THE SALIENT FLUORIDE FUEL SALT IRRADIATIONS

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Amsterdam
COMPLETE NUCLEAR INFRASTRUCTURE
**THE HIGH FLUX REACTOR (HFR)**

- High flux
- 45 MW thermal power
- Stable and constant flux profile in each irradiation position
- Main applications
  - Isotope production
  - Nuclear energy irradiation services
  - R&D
- 31 operation days per irradiation cycle, 9 cycles a year
THE HIGH FLUX REACTOR (HFR)

Ex-core region, flux control by displacement

<table>
<thead>
<tr>
<th>1</th>
<th>2</th>
<th>3</th>
<th>4</th>
<th>5</th>
<th>6</th>
<th>7</th>
<th>8</th>
<th>9</th>
<th>10</th>
<th>11</th>
<th>12</th>
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</thead>
</table>

- Reflector
- Fuel
- High Flux Position
- Medium Flux Position
- Low Flux Position
- Control Rod

<table>
<thead>
<tr>
<th>0-3, location dependent</th>
<th>5-8</th>
<th>3-5</th>
<th>1-3</th>
<th>Material DPA rate (DPA/year)</th>
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</thead>
<tbody>
<tr>
<td>0-700, displacement controlled</td>
<td>n/a</td>
<td>500-700</td>
<td>300-400</td>
<td>Linear heat rate (W/cm fresh LWR fuel)</td>
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The stable and constant flux profile in each irradiation position is a unique HFR feature
THE DUTCH NUCLEAR R&D PROGRAM

R&D themes:

- Safe Reactor Operation
- Radiation Protection
- Decommissioning
- Nuclear Technology for the future
  - SMR
  - Fusion
  - LUMOS (Learning to Understand MOIten Salts)
LUMOS

Trilateral collaboration between NRG, JRC and TUD

• Complementary competences

Molten Salt Technology fits well within R&D goals

• Improving safety
• Reducing use of resources
• Contributing to CO₂-free energy market

Program Objectives

• Obtain operational experience
• Confirm FP stability in the salt
• Investigate FP management methods
• Develop in-pile metal/graphite corrosion rig
• Waste route for spent molten salt fuel
• In-pile molten salt loop for the HFR Petten
SALIENT-01

Goals:
• Handling experience
• Salt-graphite interaction
• Fission product stability / redistribution
• Metal particle size distribution

Issues:
• Reduced salt condition \(\rightarrow\) increased graphite interaction
• Radiolytic gas production
• Graphite crucibles
• Open container (through metallic filter)
• Wall temperature maintained at ~610 °C (ThF₄-LiF), 24 TCs
• Neutron fluence monitored through activation sets
# MATERIAL SAMPLES

<table>
<thead>
<tr>
<th>nr</th>
<th>mat crucible</th>
<th>Contents</th>
<th>øin (eff)</th>
<th>øin (crucible)</th>
<th>identifier</th>
<th>Metal samples</th>
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<tr>
<td>L5</td>
<td>PCIB</td>
<td>SS</td>
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<td>8.0</td>
<td>EXP180-05</td>
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<td>LIF</td>
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<td>PCIB</td>
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<td>7.6</td>
<td>7.6</td>
<td>EXP180-01</td>
<td>nickel (Ni-201) sponge</td>
</tr>
</tbody>
</table>

Nickel added to two crucibles:
- Foil
- Sponge
FUEL POWER VS. TIME (THF$_4$-LIF)

- Fuel power increases during the irradiation (U-233 production)
- Constant wall temperature by variation of gas mixtures
ASSEMBLY

Synthesis and crucible loading at ITU

Assembly of sample holder at NRG
FLUORINE EVOLUTION (RADIOLYSIS)

- Best estimate (maximum) G-value: 0.02 (0.045) F₂-molecules / 100 eV.

- Saturation at fluorine losses of 2-8 mol-%

G-value: slope = 0.012 molecule F₂ / 100 eV

Large salt blocks, higher doserate

Fig. 2.3. Loss of Radiolytic Fluorine from MTR-47.5 Capsules.

Fig. 3. Fluorine generation curves for 1986 and 1995 irradiation experiments.
BACK-END

Fluoride salt is not an acceptable waste form (Corrosive and Unstable)

Temporary storage:
- Nickel-based container
- Inert gas
- $F_2 / UF_6$ adsorbent

NRG waste route to COVRA (government storage)
- Dissolution in strong nitric acid
- Precipitation of as hydroxides or nitrates after removal of fluoride
- Calcination to oxides
- Cementation of remaining liquid waste

General MSR waste processing: vitrification?
SALIENT-02

- Twin experiment to SALIENT-01
  - LiF – BeF₂ – UF₄ eutectic

- Not yet assembled
  - On hold

- May be rebuilt with pressure sensor
  - on-line measurement of fluorine release
NEXT: SALIENT-03/-04

Goals:
• Quantify radiolytic gas production

• Realistic chemistry (salt buffering, use of heaters)
• Metal corrosion study (Ni-based alloys)
• Influence graphite on metal corrosion
• Graphite-salt interaction
• Metal particle size distribution

• ‘Tritium release measurements’
GAMMA IRRADITION

- Space for 5 Salt capsules
- Pressure vs. dose
- Long-term experiment
- Sister experiment at TU Delft

Axial distribution of the gamma radiation over the height of the container, for the large diameter container.
OUTLOOK

• We underestimated radiolytic $F_2$ release and related chemistry
  • SALIENT-01 now non-representative
  • Significant delays / time lost
  • Safety discussions before end 2016

• 2017 priorities
  • Start of SALIENT-01
  • Start of gamma irradiations
  • Establish salt waste route to COVRA
  • First results helium bubbling @TU Delft
  • Safety Reports for SALIENT-03/-04
  • Design of the in-pile loop
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