IAEA 60 Years
Atoms for Peace and Development

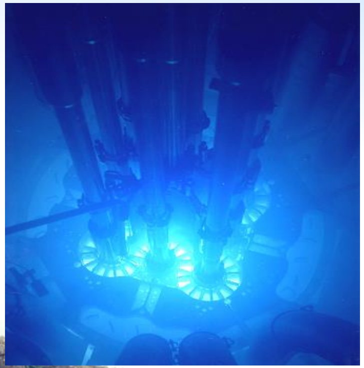
Testing Capabilities and Unique Features of High Capacity MTRs (from publication profiles)

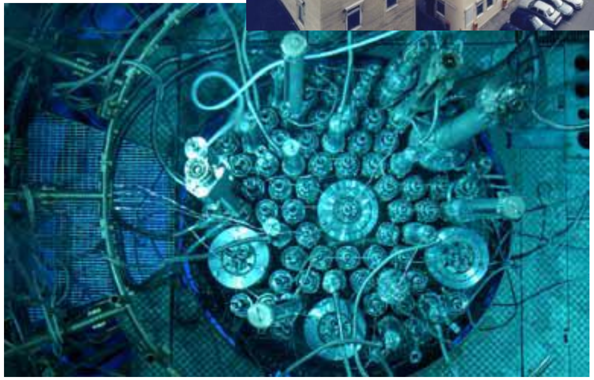
Frances Marshall
(F.Marshall@iaea.org)

Research Reactor Section
International Atomic Energy Agency
November 2017

High Capacity MTRs

- Advanced Test Reactor – USA
- Belgium Reactor 2 – Belgium
- Halden Boiling Water Reactor – Norway
- High Flux Isotope Reactor – USA
- MIR.M1 – Russia
- SM-3 - Russia





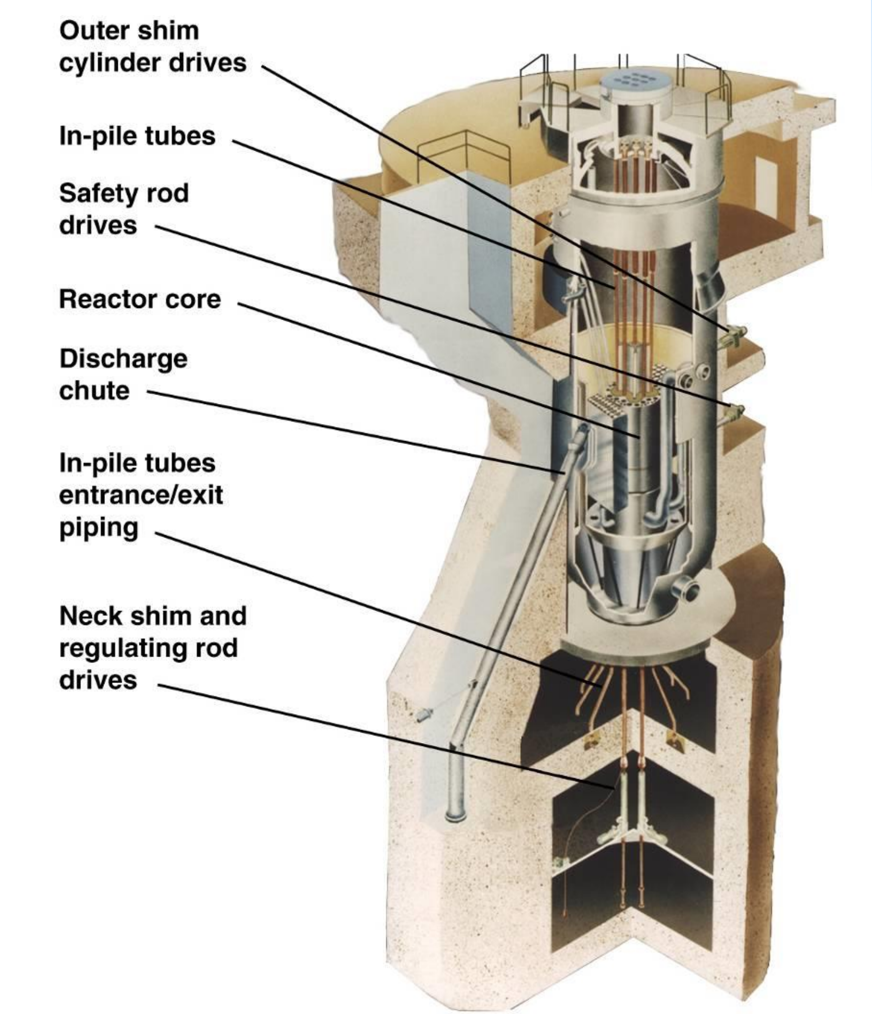
Advanced Test Reactor (USA) Overview

Reactor Type
Pressurized, light-water moderated and cooled; beryllium reflector

Reactor Vessel
3.65 m diameter cylinder,
10.67 m high stainless steel

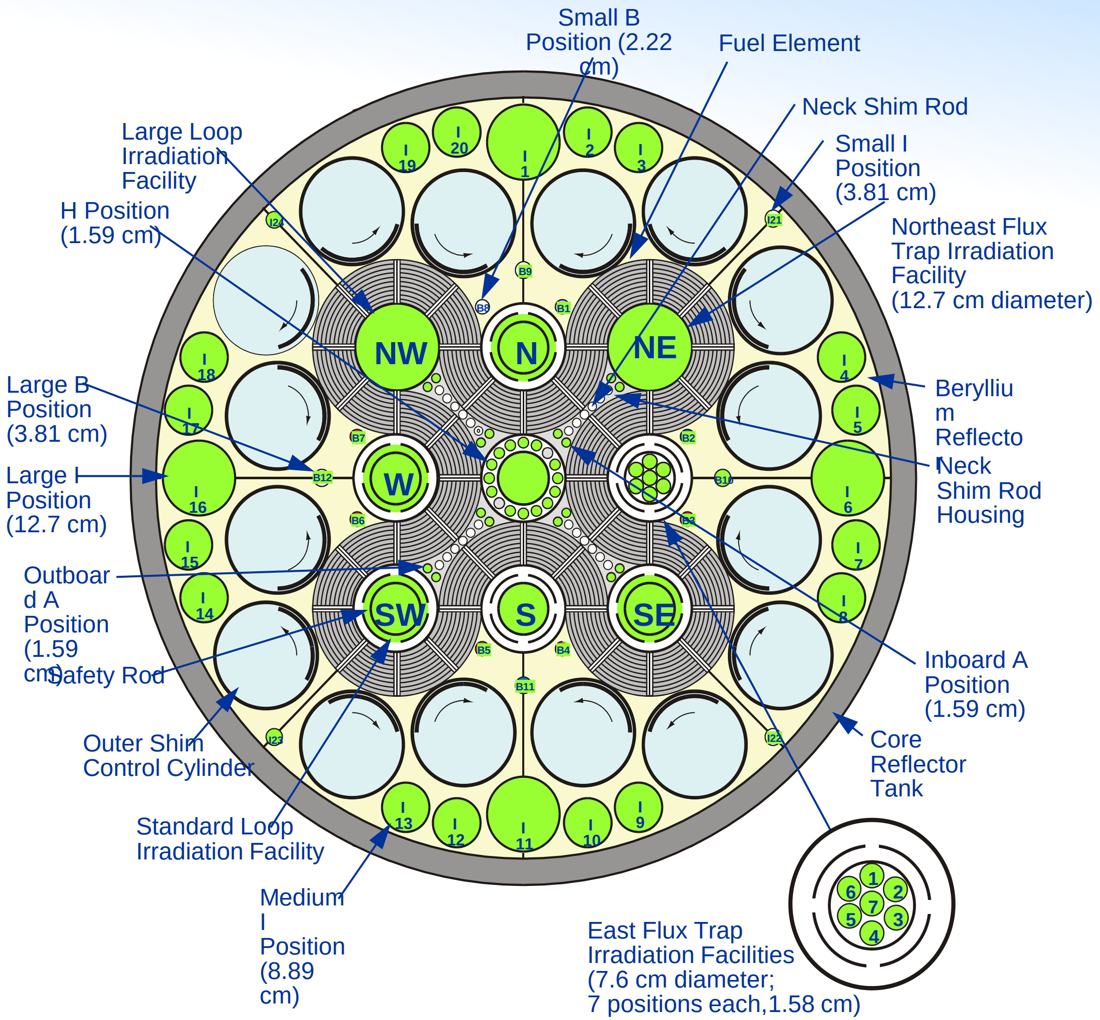
Maximum Flux, at 250 MW
1×10^{15} n/cm^2-sec thermal
5×10^{14} n/cm^2-sec fast

Operating Conditions
Pressure - 2.44 MPa
Outlet Temperature - ~72 °C
Fuel Temperature - ~240 °C



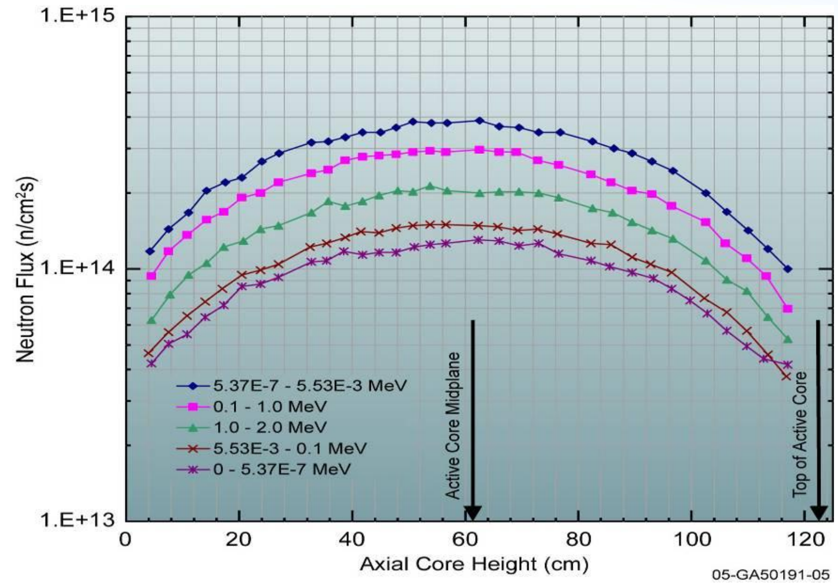
ATR Test Positions

- Test size – 1.2m length, 01.25 to 12.5 cm diameter
- 77 Irradiation Positions
- Rotating Hafnium Control Cylinders – symmetrical axial flux
- Power/Flux Adjustments (“Tilt”) across the Core - ≤ 3:1 ratio
- 4 corners (“Lobes”) can be operated at different powers – like 4 reactors operating as one



Unique ATR Design Features

- Combination of high flux and large test volumes
- Symmetrical axial power profile
- Individual experiment parameter control for multiple tests in a single irradiation position
- Individual experiment control in separate loops
- Accelerated testing for fuels – up to 20x actual operation time for some fuel types
- No design limited lifetime: expected to operate for many more years
  - Core Internals Changeout outages – new reactor internals
  - Large stainless steel reactor vessel – minimal embrittlement
- Capability to perform operating transient testing (i.e., not accidents)



Center Flux Trap Flux Profile (at 125 MW)

ATR Experiment Configurations

Simple Static Capsules

- Reflector positions or flux traps
- Isotopes, structural materials, fuel coupons or pellets

Instrumented Lead Experiments

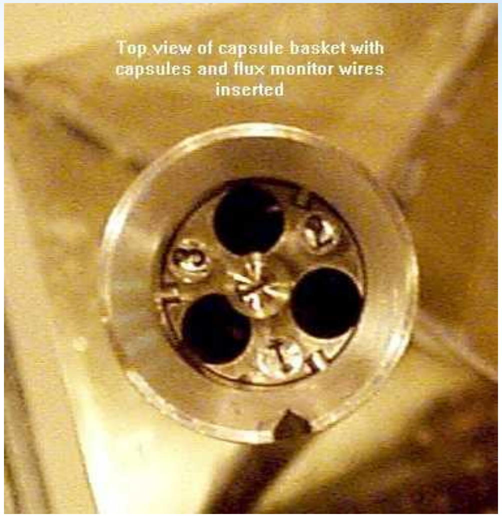
- On-line experiment measurements
- With or without temperature control
- Structural materials, cladding, fuel pins

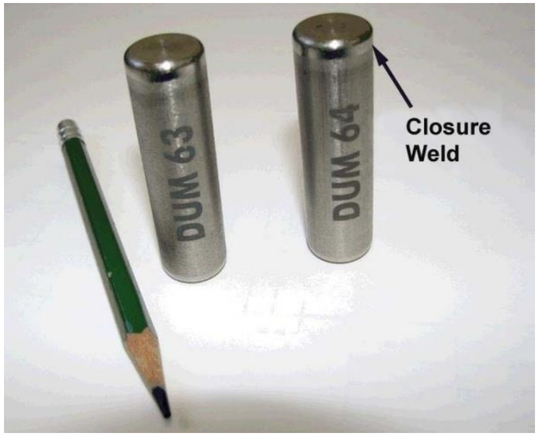
Pressurized Water Loops

- Six loops installed in flux traps
- Control pressure, temperature, chemistry
- Structural materials, cladding, tubing, fuel assemblies

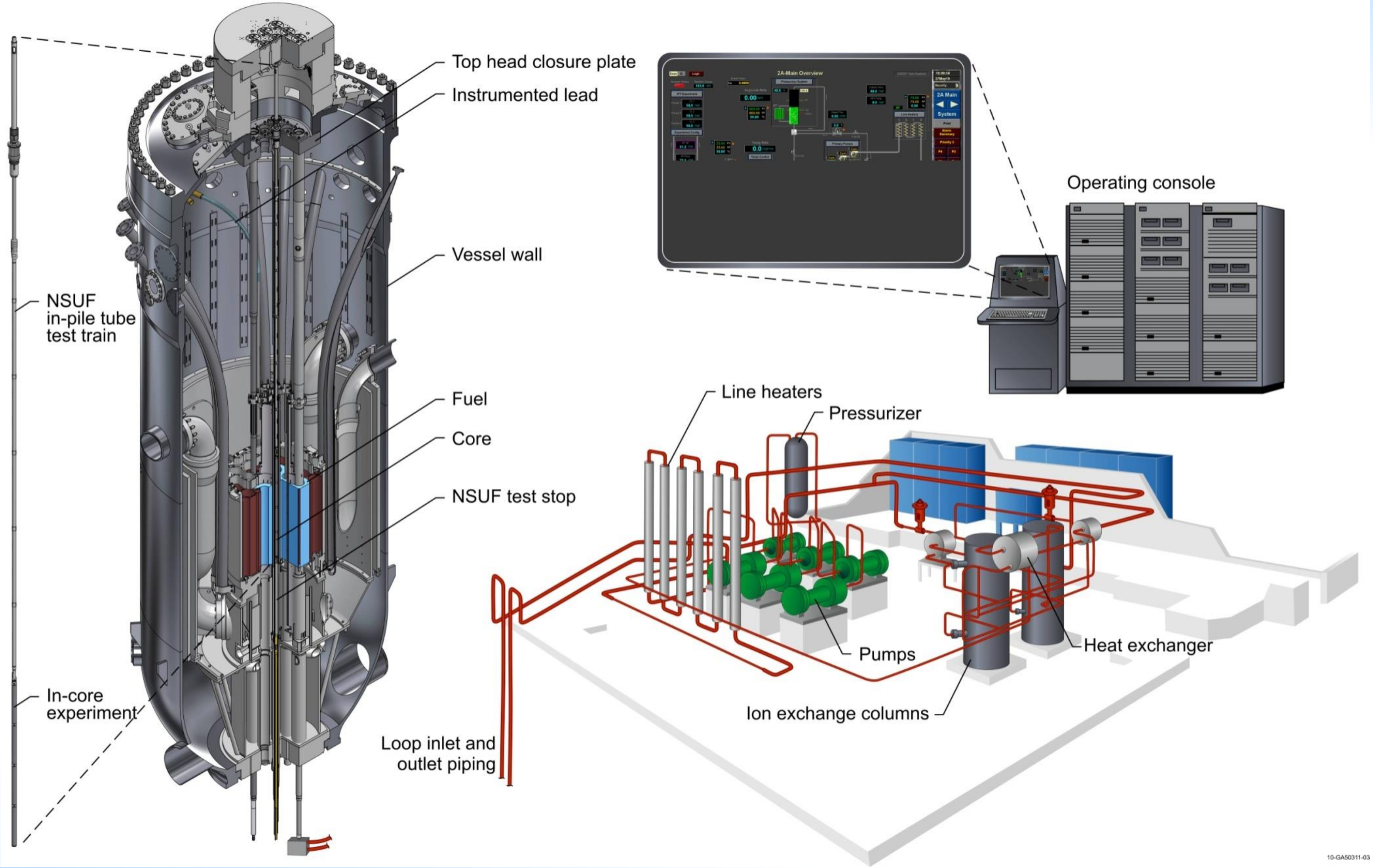
Hydraulic Shuttle Irradiation System

- 14 capsules in a set
- Inserted and removed during reactor operations



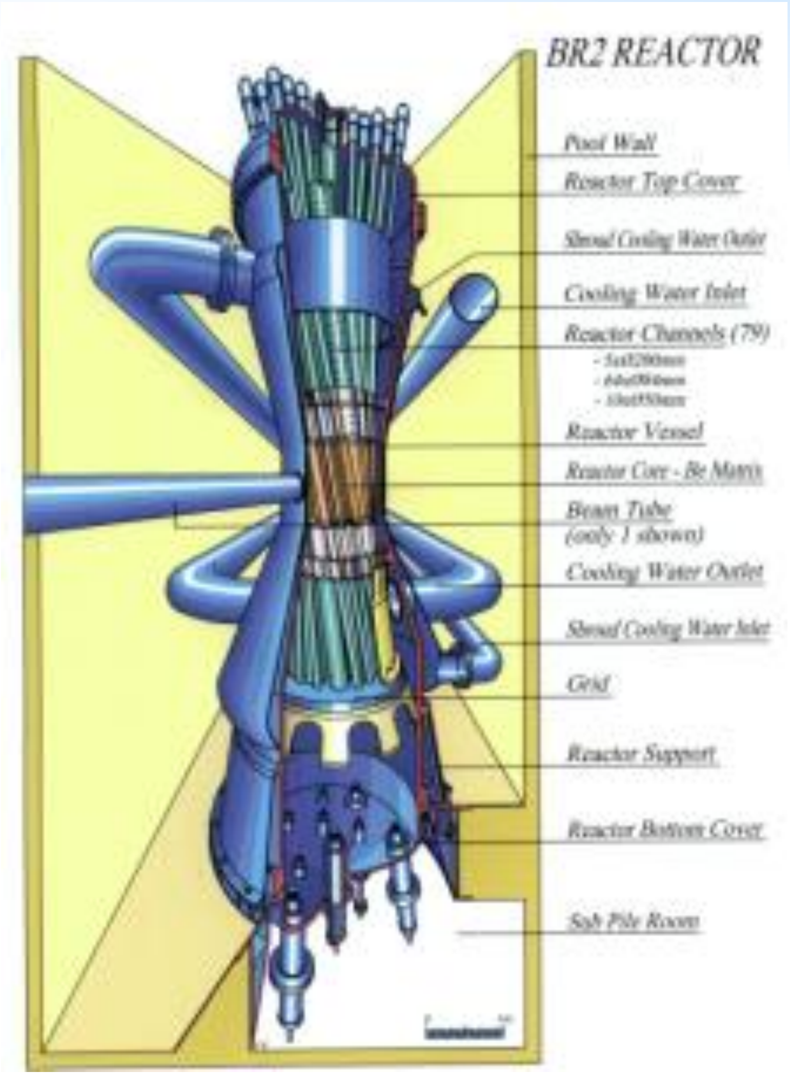


ATR Pressurized Water Loop Layout



Belgium Reactor 2 (BR2) Overview

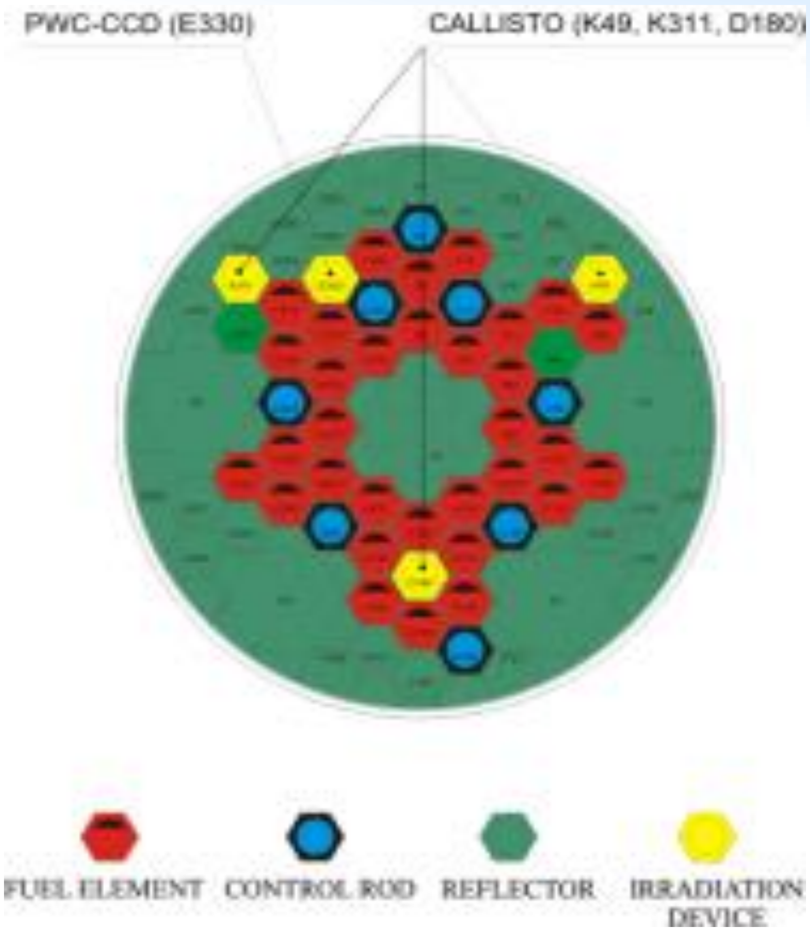
- Pool Reactor with Pressurized Water Reactor Experiment Loops
- Core Irradiation channels
  - Center vertical channel, 200 mm diameter
  - Surrounding inclined channels, 84 mm diameter
- A large number of experimental positions, including four peripheral 200 mm channels for large irradiation devices
- Irradiation conditions (temperature, pressure, environment, neutron spectrum, etc.) representative of various power reactor types
- High neutron fluxes, both thermal and fast (up to 10^{15} n·cm^{-2}·s^{-1}), at 100 MWt



BR2 Cross Section and Experiments

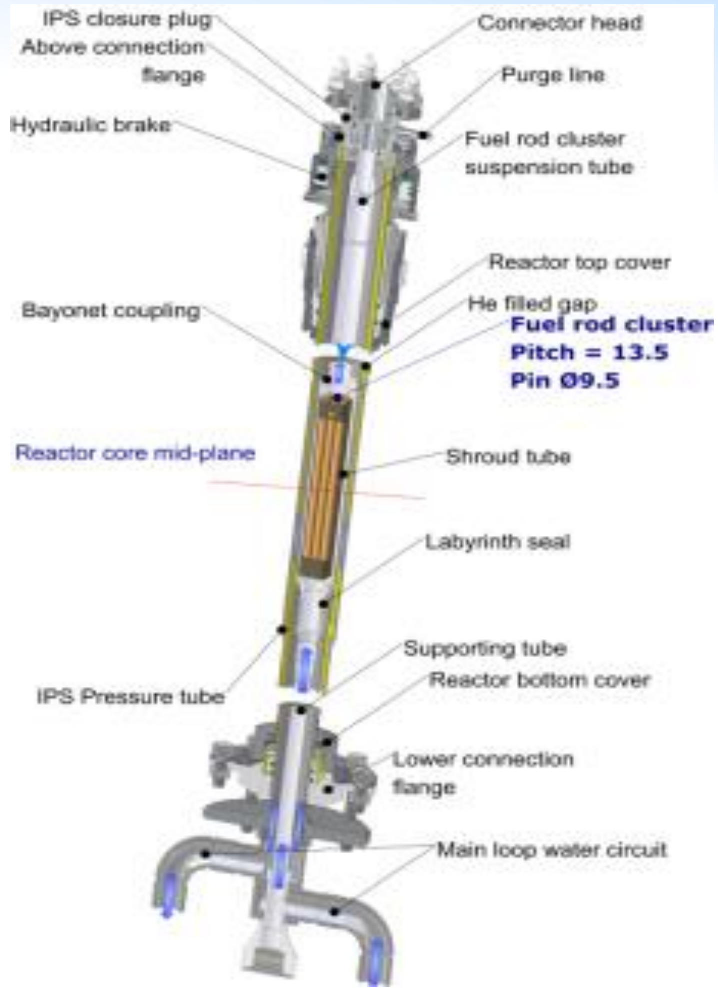
Testing Loops in BR2:

- CALLISTO — *CA*pabi*L*ity for *L*ight water *I*rradiation in *S*teady state and *T*ransient *O*peration
- MISTRAL — *M*ultipurpose *I*rradiation *S*ystem for *T*esting *R*eactor *A*lloys
- ROBIN — *RO*tating *B*asket with *I*nstrumented *N*eedles
- *LIBERTY* — *LI*fting *B*asket in the *E*xperimental *R*ig for *B*R2 *T*himble tube s*Y*stem



BR2 CALLISTO Loop

- Experiments to support predictive model validation and qualification testing under realistic power reactor operating conditions
- Three experimental rigs, called in-pile sections (IPS)
- Connected to a common pressurized cooling loop, to deliver variable pressure and temperature environments
- Investigate behaviour of advanced fuel under representative PWR operating conditions
- Assess the irradiation assisted stress corrosion cracking (IASCC) phenomena in typical light water reactor materials
- Study the corrosion process on candidate materials for future fusion reactors
- Characterise performances of high neutron dose irradiated materials for light water and fusion reactors, ADS systems
- Develop and qualify new on-line in-pile detectors



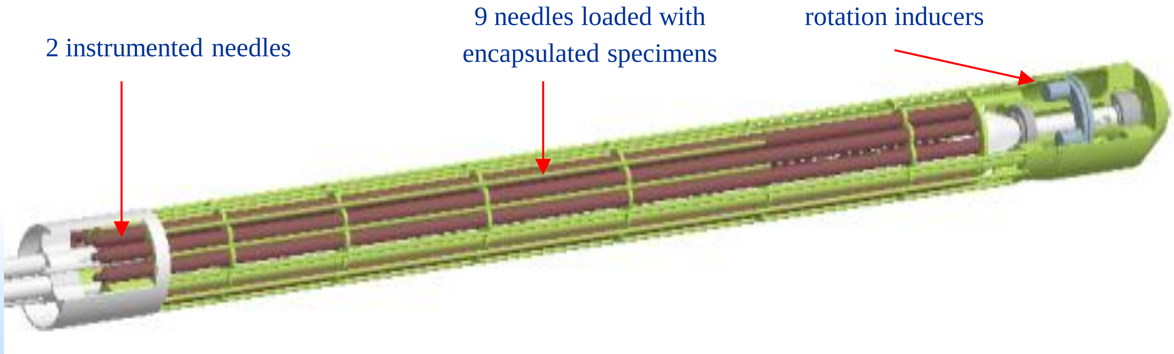
BR2 MISTRAL Test Rig

- Reusable irradiation device for research on reactor materials exposed to a high fast neutron flux at temperatures below 350°C
- Pressurised water capsule containing metallic specimens
- Loaded inside a BR2 driver fuel element
- Neutron flux (> 0.1 MeV) 2–3 × 10^{14} $n \cdot cm^{-2} \cdot s^{-1}$
- Temperature regulation in the range 160–350°C (electrical heaters)
- 0.6 dpa per 21-day cycle at 60 MWth (nominal)
- Full instrumentation
- Number of specimens and their dimension: typically, MISTRAL is designed to irradiate mini-charpy samples (4 mm × 3 mm × 27 mm) and round tensile (5 mm diameter & 27 mm long) specimens

Up to 80 specimens over 500 mm length

BR2 ROBIN Basket

- Contains specimens (typically tensile or mini-charpy) in needles in a large thimble in a standard BR2 channel, open to the reactor pool allowing devices to be loaded during reactor operation
- Contains up to nine needles with 11 mm outside diameter
- an instrumented needle containing thermocouples, a gamma-thermometer, a SPND or a fission chamber, can be loaded into ROBIN to measure parameters on-line and in real time
- To compensate for the fast flux radial gradient through the selected irradiation position, this basket can be rotated during irradiation
- Maximum fast neutron flux (E > 1 MeV) - ~ 3×10^{13} $n \cdot cm^{-2} \cdot s^{-1}$ at the central basket position
- The temperature of the specimens could be adjusted by encapsulating them into a matrix made of material that has a good thermal conductivity and or a suited density, with gas gap design



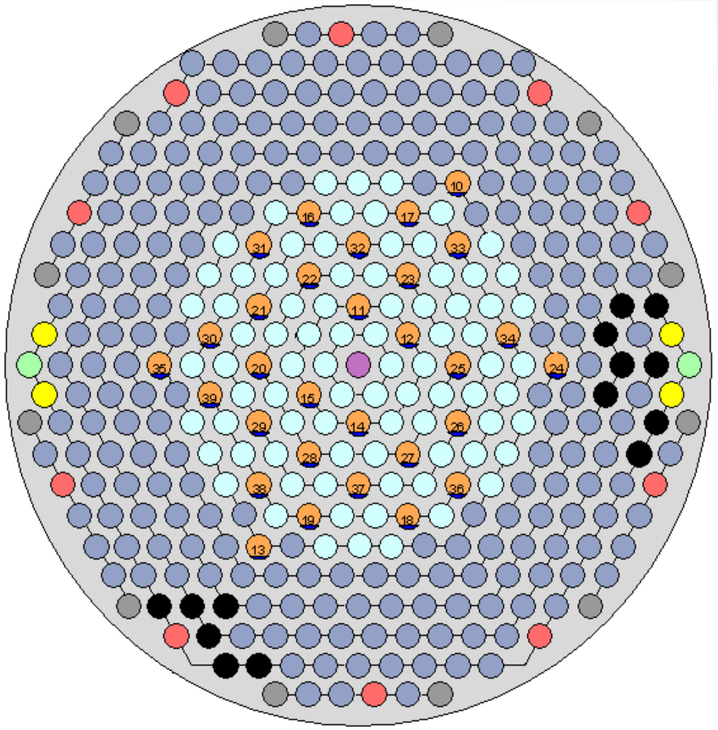
BR2 LIBERTY Basket

Fundamentally the ROBIN basket with some design improvements

- Each needle can be independently lifted up (and down) above the reactor core level when the specified fluence is reached, while the other needles remain in the neutron flux
- Each needle can be separately instrumented
- Larger specimens like the mini CT-specimens (10 mm × 10 mm) can be tested
- LIBERTY can be loaded while BR2 is in full operation
- Some electrical heating wires could be put into the needles to control the temperature of the specimens
- The specimens can be irradiated from 50°C up to 500°C and even higher (depending for instance on the needle filling material). Each of the 5 needles can have different temperatures

Halden Boiling Water Reactor (HBWR)

- Initially intended to be prototype for a boiling water reactor power plant, also intended to provide steam for a near-by paper factory
- Now focused on fuels and irradiation experiments.
- 25 MWt design, but usually operates at 18-20 MWt
- Heavy water moderated and cooled reactor with natural conditions similar to commercial water moderated and cooled reactors
- Over 300 testing positions, and can have up to 30 fuelled experiments simultaneously
- About 110 positions in the central core (light blue in core cross section)
- Height of active core 80 cm

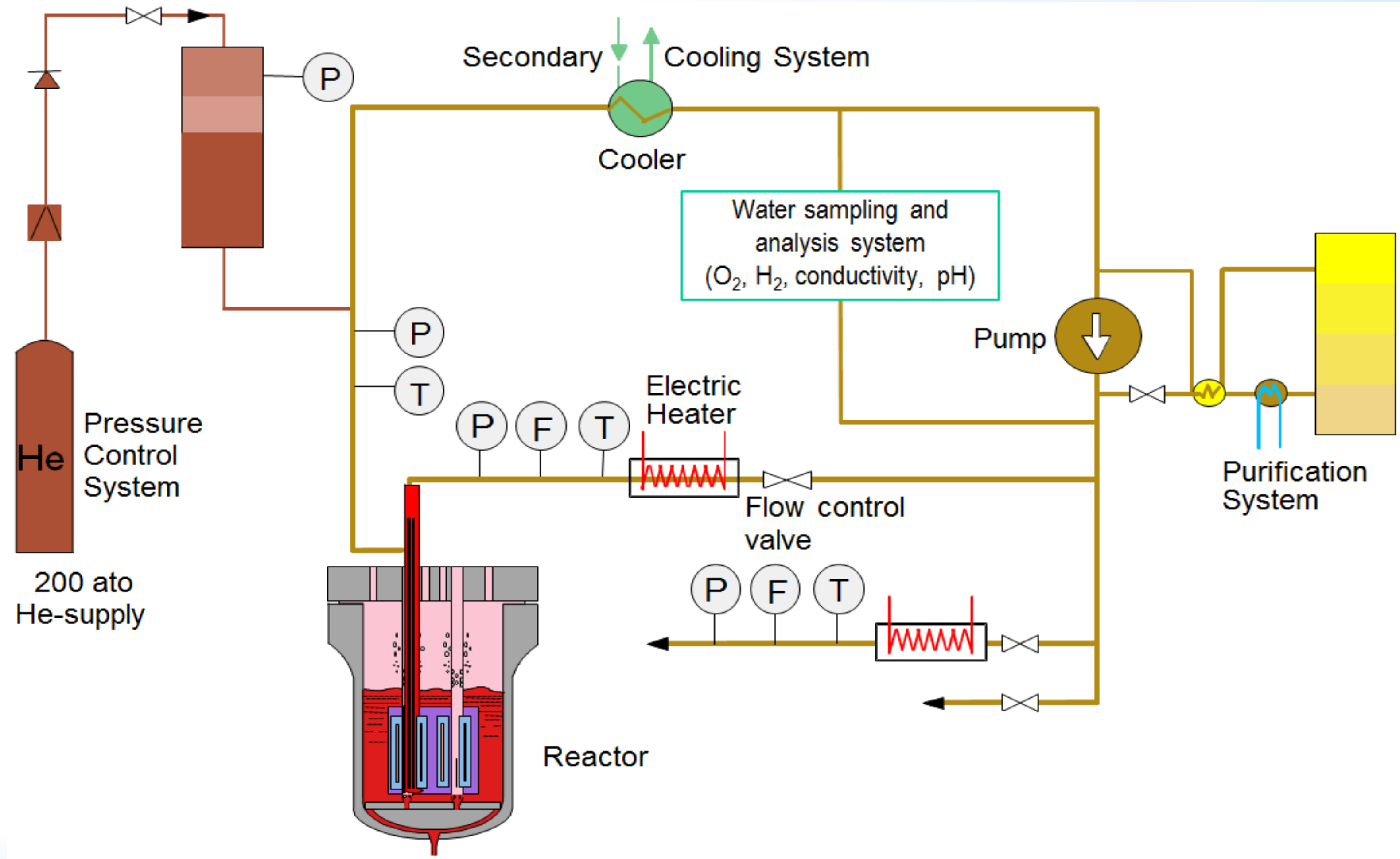


HBWR Cross Section

HBWR Test Rigs

- Loop systems for simulation of BWR/PWR/WWER/CANDU conditions;
- Pressurisation system for imposing up to 500 bar pressure on fuel rods under operating conditions
- Gas flow system
- Gas analysis system
- Hydraulic drive system
- Fuel testing instrumentation
    - Thermocouple
    - Rod pressure transducer
    - Cladding extensometer
    - Fuel stack elongation detector
    - Moveable diameter gauge
    - Neutron detectors for flux mapping in the rig to calibrate experiment power
- Material testing instrumentation:
    - DC potential drop measurement
    - Electrochemical potential sensor
    - Water conductivity cell
    - Electrochemical impedance measurement

HBWR Loops



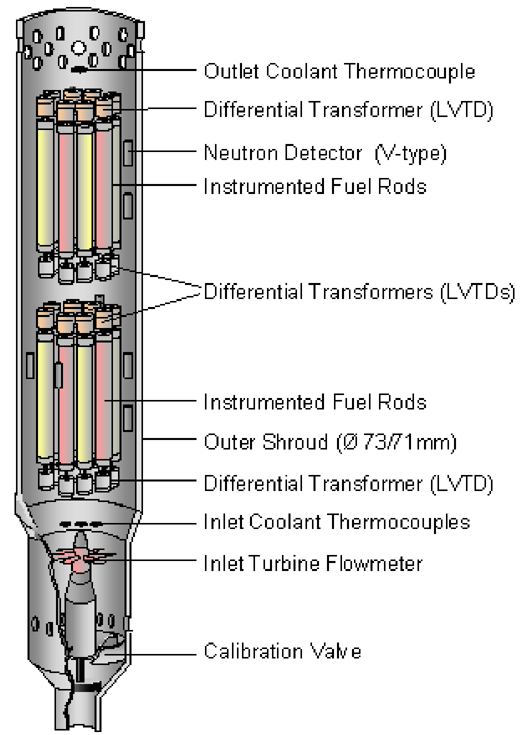
Schematic of a HBWR steady state loop system - can have up to 10 loop systems

HBWR Test Loops

- Rigs for fuel and material testing under simulated water reactor conditions are inserted into in-core pressure flasks connected to light or heavy water circulation systems.
- These systems, completely separated from the reactor cooling systems, are designed for operation at pressures and temperatures of 165 bar and 340^{o}C.
- Most simulate thermal-hydraulic and chemistry conditions of LWRs
- One loop is operated with heavy water providing prototypical CANDU reactor conditions
- The maximum heat removal capacity from a pressure flask is approximately 200 kW
- Operation with normal or hydrogen water chemistry is possible
- Boron and lithium concentrations can be varied over a wide range
- Can operate with controlled additions of water impurities such as chromium, zinc, sulphuric acid etc.
- Loop systems are used both for fuel testing as well as for core material studies, e.g. IASCC
- Have facility to disassemble NPP irradiated fuel, instrument pellets and re-assemble into test for HBWR.

HBWR Test Capabilities

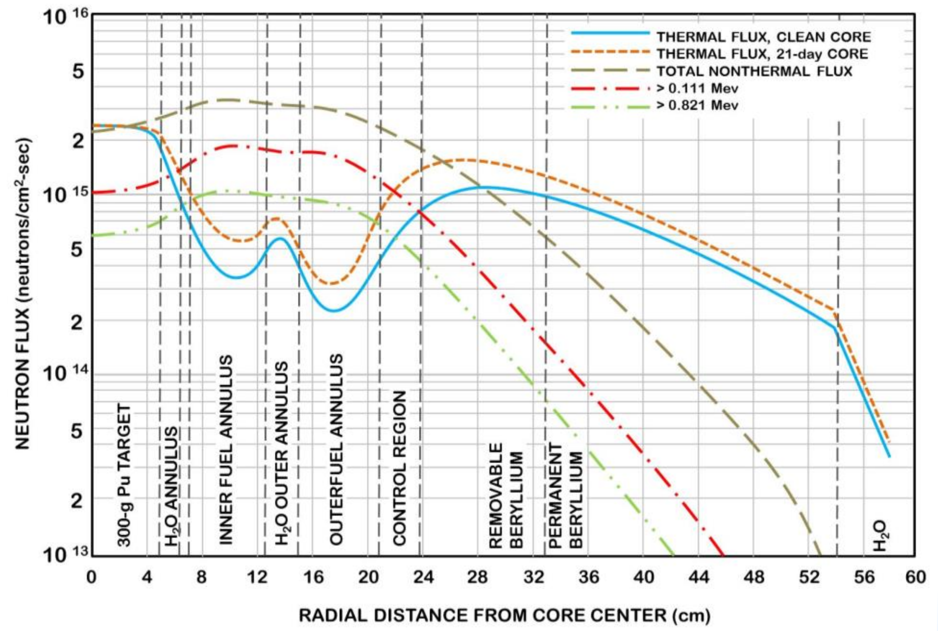
- Accidental Conditions
  - Dry-out or LOCA conditions
  - Decay heat simulated
  - Fuel rod eat-up, ballooning, and burst occur
  - Small amounts of water/steam added post-burst to enable cladding oxidation
  - System can be re-filled at hot conditions to simulate quenching
- Corrosion Investigations
  - Water loop systems with very precise chemistry control
  - Irradiation-assisted stress corrosion cracking (IASCC) – use of small bellows to apply stress during irradiation. Crack growth measured in-pile with potential drop technique
- Fuel Behavior
  - Swelling and densification
  - Gas release
  - Pellet-clad interaction

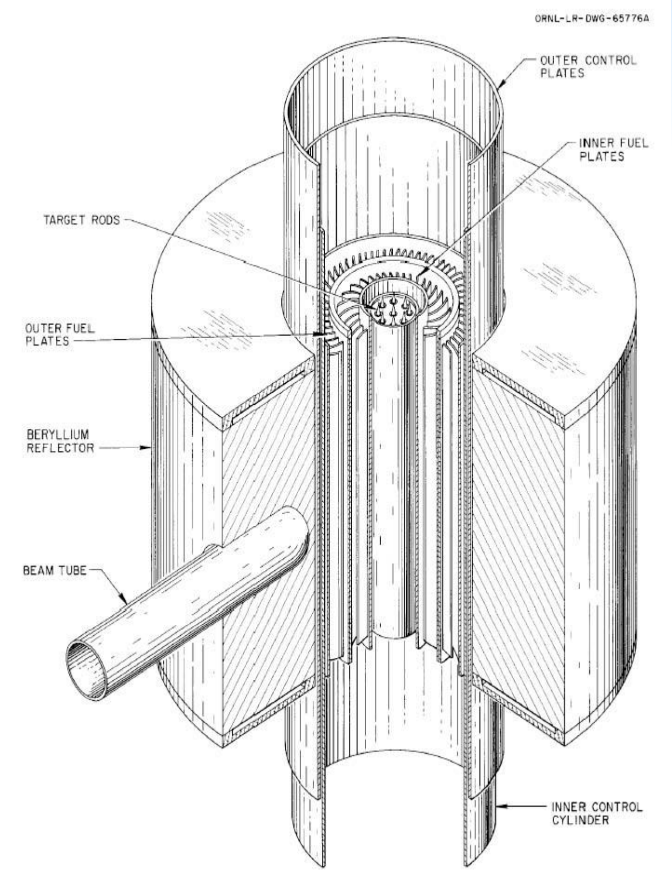


HBWR irradiation test rig for instrumented fuel rods

High Flux Isotope Reactor, USA

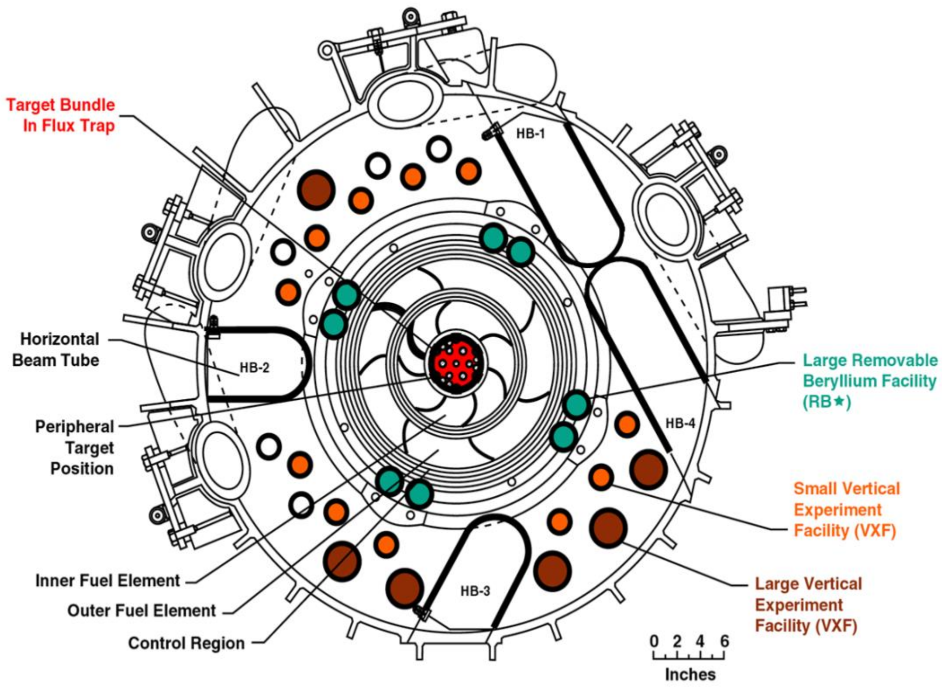
- Extremely high fluxes
  - 2.3 x 10^{15} n/cm^2-sec thermal
  - 1.2 x 10^{15} n/cm^2-sec fast
- 85 MWt
- Tank in Pool, light water-reflected and moderated, with Be reflector
- Operates 140 days/year
- Steady state operations
- https://neutrons.ornl.gov/hfir





HFIR Irradiation Missions

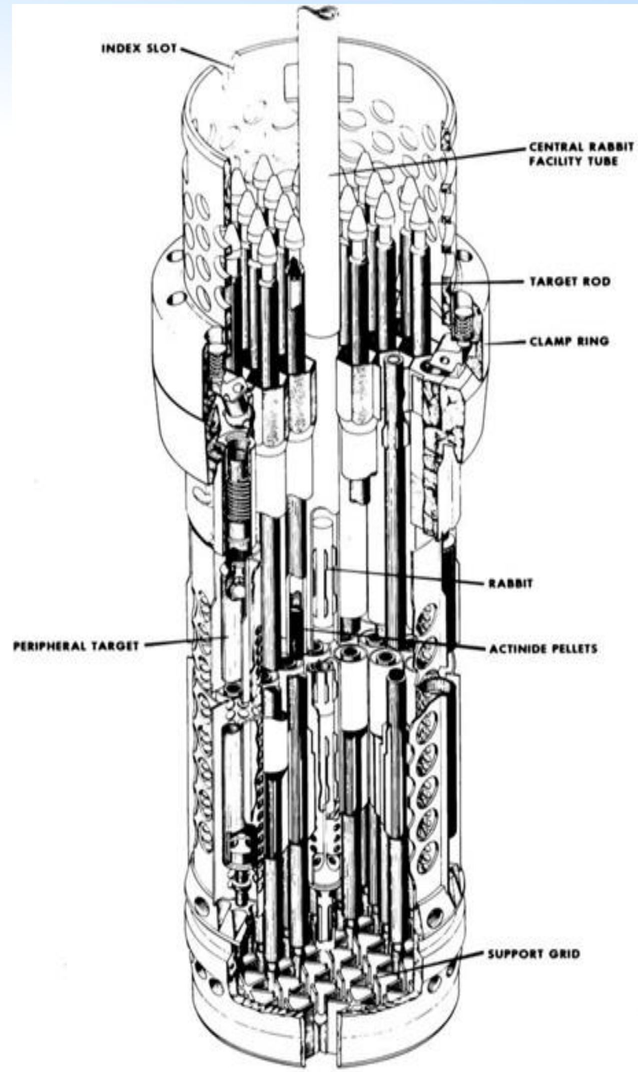
- In-core irradiations for medical, industrial, and isotope production
- Research on severe neutron damage to materials
- Neutron activation analysis (NAA) to examine trace elements and identify the composition of materials
- Gamma irradiation capability that uses spent fuel assemblies
- Four beam lines with 12 world-class instruments for condensed matter research. To use the neutron scattering capabilities of HFIR, contact the Neutron Scattering Sciences User Group at neutrons.ornl.gov



Cross section of HFIR, showing experiment positions

Experiment Options in HFIR

- High flux center flux trap
  - Up to 30 target positions (2 can accommodate instrumented experiments)
  - 6 peripheral positions at the edge of the flux trap
  - 1 hydraulic shuttle irradiation position in flux trap
  - Isotopes, fuel and material irradiations
- 21 Vertical experiment facilities
  - Instrumented lead
  - Pneumatic hydraulic tube
  - Non-instrumented capsule experiments
- Materials Irradiation Facility
  - Instrumented in center flux trap
- 2 Slant access facilities
  - Pneumatic tube for NAA
  - Additional highly thermalized spectrum environment.



Target basket in the HFIR flux trap

Extensive PIE Capabilities for HFIR

Irradiated Material Examination and Testing Facility

- Sample sorting and identification
- Sample machining using a CNC milling machine and diamond saws
- Furnace annealing
- Automated welding
- Ultrasonic cleaning
- High-temperature, high-vacuum testing
- Tensile testing with high-vacuum chamber option
- Impact testing, fatigue and fracture toughness testing of standard and subsize impact specimens
- Automated micro-hardness testing;
- Profilometry
- Scanning Electron Microscopy

Irradiated Fuels Examination Laboratory

- Full-length LWR fuel examination
- Repackaging of spent fuel
- Metrology, metallography, grinding/polishing, optical and electron microscopy
- Fission gas sampling and analysis
- Thermal imaging
- SEM/microprobe
- Microsphere gamma analyser for individual fuel particle analysis

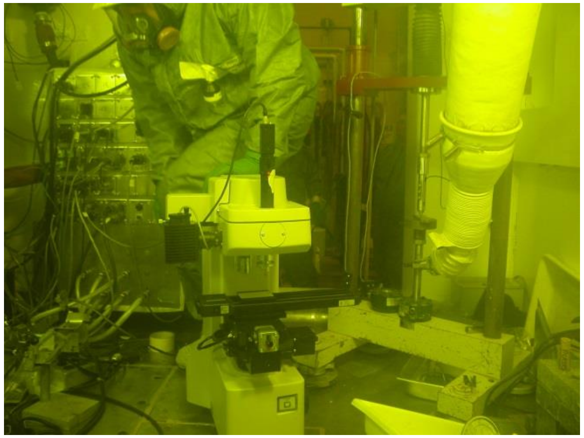
Irradiated Fuels Examination Laboratory

LAMDA Facility

Low activation materials development and analysis laboratory (LAMDA)

- For low activity samples – low activation, or small samples of high activation
- From HFIR or elsewhere
- Mechanical properties testing
- Thermal diffusivity
- Dilatometer
- Elastic modulus
- Calorimetry
- electrical resistivity
- Density measurement

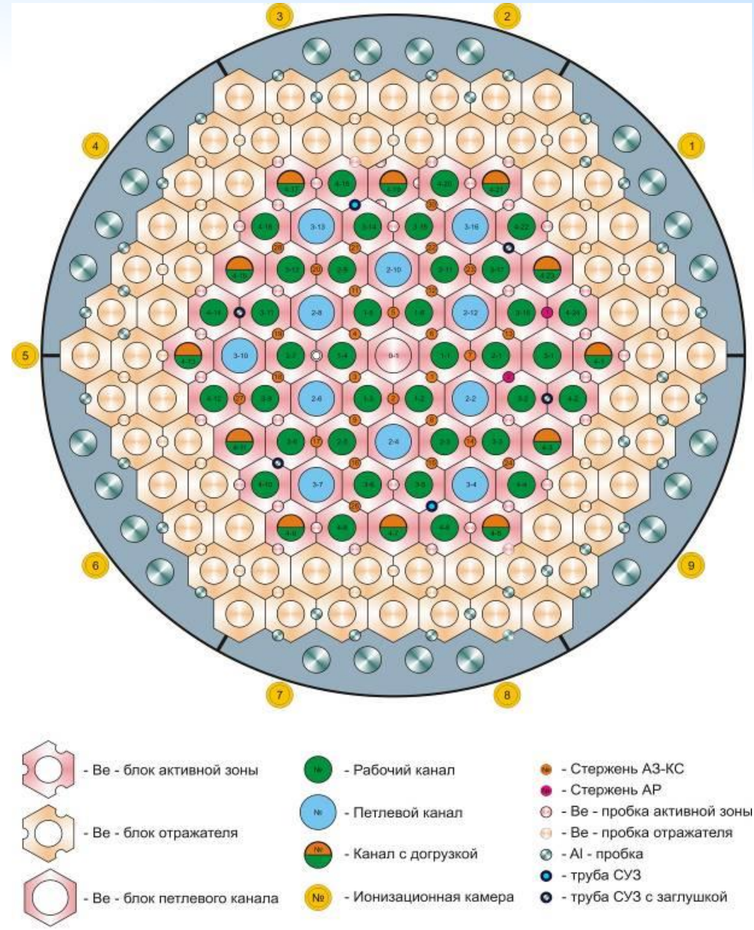
LAMBDA Facility



In-cell hardness testing, IMET

MIR.M1, Russian Federation

- Core arranged with loop channels surrounded by fuel assemblies and control rods
- Used to test structural materials and fuel assemblies from various power plant designs
- Radioisotope production



MIR.M1 core cross section

MIR.M1 Key Specifications

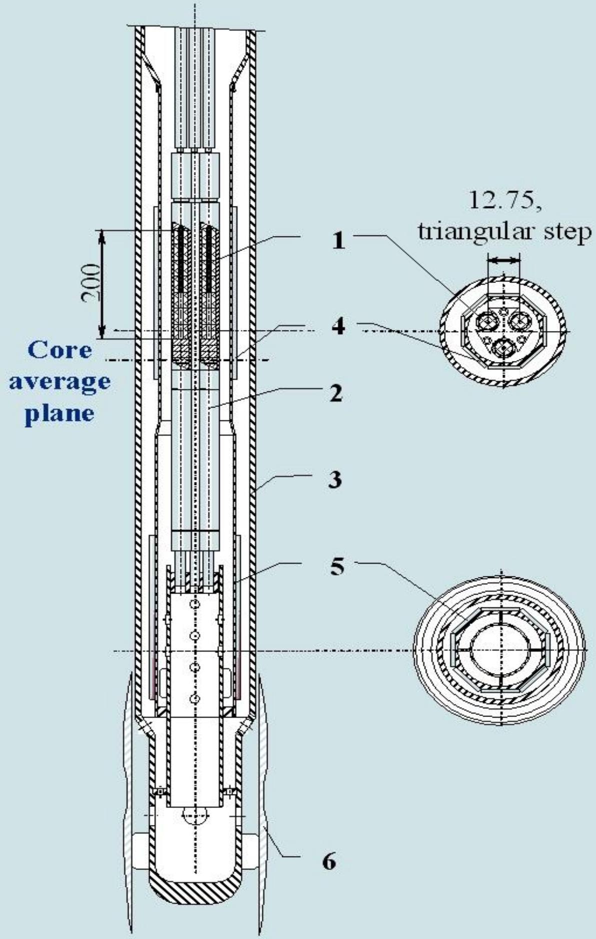
IAEA 60 Years
Atoms for Peace and Development

Maximal thermal power	100 MW
Loop channel diameter	≤148
Number of loop channels, max.	11
Thermal neutron flux density in the experimental channel	$\leq 5 \cdot 10^{14} \cdot n \cdot cm^{-2} \cdot s^{-1}$
Volumetric heat rate in the core	0.85 MW/l
Coolant:	Water
– pressure at the reactor core inlet	1.25 MPa
– temperature at the reactor inlet	≤70°C
– temperature at the reactor outlet	≤98°C
Fuel cycle duration	up to 40 days
Operating time at power,	~ 240 d/year
Planned operation time	More than 2030

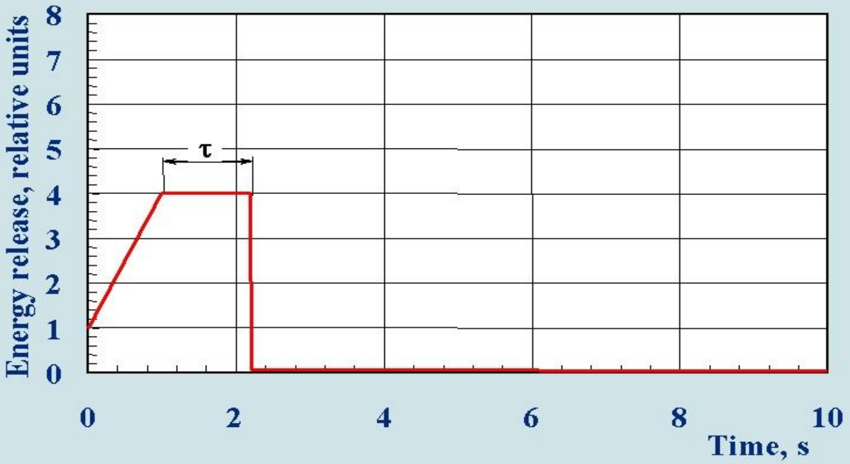
Fuel Testing Techniques in MIR.M1

- Dismountable and instrumented device for testing fuel rods ~1000 mm, containing up to 19 fuel rods
- Dismountable devices for testing short-size (~ 250 mm) fuel rods, up to four such rigs can be installed one over another in one loop channel
- Device for combined irradiation of refabricated (~1000 mm) and full-size fuel rods (≤ 3800 mm) of spent NPP fuel
- Dismountable devices for power cycling and RAMP experiments of instrumented fuel rods by displacement or rotation of the absorbing screens in the experimental channel
- Instrumented device for testing under LOCA and RIA conditions

Fuel Testing at MIR.M1



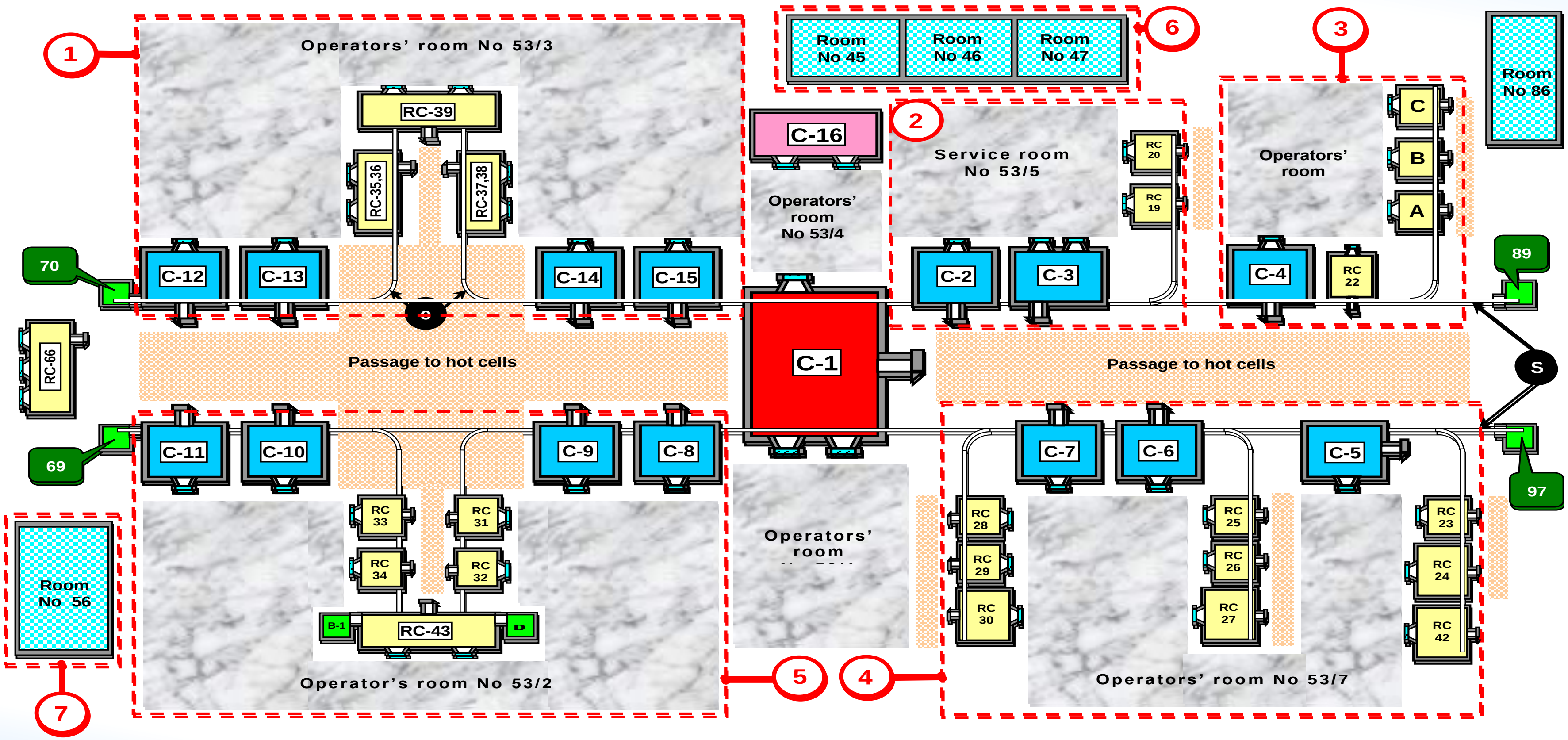
Impulse shape in the MIR reactor
(τ - exposure time at maximal LP)



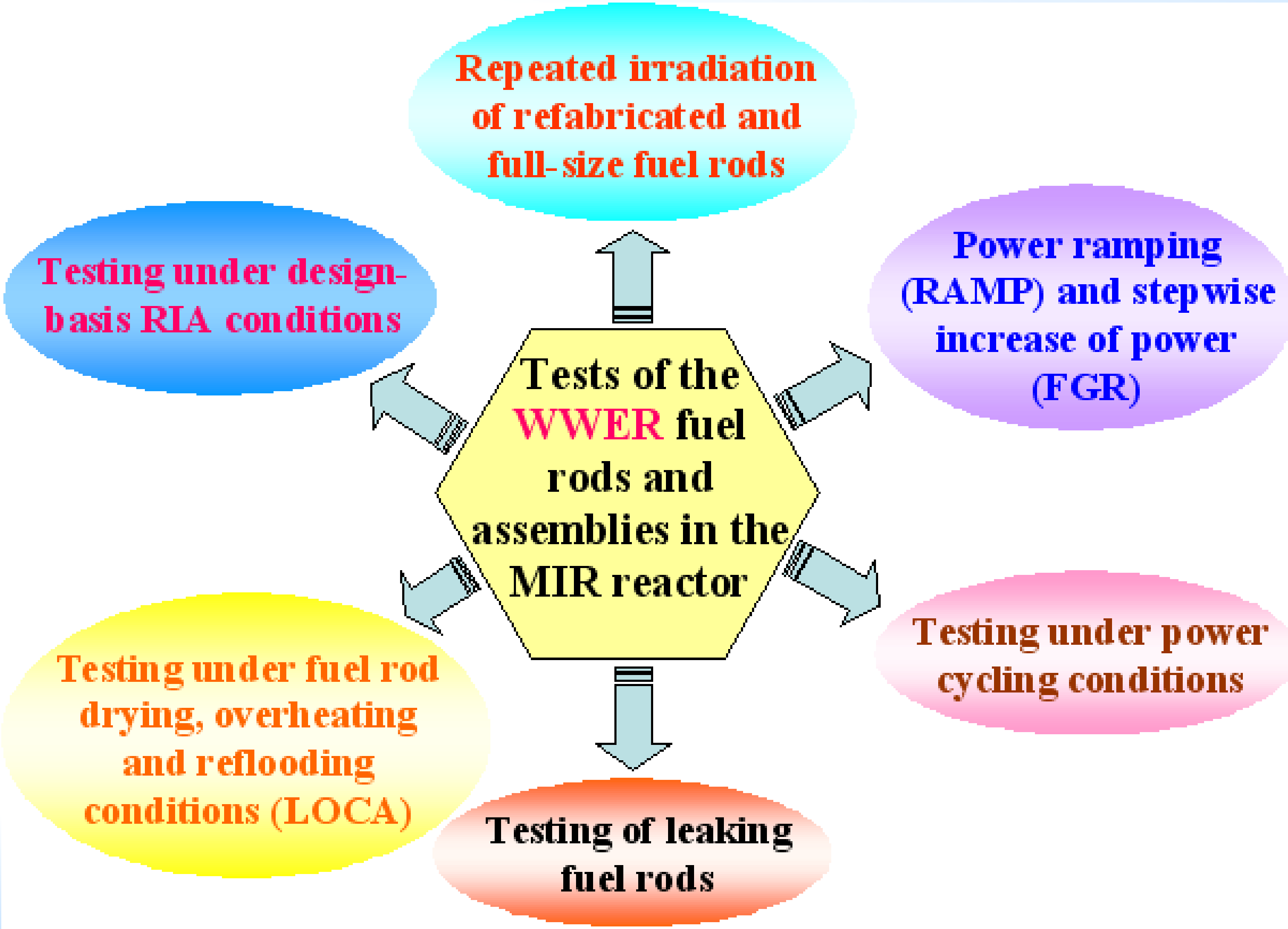
Schematic diagram of the irradiation rig designed for RIA test in the MIR reactor
1 – fuel rods, 2 - conductor pipes, 3 – shroud, 4 –upper shield, 5 –lower shield, 6 – loop channel vessel

Technique for simulation of reactivity inserted accident (RIA)

Hot Cells at MIR.M1

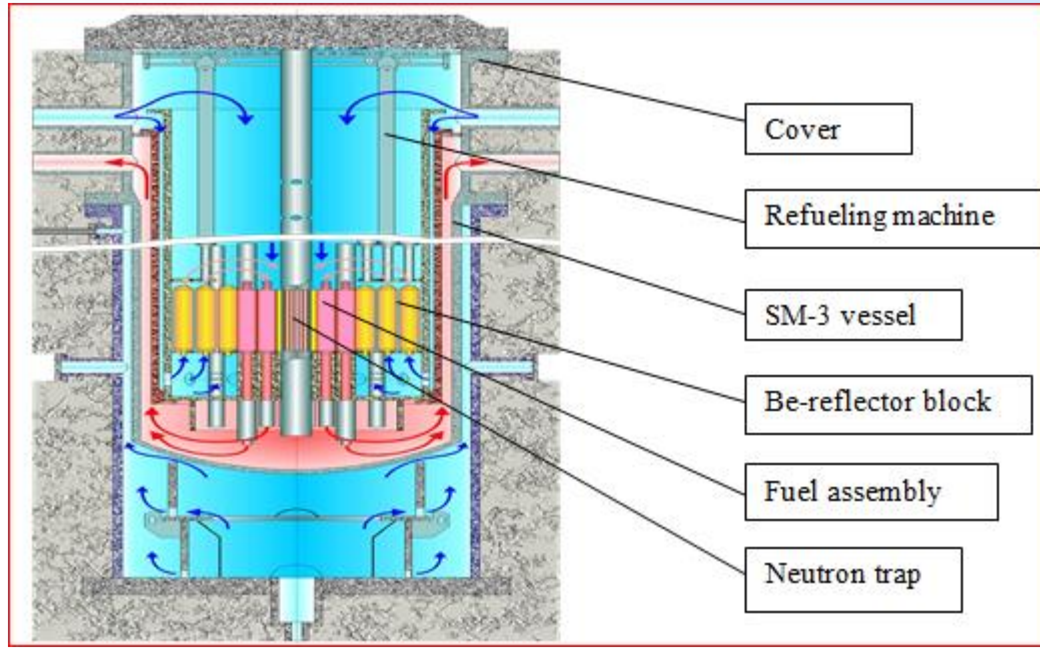


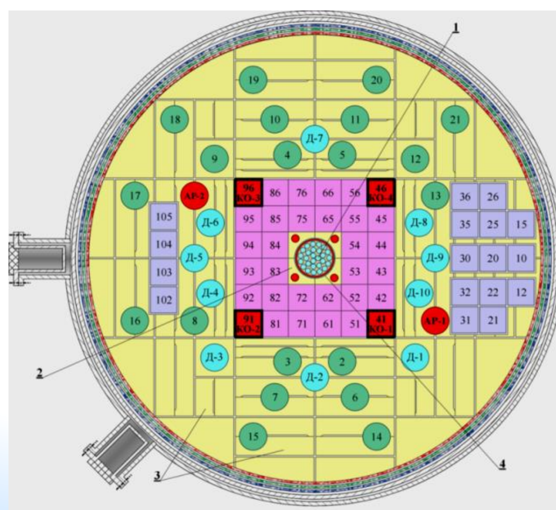
VVER Fuel Testing Program in MIR.M1



SM-3, Russian Federation

- Very high flux
  - 5E15 n/cm^2-s
- 100 MWt
- Pressure tank
- Light water cooled and moderated
- Be reflector
- Primary use is fuel and material testing
- Capsule and loop testing capabilities





1 – neutron trap
2 – beryllium liners
3 – beryllium reflector blocks
4 – central compensating element

SM-3 Irradiation Cells

Number of cells for irradiation	Up to 81	
Trap	Block option: up to 27 cells Ø 12–25 mm; Channel option: channel Ø 50 mm + 18 cells	
Core	Up to 6 and up to 4 FAs with 1 cell for targets Ø 24.5 mm	
Reflector	30 channels (of which 20 cells can be instrumented or supplied with separately coolant), Ø 64 mm	
Irradiation positions:	Neutron flux, $n \cdot cm^{-2} \cdot s^{-1}$	
	total	≥ 0.1 MeV
Trap	$\leq 5.4 \times 10^{15}$	$\leq 1.5 \times 10^{15}$
Core	$\leq 4.3 \times 10^{15}$	$\leq 2.3 \times 10^{15}$
Reflector	$\leq 1.6 \times 10^{15}$	$\leq 5.3 \times 10^{14}$

SM-3 Irradiation Loops

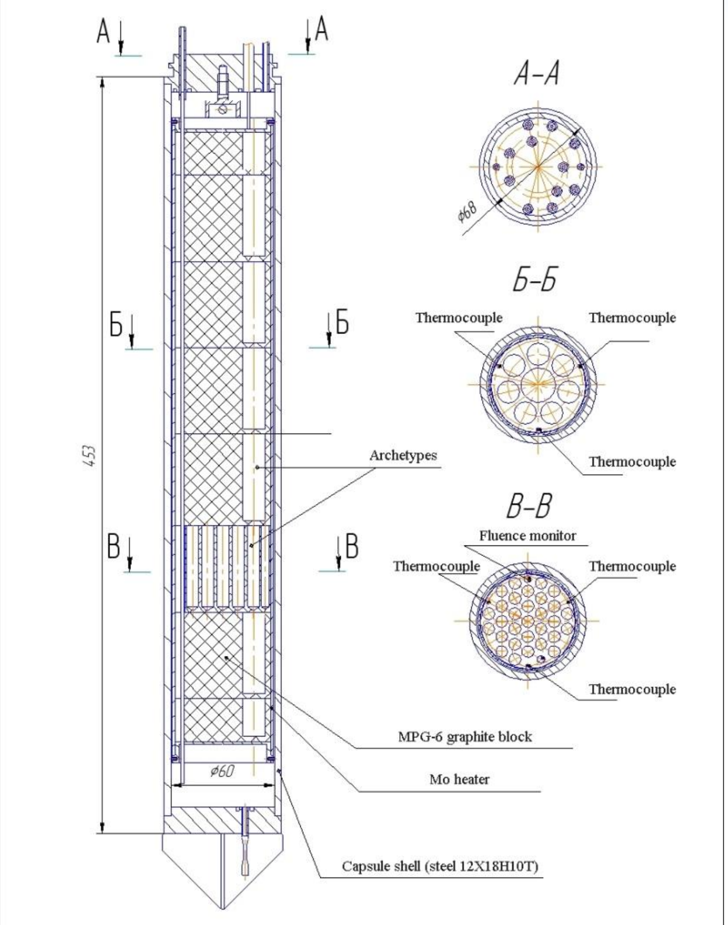
60 Years
IAEA
Atoms for Peace and Development

Design of irradiation rig	Medium	Testing parameters	
		ϕ, n·cm^{-2}·s^{-1}, (E >0.1MeV)	Kt, dpa/ year
Loop channel in the reflector	Water (≤350°C, ≤18.5 MPa)	1.0·× 10^{13}–4 × 10^{14}	0.1–6.0
Loop channel in the core	Water (≤350°C, ≤18.5 MPa)	1.2–1.5 × 10^{15}	15–18
Ampoule rig in the reflector	Boiling water (≤350°C, ≤17 MPa); heavy liquid metal (≤650°C, ≤1 MPa); supercritical water (≤650°C, ≤23 MPa); gas (He, Ne, N_2) (≤2500°C, ≤23 MPa)	5 × 10^{12}–5.3 × 10^{14}	0.1–6.0
Ampoule rig in the core	Boiling water (≤350°C, ≤ 17MPa); heavy liquid metal (≤650°C, ≤1 MPa); supercritical water (≤ 50°C, ≤23 MPa); gas (He, Ne, N_2) (≤2500°C, ≤23 MPa)	1.5 × 10^{15}–2.3 × 10^{15}	16–25

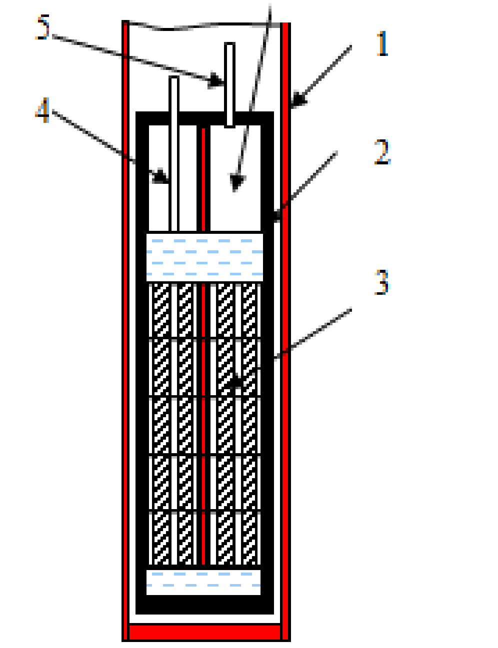
SM-3 Testing Facilities

- Long-term strength and creep tests of steels and alloys under longitudinal tension (facility ‘Neutron-8’) and internal gas pressure test at 550–800°C
- In-pile tests of different types of fuel materials at 550–2500°C
- In-pile investigation of relaxation resistance of structural materials
- In-pile investigation of creep of nuclear fuel at temperatures 700–1100°C, including pre-irradiated fuel samples to investigate the burn up effect on the creep characteristics
- In-pile tests of the core material for existing and advanced nuclear facilities at high damage rate of 1–25 dpa/year in the temperature range 100–2500°C and different environments

SM-3 Test Rigs



Device for matrix graphite and FC dummies irradiation



Capsule for irradiating vessel steel samples in boiling water

IAEA

60 Years

Atoms for Peace and Development

Thank you!

F.Marshall@iaea.org