Overview

- MARVEL Thermal-hydraulics
- Modeling
- Boundary Conditions and Assumptions
- Acceptance Criteria
- Uncertainties and Hot Channel Factors
- Deterministic Safety Analysis Results
General Thermal-hydraulic Characteristics

- **MARVEL general thermal-hydraulic (TH) characteristics**: liquid metal cooled, low-power density, closed loop, series-parallel coupled natural circulation system.
System Description

- **Key TH characteristics:**
  - Use of *natural circulation* on primary and secondary sides
    - No pumps
    - Better flow distribution
    - Higher reliability
    - Simplicity
  - 4 loops
  - Core power: 85 kW<sub>th</sub>
  - Low power densities (average values)
  - Core average NaK temperature at Hot Full Power (HFP): ~ 500 °C
  - Operating pressure in the cover gas zone: ~ 3.2 atm
System description

- Use of analytical models for preliminary system design and numerical code verification

\[ \dot{m} = \left( \frac{2 \beta T g \Delta z_C}{c_p R \rho_0} \right)^{\frac{1}{3}} \]

- Elevation difference \( \Delta z_C \) between thermal centers: \( \sim 1.1 \) m
- Minimization of circuit pressure drops \( R \)
- Predicted total NaK mass flow at Hot Full Power: \( \sim 1.5 \) kg/s

- Non-dimensional analysis
  - for deriving steady-state maps
  - thermal-hydraulic stability studies

\[ Re_{ss} = C \left[ \frac{(Gr_m) \Delta z_C}{N_G} \right]^r = 1.956 \left[ \frac{(Gr_m) \Delta z_C}{4524} \right]^{0.3636} \] [turbulent flow]
Thermal-hydraulic Modeling & Simulation Tools

- Modeling and simulation (M&S) strategy for safety analysis
  - Use **best-estimate** nuclear safety codes and commercial codes with **extensive nuclear pedigree** and **well-proven reliability**
  - Perform independent **high-fidelity** calculations using commercial computational fluid-dynamic (CFD) codes for selected system, structure, components (SSCs) for design validation
MARVEL Thermal-Hydraulic Design

- Use of INL's RELAP5-3D system thermal-hydraulic code as an M&S workhorse
- The RELAP series of codes have been developed at INL for over 50 years
  - RELAP5-3D is the flagship of nuclear reactor system analysis tools → most widely used nuclear reactor accident analysis code
  - Development still ongoing (e.g., integration into INL’s MOOSE framework)
  - Capability to model liquid metals systems
    - Several fluid properties libraries available
    - Specific correlations for liquid-metal heat transfer
    - 3-D hydraulic components, 3-D neutron kinetics
- TH model validation using MARVEL Integral Test Facility (ITF) Primary Coolant Apparatus Test (PCAT)
Boundary Conditions and Assumptions 1/2

- Core conditions from MCNP code Monte Carlo calculations
  - Core at Beginning of Life (BOL)
  - ANS-05 decay standard
  - Reactivity coefficients vs. temperature
  - Pin power peaking factors
  - Axial power peaking factor
Boundary Conditions and Assumptions 2/2

- Conservative assumptions for Beyond Extremely Unlikely events (BEU) → higher PCS and fuel temperatures
  - Gamma and neutron heating concentrated in the BeO
  - Other parameters

<table>
<thead>
<tr>
<th>Parameters</th>
<th>Best-Estimate</th>
<th>Conservative</th>
</tr>
</thead>
<tbody>
<tr>
<td>Overpower factor for the hot channel</td>
<td>1.0</td>
<td>1.15</td>
</tr>
<tr>
<td>Fuel heat transfer coefficient</td>
<td>Laminar/Turbulent</td>
<td>Laminar</td>
</tr>
<tr>
<td>Helium Stirling engine average temperature at HFP, °C</td>
<td>300</td>
<td>325</td>
</tr>
</tbody>
</table>
Acceptance Criteria

- For Extremely Unlikely (EU) events, applied to Beyond Extremely Unlikely (BEU) events
  - Fuel: from fuel mechanics analysis
  - Clad: avoid localized boiling (surface temperature < NaK saturation temperature at atmospheric pressure)
  - Bulk coolant: protect PCS integrity
  - Core: qualitative, respected if criterion 2) achieved

<table>
<thead>
<tr>
<th>Acceptance Criteria</th>
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<tr>
<td>1</td>
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<td>2</td>
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<td>4</td>
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<td>5</td>
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</tbody>
</table>

- Peak fuel centerline temperature < 925 °C
- Peak clad internal temperature < 764 °C
- Bulk coolant < 704 °C
- Core remains coolable
Deterministic Analysis Options

• RELAP5-3D is a Best Estimate code (BE)
• Safety analysis strategy: using combination of options 2+3
• Conservative assumptions for systems availability, e.g.  
  − No scram

Options for Safety analysis  
[from IAEA, SRS No. 52]
Uncertainties & Hot Channel Factors

• Hot channel factors (HCF) implemented in RELAP5-3D as safeguards against uncertainties (minimize margins)
  − protect fission product barriers (fuel, clad, PCS)
• HCF derived from references based on past experiences, analytical models, qualified references, high fidelity calculations
• HCF to be updated
  • using PCAT data
  • before going critical

\[ T_M = T_{in} + \sum_{m=1}^{M} F_m \Delta T_{m,nom} \]

Temperatures of Interest

Cladding Temperature Distribution

Thermomechanical analysis of fuel elements, S.J. Yoon, ECAR-7210
Uncertainties & Hot Channel Factors

• HCF treat in a conservative way (direct + statistical combination) uncertainties on:
  - Coolant mixing
  - Power & temperature measurements
  - Core heat transfer coefficient
  - Fuel geometry tolerances
  - Material physical properties (fuel, coolant, clad, gap)
  - Fuel nuclear properties

• Probabilistic treatment being considered for future uncertainty quantification (UQ) using RELAP5-3D/RAVEN code
Normal Operation: Steady-State 1/2

• Steady State results for 36 TRIGA fuel rods, 1.414” OD (3.59 cm), 25” (63.5 cm) tall active core
• Reactor power: 85 KWth
• All structures in thermal equilibrium
• Good steady-state temperature margins

<table>
<thead>
<tr>
<th>Parameters - Primary &amp; secondary side</th>
<th>Values</th>
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</thead>
<tbody>
<tr>
<td>NaK inlet core temperature, °C</td>
<td>471</td>
</tr>
<tr>
<td>NaK outlet core temperature, °C</td>
<td>540</td>
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<tr>
<td>NaK core temperature rise, °C</td>
<td>69</td>
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<tr>
<td>Total mass flow, kg/s</td>
<td>1.49</td>
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<tr>
<td>EGaInSn minimum temperature, °C</td>
<td>403</td>
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<tr>
<td>EGaInSn maximum temperature, °C</td>
<td>425</td>
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<td>EGaInSn temperature rise, °C</td>
<td>22</td>
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<tr>
<td>IHX EGaInSn mass flow, kg/s</td>
<td>2.6</td>
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Normal Operation: Steady-State 2/2

- Other relevant parameters

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<tr>
<th>Parameters</th>
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</thead>
<tbody>
<tr>
<td>PCS pressure drop, Pa</td>
<td>160</td>
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<tr>
<td>BeO side reflector maximum temperature, °C</td>
<td>519</td>
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<tr>
<td>PCS wall maximum temperature, °C</td>
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<td>PCS primary pressure, kPa</td>
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<td>Guard vessel to air heat losses, kW</td>
<td>4.8</td>
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<td>Air riser nominal inlet temperature, °C</td>
<td>20</td>
</tr>
<tr>
<td>Air riser outlet temperature, °C</td>
<td>36</td>
</tr>
</tbody>
</table>
Normal Operation: Load Following

- **Load-follow:**
  - Simulate reaction to imposed power change: 100/75/100 % $P_{\text{nom}}$ over ~2.5 hr period
  - All four Stirling engines in operation
  - Control system simulate reactivity insertion by control drums
    - Reactivity insertion vs. position
    - Drum rotation speed
  - Power changes imposed (simulate ±5% $P_{\text{nom}}$/min ramps)
    - PCS max temperature rate: ~0.91 °C/min (~54.5 °C/hour)
    - CD reactivity rate: ~+/-1.4 cents/min
Postulated Accident Conditions: UTOP at HFP, w/ Stirling engines

- **Unprotected Transient Overpower**
  - Step reactivity insertion (0.4$) $\rightarrow$ 1 CD out from critical position to the mechanical stops
  - No SCRAM
  - Stirling engines on $\rightarrow$ maximize energy release to the fuel
  - Reactor power peaks $\sim 3.74 \ \text{P}_{\text{NOM}}$ (318 kW) at $t = 12$ s
  - Negative reactivity feedbacks counters the power surge $\rightarrow$ system back to a steady higher power and higher temperature by $t = \sim 20$ min
  - **No safety concerns** until scram (not needed)
Postulated Accident Conditions: UTOP at HFP, w/o Stirling engines

- **Unprotected Transient Overpower**
  - Step reactivity insertion (0.4$) $\rightarrow$ 1 CD out from critical position to the mechanical stops
  - No SCRAM
  - Stirling engines off $\rightarrow$ maximize PCS temperature and pressure
  - Used for ASME D-section calculations
  - **No safety concerns** until scram (not needed)
Postulated Accident Conditions: UTOP at CZP

- **Unprotected Transient Overpower at Cold Zero Power (20 °C)**
  - Step reactivity insertion ($1.3\%$) $\rightarrow$ 1 CD out from critical position to the mechanical stops
  - No SCRAM
  - Reactor power peaks $\sim34$ $P_{NOM}$ (2.9 MW) at $t = 2$ s
  - Negative reactivity feedbacks counters the power surge
  - **No safety concerns** during first 5 minutes, reasonably also later
    - Temperatures stay safely low
  - Fast temperature ramp rate ($\sim 11$ °C/min), but max PCS temperature $< 200$ °C
Postulated Accident Conditions: ULOHS

- **Unprotected Loss of Heat Sink**
  - All 4 Stirling engines heat removal lost at \( t = 1.0 \text{ s} \)
  - No SCRAM
  - Reactor cooled only by heat losses through guard vessel only (~4.8 kW) → conservative assumption
  - Reactor shutdown by intrinsic negative reactivity
  - Return to power caused by fuel cooldown
  - Core power < guard vessel heat losses for first 24 hr
  - **No safety concerns** during at least first 24 hr
  - Beyond 24 hr, reactor power = heat losses (new equilibrium)
Postulated Accident Conditions: ULOF

- **Unprotected Loss of Flow**
  - Total blockage of all 4 downcomers at time $t = 0.0$ s (assume catastrophic damage of all 4 IHXs) →
    - not credible event
    - bounding partial loss of flow events
  - no SCRAM
  - Loss of secondary side (IHX) heat removal capabilities
  - Reactor cooled *only* by heat losses through guard vessel
  - Reactor power self-reduced
  - Hot spot clad temperature not of safety concern due to the reactor self shut-down features
  - **No safety concerns**: data shown for the first 24 hrs, beyond that reactor power = heat losses (new equilibrium)

![Core mass flow](image1)
![Hot spot temperature](image2)
![PCS & guard vessel temperatures](image3)
Postulated Accident Conditions: ULOF, no DHRAC

- Unprotected Loss of Flow and blockage of Decay Heat Removal Air Channel (DHRAC)
  - Loss of secondary side (IHX) heat removal capabilities
  - Total loss of cooling
  - Reactor power self-reduced
  - Hot spot clad temperature not of safety concern due to the reactor self shut-down features
  - No safety concerns for the first 24 hrs

Be and BeO temperatures
System pressures
PCS & guard vessel temperatures
Reactor power & heat losses
Reactivity
Temperature safety margins
Postulated Accident Conditions: ULOCA

- **Unprotected Loss of Coolant Accident**
  - MARVEL reactor avoids by-design the NaK level drop below the top of the core (core never uncovered) also during the break of the low-elevation components (downcomer, lower plenum)
  - Decay heat removal capabilities bounded by ULOF calculations
Summary

- RELAP5-3D system analysis shows reliable and stable MARVEL performances during operational transients and selected BEU transients
- **Very conservative** accident analysis shows that all **minimum safety margins are > 0**

<table>
<thead>
<tr>
<th>Transient</th>
<th>Minimum margins (°C)</th>
<th></th>
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<tbody>
<tr>
<td></td>
<td>Clad</td>
<td>Fuel centerline</td>
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<tr>
<td>UTOP- HFP</td>
<td>18</td>
<td>201</td>
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<tr>
<td>UTOP - CZP</td>
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<td>620</td>
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<td>ULOHS</td>
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<td>ULOF</td>
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Questions?