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Application of RELAP5-3D in Support of TREAT Experiment Design and Analysis

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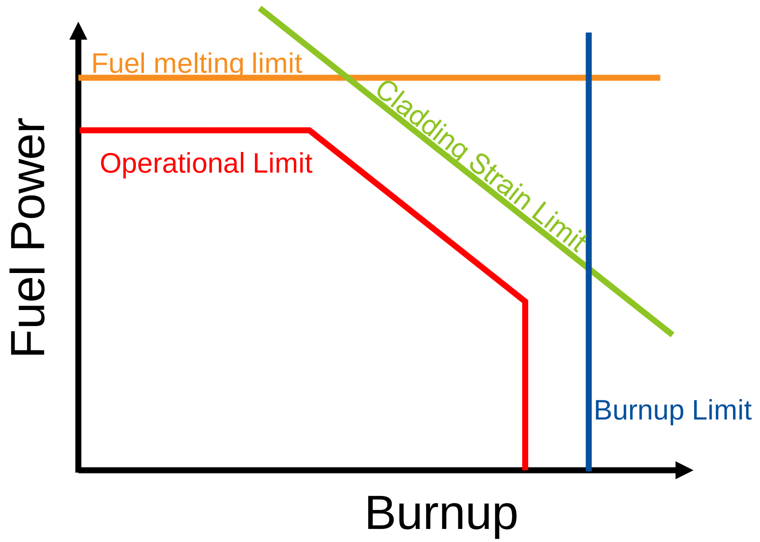
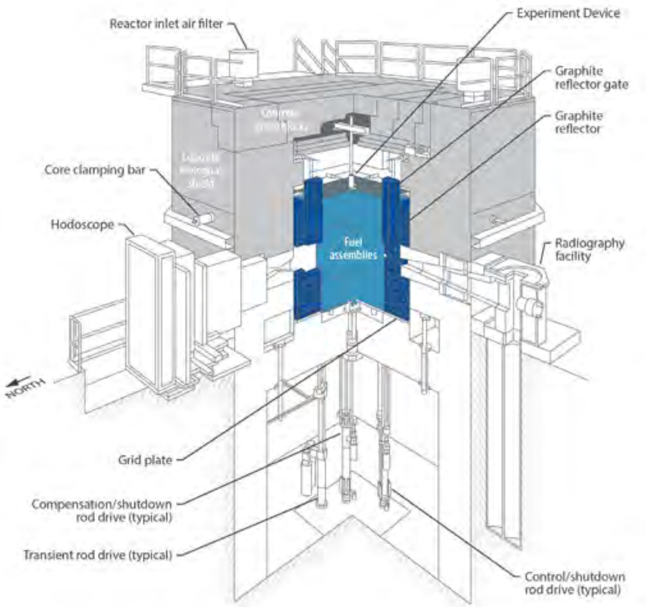
Why Transient Test Nuclear Fuels & Materials?

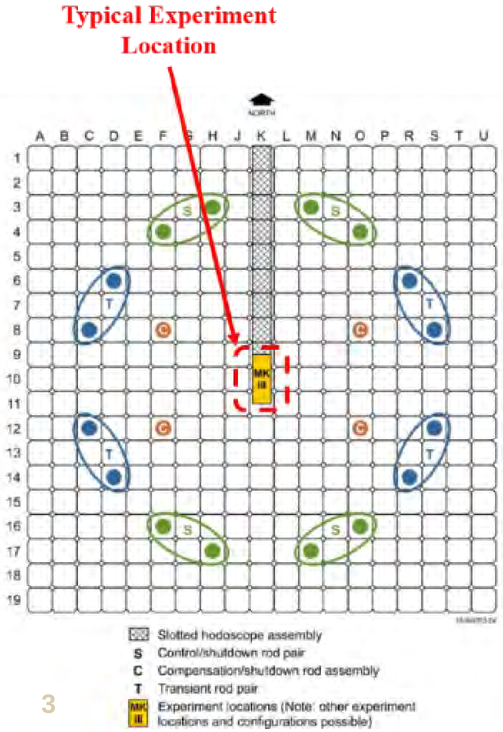
- **Transient testing is like car crash testing for nuclear fuels**
- Licensing a fuel system requires (see NUREG-0800):
  - Identification of all **degradation mechanisms and failure modes**
  - Definition of **failure thresholds** corresponding to each degradation mechanism
  - Applies to **normal operations, anticipated operational occurrences and design basis accidents**
- Many operational limits are dependent on degradation and failure thresholds
- Enables economic reactor operations via improved fuel design and performance understanding



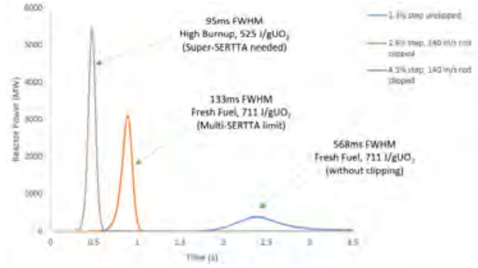
Transient Reactor Test Facility (TREAT)

- The Transient Reactor Test Facility (TREAT) main mission is to test nuclear fuels and materials in off-normal and accident conditions
  - Operated from 1959-1994, was later refurbished, and resumed operations 2017
- Zircaloy-clad graphite/fuel blocks comprise core, cooled by air blowers
  - Virtually any power history possible (within 2500 MJ max core transient energy)
  - No reactor pressure vessel/containment, facilitates access for in-core instrumentation
  - 4 slots view core center, 2 in use for fuel motion monitoring system & neutron radiography

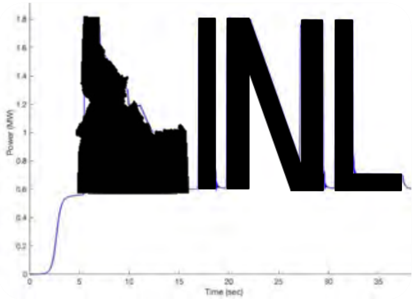


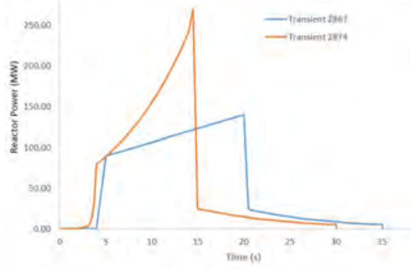


- Reactor provides brief (and typically extreme, up to 10^{17} n-cm^{-2}-s^{-1}) shaped neutron flux histories to test specimens
- Experiment vehicle does everything else
  - Safety containment, specimen environment, and instrumentation
- Collocated at INL with other complimentary facilities
  - Advanced Test Reactor (ATR) and Hot Fuel Exam Facility (HFEF)
  - Numerous fuel fabrication and characterization capabilities

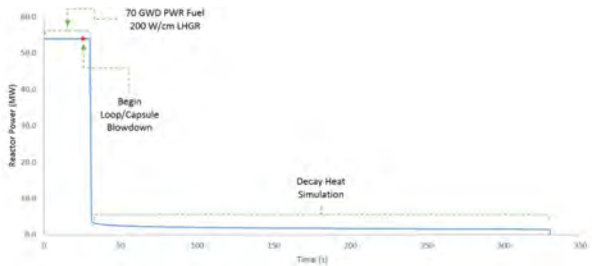


LWR RIA





SFR Transient Overpower

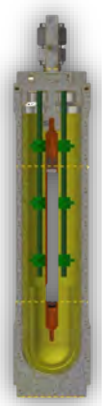


LWR LOCA

TREAT Experimental Testbeds

- **RELAP5-3D is an essential analysis tool for nearly all TREAT experiments**

General purpose



0.3 m

SETH
Separate effects static capsule

Water testbed

0.6 m

SERTTA
Static pressurized water capsule

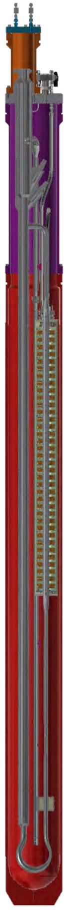
2 m

TWIST
Pressurized water blowdown capsule

Sodium testbed

1 m

THOR
Liquid-metal-bonded heat-sink capsule



3.5 m

Mark-IIIR
Flowing Na Loop

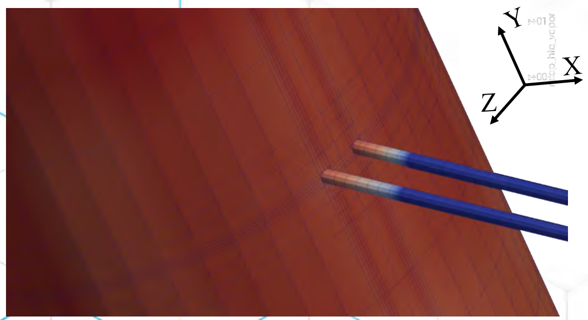
Advanced Gas Reactor & Space nuclear testbed

1 m

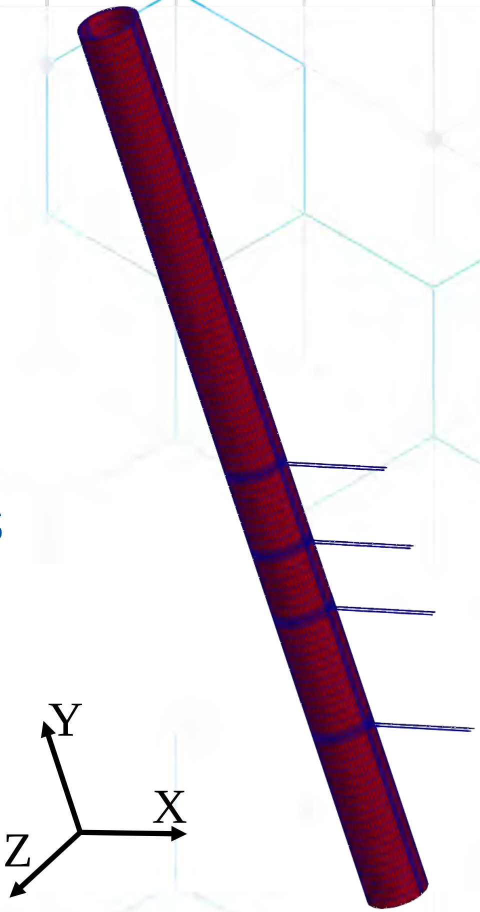
SIRIUS
Static Gas Capsule

2.5 m

Flowing Hydrogen/Helium Loop



Application: Advanced Multiphysics Code Coupling for Cladding Surface Thermocouples during Two-Phase Heat Transfer from Nuclear Fuel

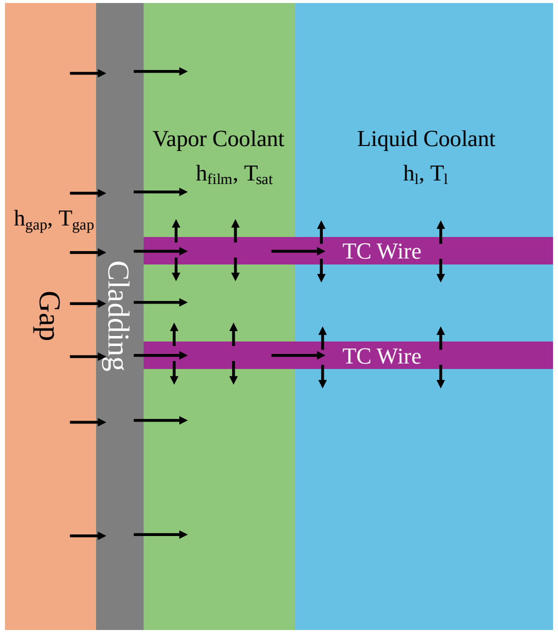


Summary

- Novel methodology for simulating the impact of outer cladding TCs during transient testing of nuclear fuels that does not require prior knowledge of the thermal-hydraulic conditions.
  - Leverages the thermal-hydraulic capabilities of RELAP5-3D and the BISON fuel performance code
  - In-memory coupling between BISON and RELAP5-3D is enabled by the MOOSE-wrapped application known as BlueCRAB
- BISON
  - 3D FEA modeling of both the fuel rod and attached TCs
  - Accounts for the inherent lack of azimuthal symmetry caused by the TCs.
  - Approach enables the ability to capture both axial and azimuthal temperature and thermo-mechanical effects due to the TCs.
- RELAP5-3D
  - Calculate the thermal-hydraulics (heat transfer coefficients, coolant temperatures, pressure) associated with the 3D representation of the fuel rod and TCs

Motivation

- Attaching thermocouples to the outer surface of the cladding is known to affect the nearby surface temperature – including the temperature measured by the thermocouple
  - What is the temperature of the cladding away from the thermocouple?
  - How does the presence of the thermocouples affect the experiment performance?
- While recognized as an important area for interpreting experimental results, little research has been performed in this area
  - NSRR analytical model [1]
  - INL FEA modelling [2]
  - Thermal-hydraulic boundary conditions (heat transfer coefficients, coolant temperatures, heat transfer regime regions, etc.) required as input parameters
- Develop a methodology using BlueCRAB to simulate the impact of outer cladding thermocouples thermo-mechanic behavior
  - Thermal-hydraulic boundary conditions calculated by RELAP5-3D
  - Thermo-mechanics simulated in BISON



T Illustration of the NSRR analytic model [1]

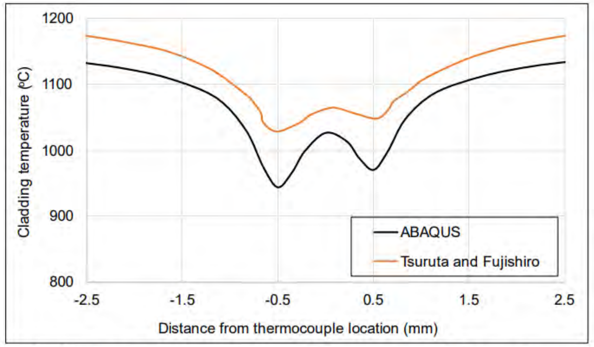


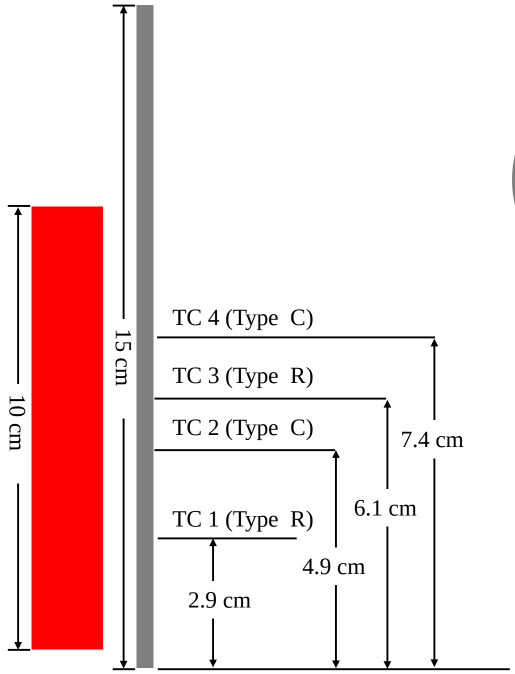
Fig. 2. Surface temperature profiles predicted from the Abaqus simulation and the reference [4] along the axial direction in the cladding.

Comparison of temperatures predicted by the NSRR analytical model [1] and INL FEA model [2]

[1] Tsuruta, Takaharu, and Toshio Fujishiro. "Evaluation of thermocouple fin effect in cladding surface temperature measurement during film boiling." Journal of Nuclear Science and Technology 21.7 (1984): 515-527.
[2] Seo, Seokbin, et al. Sensitivity study on the fin effect of thermocouple mounted on the heated surface under film boiling condition. No. INL/CON-23-73285-Rev000. Idaho National Laboratory (INL), Idaho Falls, ID (United States), 2023.

ATF-1-E Experiment

- Methodology developed and assessed using the Accident Tolerant Fuel–Reactivity-Initiated Accident–1–E (ATF-RIA-1-E) experiment
- Static water capsule fresh fuel RIA experiment performed at INL in TREAT
  - Water pre-heated to 200 °C, capsule pressurized to 2 MPa
- Fresh UO_2 fuel Zircaloy-4 cladding test rod
  - Rod pressurized to 2 MPa
  - 8 UO_2 naturally enriched fuel pellets
  - 2 insulator pellets (1 on each end)
  - Total rod length ~15 cm
- Four integral junction thermocouples welded to outer cladding surface
  - Two Type C, Two Type R
  - TC 3 & TC 4 failed prior to or during transient
  - TC is compromised of two wires spaced ~0.25 mm apart
    - Wire diameter ~0.25 mm



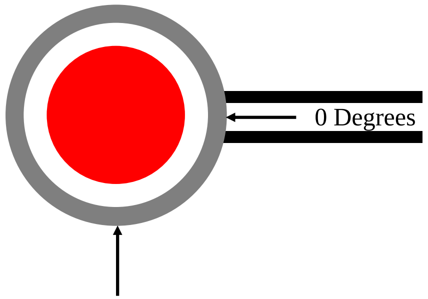
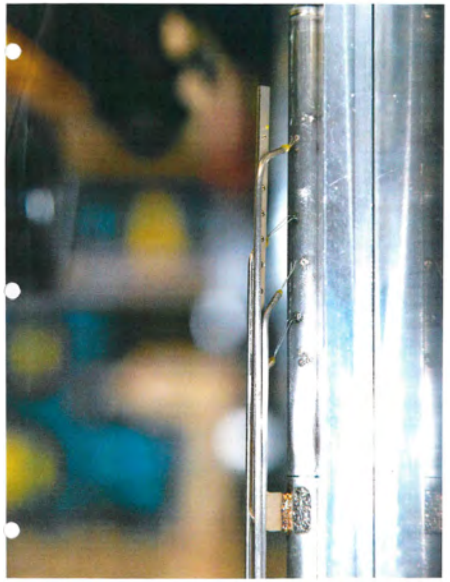
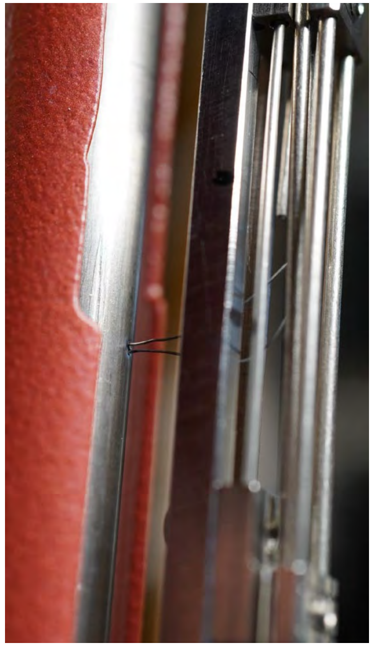


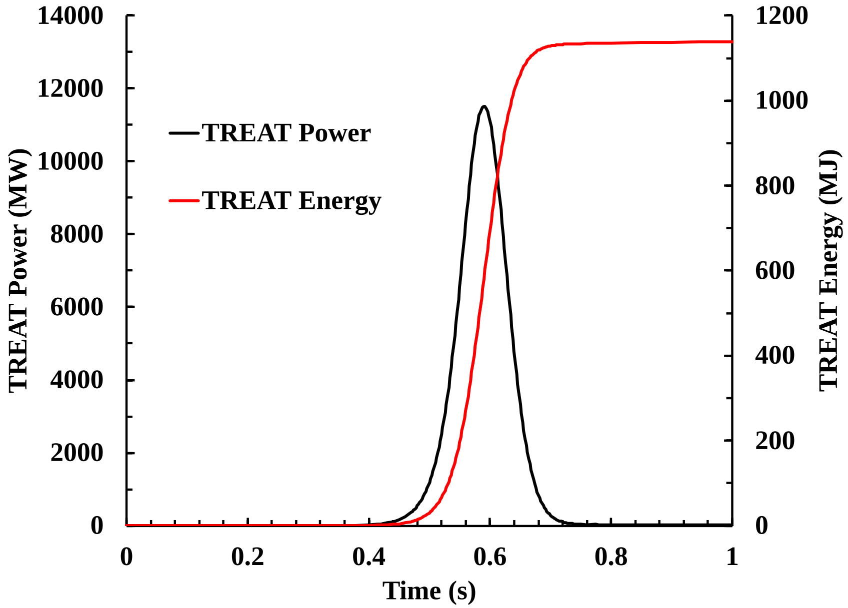
Illustration and images of ATF-RIA-1-E test fuel rod

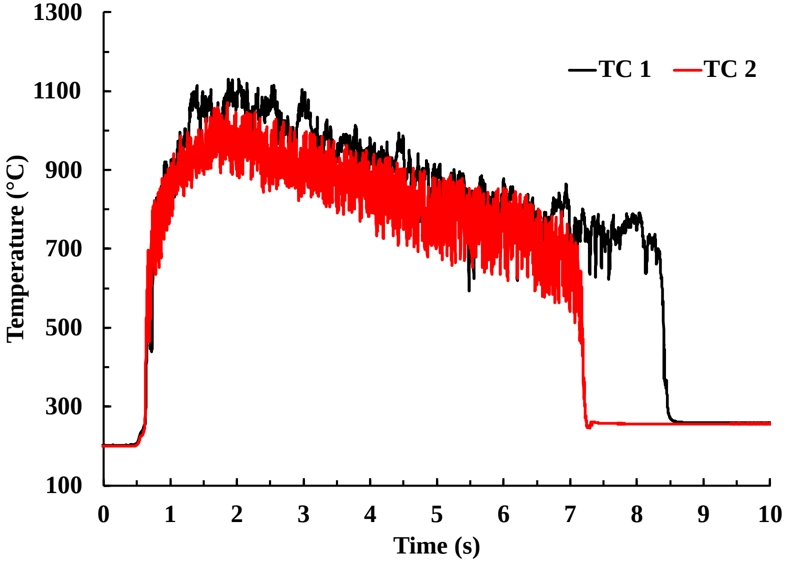


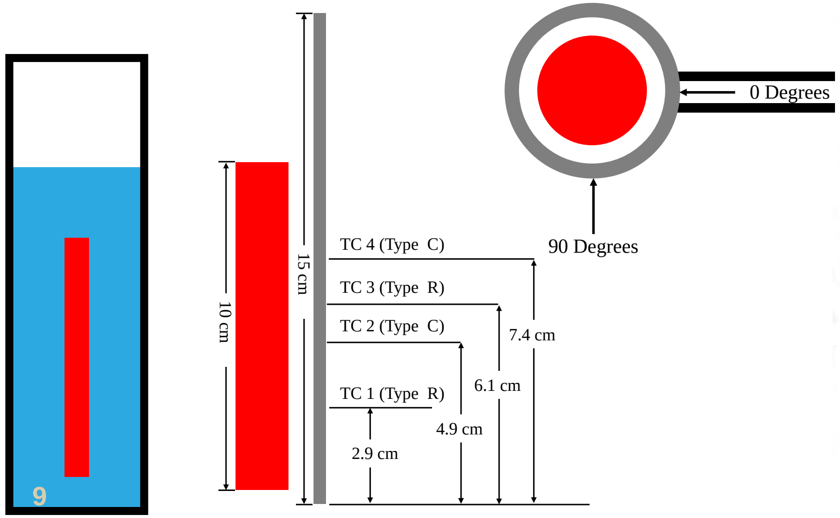


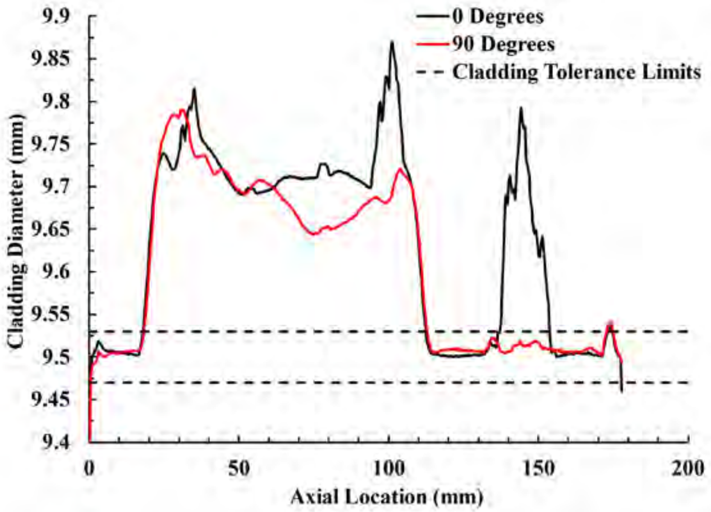
ATF-1-E Experiment

- RIA pulse full-width at half maximum: 89.8 ms
- Energy deposition: $590 \frac{J}{gm\text{−}UO_2}$
- Peak TC measured temperatures > 1000 °C
  - TC 2 rewets first at ~7.1 s, followed by TC 1 (8.4 s)
- Optical profilometry measured cladding diameter at two locations (0°, 90°)
  - Increased diameter in region of active fuel
  - Slightly peaked toward bottom

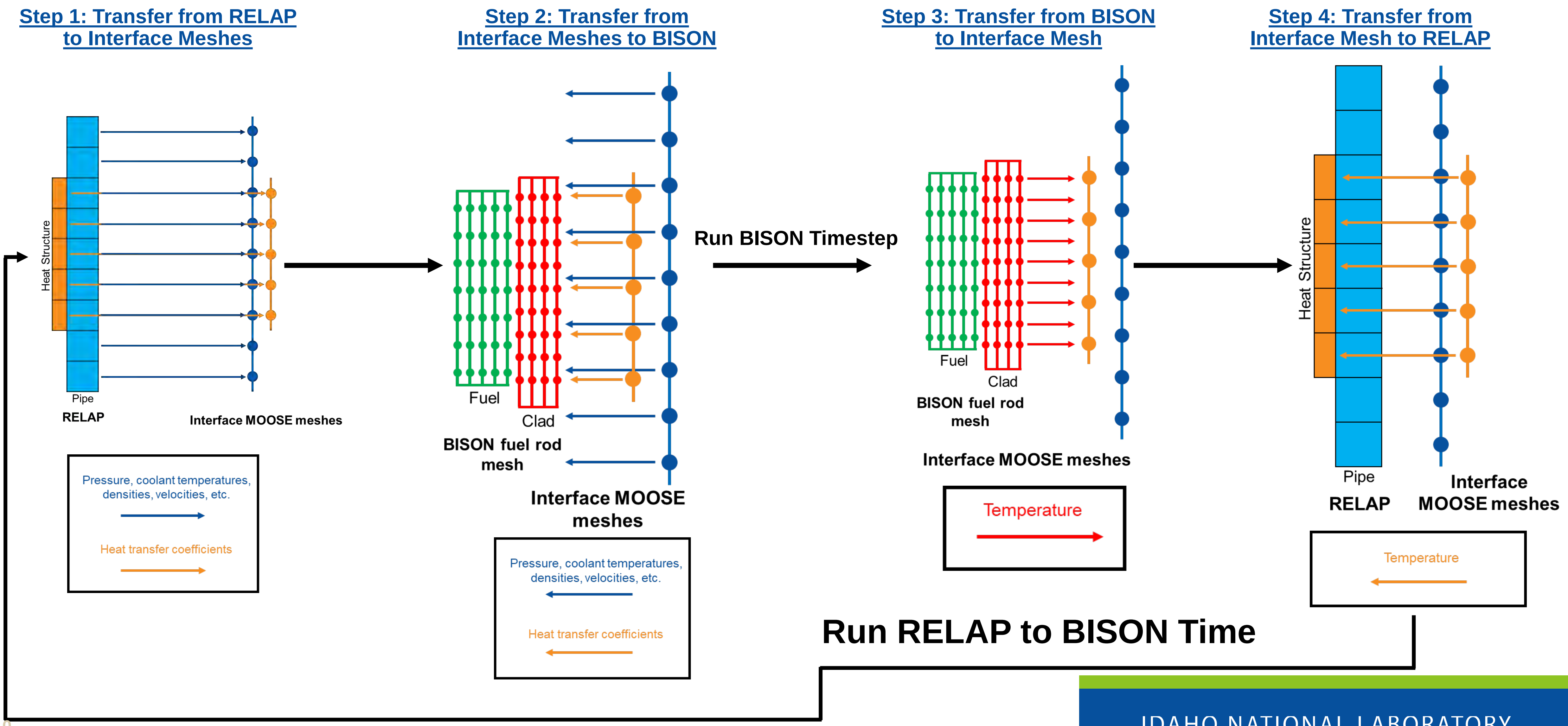






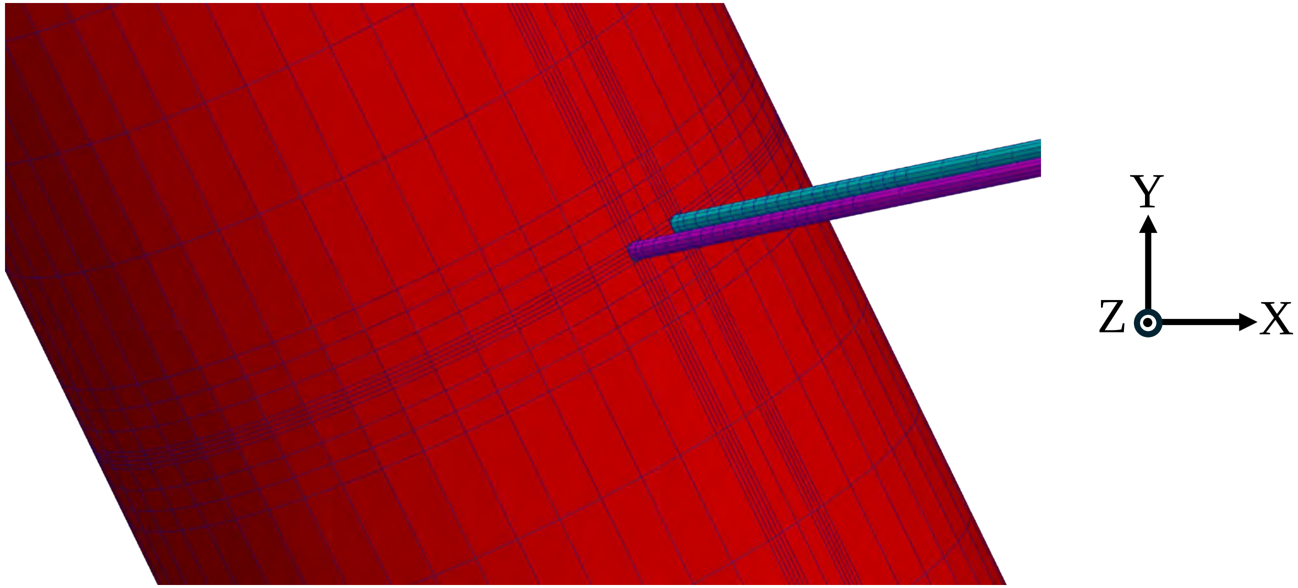


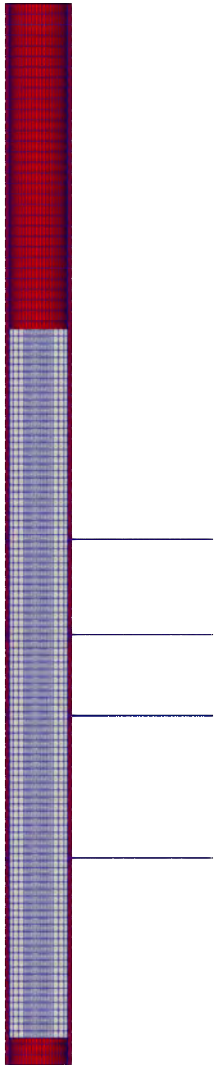
BlueCRAB Overview

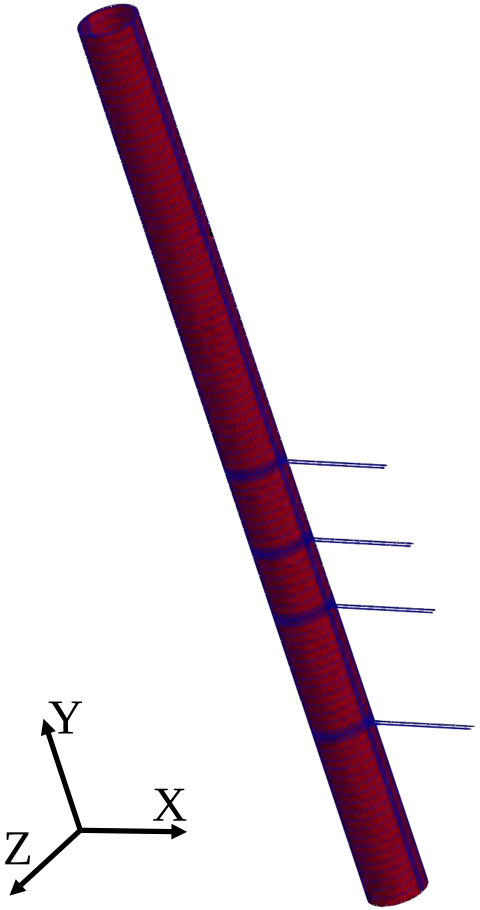


BISON Geometry and Mesh

- Geometry and mesh of the cladding tube and attached TC wires were created using the FEA software, Abaqus/CAE
- Fuel pellet mesh created in BISON using FuelPin3DMeshGenerator
- Cladding and TC wire mesh combined with fuel pellet mesh in BISON





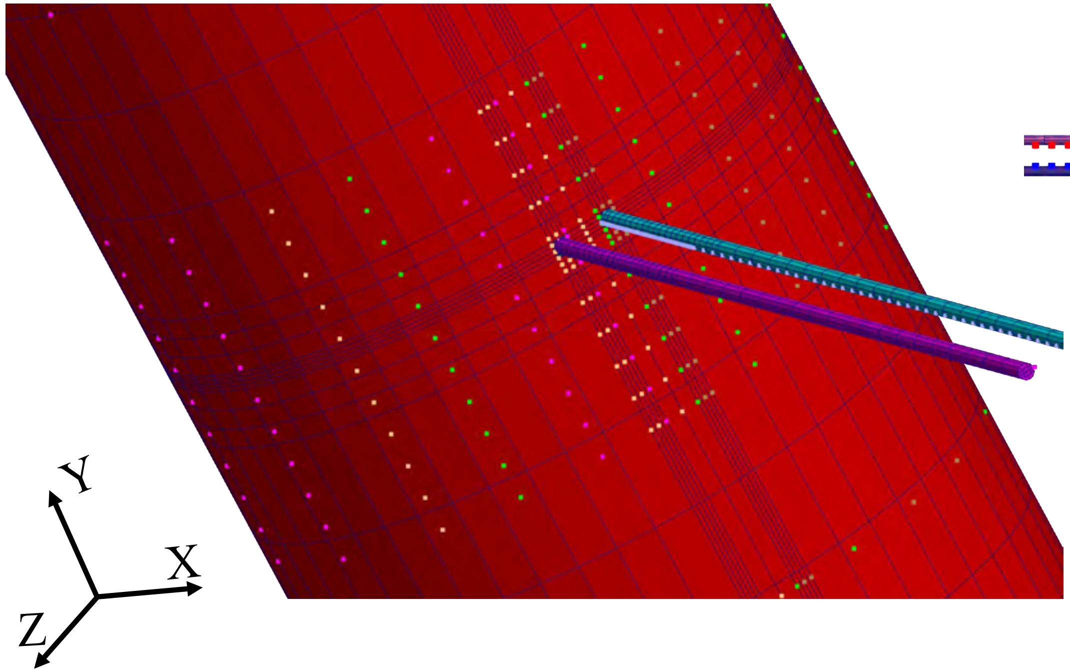


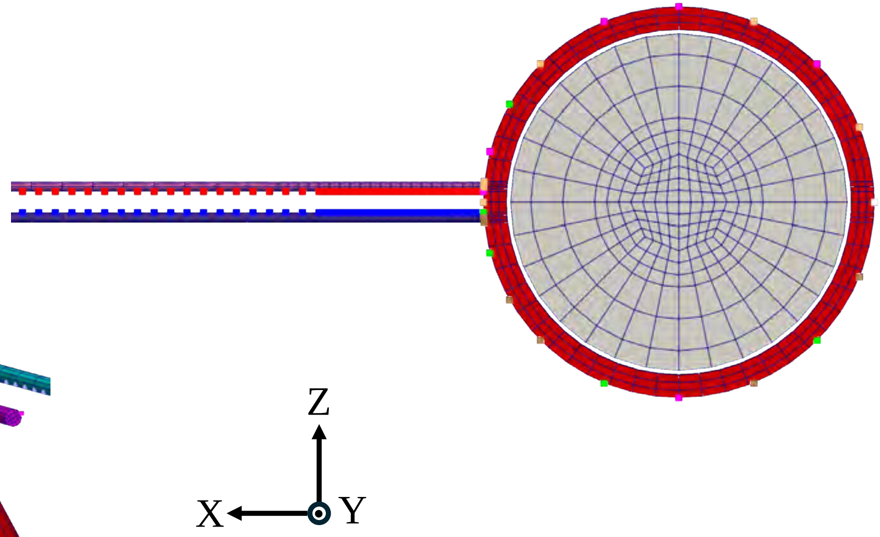
RELAP5-3D Model

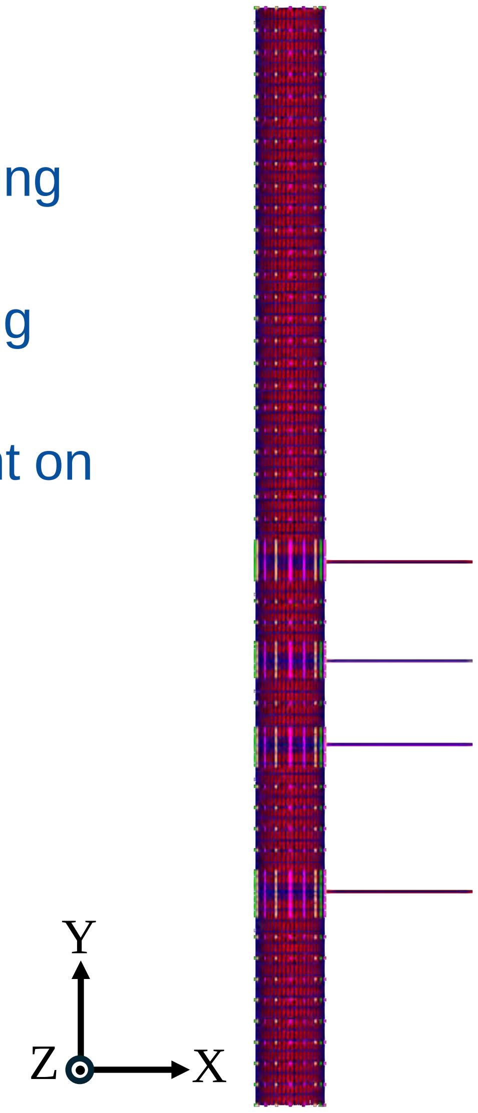
- **Capsule**
  - 1 pipe component represents static water capsule
- **Outer cladding surface**
  - 24 heat structure geometries
  - Each represents an azimuthal segment of the cladding surface
  - All heat structure geometries connected to the capsule pipe component
    - All but 1 heat structure geometry are “decoupled” from the pipe component
  - Axial resolution increased near TC wire locations
- **TC wires**
  - 8 heat structure geometries
  - Each represents the outer surface of one TC wire
  - Each heat structure geometry is connected to the pipe volume corresponding to its axial location in the capsule

BlueCRAB

- 1 interface mesh for the pipe component
- 24 interface meshes for the RELAP5-3D heat structure geometries representing the outer cladding surface
- 8 interface meshes for the RELAP5-3D heat structure geometries representing the TC wires
- Interface meshes are translated and rotated to the position that they represent on the BISON test rod

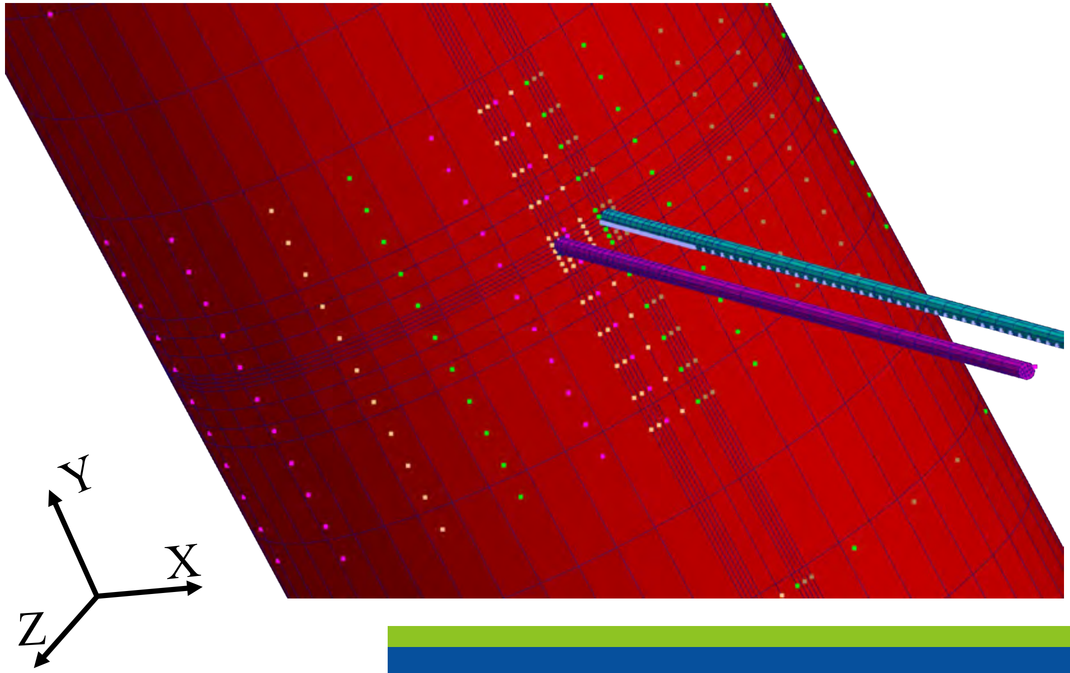






BlueCRAB

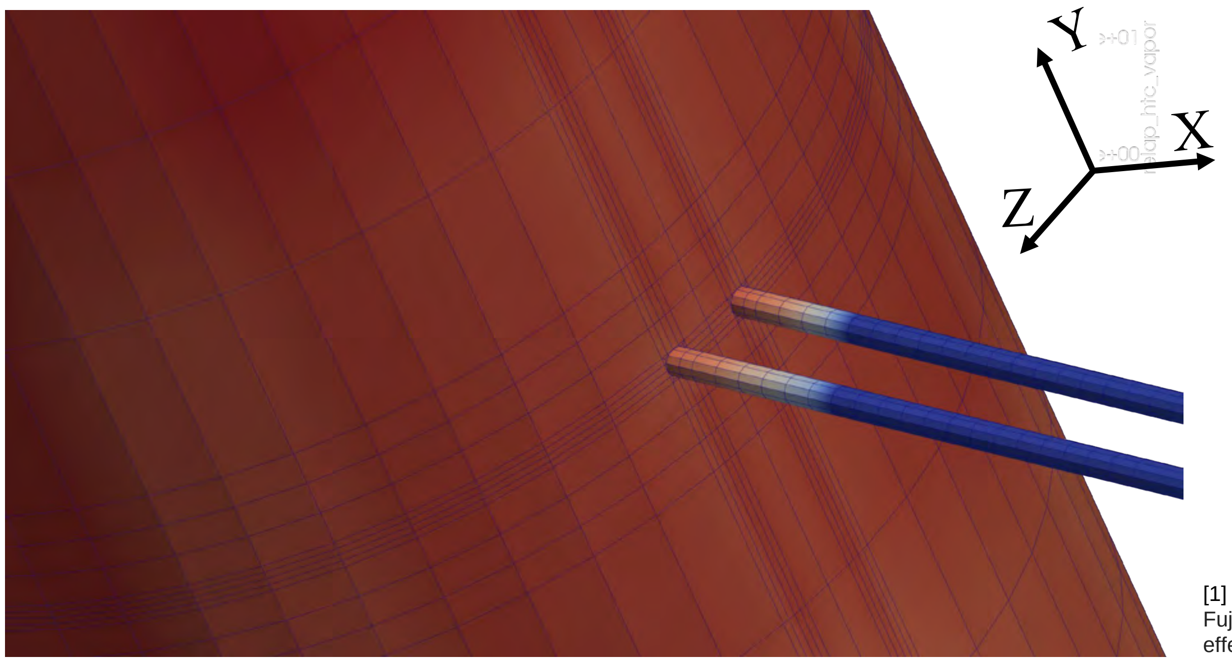
- **Outer Cladding**
  - Temperatures as a function of axial and azimuthal position are transferred from BISON to RELAP5-3D
  - RELAP5-3D calculates heat transfer coefficients based on these temperatures and transfers back to BISON
  - Results in axial and azimuthal varying heat transfer coefficients transferred back to BISON
- **TC Wires**
  - Temperatures vary along length of wire
  - Heat transfer coefficients vary along length
  - Captures change in heat transfer regime



BlueCRAB

This Work

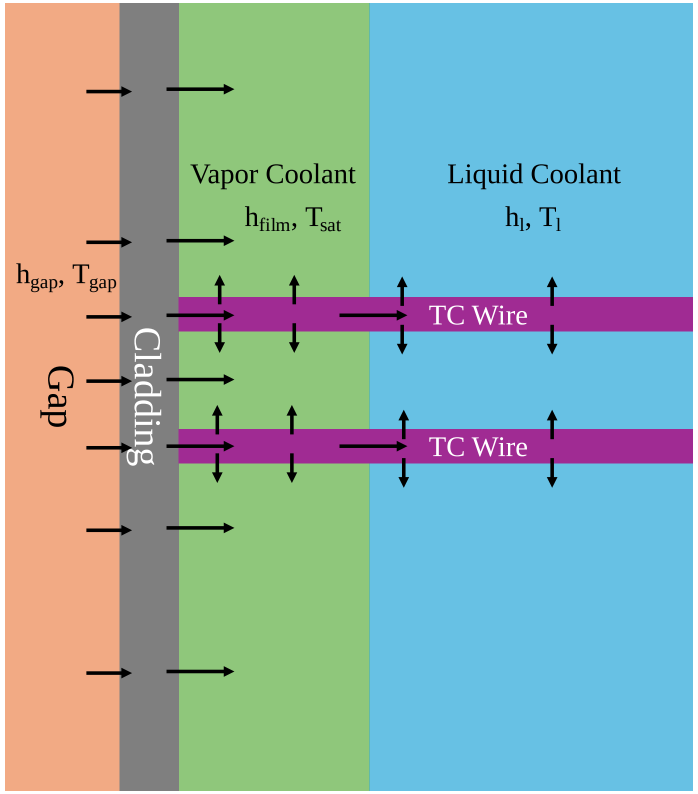
- No assumptions on the heat transfer regimes of the cladding and TC wires were made
- Heat transfer conditions are computed by RELAP5-3D based on temperatures computed by BISON.



Contour plot of relative film boiling heat transfer coefficient

NSRR Model [1]

- TH boundary conditions are required inputs.



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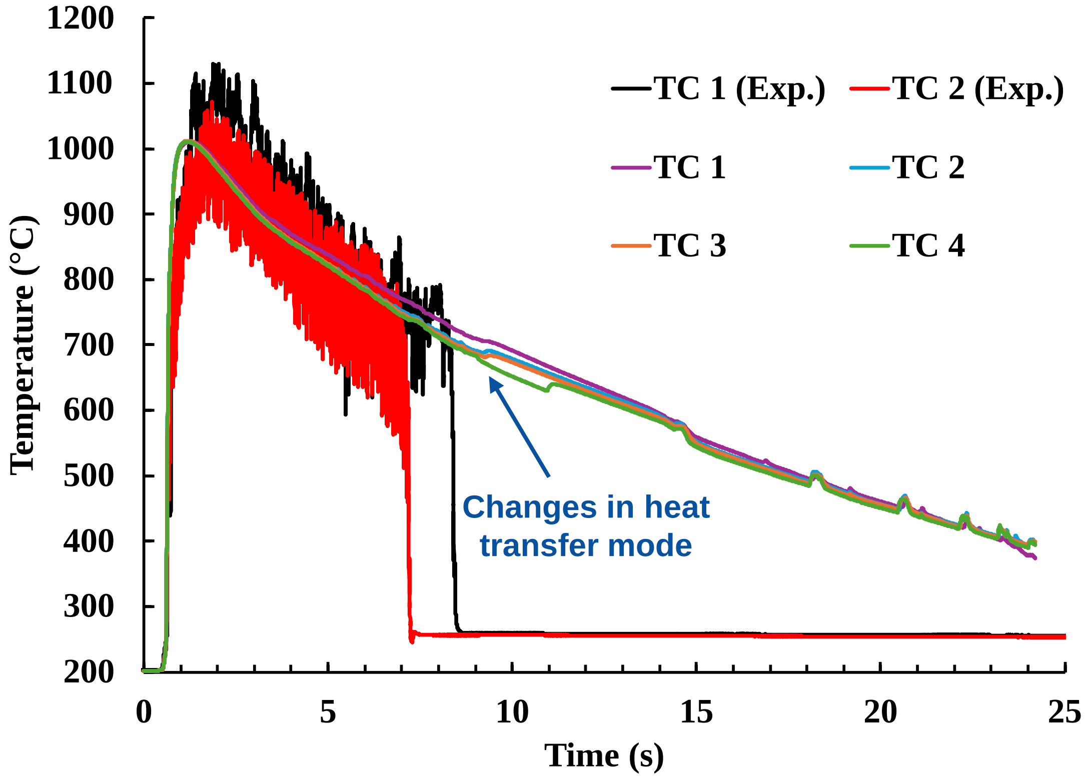
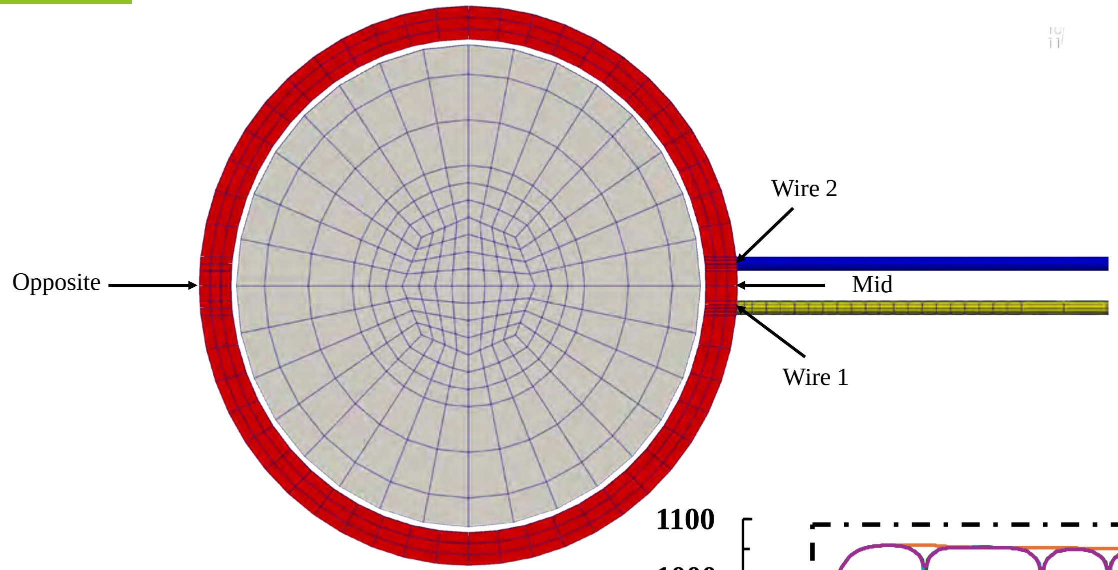
Results

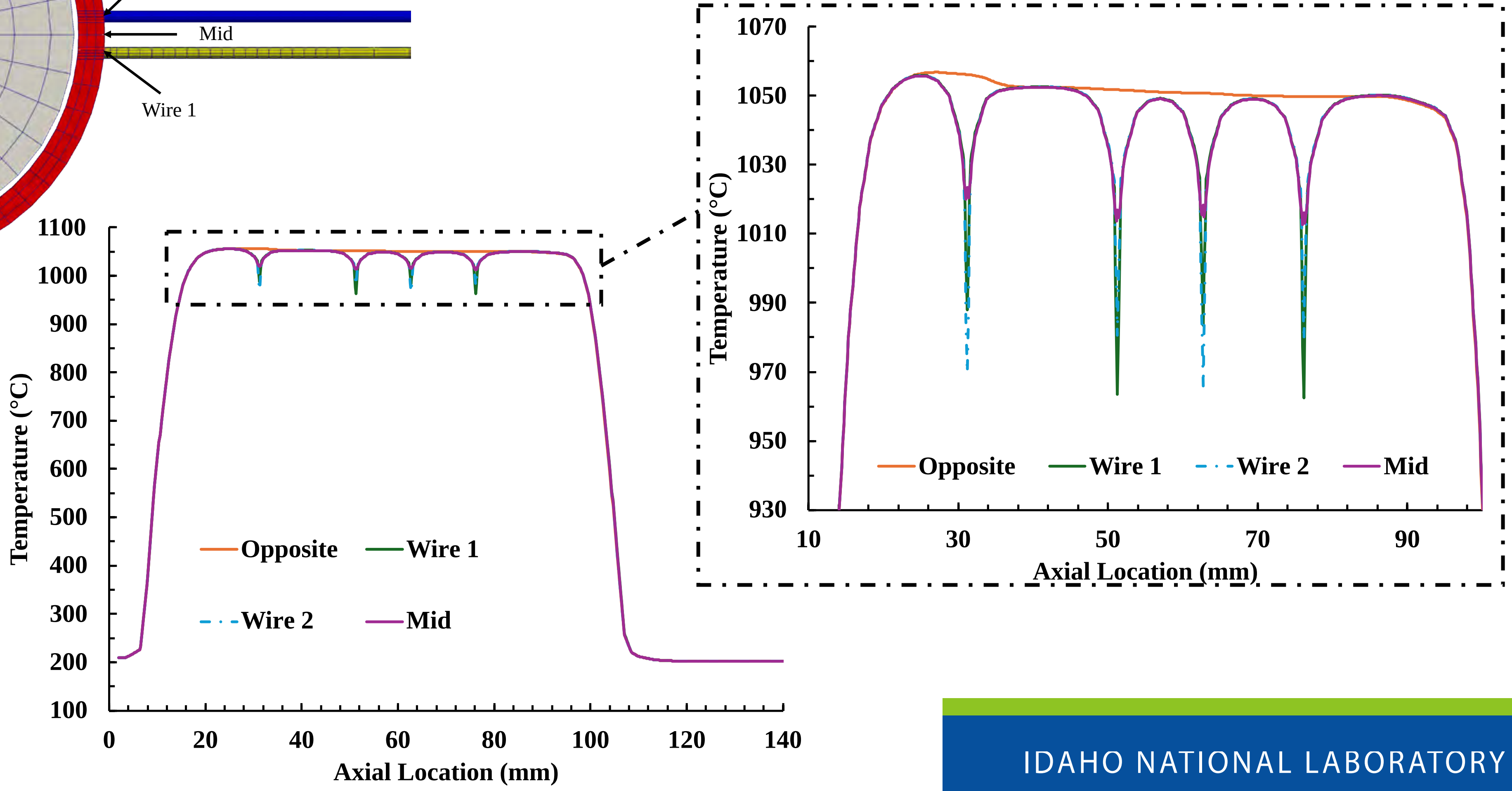
Comparison of ATF-RIA-1-E TC data and BlueCRAB Predictions

- Good agreement up until experimentally observed rewet
  - Simulation predicts slightly higher TC 1 temperatures, as observed in experiment
- After the point of rewet in the experiment, temperature fluctuations in the TC predictions are observed
  - Predicted changes between the film boiling and transition boiling heat transfer regimes along the wire near the cladding
- Although the simulation does not predict rewet until a much later point in time, the **TC temperature fluctuations**, may **provide insights into mechanisms driving rewet around the TCs**
  - Heat transfer regime on TC wire surface moves from film boiling to transition boiling, vapor film collapse/regrowth may break the vapor film on the cladding surface, allowing liquid coolant to move in and rewet the surface



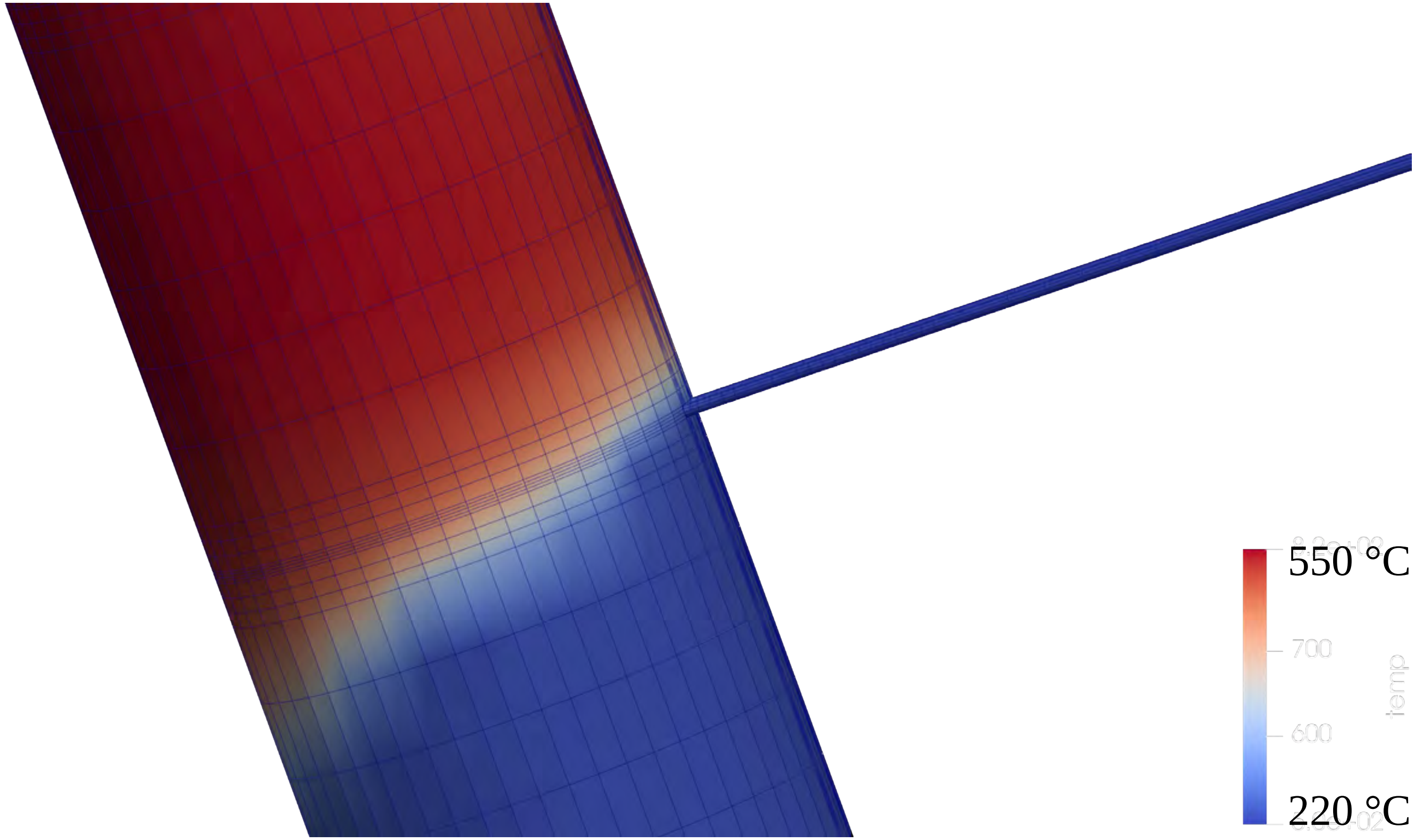
Axial Temperature Distribution 2 seconds into transient





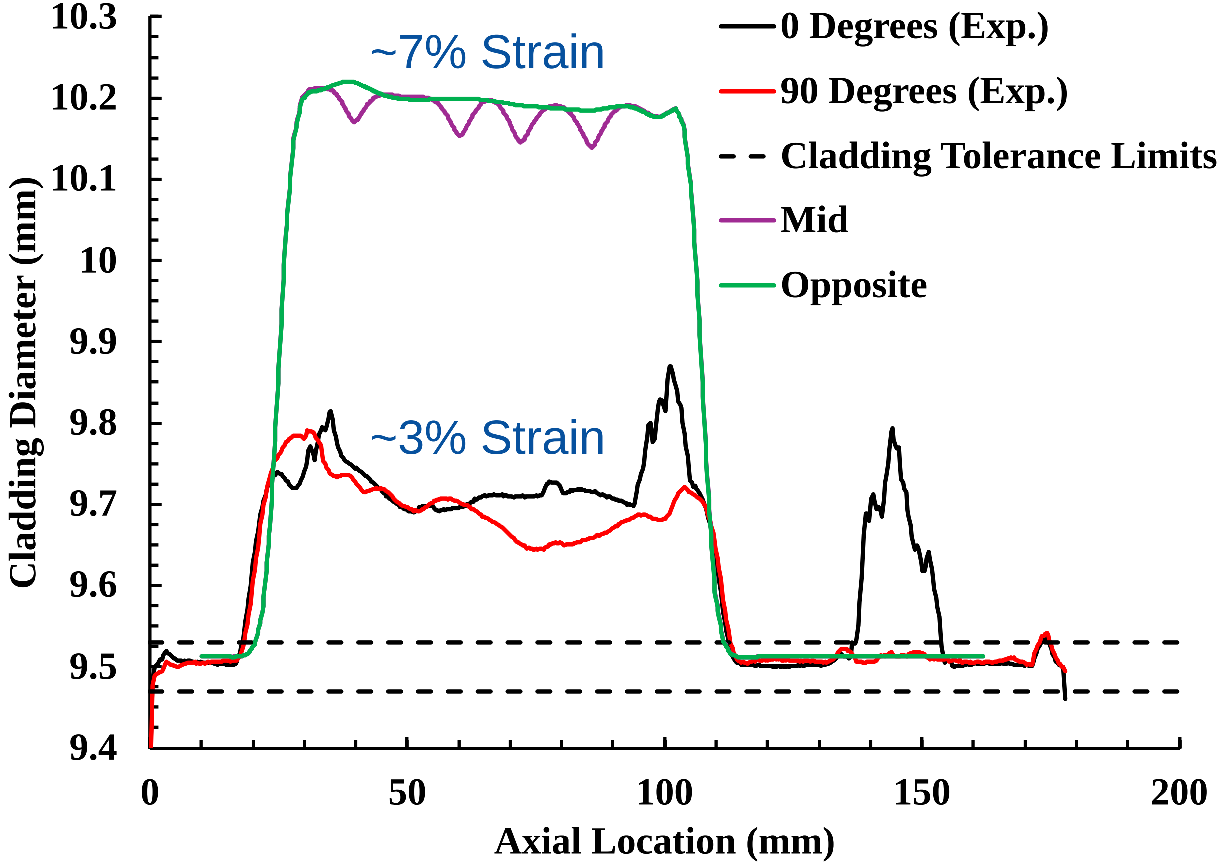
- TC wires ~80 °C less than unperturbed cladding
- Extent of the temperature decrease extends about 5 mm around the TC wire cladding interface

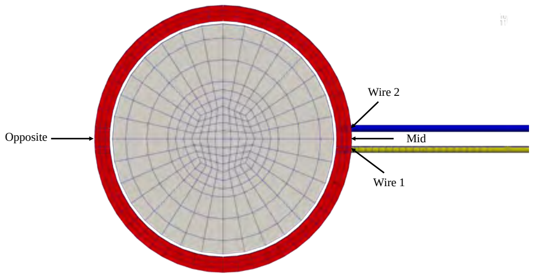
Temperature contour plot showing azimuthal and axial variation in cladding rewet

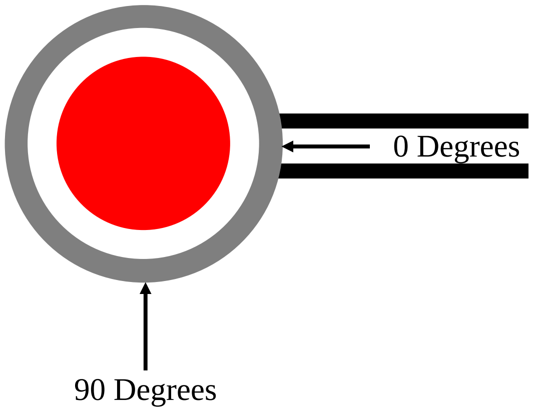


Simulation overpredicts cladding diameter, but shows qualitative agreement

- Simulations predicts cladding diameter slightly peak toward bottom of active fuel
- Predicts lower cladding strain at locations of TC wires due to lower temperature







Conclusions

- The Transient Reactor Test Facility (TREAT) at INL is used to test nuclear fuels and materials in off-normal and accident conditions
  - RELAP5-3D is an integral part in the design and analysis of experiments
- New RELAP5-3D/MOOSE (BISON) coupling capabilities are being utilized to further enhance the design and interpretation of experiments
- Novel methodology for simulating the impact of outer cladding TCs during transient testing of nuclear fuels that does not require prior knowledge of the thermal-hydraulic conditions
  - Leverages the thermal-hydraulic capabilities of RELAP5-3D and the BISON fuel performance code through BlueCRAB
- Methodology applied to the ATF-RIA-1-E experiment, and the simulation results are then compared to the experimental data
  - Simulation results show good agreement with the TC measurements and indicate that the local cladding temperature is approximately 80 ºC lower than the cladding temperature far from the TC location
  - Predictions indicate temperatures peaked toward the bottom end of the rodlet, which agrees with the TC data

Acknowledgments

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