



Advancing Reactor Technologies for Non-Electric Applications through Experimental Research

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District heating for Pevek



Haiyang NPP

District heating for Haiyang and Rushan



Temelín NPP

District heating for České Budějovice

NPPs ALREADY SUPPLY HEAT

Experimental Capabilities of the Rosatom Science Division

Fast research reactor

- Fast-spectrum irradiation
- Fuel and materials testing
- Cogeneration operating experience

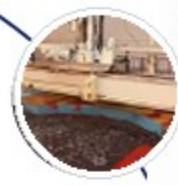


Materials research complex (hot cells)

- Mass spectrometry
- Fuel-assembly examination and repair
- X-ray diffraction analysis

Critical facilities

- Reactor neutronic characteristics
- Fuel-cycle and neutron-spectrum modelling
- Validation of codes and nuclear data



Multi-loop research reactor

- Testing of VVER, PWR and other fuel types
- NPP transients and accident conditions (LOCA, RIA, RAMP)
- Material properties and corrosion resistance



Materials testing research reactor

- Radiation chemistry, reactor technology and NAA
- Radiation effects on equipment
- Structural and biological-shielding materials



High-flux research reactor

- HTGR / TRISO fuel qualification
- MSR salts and materials
- Post-irradiation examination



Pool-type research reactors

- Neutron transmutation doping of silicon
- Radiation colouring of minerals
- ^{99}Mo production
- Reactor materials research



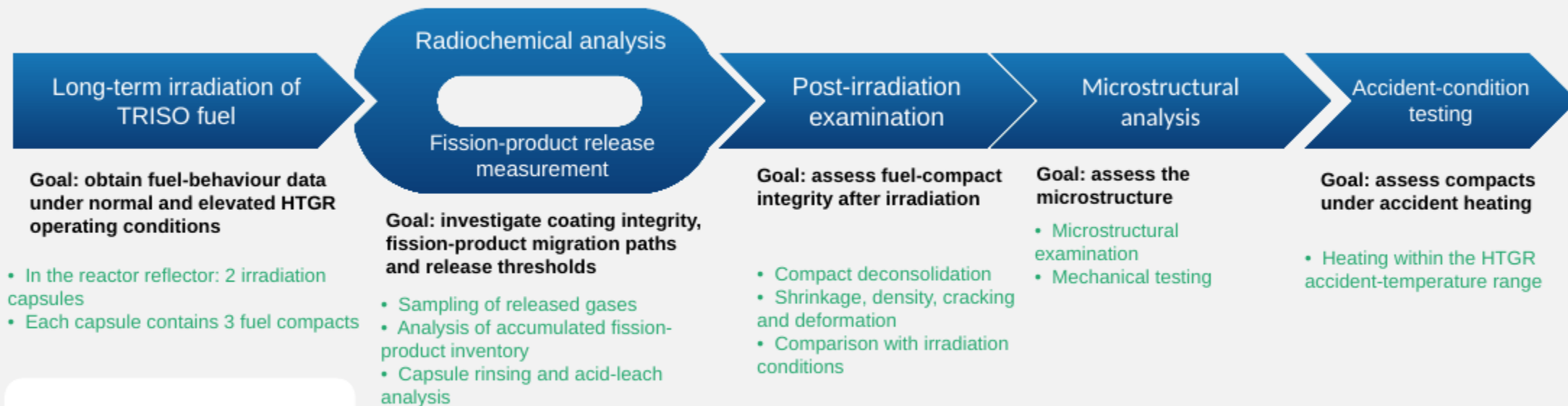
Multipurpose research reactor

- ^{99}Mo and ^{131}I production
- Neutron doping of silicon; NAA
- Radiation colouring of minerals



TRISO Fuel Qualification for HTGRs

- **Objective: reproduce representative HTGR irradiation conditions and generate fuel-performance evidence for hydrogen-production systems**



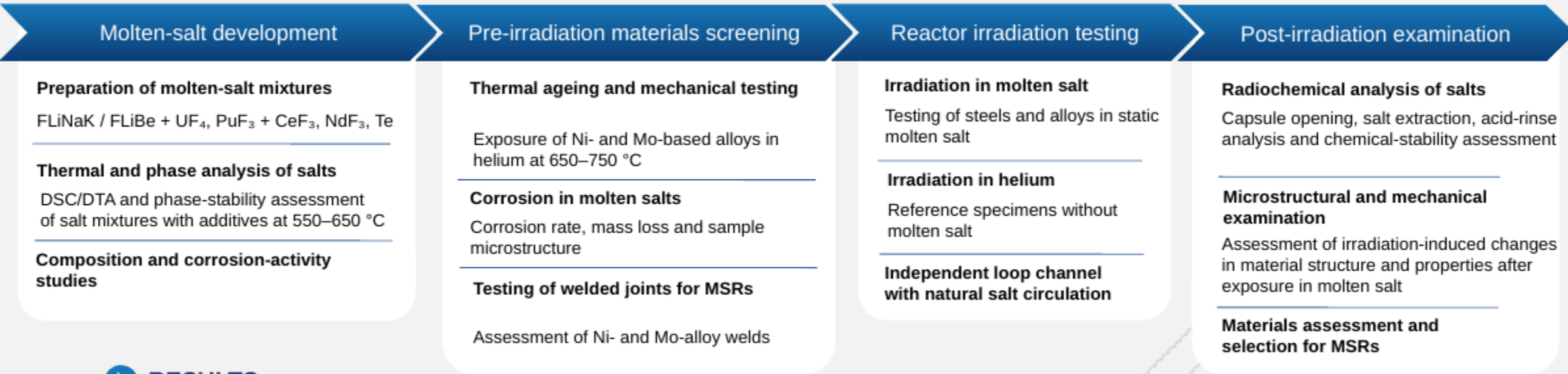
-  $T_{\text{fuel}} \leq 1300\text{ }^{\circ}\text{C}$
-  Duration: 200–400 EFPD
-  Fluence $F \leq 10^{22}\text{ cm}^{-2}$

➤ RESULTS:

- ✓ An experimental database was established for HTGR fuel qualification
- ✓ Data were obtained on fission-product retention by TRISO coatings during high-temperature transients
- ✓ Fission-product release was measured; radiochemical and microstructural examinations of the fuel and graphite matrix were completed

Materials Evidence for Molten Salt Systems

- **Objective: qualify salts, alloys, welds, irradiation technologies and PIE methods for a Russian experimental MSR**



➤ RESULTS

- ✓ Experimental data were obtained on material behaviour in irradiated FLiNaK and FLiBe melts
- ✓ The effects of salt composition and fuel additives on materials and melt chemistry were assessed
- ✓ A database was established for selecting materials and operating parameters for molten salt reactors

Experimental Research and Cogeneration

A fast research reactor combines high-flux fast-spectrum experiments with useful-energy production through a third steam-water circuit.

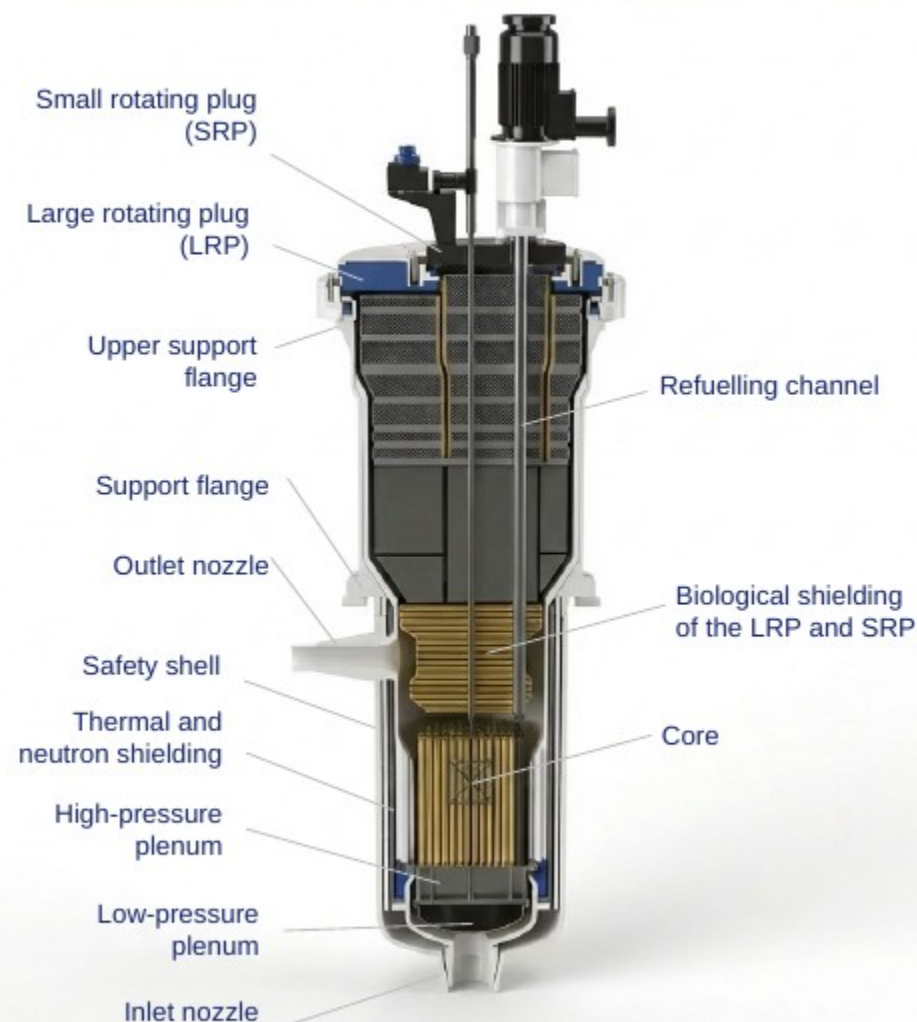


60 MWth; up to 12 MWe
The third circuit has supplied site heat since 1991

Reactor type	Sodium-cooled fast reactor
Thermal power	Up to 60 MWth
Electric output	Up to 12 MWe
Third circuit	Water / steam
Steam conditions	10 MPa; up to 480 °C
Useful-heat path	Turbine + site heating unit
Alternative heat sink	Air heat exchanger
Research role	Fuel and materials irradiation
Operating record	Since 1969; over 55 years



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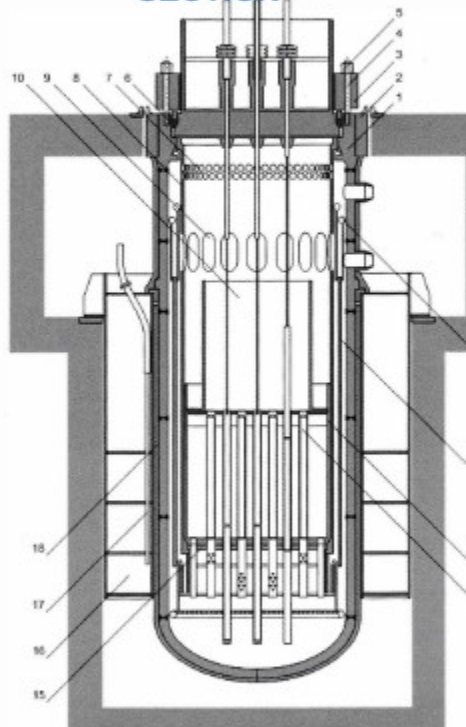


Schematic longitudinal section
of the fast research reactor

Thermal Power Increased to 256 MW

1965–1973: DIRECT CYCLE AND THERMAL-HYDRAULIC OPTIMIZATION

REACTOR VESSEL LONGITUDINAL SECTION



Integral vessel: core and natural-circulation loop
Vessel Ø 3.55 m; height 11.23 m
Core Ø 1.8 m; height 2.0 m

FROM MIXED TO DIRECT CYCLE

1965: ORIGINAL MIXED CYCLE

Reactor steam and three steam generators supplied the turbine.

STEAM GENERATORS DISCONNECTED IN THE FIRST YEAR

1966: DIRECT CYCLE

Reactor steam flowed directly to the turbine — a simple, reliable arrangement.

THERMAL-HYDRAULIC OPTIMIZATION

Higher coolant velocity + flatter axial power distribution

0.5 → 1.0 m/s: coolant velocity
80 → 170 MWth at 3.0 MPa
power at the resonance-instability boundary

1973: EXPERIMENTAL MODE

140 → 256

MWth

design → demonstrated power

+83%

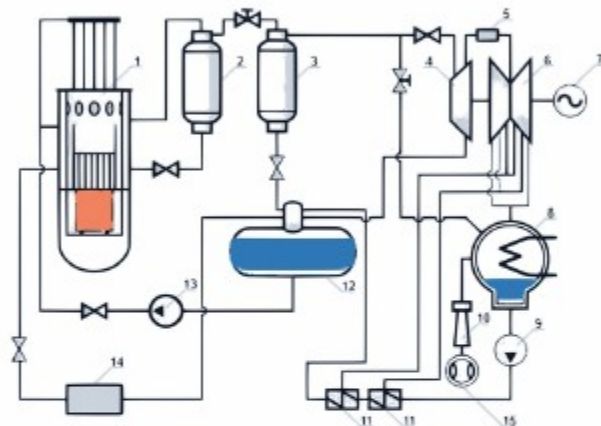
versus the original design level

28 → 52 kW/L

specific volumetric core power density

Useful Energy Utilization Increased by 80%

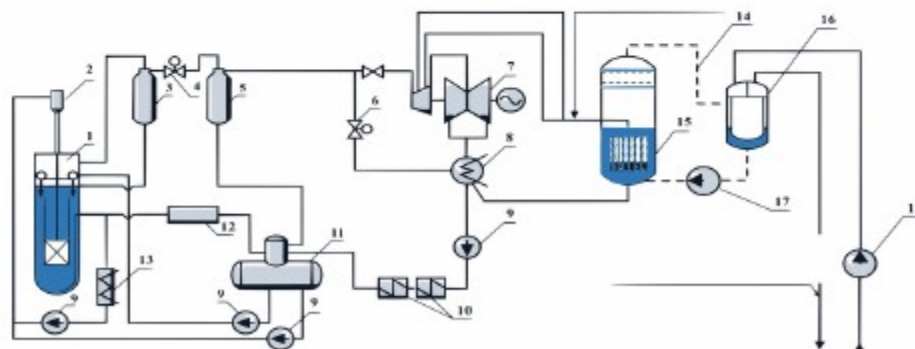
DIRECT STEAM CYCLE: SINCE 1966



COMPONENT KEY

- | | |
|----------------------------------|------------------------------|
| 1 Reactor | 9 Condensate pump |
| 2 High-pressure separator | 10 Ejector |
| 3 Low-pressure separator | 11 Low-pressure heater |
| 4 High-pressure turbine cylinder | 12 Deaerator |
| 5 Intermediate separator | 13 Feedwater pump |
| 6 Low-pressure turbine cylinder | 14 Evaporator unit |
| 7 Generator | 15 Activity-suppression unit |
| 8 Condenser | |

COGENERATION MODE: DISTRICT HEAT SINCE 1978



TOTAL USEFUL ENERGY

23% → 41.3%

electricity + delivered heat

District-heating capacity ≈35–36 Gcal/h (41–42 MWth)

Steam extraction 0.3–0.6 MPa; up to 70 t/h

District-heating water flow up to 800 t/h

Off-Gas Cleanup: >350-Fold Release Reduction

1975–1978: OFF-GAS CLEANUP AND NEUTRAL-OXYGEN WATER CHEMISTRY

1975: OFF-GAS CLEANUP



>350×

lower gaseous fission-product release activity
50 mg/m³ before the carbon beds
35 years: releases remained within the control level

1978: NEUTRAL-OXYGEN WATER CHEMISTRY

FEEDWATER O₂ CONTROL

35–50 → 200

µg/kg O₂
pH unchanged: 6.0

FEEDWATER CORROSION PRODUCTS: µg/kg

	BEFORE		AFTER	
Fe	14	→	5	2.8× lower
Cu	15	→	5	3× lower
Zn	20	→	2	10× lower

UP TO 10×

lower feedwater corrosion-product concentration
Corrosion rate: 0.02 mm/year
Protective oxides form in deposits

VK-50 Modernisation: Fuel Use, Safety and Plant Evidence

WHAT CHANGED — AND WHY

Redesigned the core and fuel assemblies

to flatten power distribution and increase fuel burnup

Accelerated absorber insertion and added three independent borated-water sources

to provide fast, diverse reactor shutdown

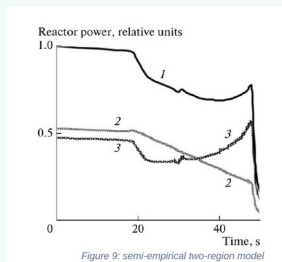
Added a 20 m³ passive accumulator and two independently powered pumps

to combine immediate and sustained emergency makeup

Added a cooldown condenser and a 100 m³ steam-suppression pool

to remove decay heat without off-site power and condense discharged steam

2009 FEEDWATER EVENT — MODEL ASSESSMENT



Calculated response · 0–50 s

FACTUAL PLANT RECORD

Feedwater ↓≈90% · protection 84 s ·
subcritical 104 s

CALCULATED AT 50 s

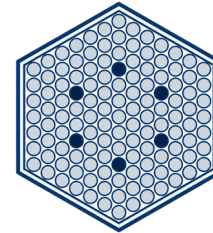
≈30% point kinetics
78% two-region model

The spatial model represented axial power redistribution that the core-average model could not reproduce

Percentages and graph are calculated, not measured power data

SAME 168 LATTICE POSITIONS — DIFFERENT FUNCTIONS

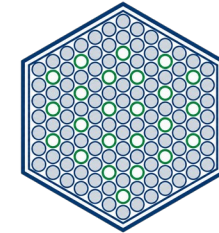
BEFORE



162 fuel rods + 6 absorber rods

W/F = 2.2

AFTER



138 fuel rods + 30 hollow Zr water tubes

W/F = 3.0

At unchanged 2.4% U-235: 13–14 → 22–23 MWd/kg U

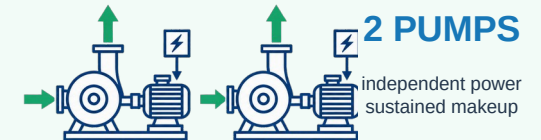
TWO COMPLEMENTARY EMERGENCY-MAKEUP ROUTES



20 m³

PASSIVE ACCUMULATOR
immediate, time-limited makeup

+



2 PUMPS

independent power
sustained makeup

RELAP5 large-break responses are calculations, not measured accident data

Experimental Evidence for Responsible Nuclear Development



Russia continues to develop nuclear energy technologies for cogeneration, hydrogen production and industrial heat.

- **TARGETED EXPERIMENTS**

Support the qualification of fuels, structural materials and operating regimes

- **EXPLORATORY R&D**

Addresses unresolved engineering questions

- **OPERATING DATA ANALYSIS**

Helps validate models and reveal full-scale plant behaviour

Together, these capabilities support evidence-based engineering decisions and the safe, sustainable development of nuclear technology.