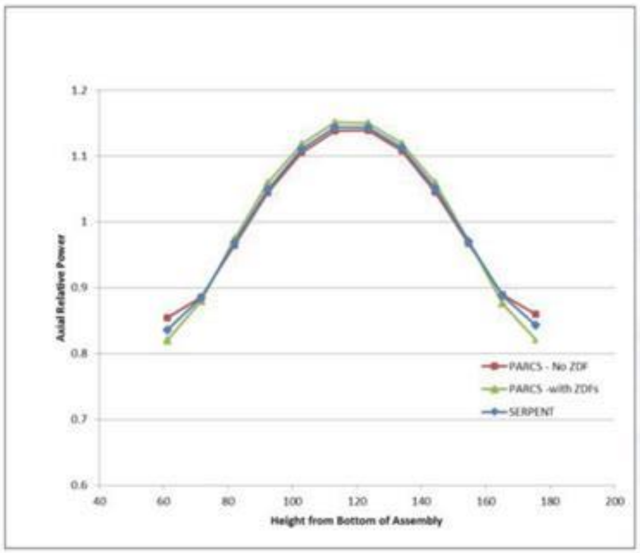
Single Assembly Axial Power Shape



- Serpent
  - k_{eff}:1.42291 ±29pcm
- PARCS without ZDF:
  - 1.44389
- PARCS with ZDF:
  - 1.42201

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Ongoing Work

- Generate Monte Carlo MCC solutions with MCNP (ANL Model) and with OPENMC (MIT Model)
- Compare solutions on MCC for all Monte Carlo solutions w/ INL and ANL
- Perform same analysis for M8CAL w/ SERPENT and SERPENT/PARCS as performed with MCC
- Complete UQ/Sensitivity Analysis on MCC and M8CAL
- Complete work on the PARCS model of M8CAL
- Perform V&V on the coupled PARCS/AGREE thermal-fluids model for TREAT steady-state and transient applications.
- Complete IRPhE Benchmark Specifications

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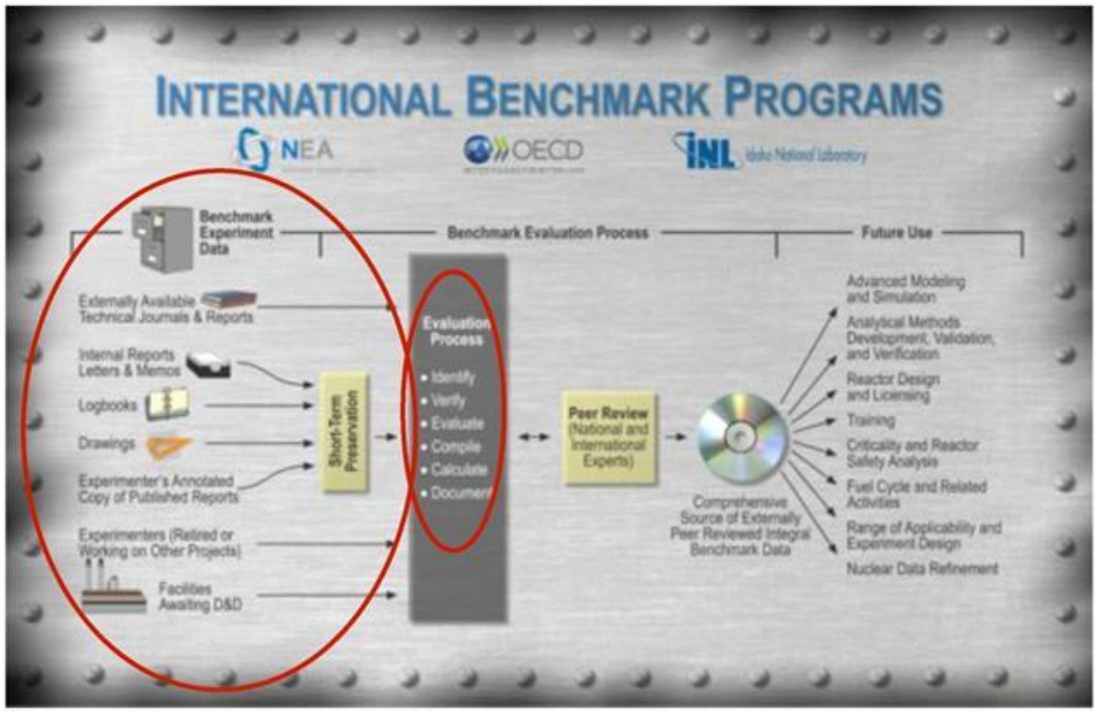
INL Benchmark Efforts

Progress of INL Benchmark Efforts



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Current Status for INL = Delayed

- Limited personnel time available
  - TREAT restart efforts
  - TREAT experiment design
  - IRPhEP benchmark project schedule/limitations
- Current activities
  - Data acquisition
  - Detailed model completion
  - Benchmark model description

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Baseline Assessment of TREAT for Modeling and Analysis Needs

- INL/EXT-15-35372
- October 2015
- One-Stop-Shop
  - Drawings
  - Materials
- Modeling and simulation
- NOT A BENCHMARK
  - But still very useful
    - For doing a benchmark

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TREAT Restart Benchmark Needs

- Three Core Configurations
  - Minimum Critical Mass (Small)
  - TBD (Medium)
  - M8CAL (Large)
- Experimental Data Validation
  - Criticality
  - Control Rod Worth
  - Excess Reactivity
  - Shutdown Margin
- Purpose
  - Understand uncertainties in the TREAT core that would impact operations
  - Validation of nuclear data and codes utilized to support TREAT operations
- Publication in IRPhEP Handbook

Minimum Critical Mass Core

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TREAT Minimum Critical Mass Core Loading

- 122 Standard Fuel Assemblies
- 11 Thermocoupled Fuel Assemblies
- 8 Control Rod Assemblies
  - Shortened Control Rods
    - 18 in., ~46 cm, reduction
- 220 Dummy Assemblies
  - 1 Source Assembly
  - *~40 Zr-Clad*
  - *~179 Al-Clad*
- 361 Assemblies Total

TREAT Minimum Critical Mass Core Loading

	A	B	C	D	E	F	G	H	J	K	L	M	N	O	P	R	S	T	U
1	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A
2	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A
3	A	A	A	A	A	A	Z	Z	Z	A	Z	Z	Z	A	A	A	S	A	A
4	A	A	A	A	A	Z	F	F	F	Z	F	F	T1	Z	A	A	A	A	A
5	A	A	A	A	Z	F	F	F	F	F	F	F	F	F	Z	A	A	A	A
6	A	A	A	Z	F	F	F	C	F	F	F	C	F	F	F	Z	A	A	A
7	A	A	Z	F	F	F	F	F	F	T2	F	F	F	F	F	F	Z	A	A
8	A	A	Z	F	F	C	F	F	F	F	F	F	F	C	F	F	Z	A	A
9	A	A	Z	F	F	F	F	F	F	F	F	F	F	F	F	F	Z	A	A
10	A	A	A	Z	F	F	F	F	F	T1	T3	T3	T3	T4	T3	Z	A	A	A
11	A	A	Z	F	F	F	F	F	F	T4	F	F	F	F	F	F	Z	A	A
12	A	A	Z	F	F	C	F	F	F	T4	F	F	F	C	F	F	Z	A	A
13	A	A	Z	F	F	F	F	F	F	F	F	F	F	F	F	F	Z	A	A
14	A	A	A	Z	F	F	F	C	F	T3	F	C	F	F	F	Z	A	A	A
15	A	A	A	A	Z	F	F	F	F	F	F	F	F	F	Z	A	A	A	A
16	A	A	A	A	A	Z	F	F	F	Z	F	F	F	Z	A	A	A	A	A
17	A	A	A	A	A	A	Z	Z	Z	A	Z	Z	Z	A	A	A	A	A	A
18	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A
19	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A

A	Al-Clad Dummy Assembly
C	Control Rod Fuel Assembly (Short Poison Section)
F	Standard Fuel Assembly
S	Zr-Clad Source Assembly
T1	4-Fuel Thermocouple Assembly
T2	Reflector-Fuel Gradient Thermocouple Assembly
T3	Midfuel, Reflector & Skin Thermocouple Assembly
T4	Transient Thermocouple Assembly
Z	Zr-Clad Dummy Assembly

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Comparison to First Start-Up Core

- July 7, 1959
- 133 Standard Fuel Assemblies
- 16 Thermocoupled Fuel Assemblies
- 8 Control Rod Assemblies
  - Standard Control Rods

TREAT Original Startup Core Loading

	A	B	C	D	E	F	G	H	J	K	L	M	N	O	P	R	S	T	U
1	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A
2	A	A	A	A	A	A	A	A	Z	Z	Z	A	A	A	A	A	A	A	A
3	A	A	A	A	A	A	Z	Z	F	T4	F	Z	Z	A	A	A	S	A	A
4	A	A	A	A	A	Z	F	F	F	F	F	F	T1	Z	A	A	A	A	A
5	A	A	A	A	Z	F	F	F	F	F	F	F	F	F	Z	A	A	A	A
6	A	A	A	Z	F	F	F	C	F	F	F	C	F	F	F	Z	A	A	A
7	A	A	Z	F	F	F	F	F	F	T2	F	F	F	F	F	F	Z	A	A
8	A	A	Z	F	F	C	F	F	F	F	F	F	F	C	F	F	Z	A	A
9	A	Z	F	F	F	F	F	F	F	F	F	F	F	F	F	F	F	Z	A
10	A	Z	T3	F	F	F	F	F	F	T1	T3	T3	T3	T4	T3	T3	T1	Z	A
11	A	Z	F	F	F	F	F	F	F	T4	F	F	F	F	F	F	F	Z	A
12	A	A	Z	F	F	C	F	F	F	T4	F	F	F	C	F	F	Z	A	A
13	A	A	Z	F	F	F	F	F	F	F	F	F	F	F	F	F	Z	A	A
14	A	A	A	Z	F	F	F	C	F	T3	F	C	F	F	F	Z	A	A	A
15	A	A	A	A	Z	F	F	F	F	F	F	F	F	F	Z	A	A	A	A
16	A	A	A	A	A	Z	F	F	F	F	F	F	F	Z	A	A	A	A	A
17	A	A	A	A	A	A	Z	Z	F	T3	F	Z	Z	A	A	A	A	A	A
18	A	A	A	A	A	A	A	A	Z	Z	Z	A	A	A	A	A	A	A	A
19	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A

A	Al-Clad Dummy Assembly
C	Control Rod Fuel Assembly (Short Poison Section)
F	Standard Fuel Assembly
S	Zr-Clad Source Assembly
T1	4-Fuel Thermocouple Assembly
T2	Reflector-Fuel Gradient Thermocouple Assembly
T3	Midfuel, Reflector & Skin Thermocouple Assembly
T4	Transient Thermocouple Assembly
Z	Zr-Clad Dummy Assembly

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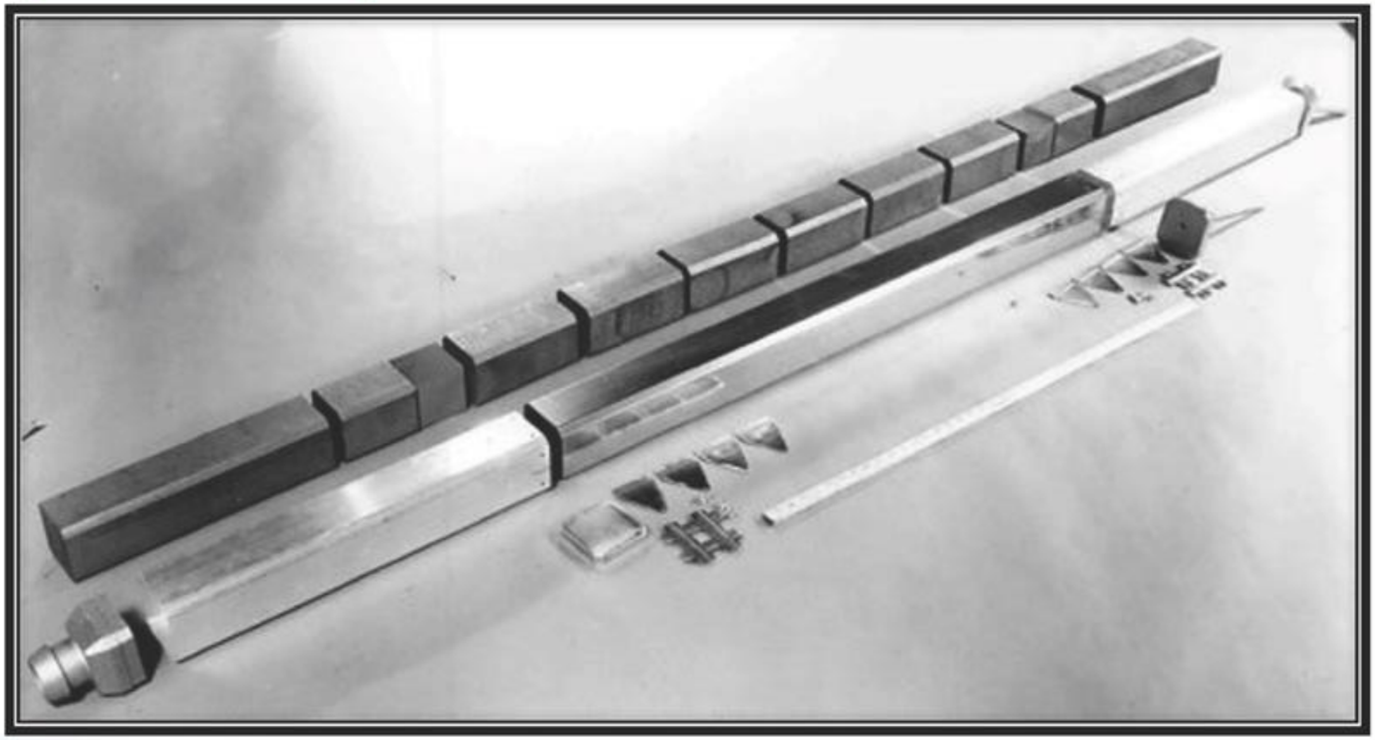
TREAT Standard Fuel Assembly



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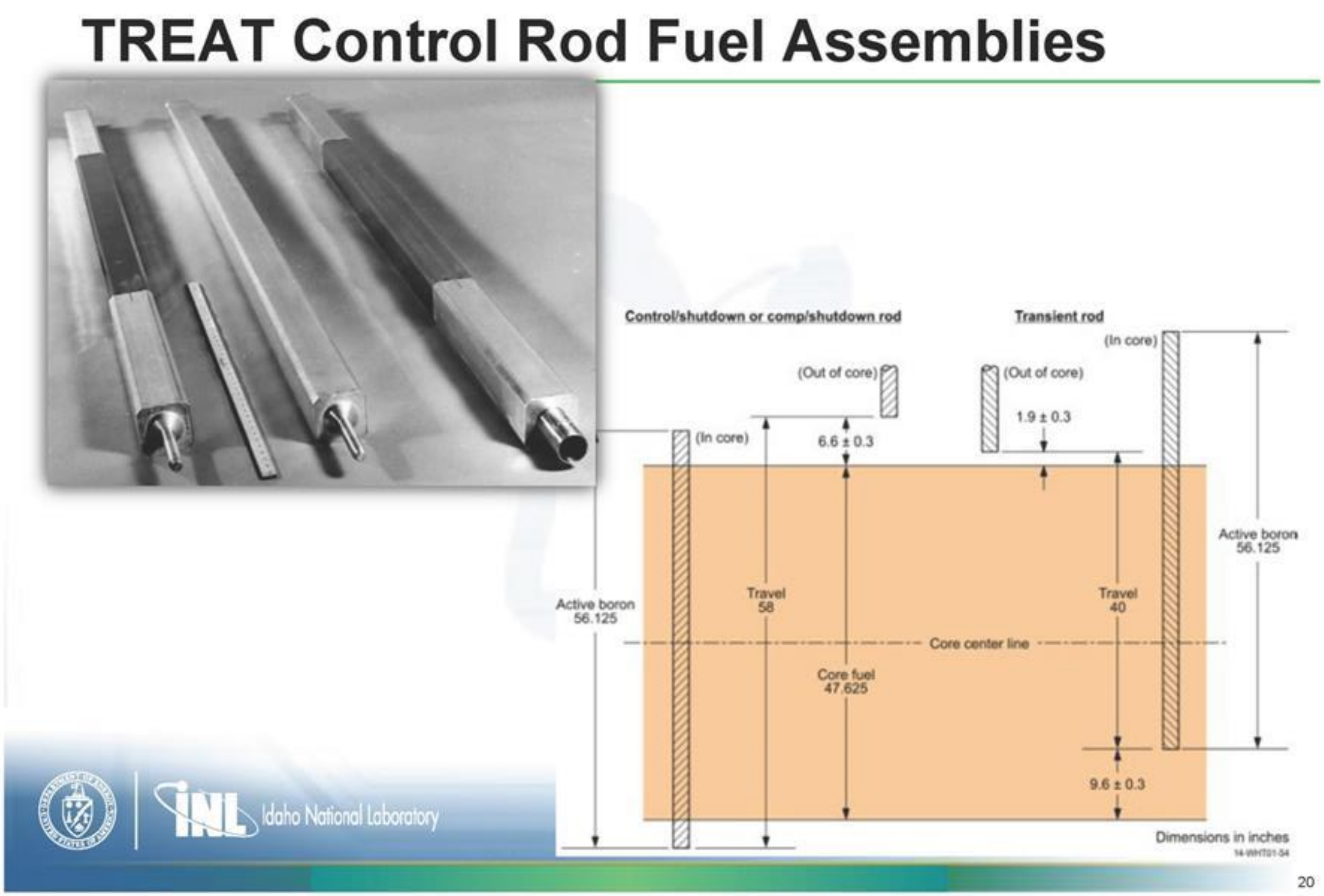
TREAT Control Rod Fuel Assemblies
Control/shutdown or comp/shutdown rod
Transient rod
(In core)
(Out of core)
(Out of core)
(In core)
6.6 ± 0.3
1.9 ± 0.3
Active boron
56.125
Travel
58
Core center line
Core fuel
47.625
Travel
40
Active boron
56.125
9.6 ± 0.3
Dimensions in inches
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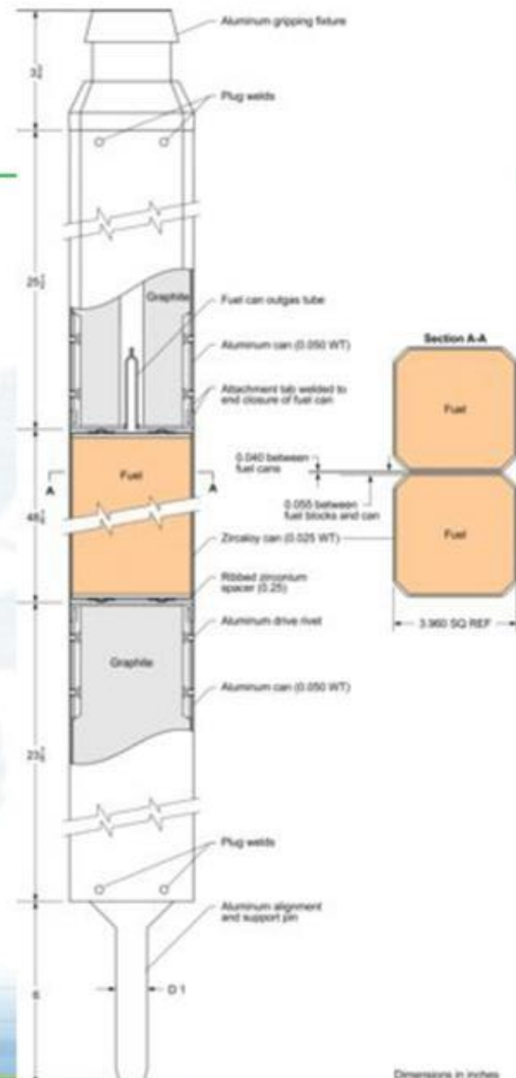
Standard Fuel Assembly

- **MCNP Model (Left)**
- **Drawing (Right)**
- **Challenges**
  - ❖ Modeling octahedral chamfers
    - ○ Graphite reflectors and top aluminum fixture
    - ○ Unique MCNP geometric errors



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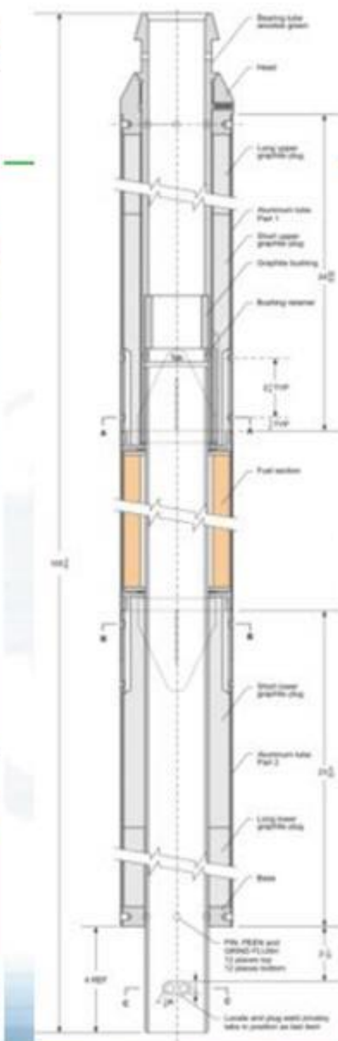
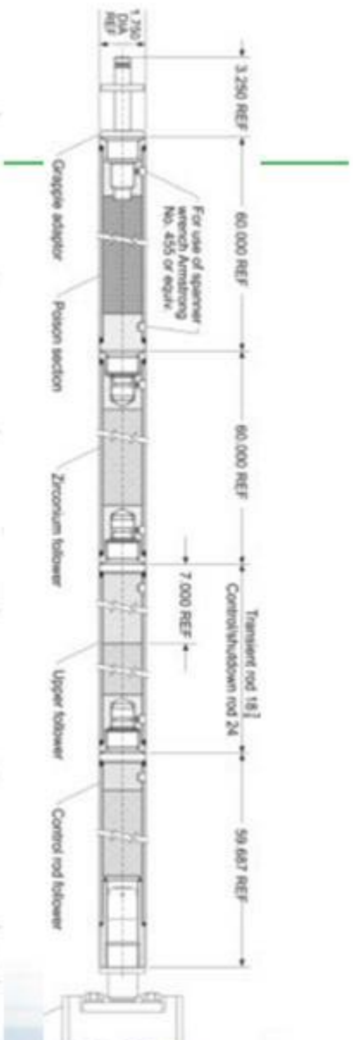
Control Rod Fuel Assembly

- ➢ MCNP Model (Left)
- ➢ Assembly Drawing (Middle)
- ➢ Control Rod Drawing (Right)
  - ❖ Most remaining assemblies are variants of standard and control rod assemblies





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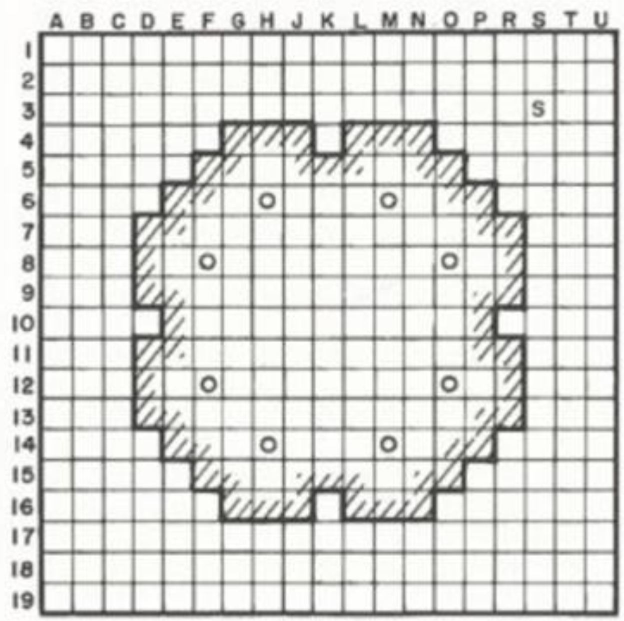
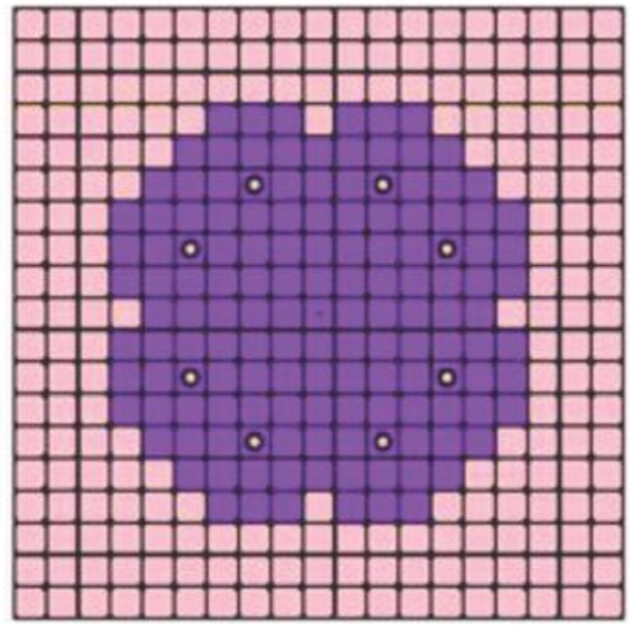
TREAT Minimum Critical Mass Core Layout (Current Progress)





Slightly supercritical : 5185 g U^{235}

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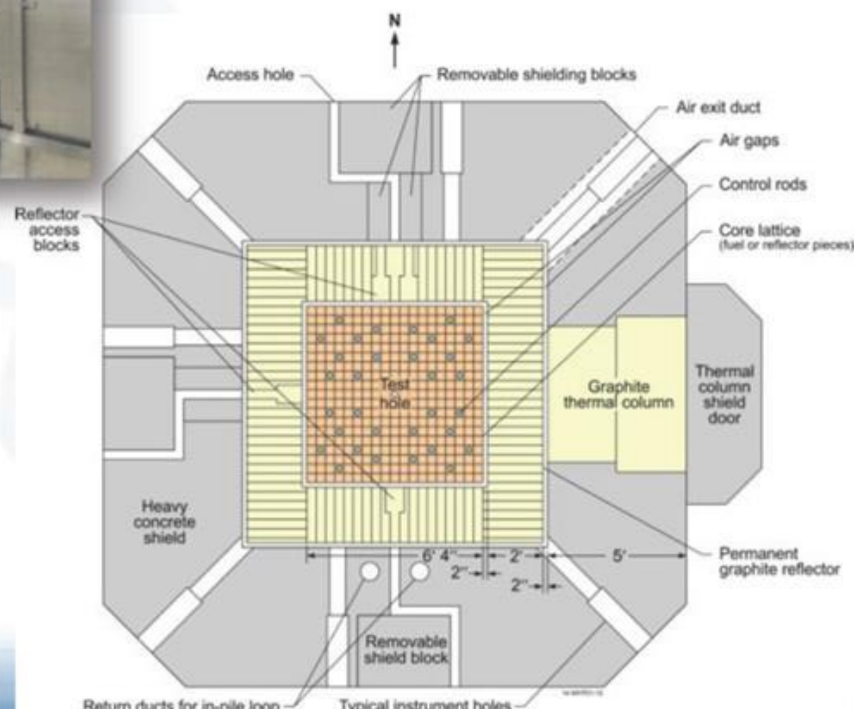
Path Forward on Minimum Critical Mass Core

- Development of detailed model components
  - Thermocouple fuel assemblies (4 types)
  - Source fuel assembly
  - Aluminum-clad dummy assemblies
  - Permanent reflector
  - Core support structure
  - Shielding
  - Detectors
- Benchmark development
  - Complete models
  - Biases and simplifications
  - Uncertainty/ sensitivity analyses
  - Internal review
- Submission to IRPhEP

Remaining Core Components to Model



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M8CAL Core

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TREAT M8CAL Core

	A	B	C	D	E	F	G	H	J	K	L	M	N	O	P	R	S	T	U
1	Z	Z	F	F	F	F	F	F	F	X	F	F	F	F	F	F	F	Z	S
2	Z	F	F	F	F	F	F	F	F	X	F	F	F	F	F	F	F	F	Z
3	T3	F	F	F	F	F	F	CS	F	X	F	CS	F	F	F	F	F	F	F
4	F	F	F	F	F	CS	F	F	F	X	F	F	F	CS	F	F	F	F	F
5	F	F	F	F	F	F	F	F	F	X	F	F	F	F	F	F	F	F	F
6	F	F	F	CT	F	F	F	F	F	X	F	F	F	F	F	CT	F	F	F
7	F	F	F	F	F	F	F	F	F	X	F	F	F	F	F	F	F	F	F
8	F	F	CT	F	F	CC	F	F	F	X	F	F	F	CC	F	F	CT	F	F
9	F	F	F	F	F	F	F	F	F	X2	F	F	F	F	F	F	F	F	F
10	F	F	F	F	F	F	F	T3	F	MK	F	T3	T3	T3	F	F	F	F	F
11	F	F	F	F	F	F	F	T1	F	Z2	F	T1	F	F	F	F	F	F	F
12	F	F	CT	F	F	CC	F	T3	F	F	F	T3	F	CC	F	F	CT	F	F
13	F	F	F	F	F	F	F	T3	F	T3	F	T1	F	F	F	F	F	F	F
14	F	F	F	CT	F	F	F	F	T1	T3	F	F	F	F	F	CT	F	F	F
15	F	F	F	F	F	F	F	F	F	T3	F	F	F	F	F	F	F	F	F
16	F	F	F	F	F	CS	F	F	F	F	F	F	F	CS	F	F	F	F	F
17	F	F	F	F	F	F	F	CS	F	F	F	CS	F	F	F	F	F	F	F
18	Z	F	F	F	F	F	F	F	F	F	F	F	F	F	F	F	F	F	Z
19	Z	Z	F	F	F	F	F	F	F	F	F	F	F	F	F	F	F	Z	Z

	A	Al-Clad Dummy Assembly
20	CC	Compensation Rod Fuel Assembly
	CS	Control/Shutdown Rod Fuel Assembly
	CT	Transient Rod Fuel Assembly
303	F	Standard Fuel Assembly
	MK	MK III Test Vehicle (M8CAL)
	S	Zr-Clad Source Assembly
15	T1	4-Fuel Thermocouple Assembly
	T2	Reflector-Fuel Gradient Thermocouple Assembly
	T3	Midfuel, Reflector & Skin Thermocouple Assembly
	T4	Transient Thermocouple Assembly
	X	48" Access Hole Dummy Assembly
	X2	48" Access Hole Dummy Half-Assembly
	Z	Zr-Clad Dummy Assembly
	Z2	Zr-Clad Dummy Half-Assembly

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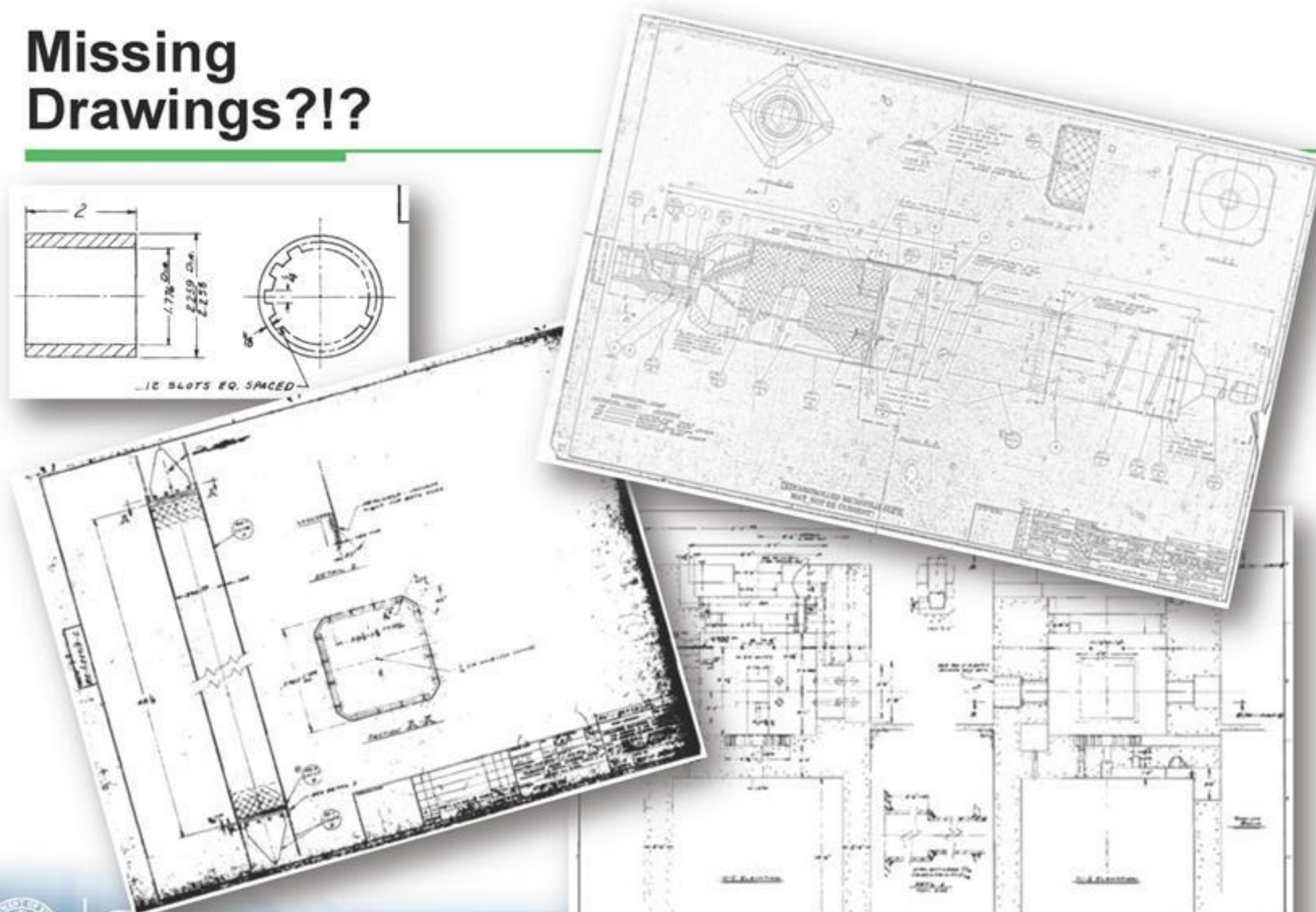
Missing Drawings?!?



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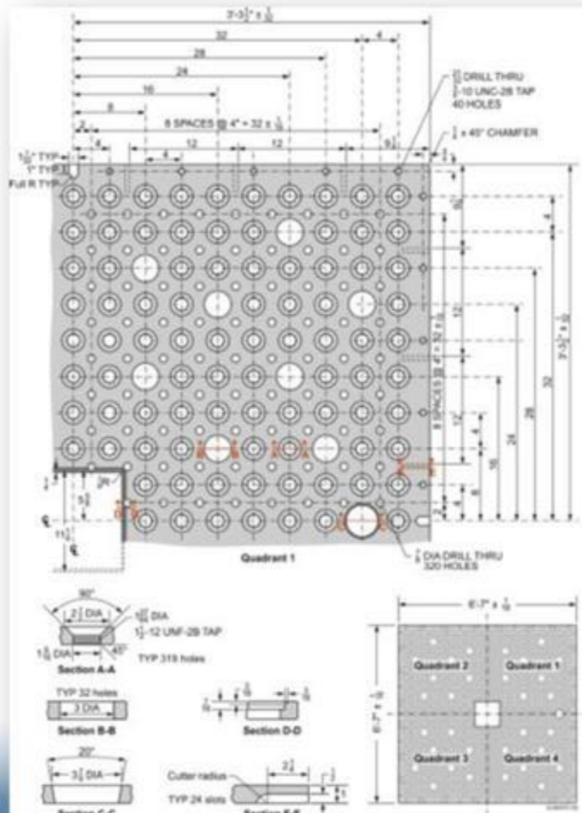
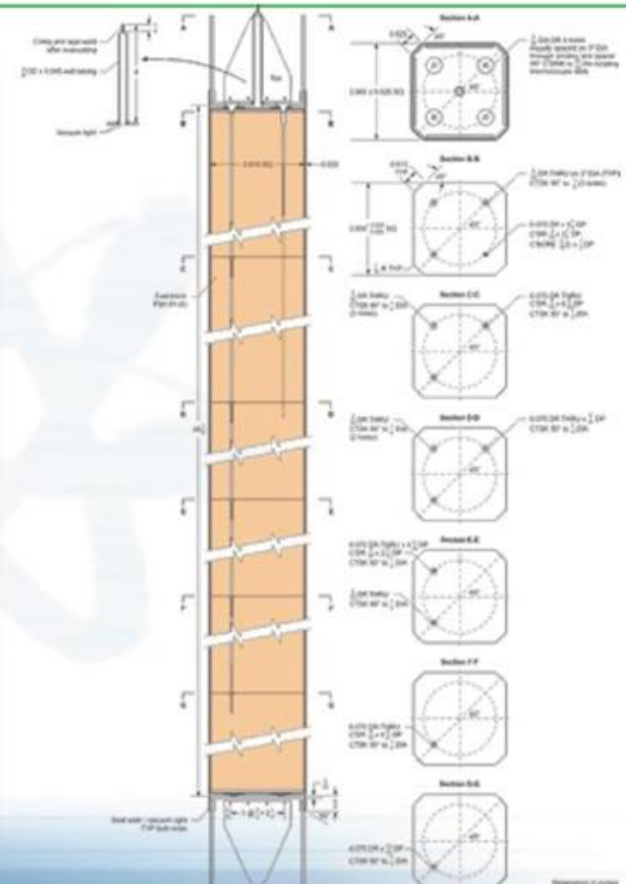
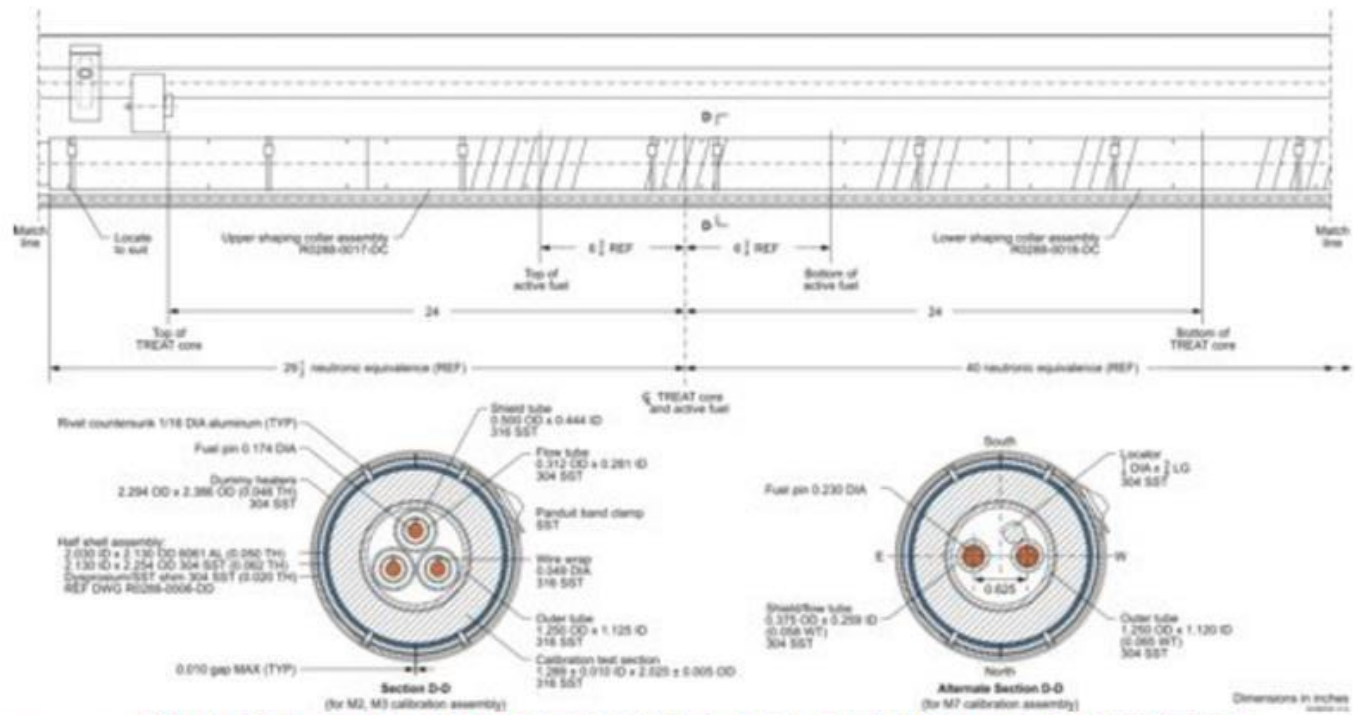
Digitization of Various Drawings





Could Substitute Similar Series Drawings



MIDDLE SECTION OF M2/3/7-CAL VEHICLE ASSEMBLY

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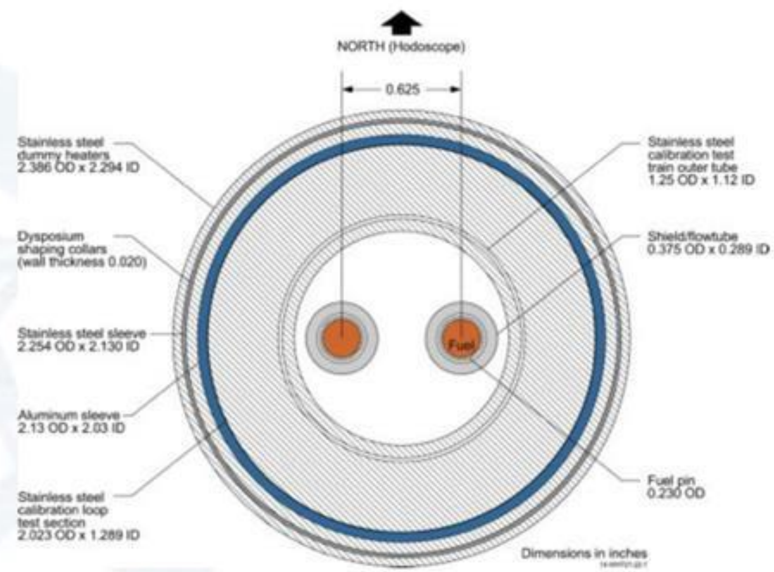
However: Missing Component Dimensions

- **Missing calibration loop design data**
- **Missing support structure details to hold M8CAL fuel pins and wires**
- **<u>Some loop components might still be in TREAT</u>**
  - **M-series can still in-core**



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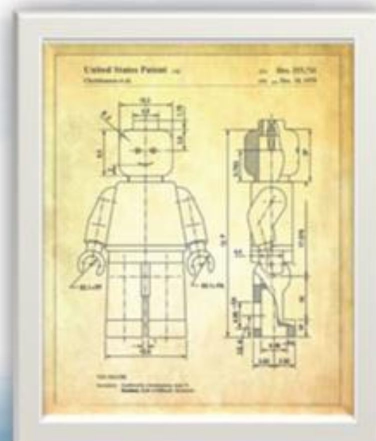
Composition of M8CAL Fuel Pins

- ➢ Appendix data from X425 Casting Campaign
  - ❖ ANL-IFR Report
  - ❖ Data to evaluate uncertainties and impurities
    - ○ T-433 fuel rod
    - ○ T-462 fuel rod
  - ❖ Other useful IFR reports may exist
- ➢ Details regarding flux monitor wire compositions are sparse
  - ❖ Some drawings, but not clear if all flux wires were designed the same



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Thermocouple Fuel Assemblies

- Five types of thermocouple assembly design strategies
- Three types of thermocouple installations
- Assuming <u>manufacturing standard</u> design and uncertainties

- Type A
  - Chromel-alumel
  - SS304 sheath with MgO insulation
- Type B
  - Chromel-alumel
  - 28-gauge wire in asbestos-glass insulation
- Type C
  - Fast-response chromel-alumel
  - 28-gauge wire directly attached to fuel blocks

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Path Forward on M8CAL Core

- After MinCrit Core Completion
- Development of detailed model components
  - New core arrangement
  - Access hole assemblies
  - Half assemblies (two types)
  - MK3 calibration loop
  - Fuel pins and flux wires
- Benchmark development
  - Develop models
  - Biases and simplifications
  - Uncertainty/ sensitivity analyses
  - Internal review
- Submission to IRPhEP

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Publication in IRPhEP

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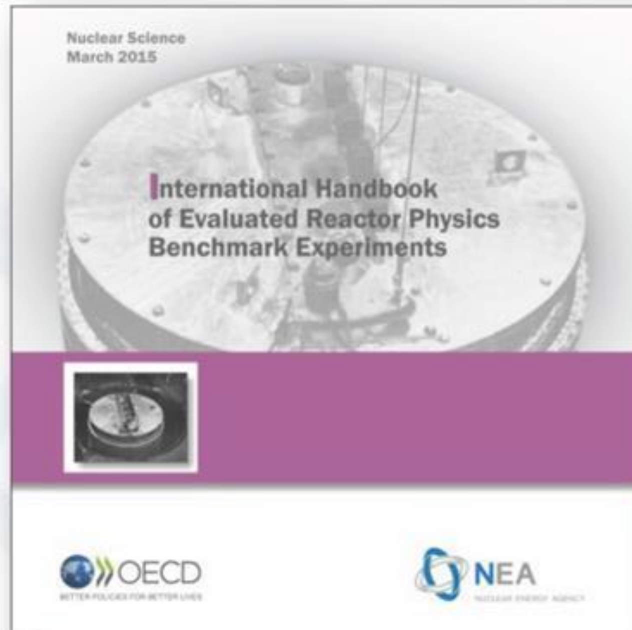
International Handbook of Evaluated Reactor Physics Benchmark Experiments

March 2015 Edition

- 20 Contributing Countries
- Data from 143 Experimental Series performed at 50 Reactor Facilities
- Data from 139 are published as approved benchmarks
- Data from 4 are published in DRAFT form
- Handbook available to OECD member countries, all contributing countries, and to others on a case-by-case basis



http://irphep.inl.gov/
http://www.oecd-nea.org/science/wprs/irphe/

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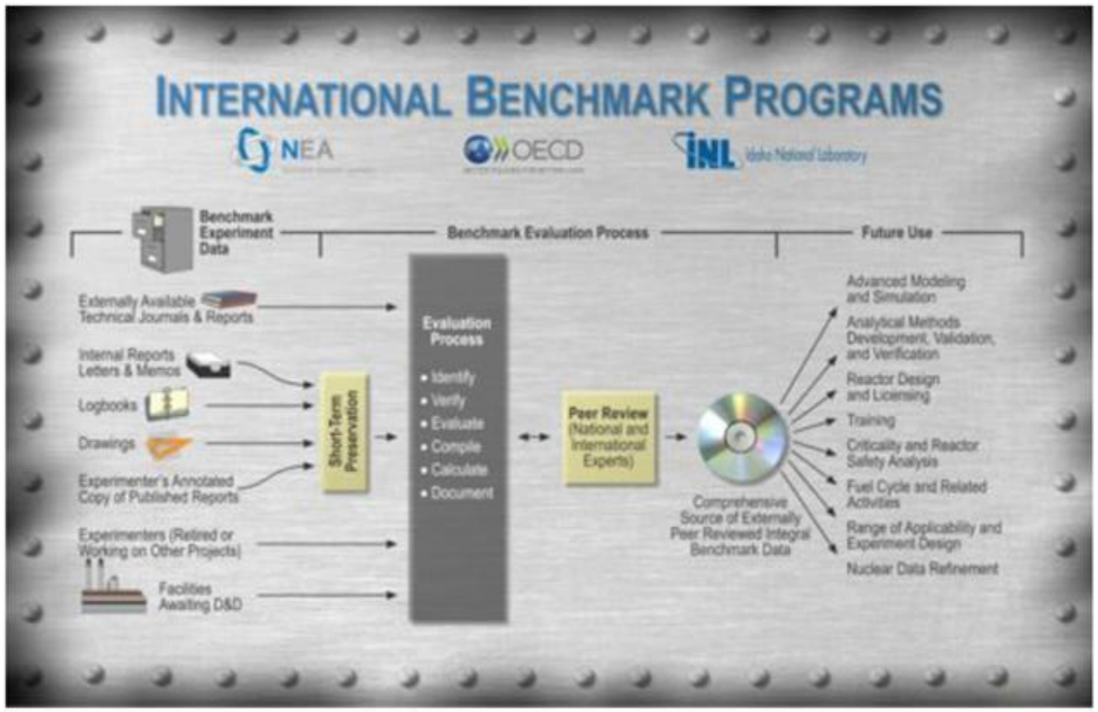
IRPhEP Benchmark Evaluation Process



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IRPhEP Benchmark Structure (aka TREAT-FUND-RESR-00X)

- **Identification Number**
  - (Reactor Name) - (Reactor Type) - (Facility Type) - (Three-Digit Numerical Identifier)
  - Measurement Type(s)

Reactor Type		Facility Type		Measurement Type	
Pressurized Water Reactor	PWR	Experimental Facility	EXP	Critical Configuration	CRIT
VVER Reactors	VVER	Power Reactor	POWER	Subcritical Configuration	SUB
Boiling Water Reactor	BWR	Research Reactor	RESR	Buckling & Extrapolation Length	BUCK
Liquid Metal Fast Reactor	LMFR			Spectral Characteristics	SPEC
Gas Cooled (Thermal) Reactor	GCR			Reactivity Effects	REAC
Gas Cooled (Fast) Reactor	GCFR			Reactivity Coefficients	COEF
Light Water Moderated Reactor	LWR			Kinetics Measurements	KIN
Heavy Water Moderated Reactor	HWR			Reaction-Rate Distributions	RRATE
Molten Salt Reactor	MSR			Power Distributions	POWDIS
RBMK Reactor	RBMK			Nuclide Composition	ISO
Space Reactor	SPACE			Other Miscellaneous Types of Measurements	MISC
Fundamental Physics	FUND				

Can add new types as needed

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Path Forward via IRPhEP

- After Evaluation & Internal Review
- Independent Review
- IRPhEP Annual Technical Review Meeting
  - April 2017, or October 2017
- Response to Action Items from IRPhEP Review
- Publish in IRPhEP Handbook
  - ~6 months after meeting

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Where Do We Stand

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Measured Data Available from Previous Tests

- Rod Positions
- Transient Tests
  - Power, Temperature, Rod Positions
- Experiments
  - PCF
  - TCF
  - Above Supporting Data

- Slowly but surely digging it all out

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QA Pedigree of Available Data?

- Comprehensive data sets not yet fully available
  - Missing
    - Drawings
    - Measurements
    - Compositions
    - Dimensions
    - Calibrations
    - Uncertainties
    - Measurement Biases
  - Impacts Schedules

- Purpose of data
  - Originally
    - Validate basic computational methods for core operations and irradiation tests
  - Now
    - Multi-physics, complex core-experiment dynamics

What Data Would We Like/Need from Start-Up Testing?

- Operations
- Experimentation
  - Specific to Stakeholders
- Safety Analyses
- Methods Development
- Fundamental Physics
- Science-Based Missions
- Standards

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¿Questions?

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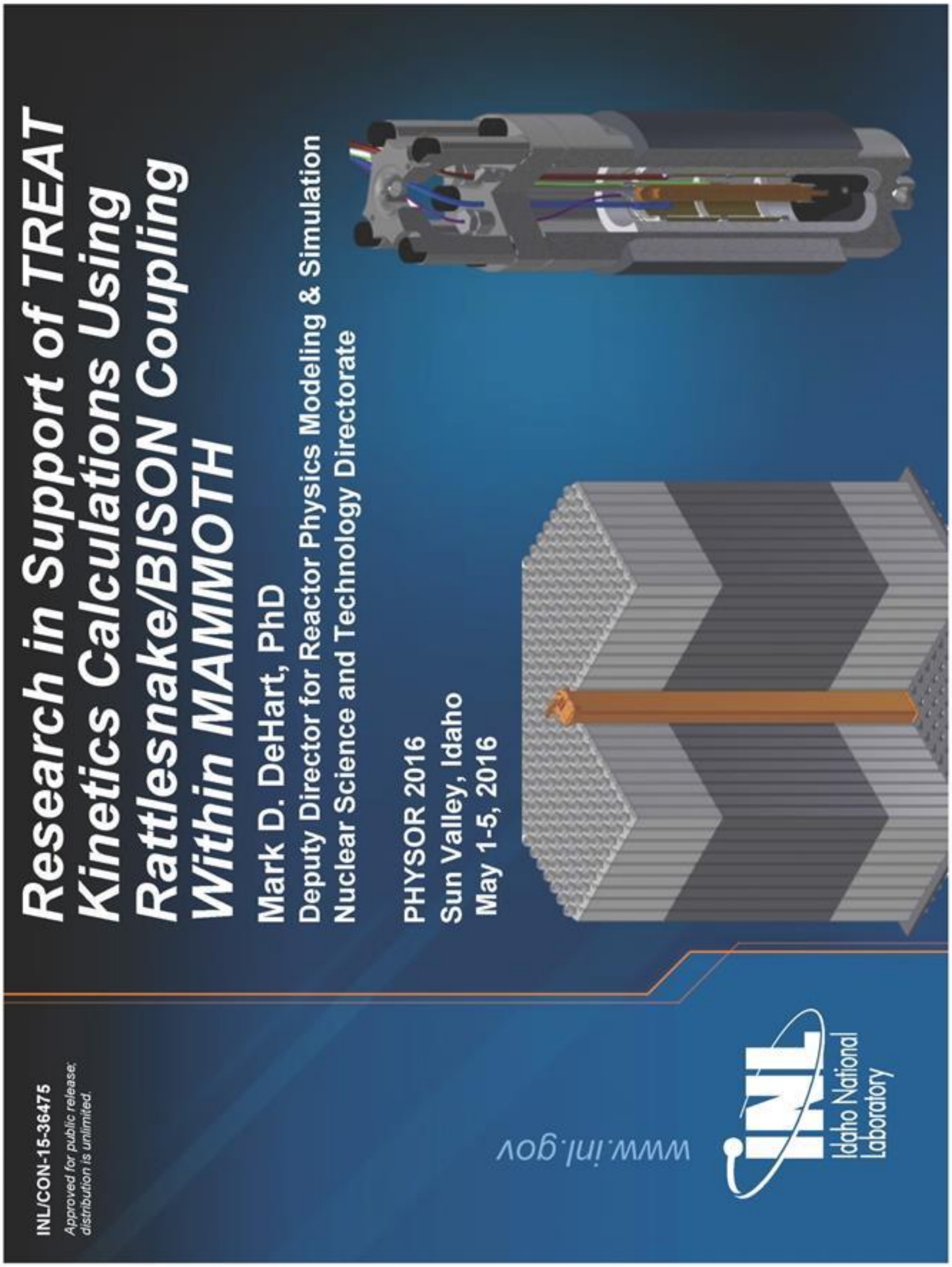
Research in Support of TREAT Kinetics Calculations Using Rattlesnake/BISON Coupling Within MAMMOTH
Mark D. DeHart, PhD
Deputy Director for Reactor Physics Modeling & Simulation
Nuclear Science and Technology Directorate
PHYSOR 2016
Sun Valley, Idaho
May 1-5, 2016
INL/CON-15-36475
Approved for public release; distribution is unlimited.
www.inl.gov
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TREAT's mission is to deliver transient energy deposition to a target or targets inside experiment rigs.

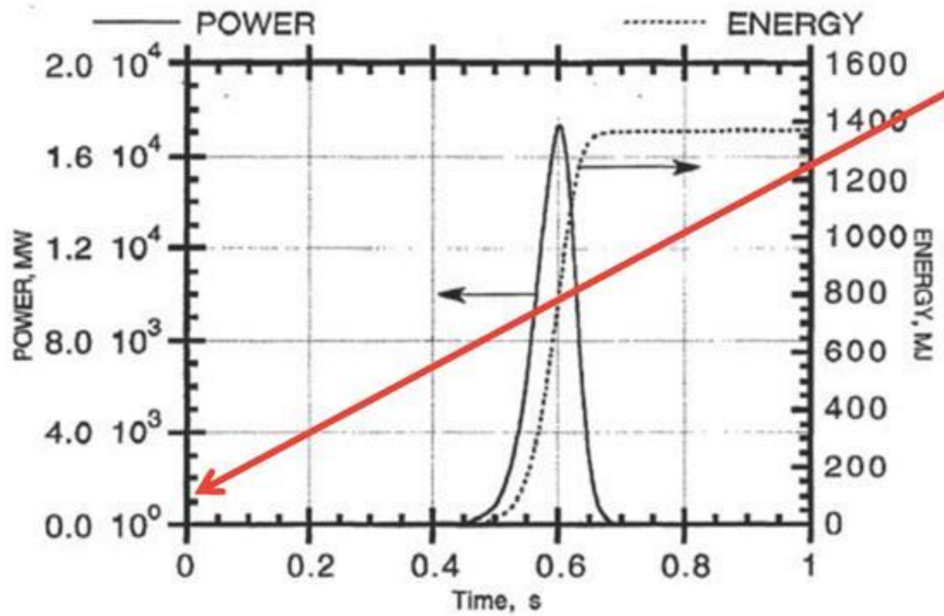

FIG. 5. Plot of TREAT reactor power and energy for hypothetical RIA-type transient resulting in 1400-MJ pulse with a 72-msec FWHM capable of depositing 1200 kJ of energy per kg of fuel (290 cal/g).

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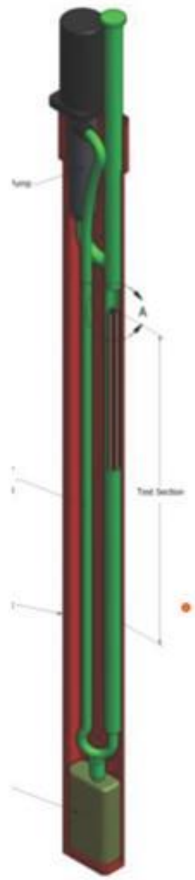
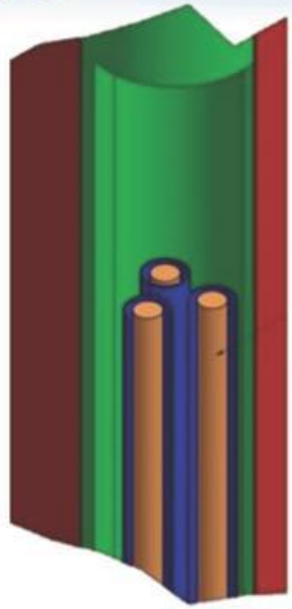
TREAT Experiments





- TREAT's mission is to deliver transient energy deposition to a target or targets inside experiment rigs.
- Historically, failure conditions were determined by a number of transient experiments.
- In these experiments, very little predictive capability for core performance existed, and experiment models were somewhat limited.
- A number of pre-experiment tests (calibration tests) were required prior to the actual measurement
  - Steady state
  - Transient (low power, high power)

TREAT and Temperature Feedback

- There is strong nonlinear coupling between the thermal feedback and the neutron radiation field distribution in TREAT.

$$\phi(r,E,t) \longrightarrow T(r,\phi,t) \longrightarrow \sigma(r,E,t)$$

- The best current practice is to apply a split operator approach the radiation transport equations and the heat transport equations. ANL is currently doing TREAT analysis with MCNP and a point kinetics solution with very coarse meshing (9 temperature regions in the core).
- This will result in a reduction of accuracy and is not unlike analysis methods performed in the early 90's. This required numerous calibration transients prior to initiating an experiment series
- Experience to date indicates that the evolution of T as a function of time and is also a nonlinear function due to temperature dependent thermal properties of graphite.
- Poor characterization of core power transients will lead to the inability to accurately quantify fuel behavior.

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Modeling TREAT with MAMMOTH

- MAMMOTH has been built using the MOOSE framework (Multi-physics Object Oriented Simulation Environment)
- MOOSE allows implicit, strong, and loose coupling of MOOSE animal solutions
- MAMMOTH is the MOOSE-based multi-physics reactor analysis tool.
- At present, TREAT core simulation efforts rely on BISON (fuel performance), Rattlesnake (time-dependent neutron transport) and MAMMOTH.
- LWR-type pin experiments are being evaluated using RELAP-7 as well.

[redacted]

- Note that MAMMOTH is a single executable code with multiple personalities all co-existing.
- All codes are based on FEM – MOOSE routines perform all solutions.
- All data from all codes is available to the solver(s) used.

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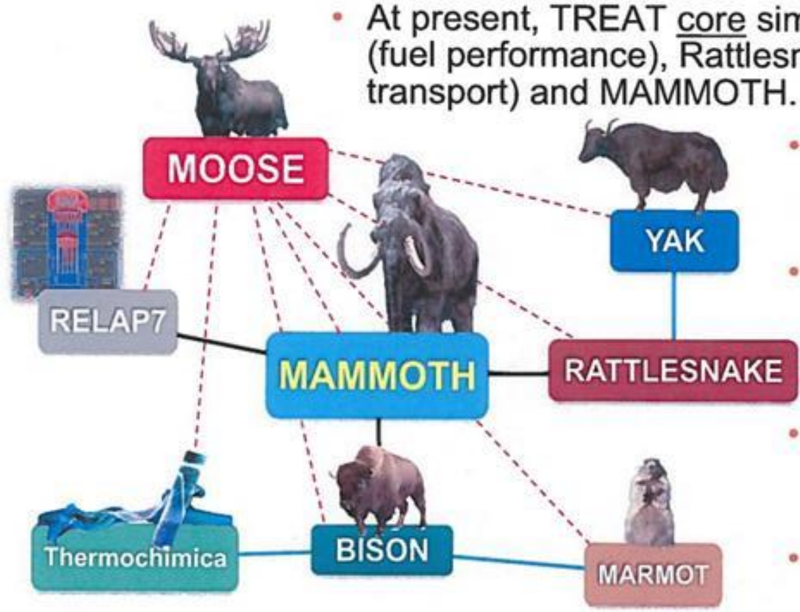

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The Magic of MOOSE

- MOOSE itself “simply” takes the equations associated with a given analysis and automatically expands them into the corresponding set(s) of finite element equations for user-specified mesh(es).
- These equations are all interdependent and can potentially result in a very large matrix, but but one that will yield a fully implicit solution.
- The Jacobian-free Newton Krylov method is generally used for solving the coupled equations – such matrices are too large to invert.
- Individual “physics” can be solved independently if desired (JFNK or other), then iterations performed between the two solutions until both converge (tight coupling)
- JFNK provides an extremely robust solution method for stiff, highly nonlinear, and tightly coupled problems
  - Provides the convergence of Newton’s method without the need to form a Jacobian (saves time and memory)
  - Directly supports advanced preconditioning strategies (physics-based and multilevel)
  - Implicit method is unconditionally stable
- JFNK solvers are readily available in PETSc; PETSc is incorporated into MOOSE and all of its solution methods are available

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Evolution of Ability to Perform Full Core Kinetics for TREAT

- Collection of core physical data (easier said than done).
- Cross section and delayed neutron data preparation and testing
- Methods evaluations (S_n, P_n and diffusion)
- Infinite medium tests
  - k-eff
  - Comparison of spatial and point kinetics without feedback
- Mesh development
  - Single element
  - Minimum critical core
  - Calibration transient 15 (1.55% Δk)
- Thermal model and material properties
- Single element testing with and without feedback
- Evaluation of homogenization approaches
- Minimum critical core eigenvalue calculations
- Full core transients with feedback

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First Steps

- Core data is not located in a single report, repository or set of drawings; some reports/drawings are inconsistent with other available data.
  - INL report "Baseline Assessment of TREAT for Modeling and Analysis Needs," by John Bess and Mark DeHart, INL/EXT-15-35372, was released this month.
  - ~500 pages of measurements, specifications, updated (redrawn) drawings and illustrations
- Cross section evaluations showed that due to the mfp of neutrons in graphite, reflectors regions and control rods must be taken into account in generating fuel cross sections (and vice versa). Cross section generation requires three dimensional flux solutions.
- Infinite media fuel calculations were performed to ensure that S_n, P_n and diffusion cross sections were being generated consistently.
- The space-time transport solution was compared to an equivalent point kinetics solution for simple and increasingly complex transients

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First Steps

- Core data is not located in a single report, repository or set of drawings; some reports/drawings are inconsistent with other available data.
  - INL report "<u>B</u>aseline <u>A</u>ssessment of <u>T</u>REAT for <u>M</u>odeling and <u>A</u>nalysis <u>N</u>eeds," by John Bess and Mark DeHart, INL/EXT-15-35372, was released this month. Also known as the BATMAN report.
  - ~500 pages of measurements, specifications, updated (redrawn) drawings and illustrations
- Cross section evaluations showed that due to the mfp of neutrons in graphite, reflectors regions and control rods must be taken into account in generating fuel cross sections (and vice versa). Cross section generation requires three dimensional flux solutions.
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- The space-time transport solution was compared to an equivalent point kinetics solution for simple and increasingly complex transients

Model Development

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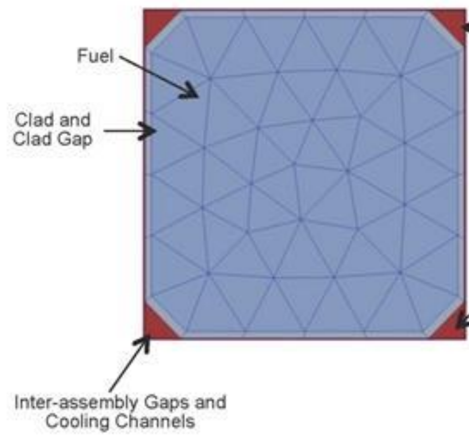
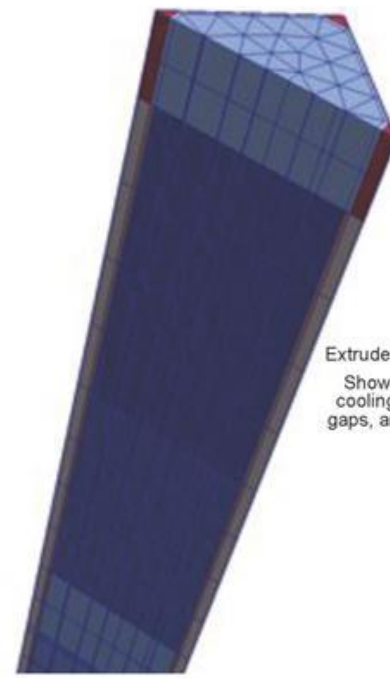
- First developed a rough model of a single element and used for infinite lattice calculations



Vertical Air Gaps
- Post-transient cooling
- Neutron streaming/leakage
- Cannot



Extruded geometry with 3D mesh:

Shows: - Interassembly gaps, cooling channels, clad and clad gaps, and axial levels of standard assembly

- Used to study modeling parameters
  - Mesh convergence
  - Cross sections
  - Homogenization approaches
  - **Streaming effects**
  - **Void treatments**
  - Comparison to Monte Carlo solutions
- Also used for first coupled calculations

Coupled Physics in MAMMOTH

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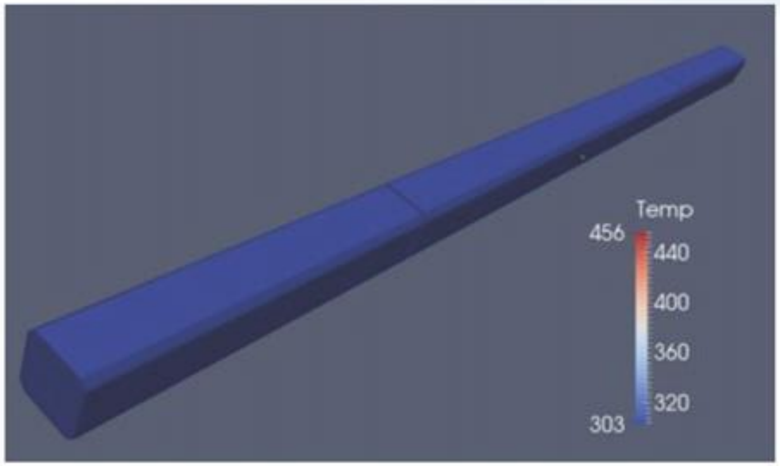
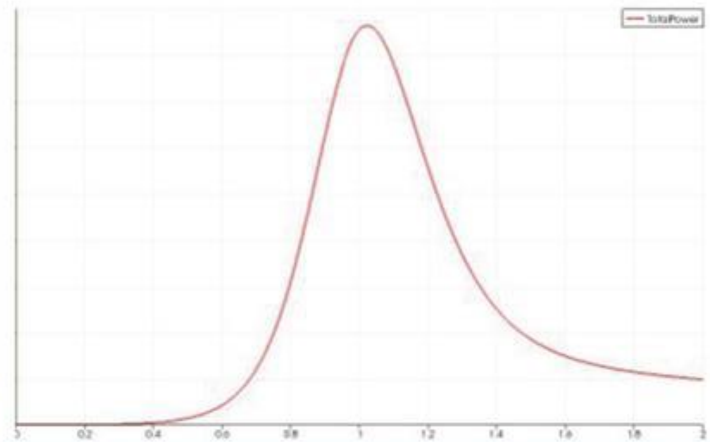
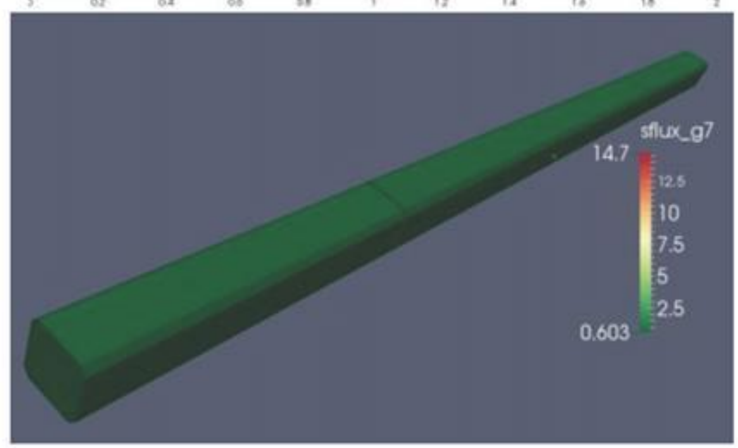


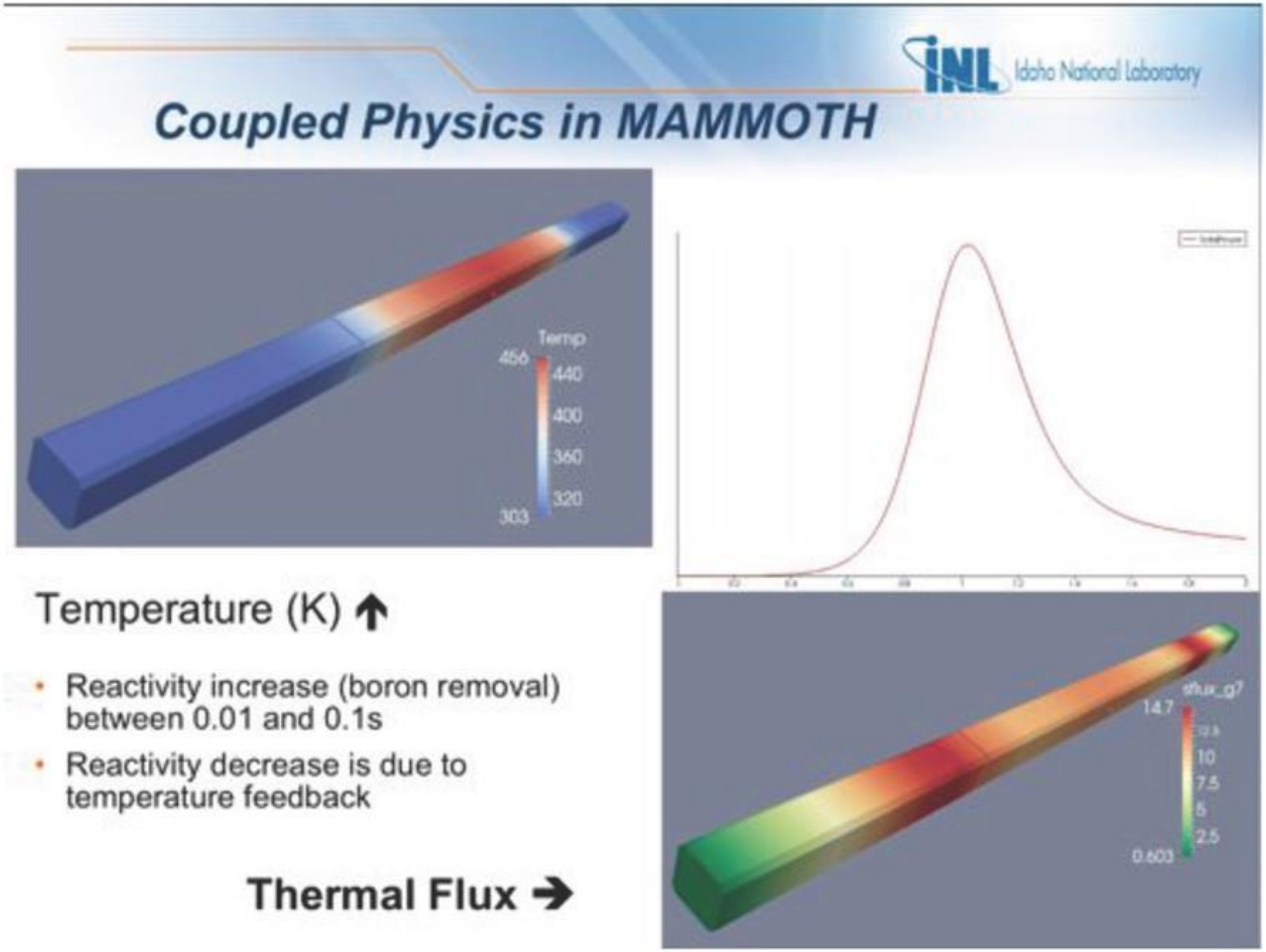


Temperature (K) ↑

- Reactivity increase (boron removal) between 0.01 and 0.1s
- Reactivity decrease is due to temperature feedback

Thermal Flux →

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Coupled Physics in MAMMOTH
Temp
456
440
400
360
320
303
Temperature (K) ↑
• Reactivity increase (boron removal) between 0.01 and 0.1s
• Reactivity decrease is due to temperature feedback
Thermal Flux →
sflux_g7
14.7
10
7.5
5
2.5
0.603

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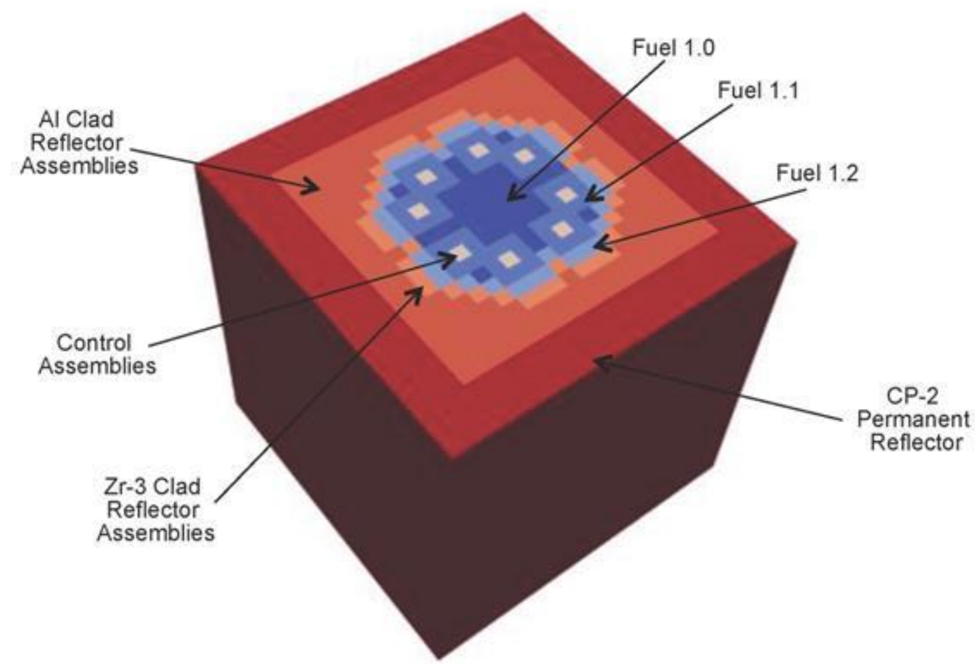
Minimum Critical Core

Work performed by Mr. Anthony Alberti, Oregon State University, for his Masters Degree (now continuing MAMMOTH work for his PhD)



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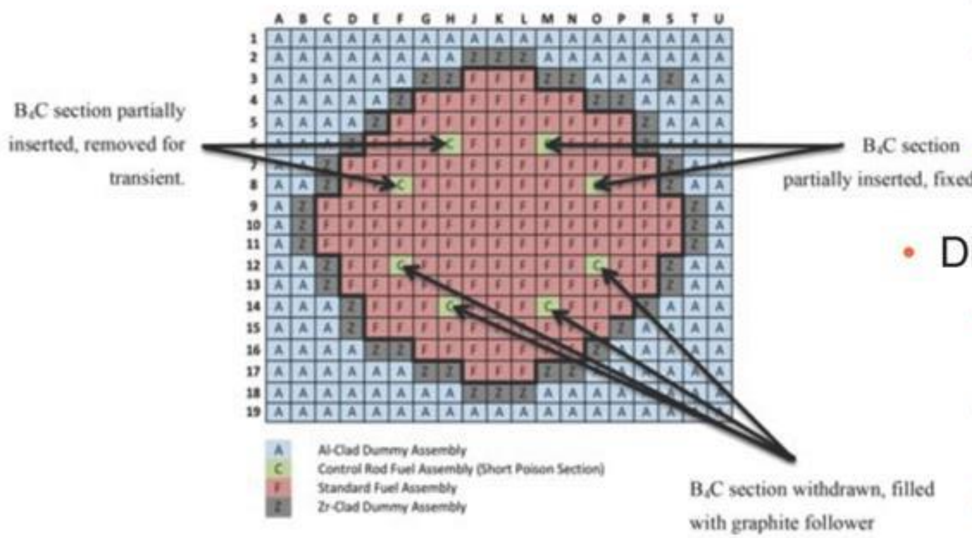
159 Element "Small Core" Configuration

- Developed solid understanding of modeling issues
  - Infinite media kinetics
  - Infinite lattice eigenvalue and transients
  - Simulation of TREAT Test 15 pre-operation transient testing.



- Advantages
  - Simple core
  - No in-core experiments or slots
  - Detector current data available
- Disadvantages
  - Exact rod movement not known
  - Asymmetric (control rod positioning)
  - Old instrumentation
- Starting point for transient validation

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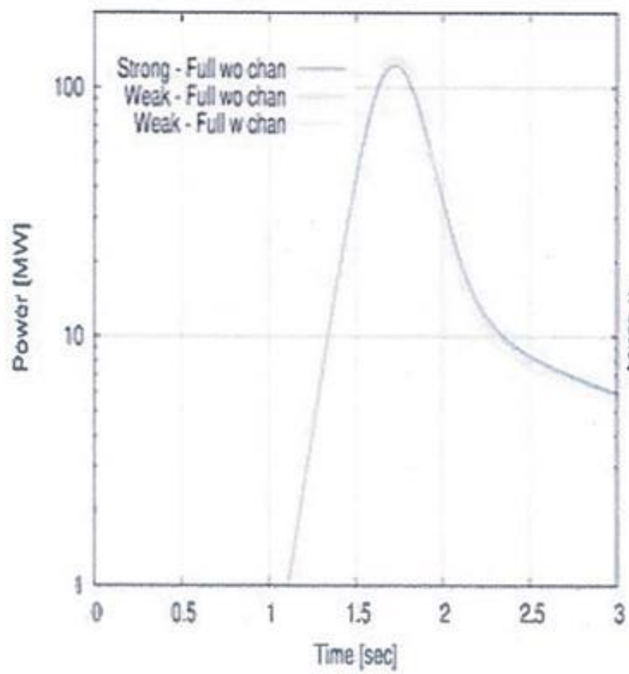
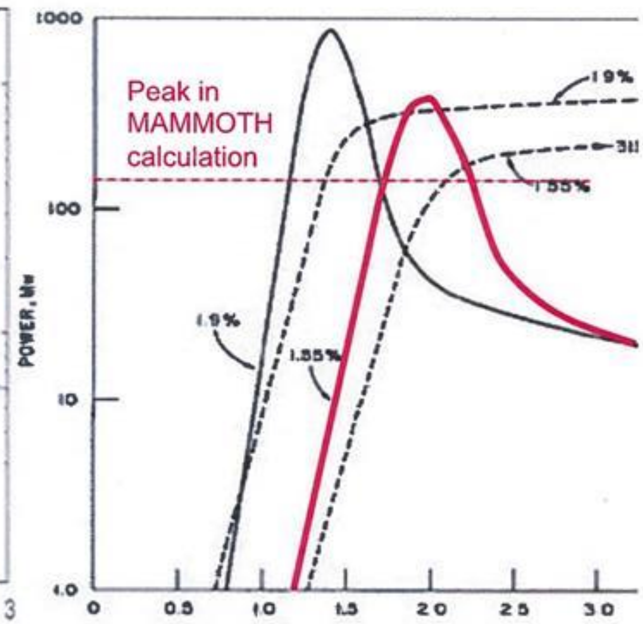
First Attempt at a Transient simulation (1.5% Δk)





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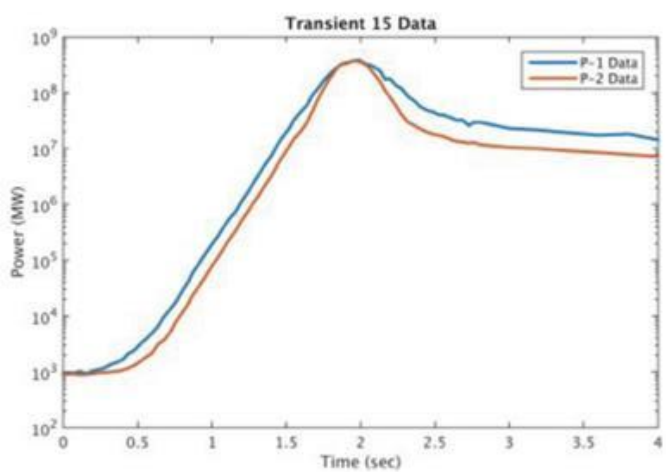
Neutron Kinetics – "Real" data

- Real Data
  - Transient 15: ANL-6173 Listed period = 0.105 sec and reactivity = 1.55%Δk/k
  - Original chamber current data was re-evaluated to determine appropriate bounds to place on these measurements
    - Period is the measured quantity, not reactivity
    - Chamber P-1 tented towards longer periods while P-2 tended toward shorter periods



Period	Reactivities
0.103 sec (min)	0.01552
0.1075 sec (most probable)	0.01515
0.112 sec (max)	0.01481

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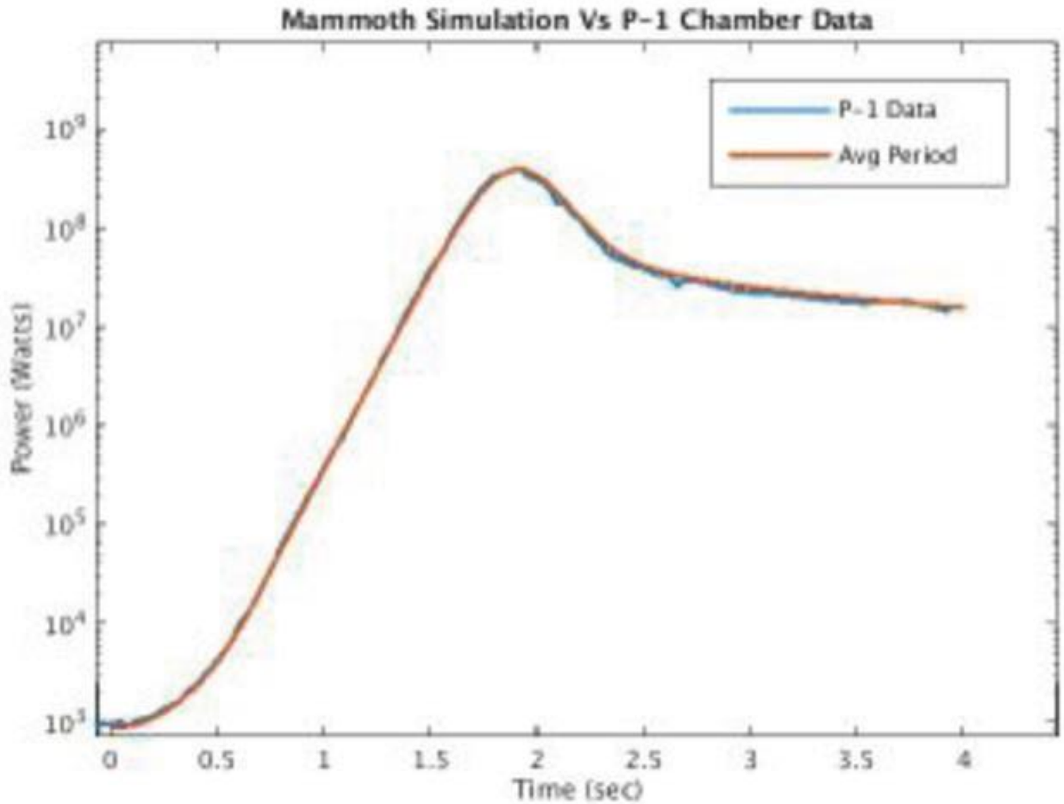
Combined Kinetics and Feedback in Mammoth

- P1 Data (shifted in time by 0.07 sec) vs Average Period Result using Mammoth



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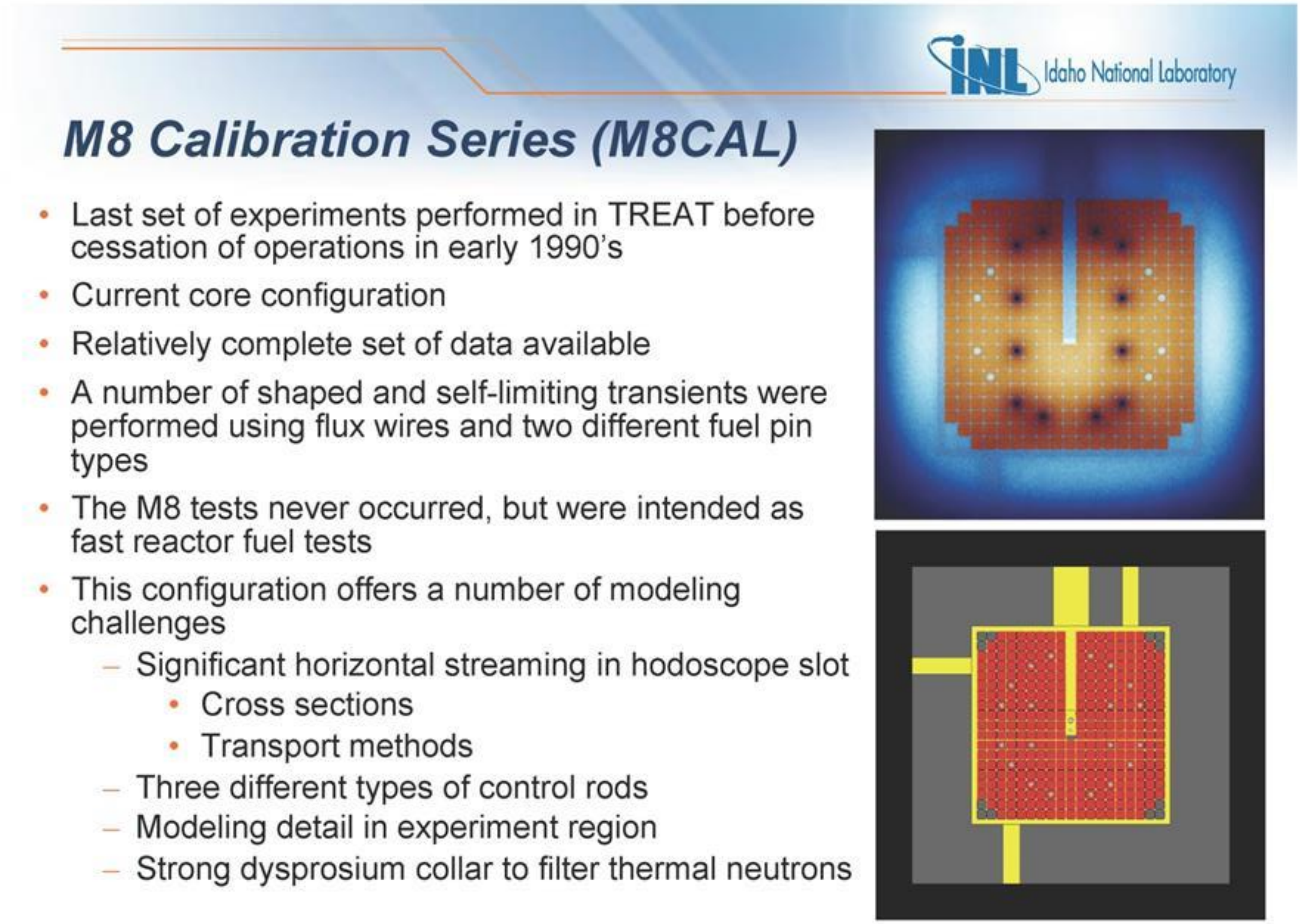
Combined Kinetics and Feedback in Mammoth

- ANL – 6173 (Trans 15)
- Peak Power = 380MW
- Integral Power = 315 MW-sec or (MJ)
- ΔT at core center = 176 °C (K)
- Note: We have no uncertainties from the data on these values

Period	Peak Power (MW)	Peak Power (% Diff)	Integral Power (MJ)	Integral Power (% Diff)	ΔT max (Kelvin)	ΔT max (% Diff)
Min (0.1033 sec)	425	11.7	291	7.6	180	2.2
Avg (0.1082 sec)	384	1.1	281	10.7	174	1.3
Max (0.1126 sec)	355	6.5	268	14.9	166	5.8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M8 Calibration Series (M8CAL)
Last set of experiments performed in TREAT before cessation of operations in early 1990's
Current core configuration
Relatively complete set of data available
A number of shaped and self-limiting transients were performed using flux wires and two different fuel pin types
The M8 tests never occurred, but were intended as fast reactor fuel tests
This configuration offers a number of modeling challenges
– Significant horizontal streaming in hodoscope slot
• Cross sections
• Transport methods
– Three different types of control rods
– Modeling detail in experiment region
– Strong dysprosium collar to filter thermal neutrons

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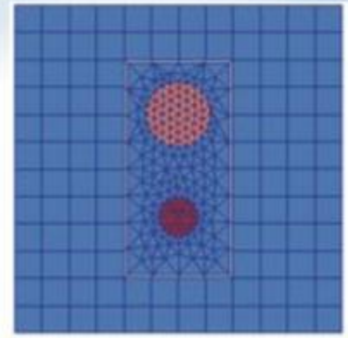
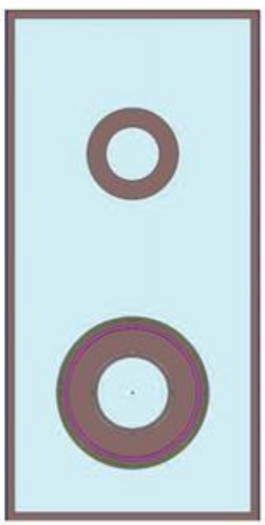
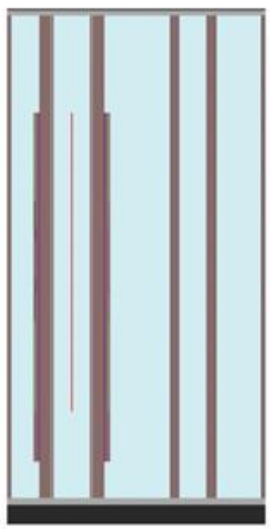
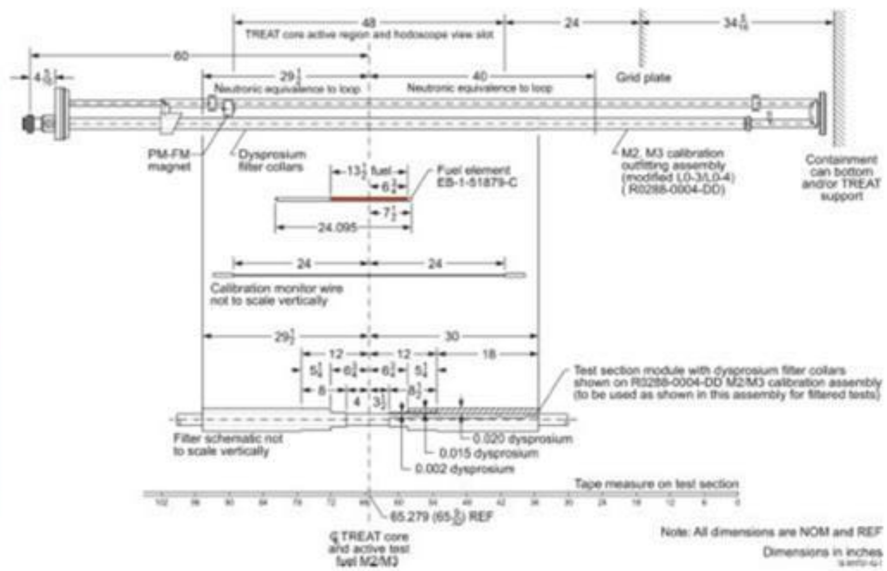
M8CAL

- Serpent calculations to generate cross sections
- Designed M8-Calibration core
- Dy collar – Equivalent to a control rod
  - ~1.0468 with air inside M8CAL vehicle region
  - ~0.9964 – 0.999989 (Depending on critical CR positions) with M8CAL vehicle modeled









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M8CAL Steady State Calculations

	keff	pcm
Serpent (reference)	0.998840	-
Rattlesnake (Serpent Diff Coef.)	0.845496	-15352.2
Rattlesnake (Rattlesnake Diff Coef. – SPH Correction)	1.029792	3098.8

Next steps

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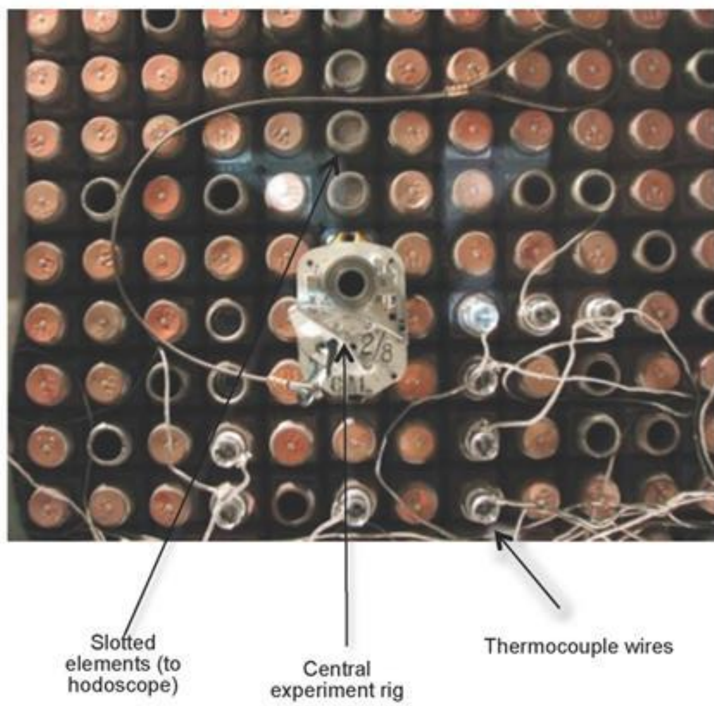
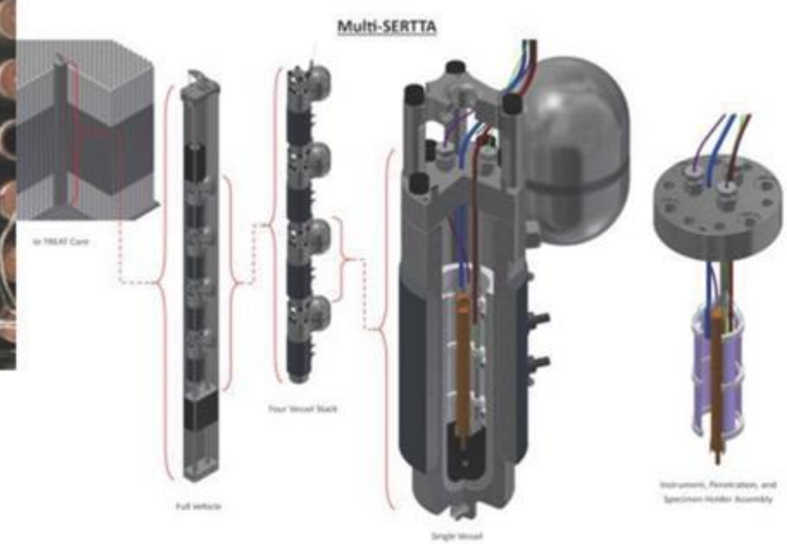
- Resolve differences in steady state predictions and measurements in M8CAL
- Begin transient simulations for M8CAL measurements.
- Continue validation efforts
- Improvements in cross section methods



- Begin working more closely with experiment design and core operations staff to begin planning measurements to assist in methods validation.



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Questions?

INL-MIS-2016-38400
Approved for public release; distribution is unlimited.

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Reactivation of the TREAT Hodoscope

David L. Chichester
Scott M. Watson
James T. Johnson
Scott J. Thompson
Jay D. Hix
Robert S. Schley
Lee O. Nelson
Daniel M. Wachs
Richard S. Bondurant (Areva)

May 2016

Idaho National Laboratory

Nuclear Science User Facilities

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Refurbishment Activities Overview

- Fuel motion monitoring system (FMMS) description and refurbishment plan
- Fast-neutron sensors
- Data acquisition system
- Summary

TREAT Transient #1022

Time

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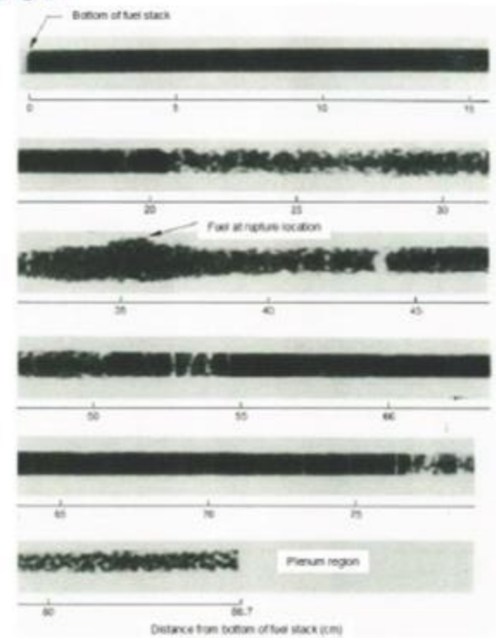
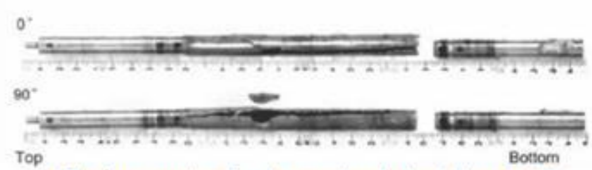
How Does Fuel Fail in a Transient?

- The fuel rapidly heats up
  - Cracking → rubbleization → melting
  - Fission gasses, rapidly released from the fuel matrix, cause a pressure pulse
  - Fission products diffusing into the cladding form eutectics
- The cladding is breached
  - Mechanical stress and thermal conductivity degrade cladding
  - Pellet–cladding mechanical interaction
  - Pellet–cladding chemical interaction
  - Gas-pressure leads to burst cladding
- Fuel debris enters the coolant region
  - Loss of local cooling accelerates damage evolution
  - Debris transport induces damage elsewhere
- High-temperatures lead to cladding/steam catalysis (Zr cladding) and hydrogen production



Neutron radiography of a failed fuel rod, showing internal rubbleization



Photograph of a breached cladding rod

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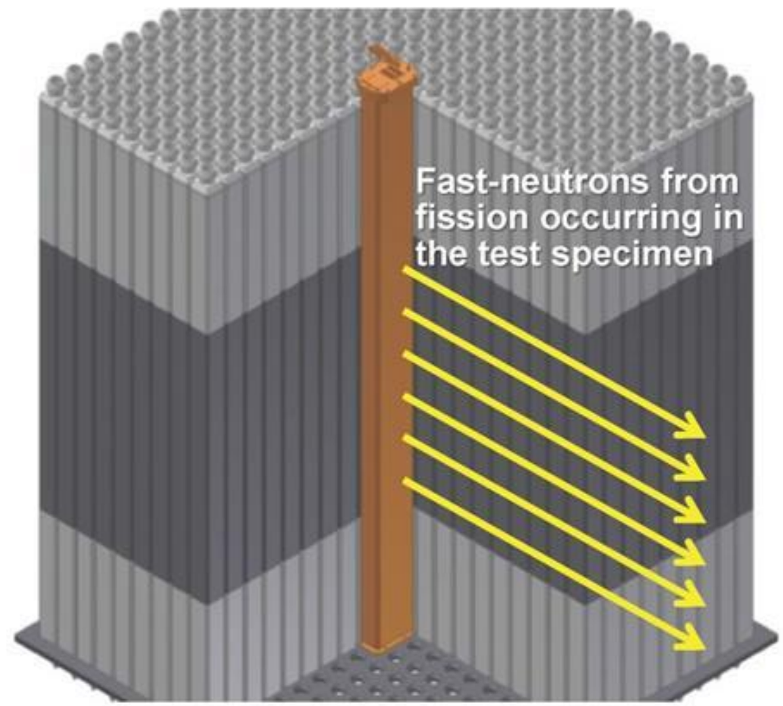
Fuel Motion Monitoring

- Monitoring fuel movement and relocation is an important diagnostic for assessing accident tolerant fuel
- Fast-neutron hodoscope
  1. A row of empty elements is placed between the fuel and the outside edge
  2. A fast-neutron collimator with hundreds of small slits is placed external to the core, each viewing a small area of the test vehicle
  3. Fast neutron detectors are placed at the outside surface of the collimator
  4. Fission is induced in the fuel sample by the reactor during the transient
  5. Fission-neutrons from the test specimen leak from the core, through the slits, and are measured in the detectors



Three-dimensional rendering of the TREAT core, illustrating the fuel elements (grey) and a transient test vehicle (orange)

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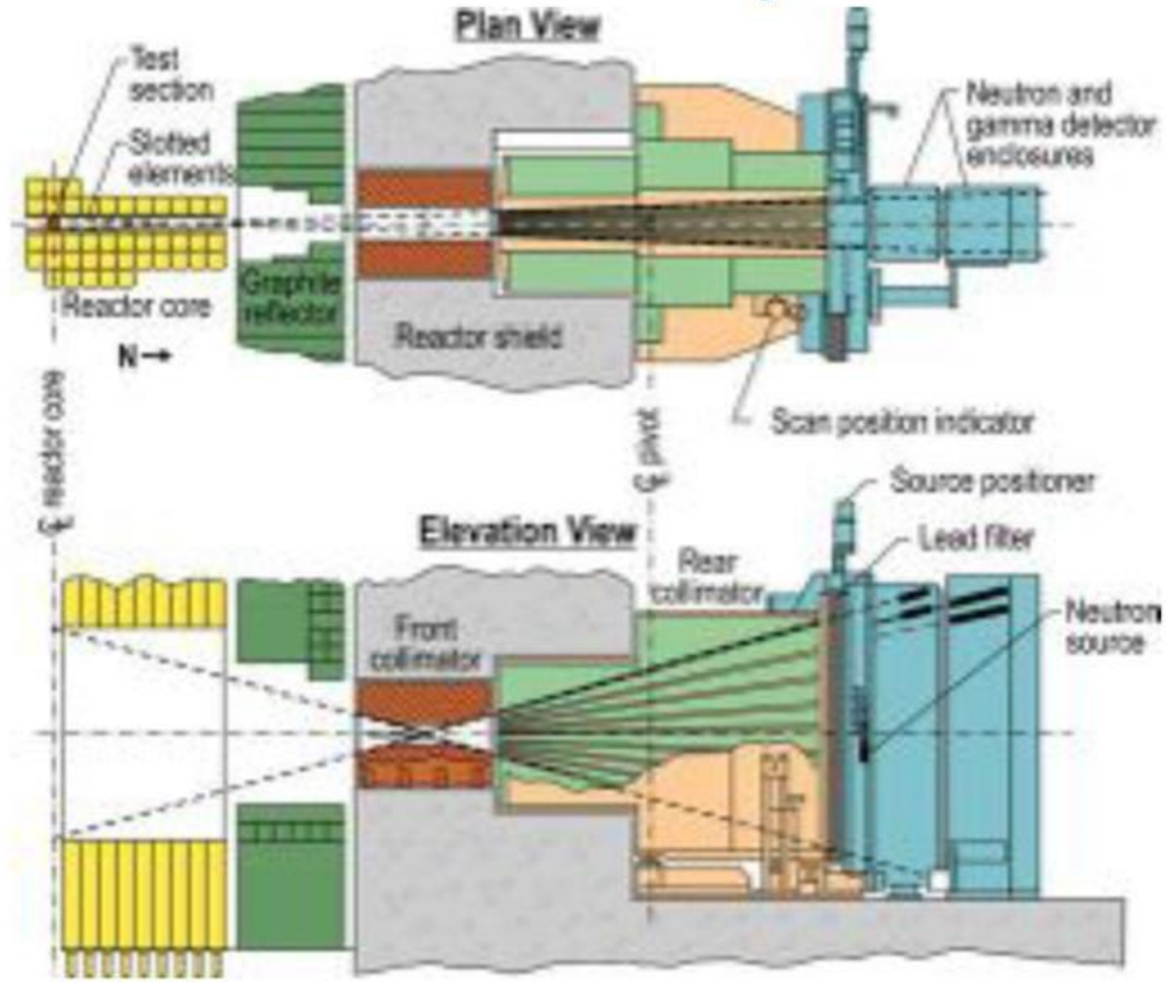
TREAT Fast-Neutron Hodoscope



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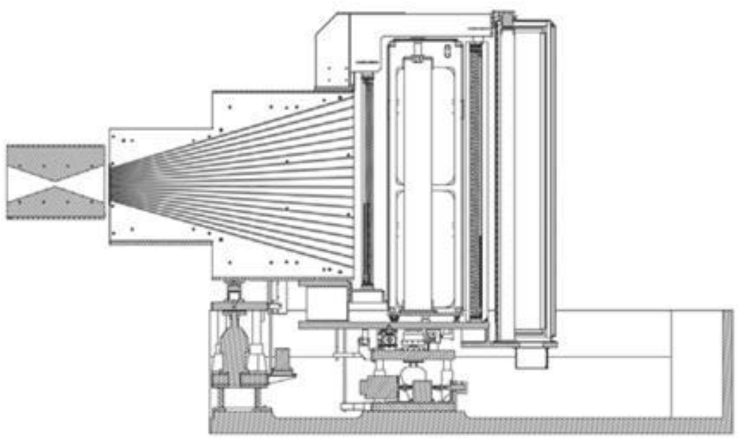
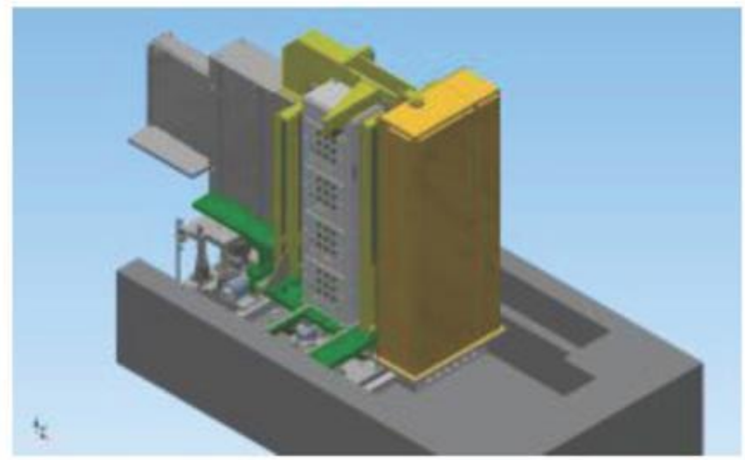
TREAT Fast-Neutron Hodoscope



2-D cross-section line drawing of the FMMS



Perspective 3-D view of the FMMS, from the northeast

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TREAT Fast-Neutron Hodoscope

View of the Hodoscope system from the side

View of the Hodoscope system from the rear, with the detector panel open

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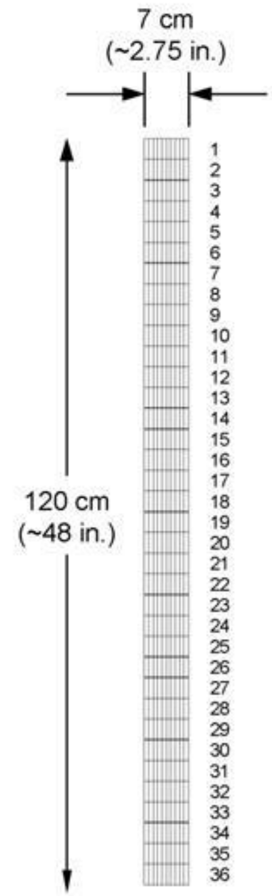
Hodoscope *"Imaging"*

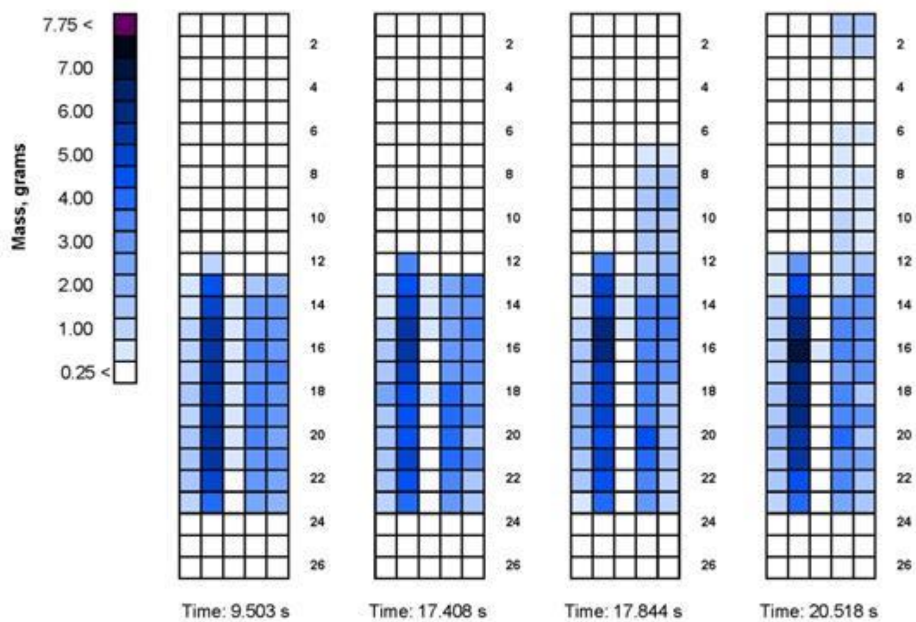
- At the test-item location (the center of the TREAT reactor), the system has a field of view 120-cm tall by 6.7-cm wide area
- The collimator has 360 pixels
  - 36 vertical pixels
  - 10 horizontal pixels.
- Channel spacing:
  - Vertical = 34 mm
  - Horizontal = 7 mm
- Detectable motion
  - Horizontal = 0.2 mm
  - Vertical = 6 mm
- For typical experiment fuel loadings, each pixel has a sensitivity of ~0.1 g of fuel
- Reactor power
  - Minimum = 10 kW
  - Maximum = 20,000 kW



Hodoscope field of view at the test item location

Example Hodoscope Data

- Hodoscope results are often visualized using a symbol or color pictogram
- In this pictogram darker colors indicate areas with more fuel, lighter colors are areas with less fuel
- There were two fuel pins in this experiment, a left pin and a right pin
- The pin on the left was mostly centered on a column of hodoscope pixels, the pin on the right was in-between two columns



Snap-shot views of data from a Hodoscope experiment

This data shows the simultaneous response of two fuel pins to a transient. The pin on the right shows significant axial fuel relocation has occurred at 17.844 seconds. This observation establishes the failure point and the progression of fuel movement after the breach.

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FMMS Refurbishment Needs

- The FMMS detectors date from circa 1960s and are no longer functional
- The FMMS data acquisition system dates from circa 1980s and is on longer functional
- Work is needed to:
  - Inspect, repair/replace, and qualify the detectors
  - Design, assemble, and qualify a new data acquisition system
  - Inspect, repair/replace, and qualify the hodoscope electromechanical systems
  - Develop and qualify shot control system
  - Develop 3-dimensional engineering models and radiation transport simulation models
  - Regain institutional knowledge about conducting transient experiments and interpreting FMMS data

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Fast Neutron Sensors

Idaho National Laboratory

Fast Neutron Sensors

Proton Recoil Scintillator Detector

Proton Recoil Proportional Counter

- Experiments at TREAT are one-of-a-kind events, very-high reliability is required for measurements
- There are 360 detector slots in the hodoscope collimator
- The Hodoscope employed multiple, redundant detector systems to ensure reliability
  - Proton recoil scintillator detector (PRSD) – ZnS/epoxy matrix sandwiched between Lucite hemi-cylinders
  - Proton recoil proportional counters (PRPC) – methane, operated at 5 kV
- During a test, data was recorded from >720 detectors every 0.001 seconds
- Maximum instantaneous data rate is expected up to 400,000 events s^{-1}

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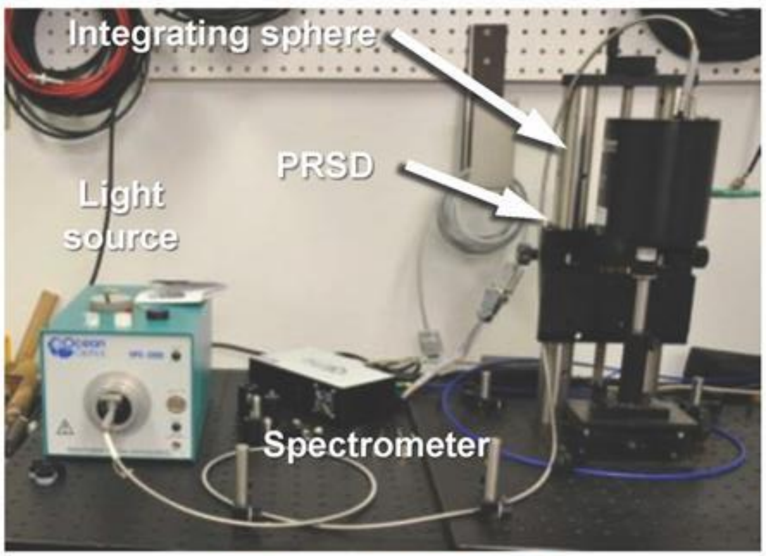
PRSD Scintillators – First Evaluation

- Comprehensive examination of 106 PRSD scintillators recovered from ‘spare parts’ found in the I&C shop
  - Visual inspection
  - Method development for paint stripping and cleaning
  - Fluorescence inspection using CCD-based UV-Vis spectrometer (OceanOptics QEPRO-QE spectrometer) with Xe and LED light sources

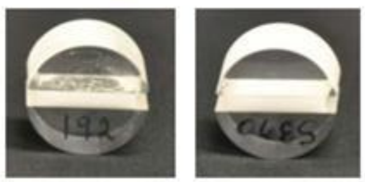


Spectrometer testing setup

PRSD scintillator under ultraviolet illumination

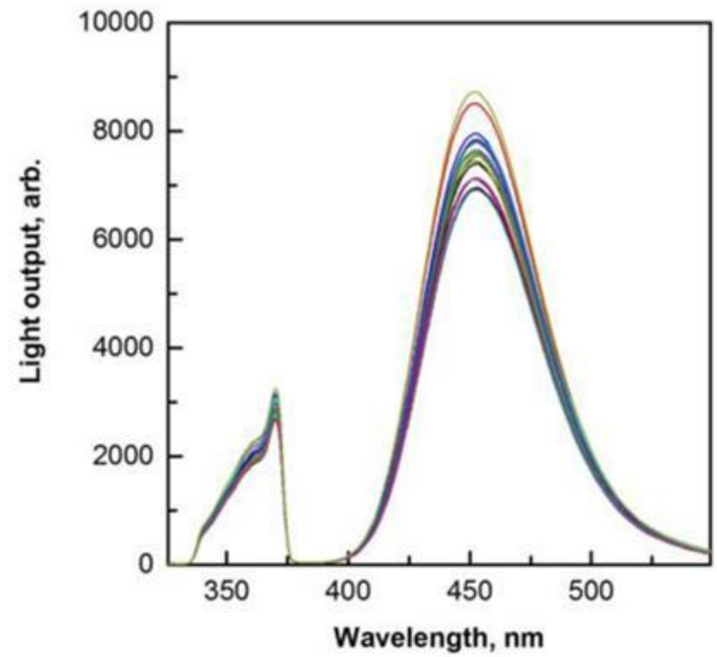
PRS Performance Observations

- Consistent emission spectra but variable light output
- Large variability in scintillator conditions
  - Delamination



  - Yellowing

- Low-performance scintillators were culled from the sample set
- Work is underway to refinish scintillators with air-brush recoating using EJ-510 reflective paint (TiO_2)



Spectral output from 16 PRSD scintillators

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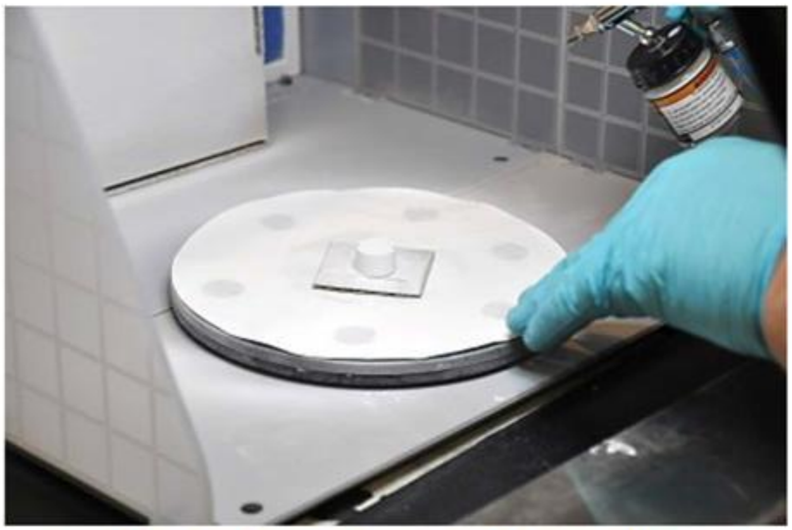
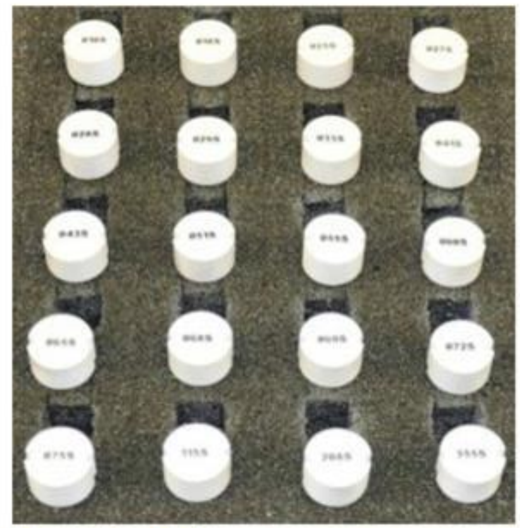
PRS Refurbishment



Paint booth and airbrush being used to spray PRS with scintillation paint inside fume hood at laboratory in the IRC



Refurbished PRSs ready for advancement for detector integration and eventual installation in the hodoscope

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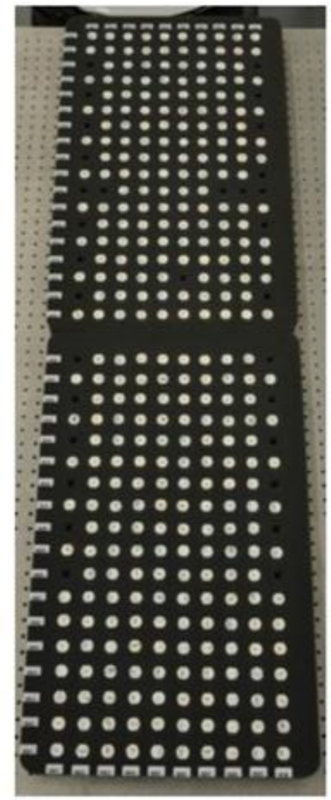
Next Step for PRS Refurbishment

- Recovered 326 PRSs from the TREAT hodoscope
- Several recovered PMTs were found with moisture inside and were determined to have lost vacuum by breach of the glass gate created during manufacture
- Identified that several different methods were used for coupling PRSs to PMTs in the installed array
- 99 PRSs identified, by visual inspection and under 365-nm illumination, as candidates for refurbishment



Removal of PRSDs from the hodoscope's rear detector cabinet (left) and the decoupled PRSs staged in a foam template in BCTC, showing as-found placement (right)

Photomultiplier Tube (PMT) Evaluation

- The PMTs installed at TREAT are not functional and must be replaced
  - Original PMT vendor (AMPEREX) is no longer in business, alternative is needed
  - Candidate photomultiplier tubes were selected based on similarity to the original XP1110 performance specifications
- Tube and socket must have close to the same physical dimensions as the XP1110 to be used in original phenolic PRS/PMT assembly holders
- Candidate tubes were evaluated using the following parameters:
  - Linear gain response
  - Dark current
  - Signal to noise ratio
  - Linearity as a function of photon energy
- A temporary data acquisition system was assembled to allow us to develop a method for characterizing the PMTs

Original Amperex XP1110 PMT and base, with attached PRS and phenolic holder

Candidate replacement PMT examples manufactured by Hamamatsu

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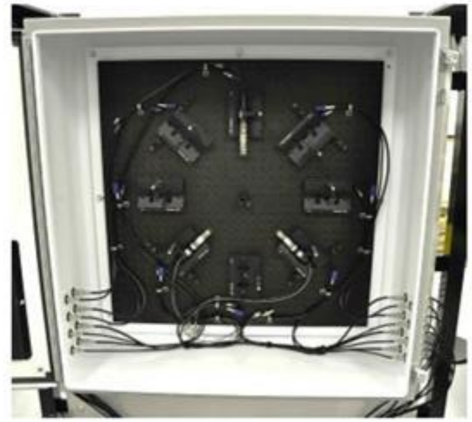
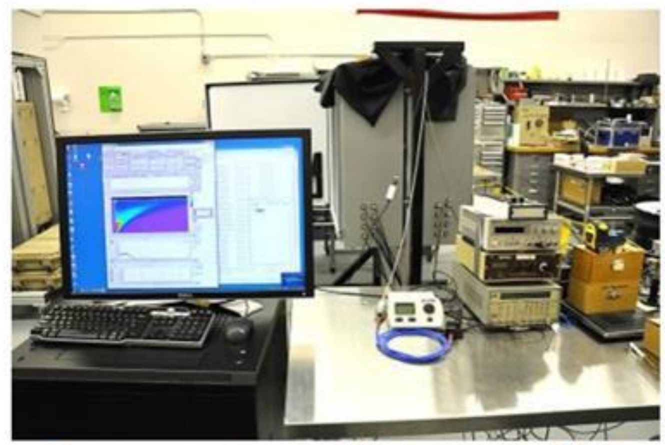
PMT Evaluation Station

- Thorlabs USB-controlled LED Driver and high-stability 420-nm LED
- Custom software written to operate PMT high-voltage supply and analyze many parameters at various light and bias voltage conditions
  - Ability to ramp light output automatically
  - Ability to ramp PMT bias voltage automatically
- Components for a permanent system are now being acquired to assemble a stand-alone system for full-production for acceptance testing and to evaluate and performance-check each PMT before it is integrated with PRS to create a PRSD



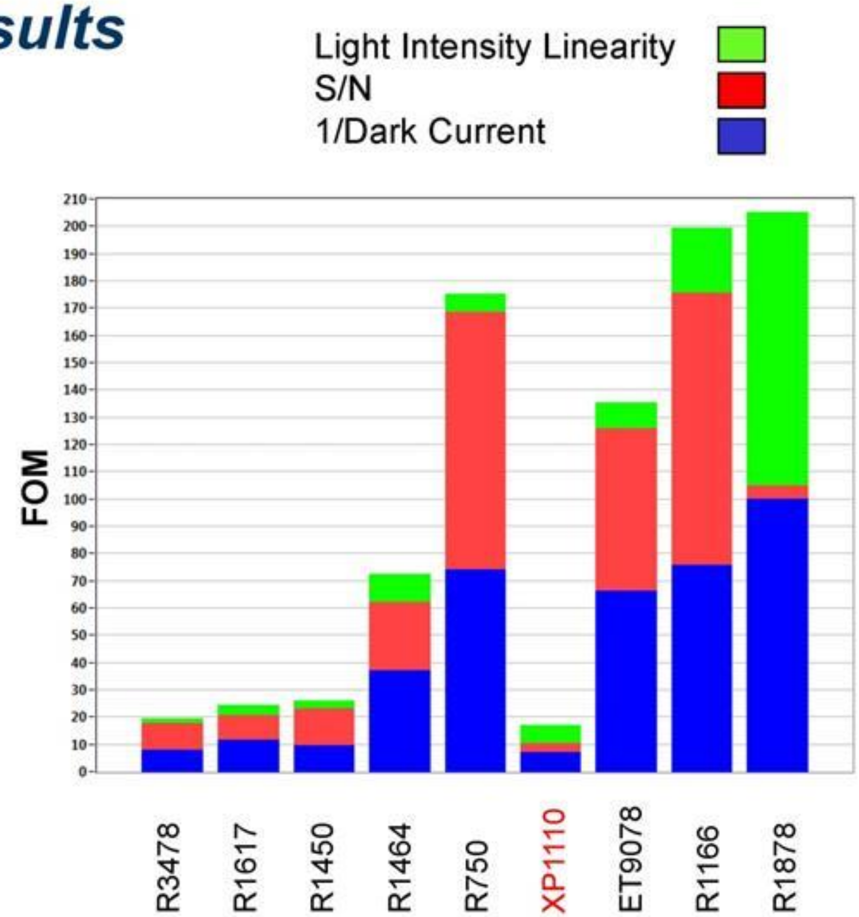
View inside light-tight enclosure used for PMT evaluations



Temporary system used to characterize initial candidate PMT's as a function of photon energy and bias voltage linearity

PMT Evaluation Results

- Figures of merit generated to compare and assess the PMTs
- Optimal total FOM is based on magnitude and balance among the three FOMs
- Final choice: R1166



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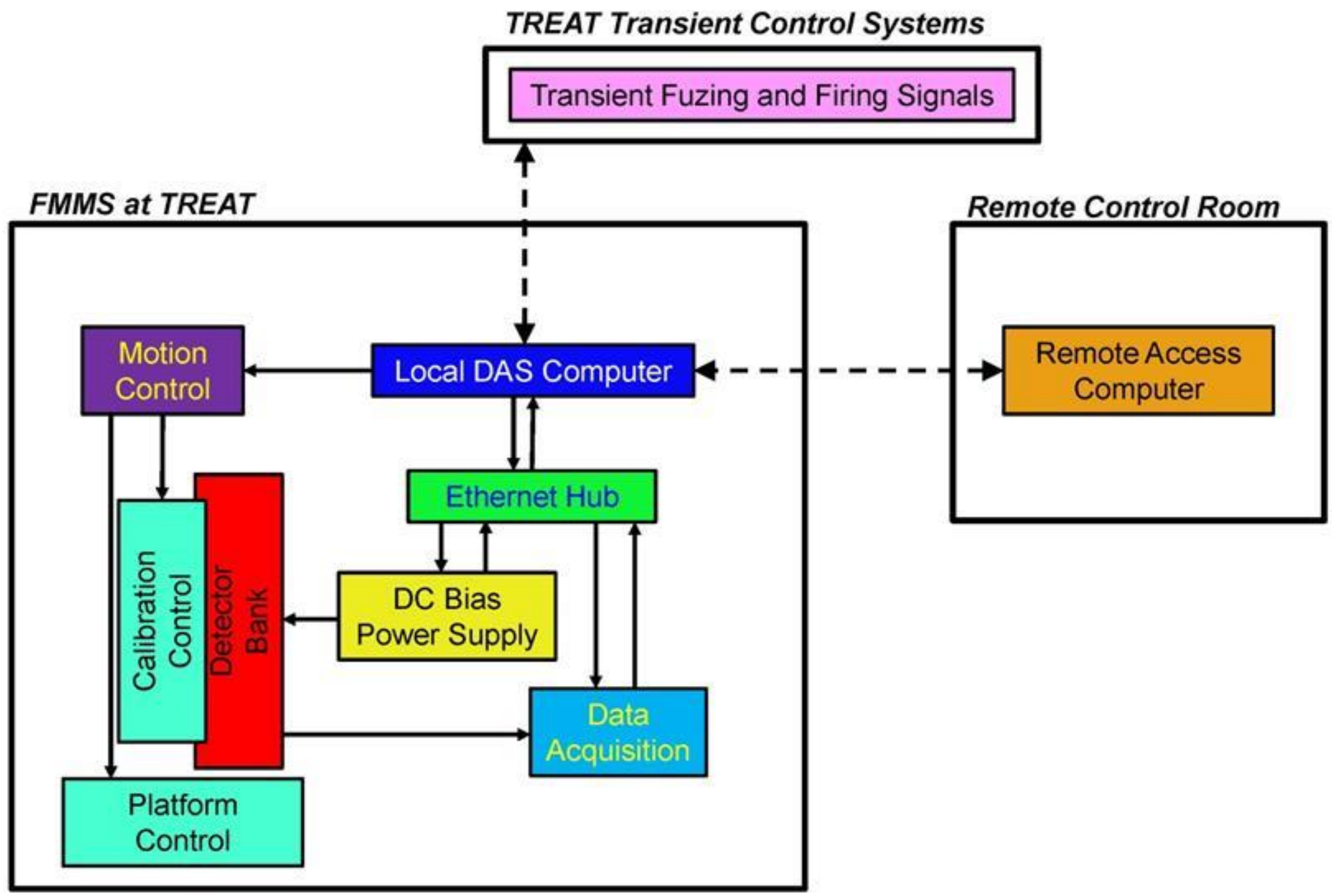
Data Acquisition System

FMMS DAS Architecture



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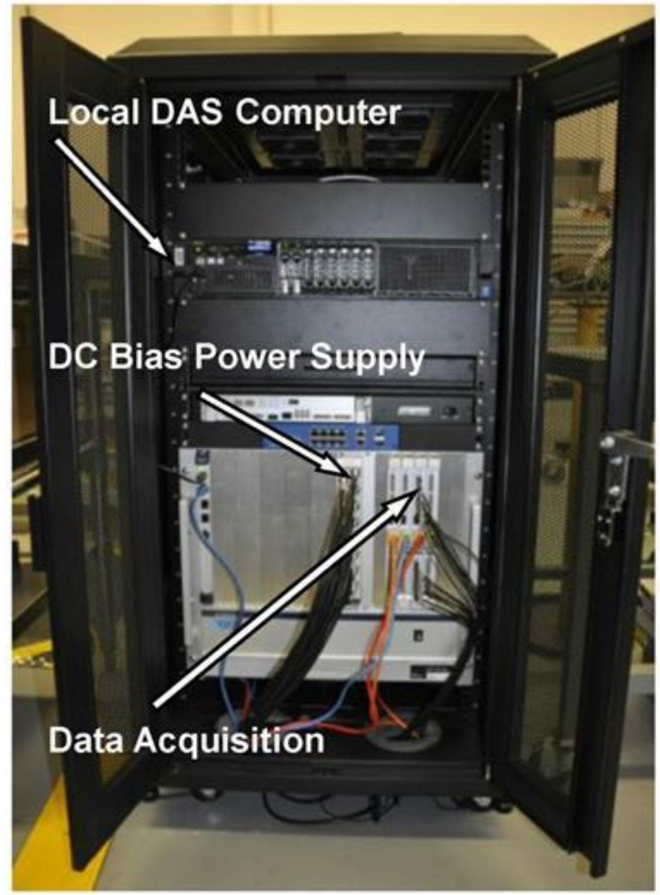
Initial DAS Installation

- DAS cabinet (half-sized for prototype)
  - Forced air cooling
  - Can hold two VME crates
- WIENER VME crate – each crate can support 96 channels (HV and data)
- Local DAS Computer
  - Dell Precision Rack 7910
  - Four Quad-Core Processors
  - 128 GB RAM
- PMT high-voltage supply
  - ISEG Mpod HV Module
  - 3 kV/3 mA
  - 16 channels/module
- Data acquisition
  - Struck 3316
  - 250 MS/s
  - 14-bit
  - 16 channels/card



The prototype DAS assembled in our laboratory, configured to control 16 PRDS (HV and data)

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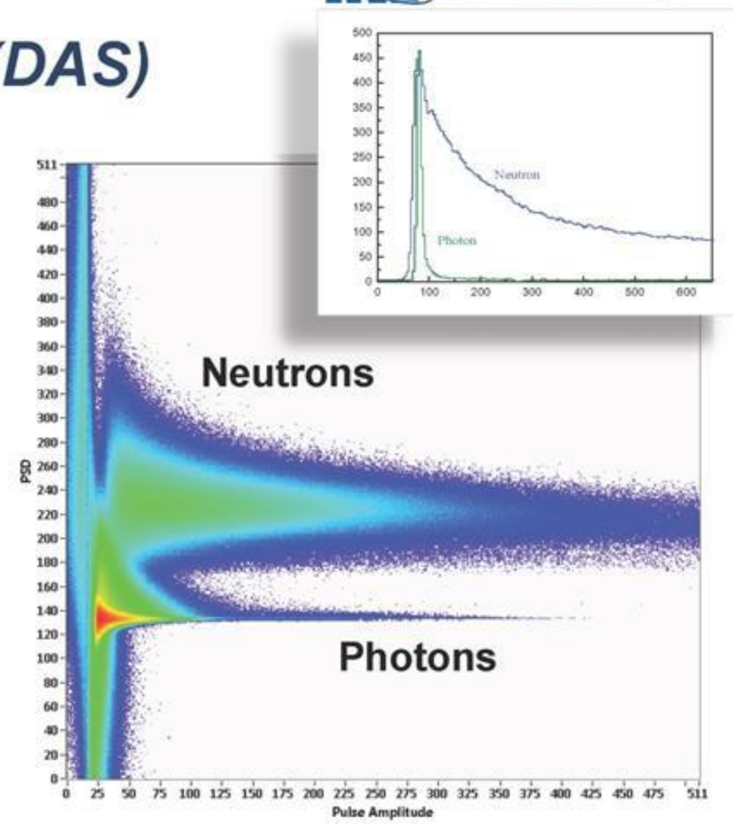
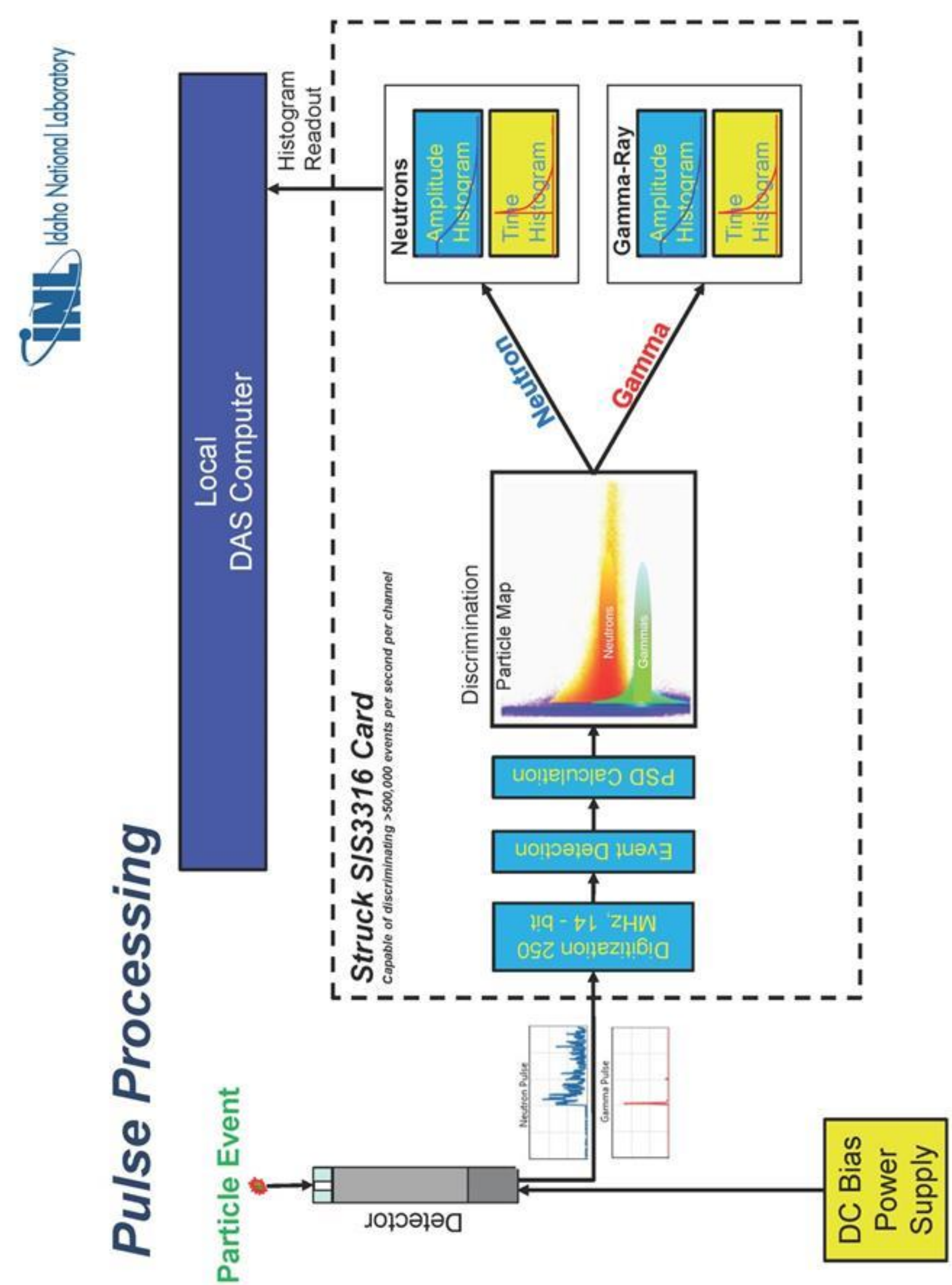
Data Acquisition System (DAS)

- Using STRUCK SIS3316-250-14 (16 channel 250 MSPS 14 bit) digitizers
  - Input range 2 or 5 volts
  - Global start/time reset input
  - Particle-energy determination
  - Particle-type determination via pulse-shape discrimination (PSD)
  - Two internal 2-D histograms per channel: one used for particle type mapping and one used as the discrimination look up table
  - Programmable time bin size
- High event rates known to degrade PSD performance → original hodoscope implementation did not use PSD (poor n/γ S/R)



Preliminary PSD testing of a PRSD scintillator using DAS prototype



Pulse Processing
Idaho National Laboratory
Particle Event
Detector
Neutron Pulse
Gamma Pulse
DC Bias Power Supply
Struck SIS3316 Card
Capable of discriminating >500,000 events per second per channel
Digitization 250 MHz, 14 - bit
Event Detection
PSD Calculation
Discrimination
Particle Map
Neutrons
Gammas
Neutron
Gamma
Neutrons
Amplitude Histogram
Time Histogram
Gamma-Ray
Amplitude Histogram
Time Histogram
Histogram Readout
Local DAS Computer

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Summary

- TREAT reactor will resume transient testing in 2018
- Monitoring the location and movement of fuel during transient tests provides critical information needed to understand accident progression and fuel performance margins
- Fast-neutron imaging, using a large steel hodoscope, is used at TREAT to infer the location of fuel in the reactor's core during transient tests
- Work is underway to return the TREAT hodoscope to operation by refurbishing the PRS detectors and their data acquisition system

Forward! To Transient #2885!

INL

Idaho National Laboratory

CABRI Hodoscope

Transient Test Reactor Physics Workshop

Sun Valley – May 5th 2016

IRSN
INSTITUT
DE RADIOPROTECTION
ET DE SÛRETÉ NUCLÉAIRE

Faire avancer la sûreté nucléaire

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IRSN/PSN-RES/SEREX/L2EP

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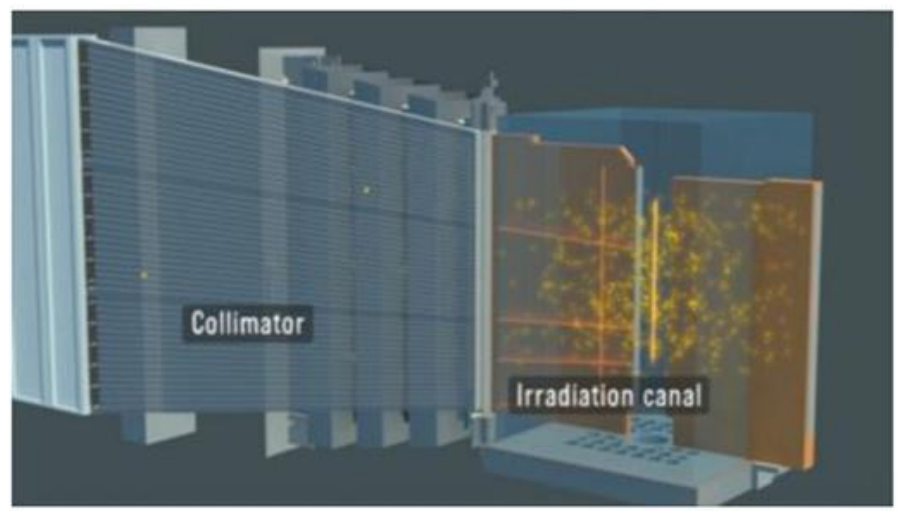
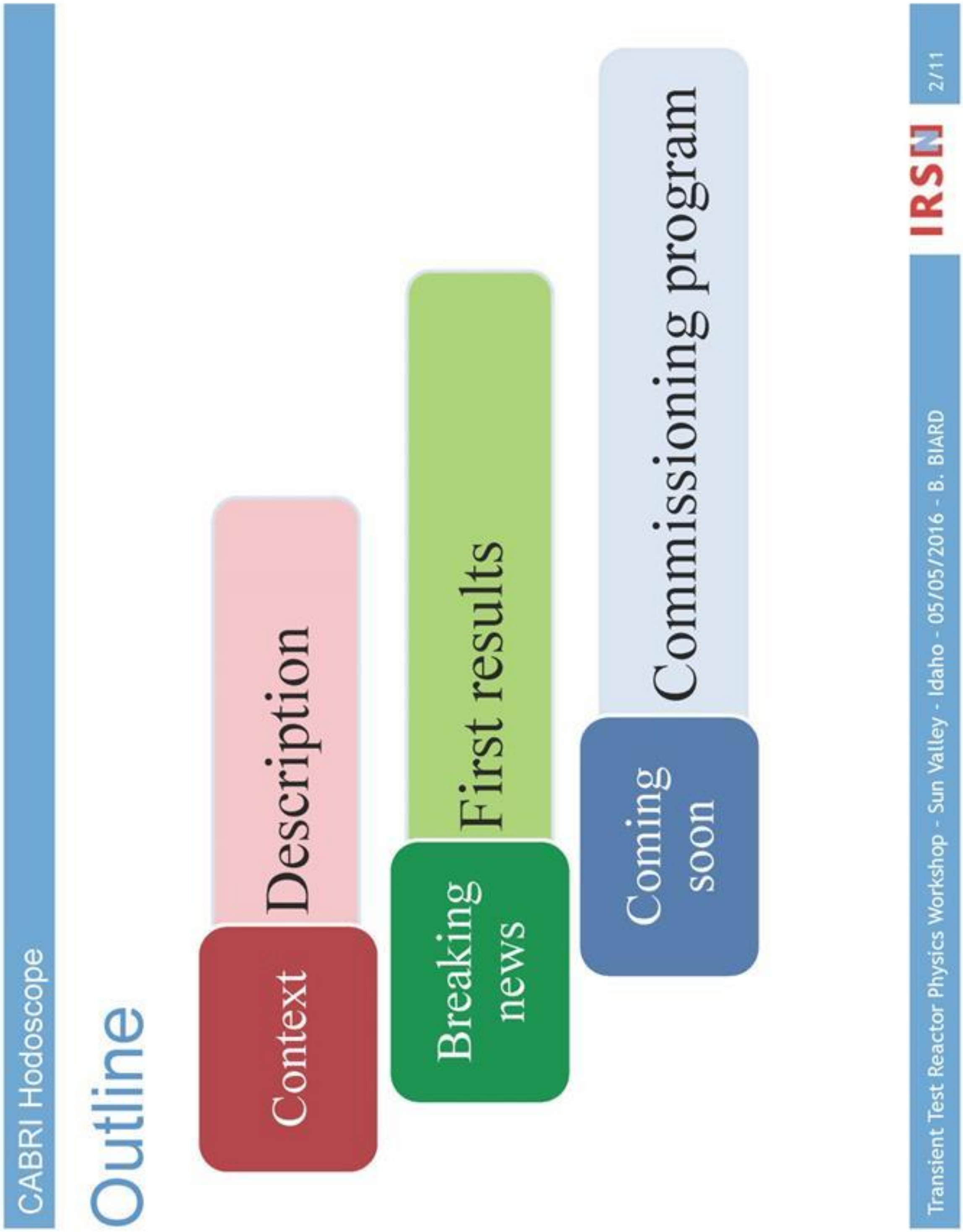



CABRI Hodoscope
Outline
Context
Description
Breaking news
First results
Coming soon
Commissioning program
Transient Test Reactor Physics Workshop - Sun Valley - Idaho - 05/05/2016 - B. BIARD
IRSN
2/11

Some contextual facts

A major renovation of the whole CABRI facility

- A new water loop to comply with

➔ Which impact for the measurement ? Assessment and adaptation required

- Up to date safety requirements: new seismic constraints, fire and thunder protection, post-Fukushima complementary safety surveys ...

➔ Major civil engineering and mechanical work in the hodoscope immediate vicinity (for the hodoscope itself and for the surrounding elements)

➔ Aggressive environment for sensible parts

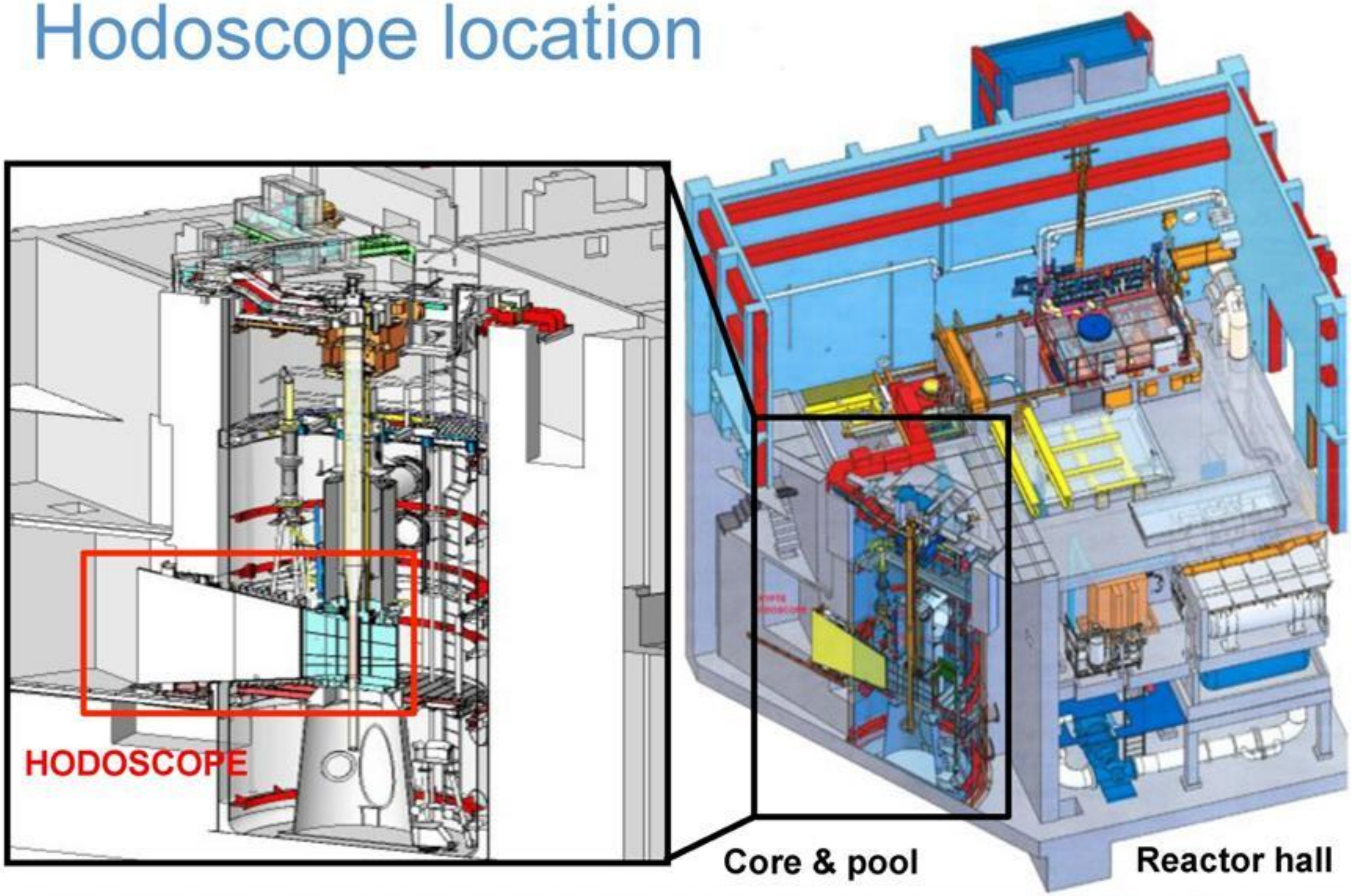
A long standby ....

- Last RIA test by the end of 2002, last transient and hodoscope functioning in 2003
- First criticality in the water loop configuration October 2015

➔ Obsolescence ? Available support ? Possible upgrades ?

➔ How to maintain skills availabity ?

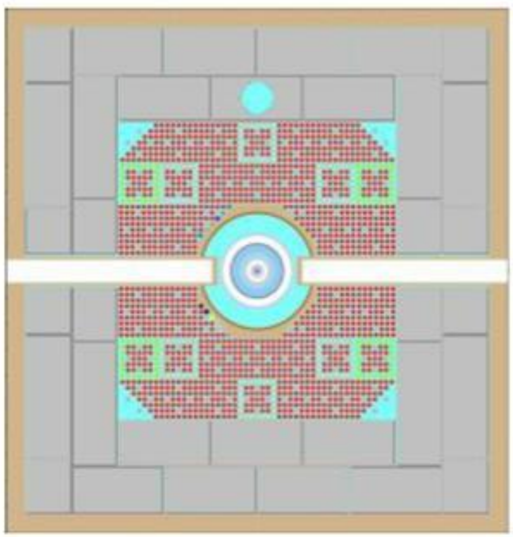
Hodoscope location

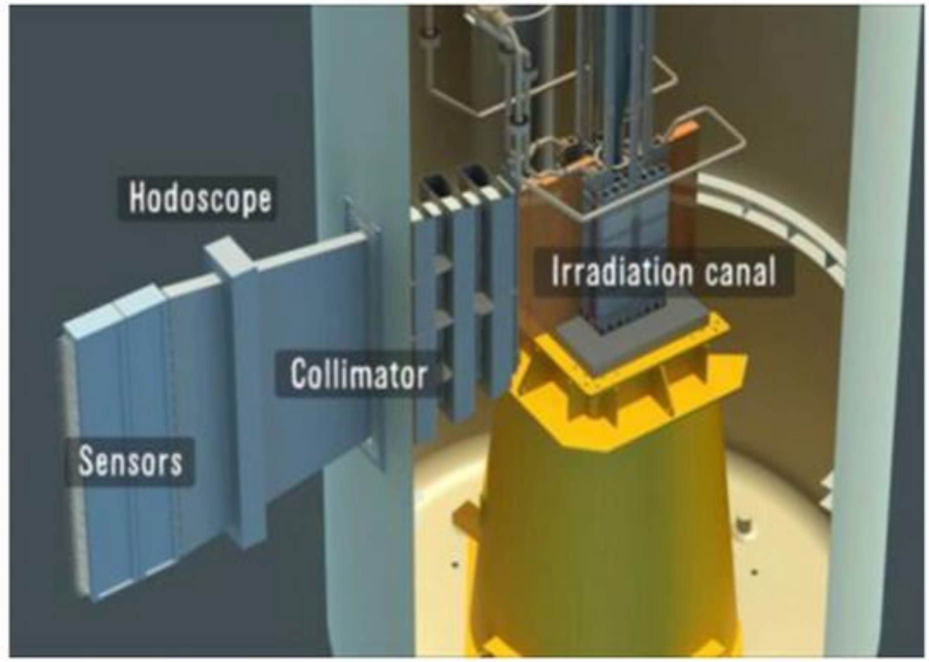


CABRI Hodoscope: on-line fuel motion diagnostics

Rapid neutron detectors :

- On-line fuel motion detection (displacement, ejection, relocation)
- test rod fissile length
- driver core power profile
- pulse width





306 counting tracks:

- 51rows x 3 columns collimator
- 153 Fission Chambers
- 153 Proton Recoil counters
- up to 1 ms acquisition rate

Collimator

Front view (facing the fuel) →

Housed in a sheath connected to the core by a bellows

Length : 3 m

Weight : 5300 kg

Material : Stainless steel

Motorized in rotation and verticality

51 rows x 3 columns ==> 153 windows

Distance to core axis : 1m

<u>Channel size :</u>

Front side : 7.5 x 15 mm

Back side : 10 x 20 mm

↑ *Back view (detector side)*

Field of view at core axis: 10.2x20.4mm

collimator → detector → electronics ↔ Control station

Detector bank

Collimator (back side)

153 Fission Chambers (CF)

(^{237}Np coating, Argon gas)

Low efficiency, low dead time

→ high saturation level

→ power transient measurement

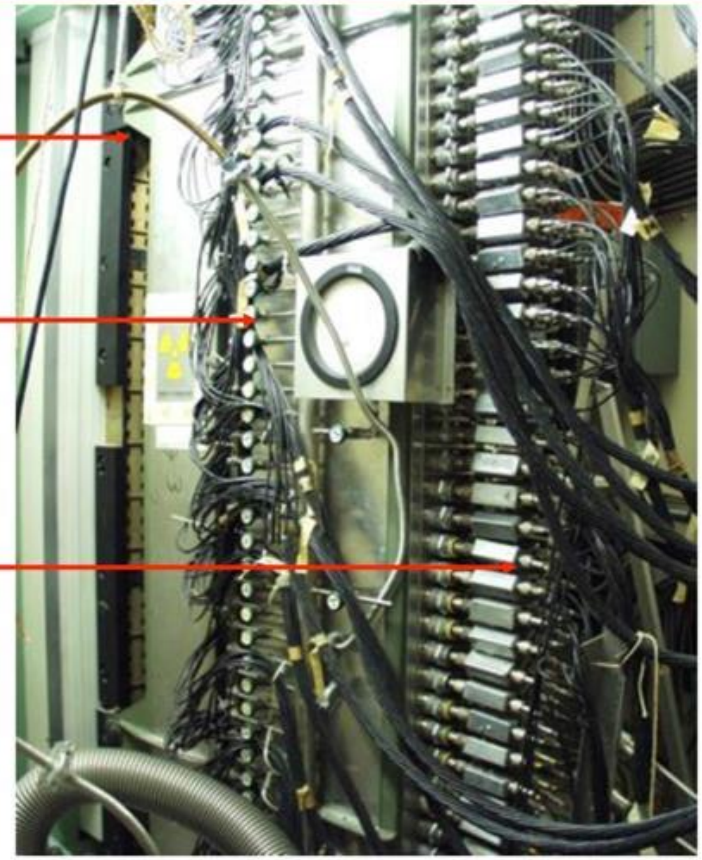
153 Proton Recoil Counters (PR)

(CH_4 ionization chamber)

Higher efficiency, higher dead time

→ reduced noise,

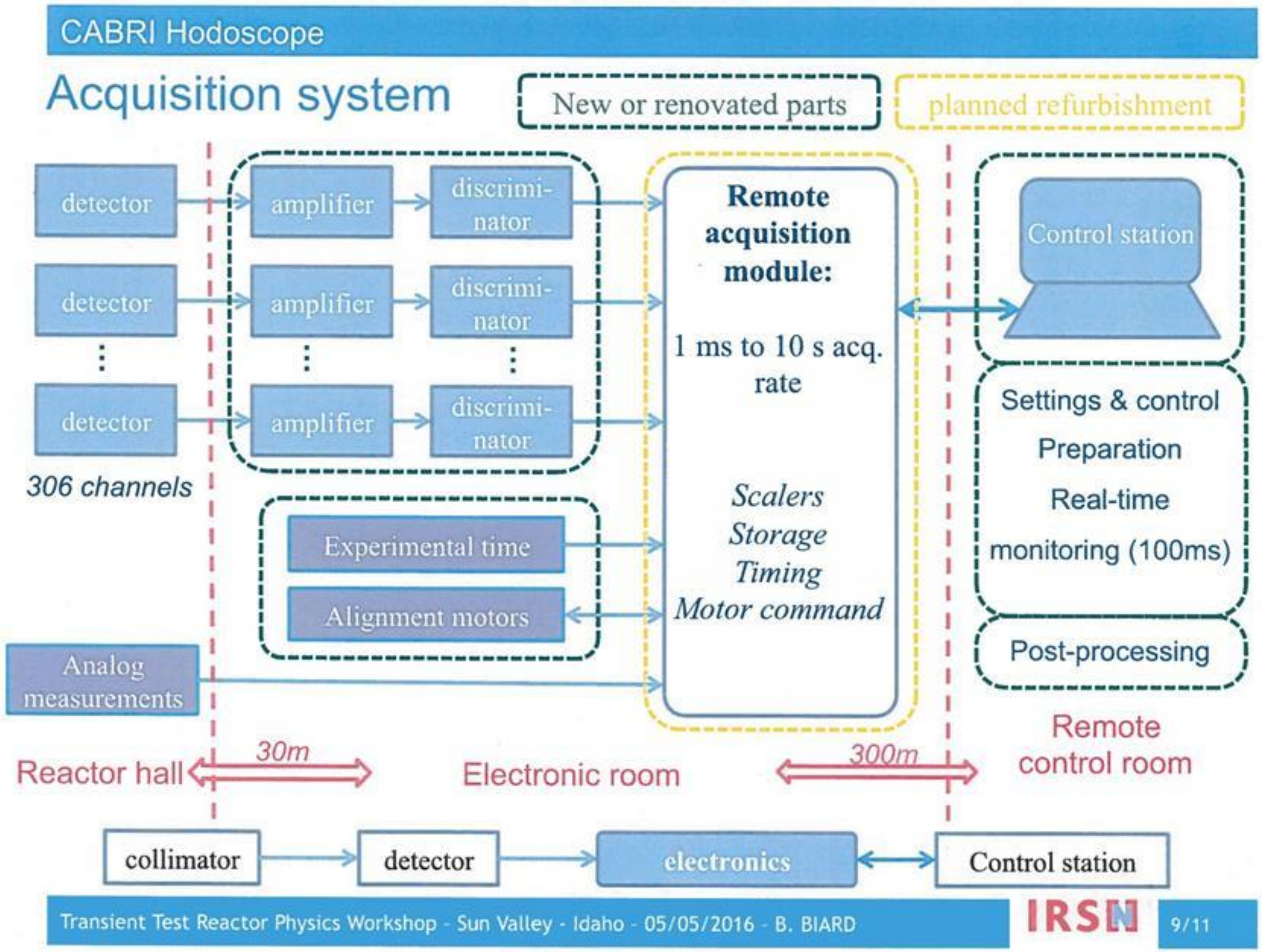
→ "low" power measurement



collimator → **detector** → electronics ↔ Control station

Detector characteristics

	Fission Chamber	Proton Recoil counter
Nature	^{237}Np deposit on SS foils (~400mg)	CH_4 ionization chamber tungsten wire anode
Filling gas	Argon 5 bar	CH_4 1.2 bar (adjustable)
Efficiency	$6\ 10^{-4}$ / neutron	$2\ 10^{-2}$ / neutron
Max count rate	$5\ 10^{6}$ cts/s	$2\ 10^{6}$ cts/s
S/B *(expected identical for water or Na loop)*	0.15	0.06
Saturation	> 20,000 MW	~50 MW
Dead time	~80 ns	~240 ns
Main purpose	Higher saturation threshold: **transient measurement**	Lower noise: **low power measurement**


CABRI Hodoscope
Acquisition system
New or renovated parts
planned refurbishment
detector
detector
detector
306 channels
amplifier
amplifier
amplifier
discrimi-nator
discrimi-nator
discrimi-nator
Experimental time
Alignment motors
Analog measurements
Remote acquisition module:
1 ms to 10 s acq. rate
Scalers
Storage
Timing
Motor command
Control station
Settings & control
Preparation
Real-time monitoring (100ms)
Post-processing
Reactor hall
30m
Electronic room
300m
Remote control room
collimator
detector
electronics
Control station
Transient Test Reactor Physics Workshop - Sun Valley - Idaho - 05/05/2016 - B. BIARD
IRSN
9/11

Acquisition system

Electronic cabinets

Deported acquisition module

collimator → detector → electronics ↔ Control station

Renovation of the acquisition electronics

- **Lack of availability** for some **amplifier/discriminator** components combined to limited spare parts: how to repair in case of trouble ?

→ Full-discrete design prototypes: must match the actual size and connections and present **same or better characteristics**

→ Tests within ISIS reactor : OK

→ **On-site tests**

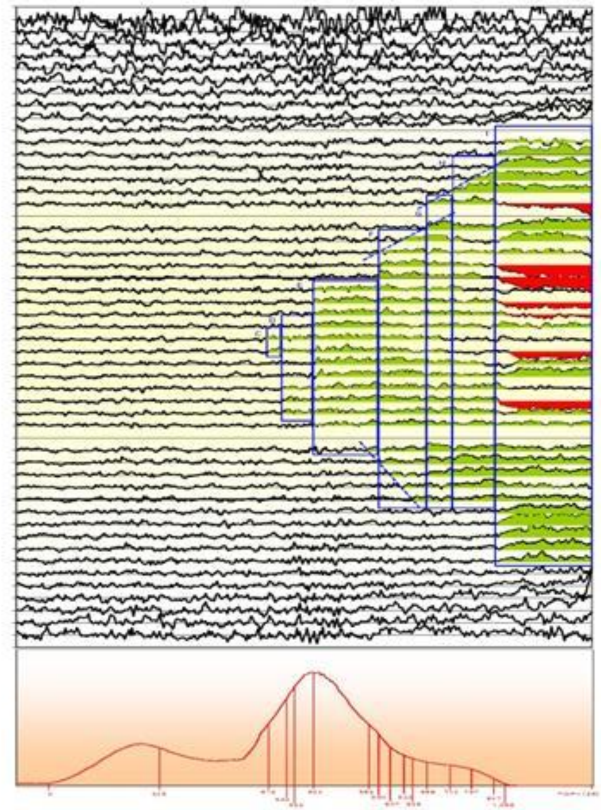
- → **Integration tests** (electronic pulse generator and neutron source) successfully completed: same resolution and amplitude, lower dead time and electric consumption
- → **Steady-state low power** CABRI functioning (up to 80kW) : OK
- → Steady-state High power and **transient tests** (start-up) : to be realised

Signal processing improvement

- Dead time characterization thanks to an experimental campaign in SILENE
  → better knowledge for a better correction

- Signal processing
  → wavelet analysis algorithm

- Signal to mass conversion
  → multigroup neutronic calculation with MORET (Monte-Carlo code)



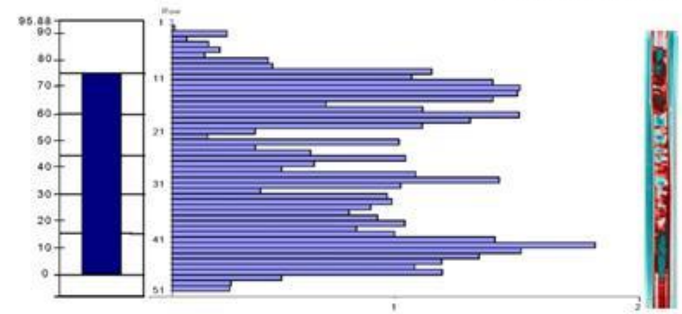
collimator → detector → electronics ↔ Control station

Signal to mass conversion

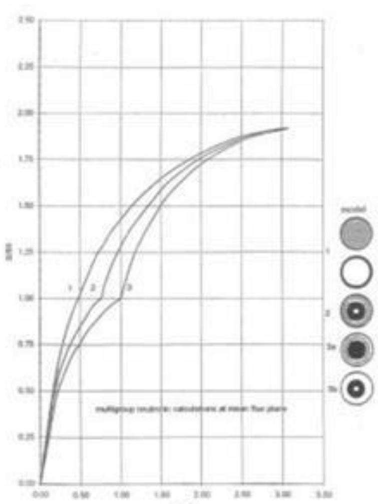
Comparison with radiography (IRIS measurements)

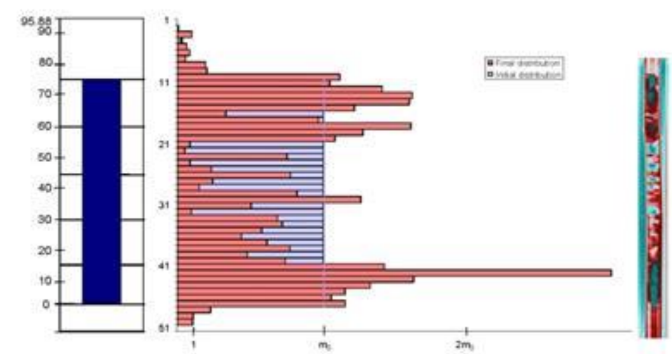
100 kW post-transient power plateau

➔ Normalized integrated signals



Abacus issued from MonteCarlo MORET calculation (peculiar to each rod geometry and device and test channel configuration)



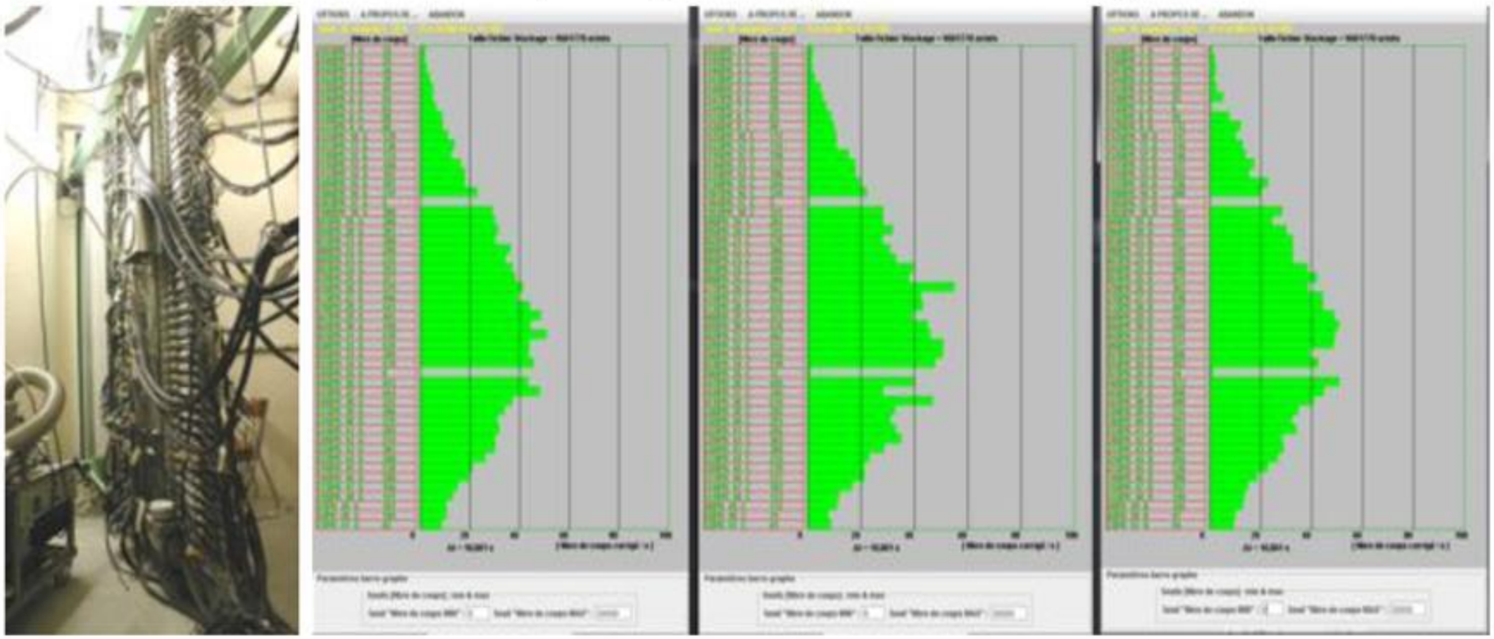


Final distribution of the fuel mass in the rod

collimator → detector → electronics ↔ Control station

Hodoscope first results

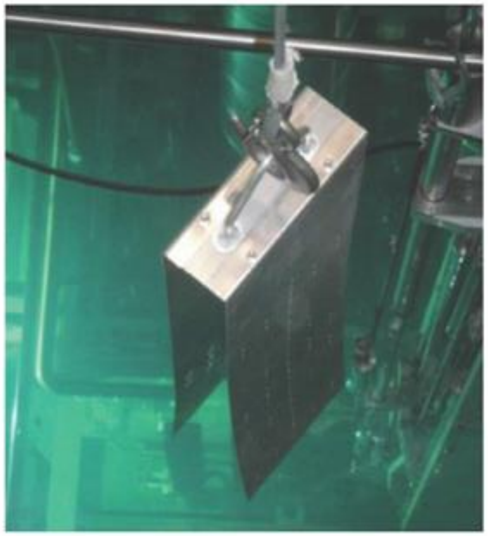
- 12 years between last CABRI core functioning (2003 – Sodium loop) and first criticality in the Water Loop configuration (October 20th 2015)

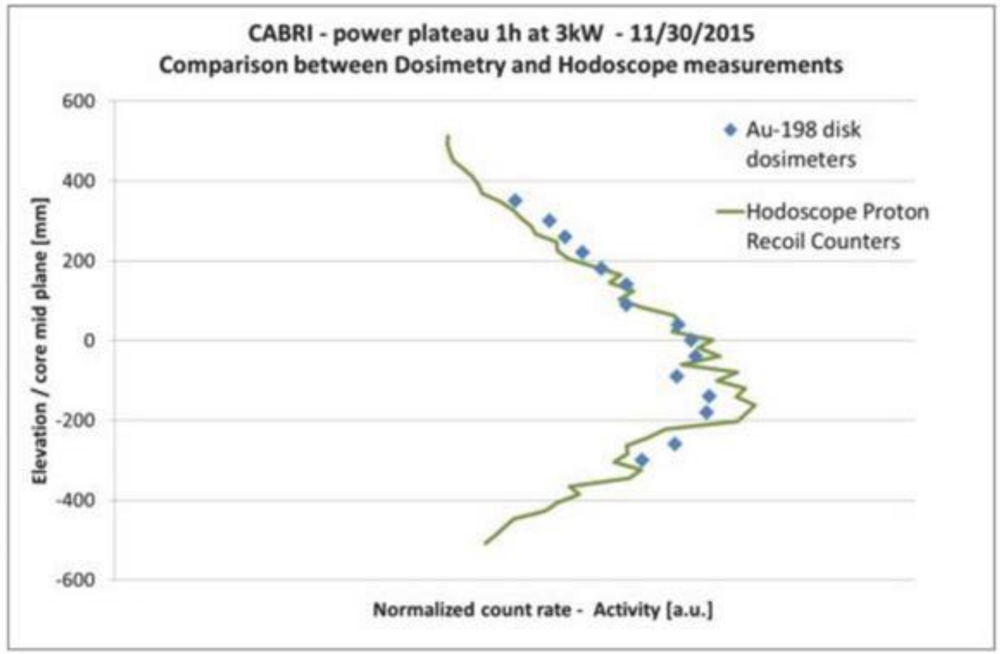


→ Good behavior of all the command & control system and of the detectors (PR and CF) at very low power level (3kW to 80 kW)

→ Determination of lower threshold for detection

Hodoscope first results





→ Good qualitative agreement between raw data count from hodoscope detectors and dosimeters

Next: commissioning program

Global Functioning

- Monitoring of power plateaus and transients (start-up)
  → check the response of all detectors on the whole power and acquisition rate range (linearity, saturation, offset, dead time…)
  → check the command & control system (real-time monitoring, automatic recording sequence, redundancies, external acquisition…)

Alignment

- Distance between collimator and tested rod (axis of the test cell)
- Verticality
- Plus: 1st thermal balance (test device with a fuel rod)

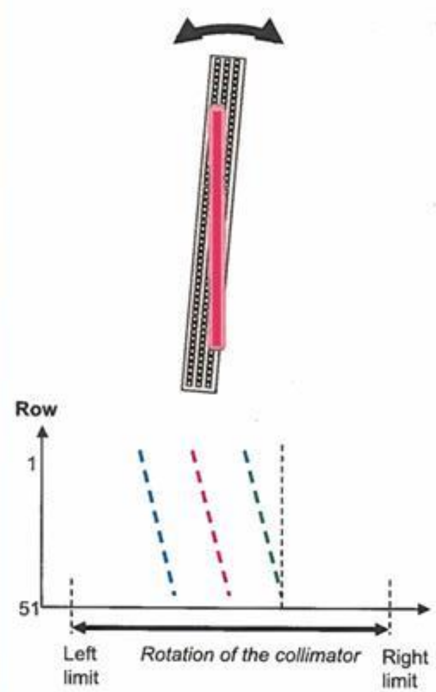
Calibration

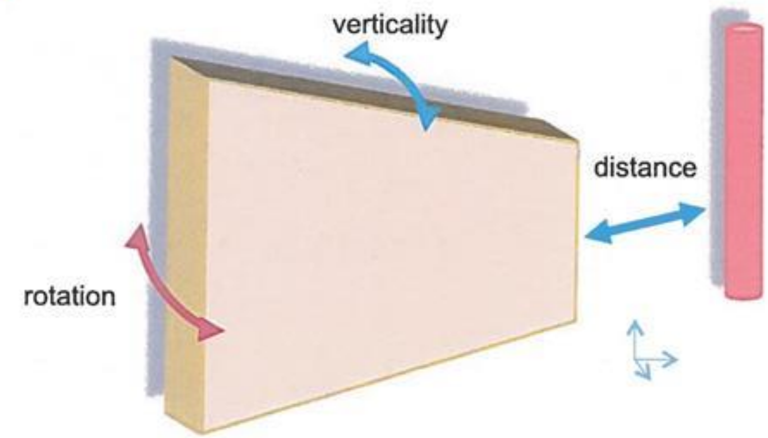
- Power plateau performed on the hodoscope calibration device
- Transfer and quantitative gamma scanning in IRIS

CABRI Hodoscope

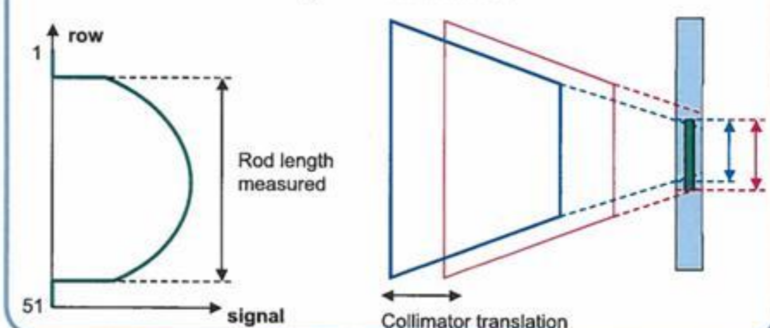
Hodoscope alignment

Verticality of the collimator
Criteria: signal maxima position

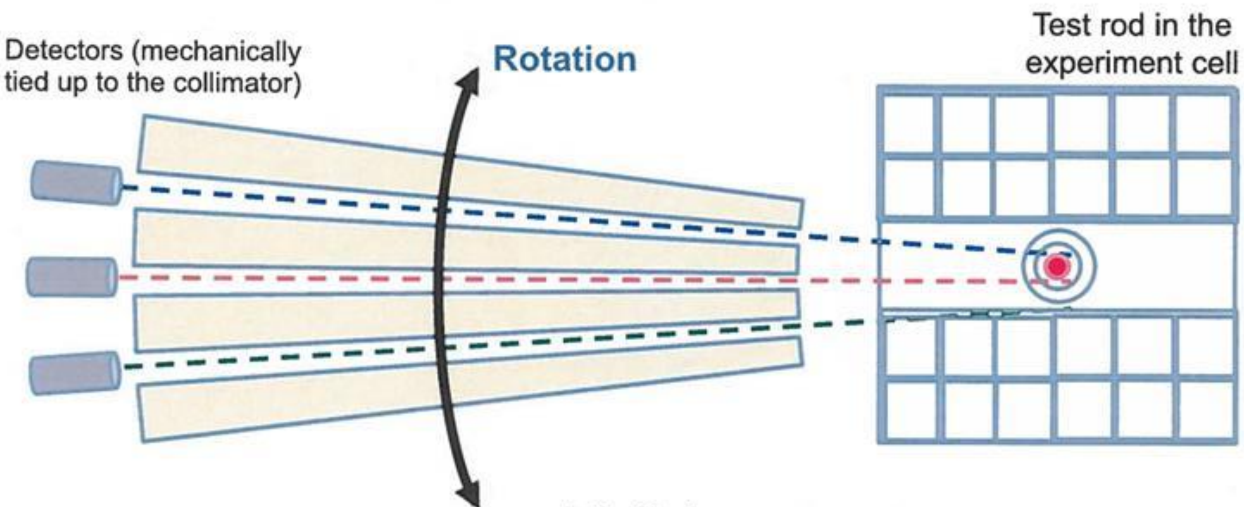




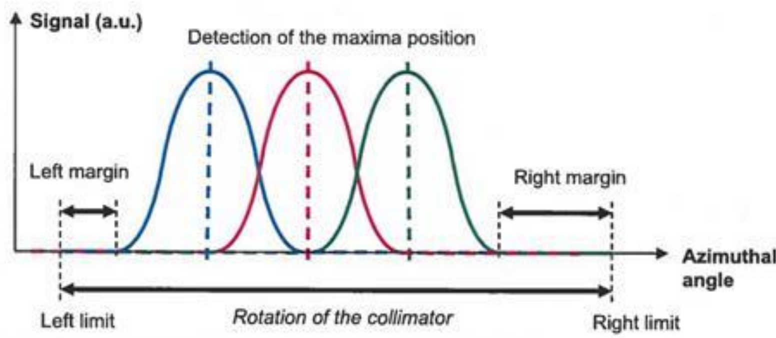
Collimator to test cell distance adjustment
Criteria: correct rod length measurement



Azimuthal positioning of the collimator



The collimator is motioned during CABRI core steady-state functioning (typically ~100kW)
→ Determination of the split and maxima positions
Then it is stopped in the split position for triggering the pulse



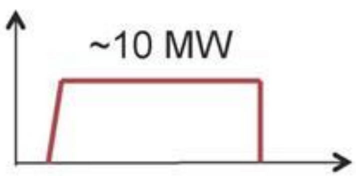
Hodoscope calibration device

- Almost standard test vehicle:
  - Actual size and materials
  - Restricted instrumentation (no transient sensors, less thermocouples)
  - Fresh UO_2 fuel rod

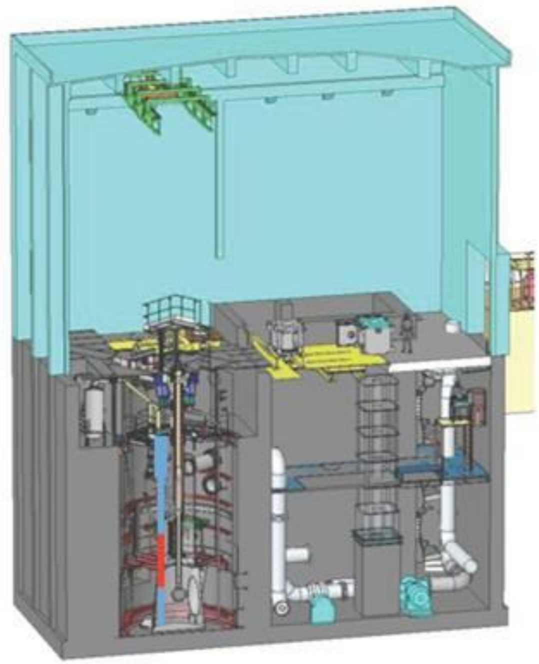
- May be positioned in the test cell in the center of the CABRI core or in IRIS NDE facility

Calibration procedure

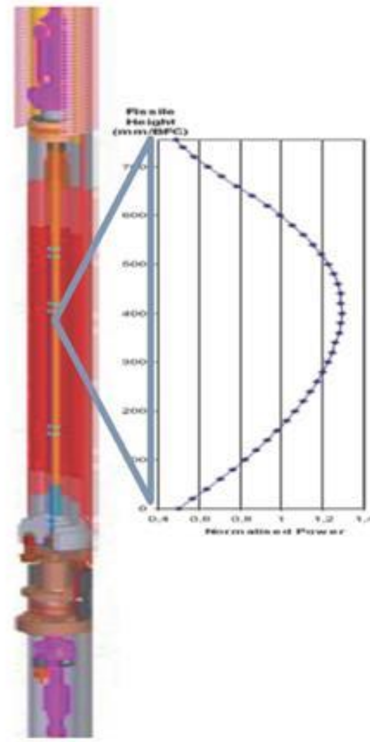
Steady-state
power irradiation



Test device recovery
and transfer to IRIS



Quantitative
gamma-scanning



→ Comparison of hodoscope recording and gamma profile

Thank you for your attention