INL/MIS-23-75662-Revision-0

Deployment of additional LWR test loops and BWR / Ramp testing in ATR I-Loop

December 2023

Nate Oldham

Changing the World's Energy Future

Idaho National Laboratory

INL/MIS-23-75662-Revision-0

Deployment of additional LWR test loops and BWR / Ramp testing in ATR I-Loop

Nate Oldham

December 2023

Idaho National Laboratory
Idaho Falls, Idaho 83415

http://www.inl.gov

Prepared for the
U.S. Department of Energy
Under DOE Idaho Operations Office
Contract DE-AC07-05ID14517

2023 December 05

Nate Oldham*
Irradiation Test Design Engineer

Deployment of additional LWR test loops and BWR / Ramp testing in ATR – I-Loop

* nate.oldham@inl.gov

Contents

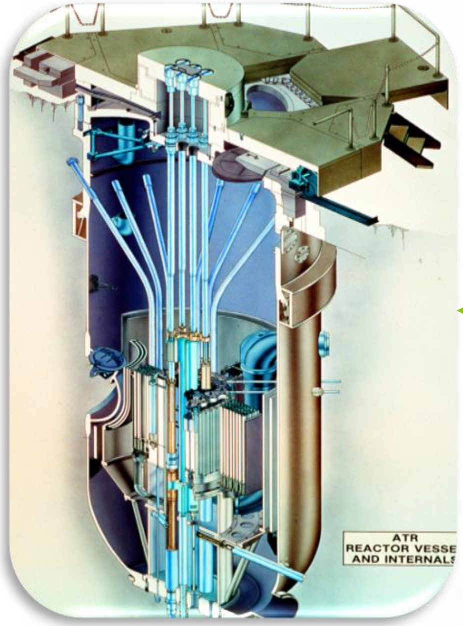
- Advanced Test Reactor (ATR) Overview
  - Location
  - Core / Flow
  - How ATR Works
- I-Loop Facility
  - ATR Water Loop - Typical
  - Expanding Water Loop Capacity in ATR - I-Loop Origin
  - I-Loop Overall Objectives
  - I-Loop Tube – In-Vessel Equipment
  - I-Loop – X-Core Equipment
  - I-Loop Typical Experiment
- BWR Testing
  - I-Loop Tube Flow
  - BWR Test Train
  - BWR Tier / Holder
  - BWR Neutronics (Enrichment and Heat Rate)
  - BWR Thermal Hydraulics
- Ramp Testing
  - Background “What is PCI Ramp Testing”
  - Summary of PCI
  - Methods for Manipulating Specimen Power
  - PCI Ramp Test Design
  - He-3 Neutronics
  - Proposed Ramps

ATR Location

- Located in South East Idaho on the INL Site
- Facility is the Advanced Test Reactor Complex
- Constructed in 1960’s
- Operational in 1967

ATR Complex

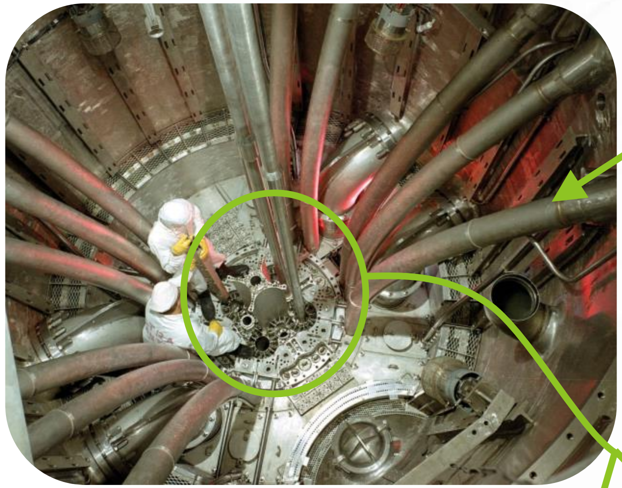
ATR Main Floor



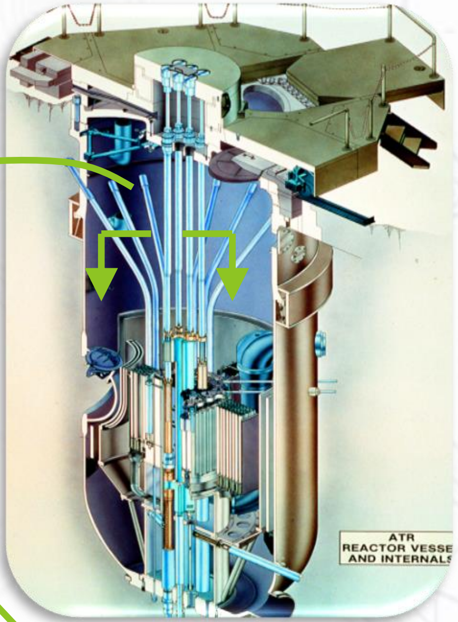
ATR Vessel

ATR Core / Flow

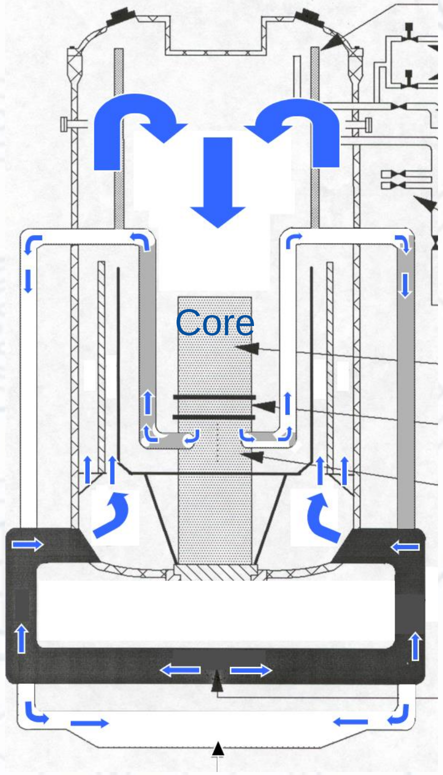
- Advanced Test Reactor (ATR) is a materials and fuels test reactor
  - Pressurized light water reactor
    - 51ºC (125ºF)
    - 2.4 MPa (355 psig)
  - Down flow through core
  - Beryllium reflector around core
  - 250 MW thermal power
  - ~1.2 m dia x 1.2 m high core
  - ~60 day cycles
  - ~200 EFPDs (days) per year
  - Clover leaf fuel design



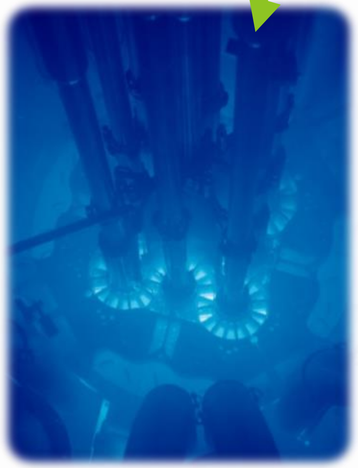
ATR Core Top View (1965)



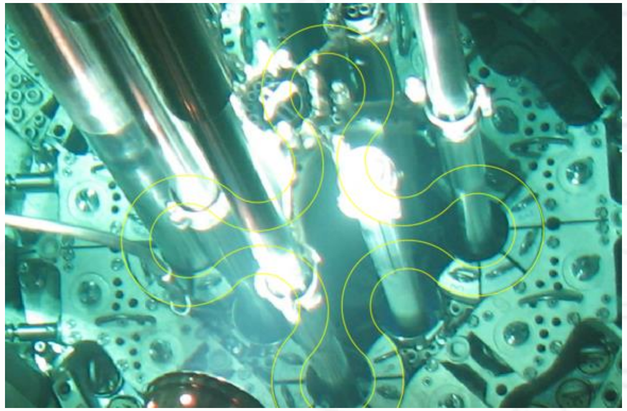
ATR Vessel



ATR Primary Coolant



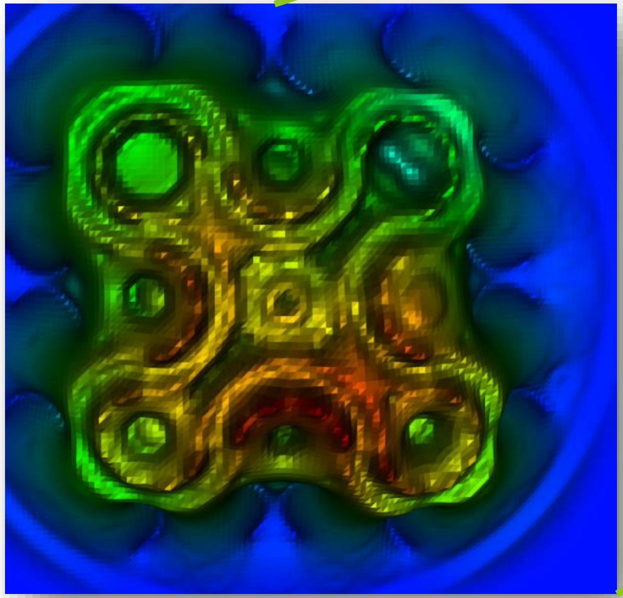
ATR Operating



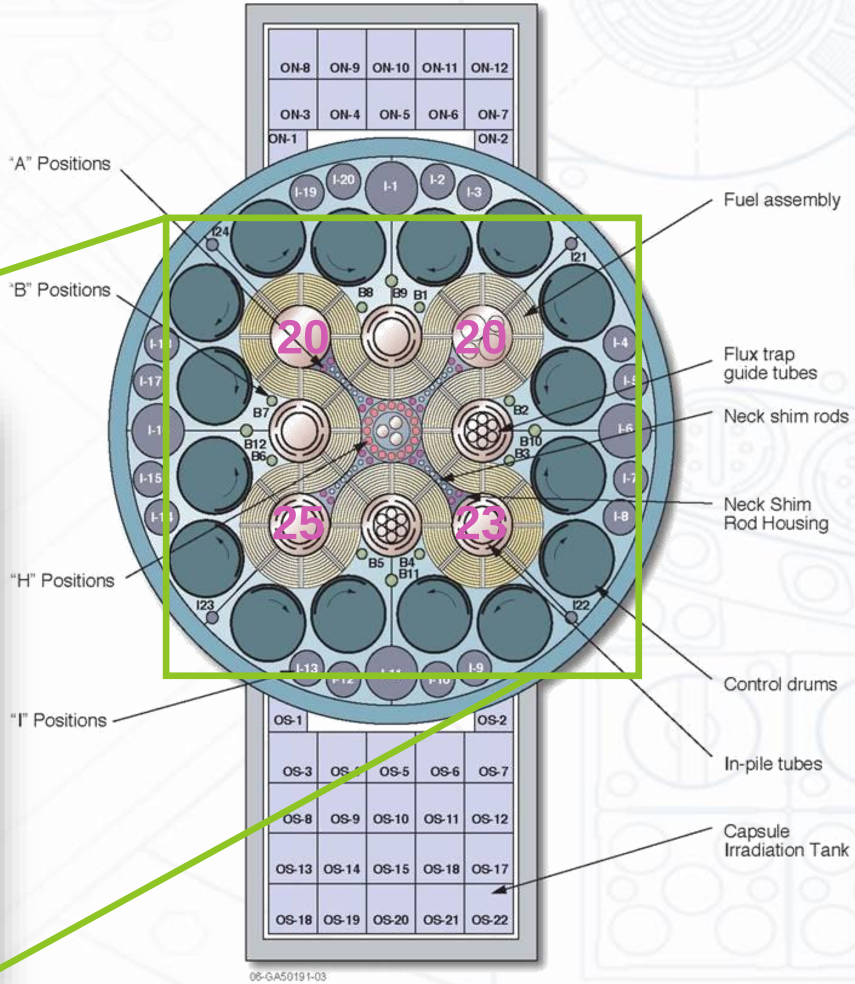
ATR Outage

How ATR Works

- Cloverleaf fuel pattern
  - 40 individual fuel elements
  - Fuel pattern snakes around the nine flux trap positions
- Corner flux trap positions are called ‘lobes’
  - Each lobe is a mini reactor
  - Lobe power is controlled by four control drums
  - Control drums have a hafnium plate wrapped 120° (neutron absorber)
  - Control drums rotate to move the hafnium exposure to the core



Total Flux Plot
By Josh Peterson-Droogh



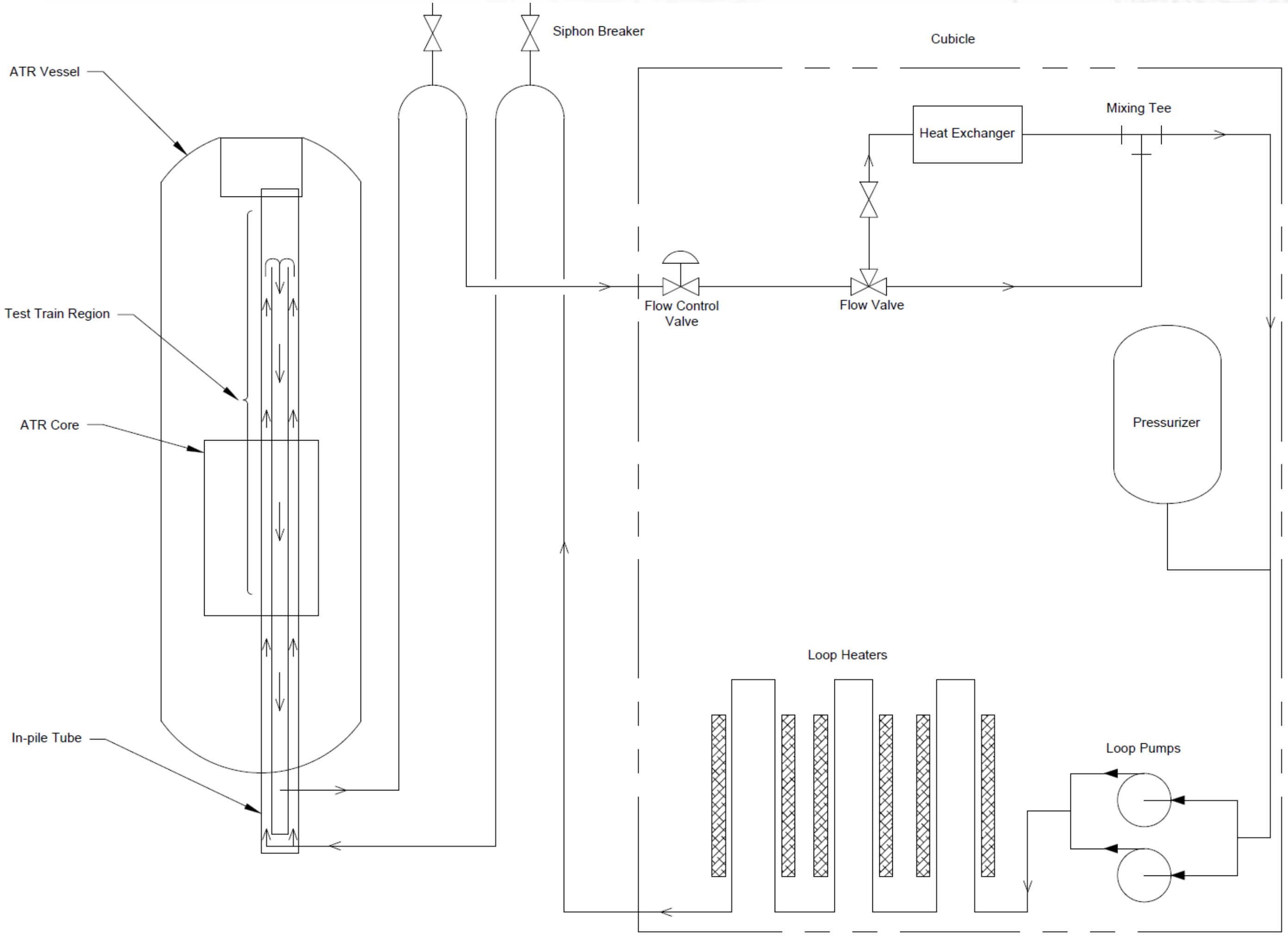
ATR Core Cross Section

CYCLE:	171B-1Xe		
*Form has approved IWAP and F&OR.			
BOLD, RED TEXT = TO BE EVALUATED	EFPDs to CIC		
Red text = change made	5/3/2023	Outage Scope Freeze Date	
	Outage	Press Up	EFPD
Scheduled Duration (days):	2	1	59
Start:	7/26/2023	7/28/2023	7/29/2023
Start Day of the Week:	Wednesday	Friday	Saturday
Finish:	7/28/2023	7/29/2023	9/26/2023
NE Lobe Power, MW			20.0 +/-1.0
SE Lobe Power, MW			23.0 +/-3.0
SW Lobe Power, MW			25.0 +1.0/-5.0
NW Lobe Power, MW			20.0 +/-3.0

ATR Integrated Strategic Operational Plans (ISOP)

Contents

ATR Water Loops - Typical



ATR Typical Loop Flow Diagram

Function:

- Used to conduct irradiations in a controlled environment
- Operates independently from ATR primary coolant
- Capable of higher pressures, temperatures, and water chemistry

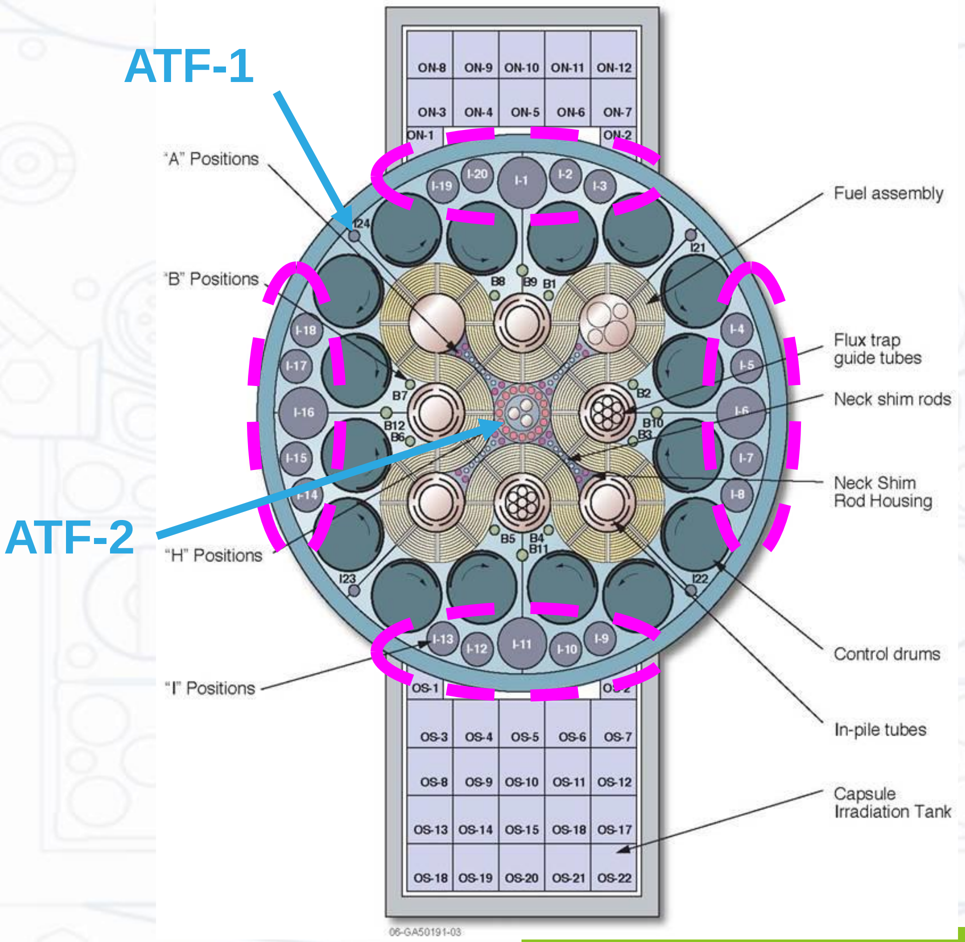
Components:

- In-vessel (in-pile) equipment
  - In-Pile Tube or I-Loop Tube – device that passes through the reactor core
  - Houses an experiment specimen
  - Coolant flows upward in an annulus then turns around to flow downward around the test train
- **X-core (out-of-pile) equipment:**
  - Mostly located in a basement cubicle
  - Consists of piping, valves, heat exchangers, heaters, and pumps

Expanding Water Loop Capacity in ATR

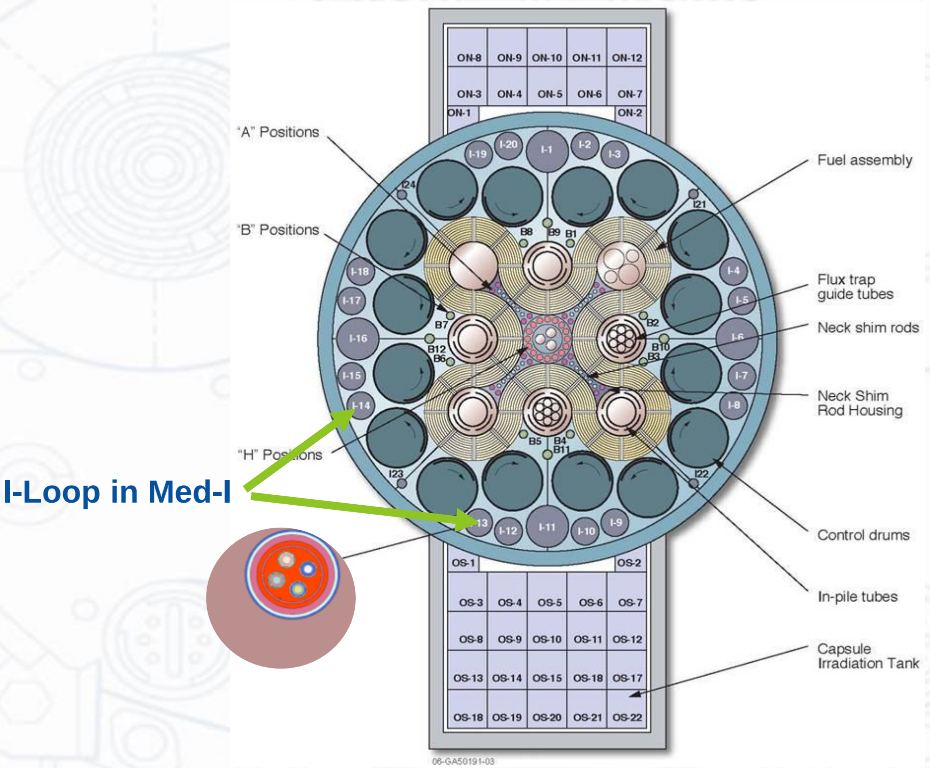
- U.S. facilities, ATR, HFIR, MITR, currently irradiate materials and other science-focused specimens
- Halden Reactor in Norway also performed irradiations
- Pressurized water loop in center flux trap with ATF-2 experiment – already highly prescribed
- Theoretically possible to add an additional flux trap loop
  - But oversubscription, nuclear interactions with boiling effects, and current lack of diverse capabilities make option less desirable
- Except for the relatively underutilized reflector positions (“I” positions)
  - ATR reflector flux similar to Halden central core
  - Possibility of two-phase flow conditions (BWR) and local power control manipulation

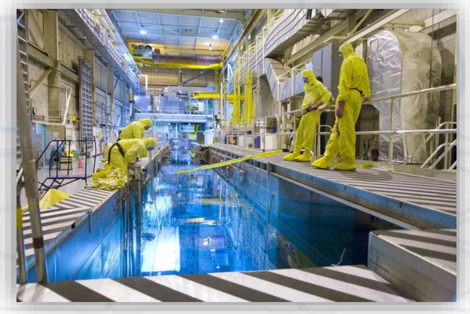
Advanced Test Reactor



I-Loop Overall Objectives

- Test Section
  - Located in Med-I position
  - Experiment holder 2.67 cm x 2.67 cm
  - Experiment section ~120 cm long
  - Can accommodate 2 x 2 array of tensile specimens or fuel rods
- Coolant
  - Up-flow (PWR and BWR)
  - Flow ~1.5 liters/second (25 gpm)
  - Temperatures up to 320 °C (600 °F)
  - Pressures up to 15.5 MPa (2300 psig)
  - Customizable water chemistry e.g. borated PWR coolant
- Testing Options
  - Instrumentation for temperature, pressure, elongation, swelling, and creep measurement
  - Ramp testing - designed with helium-3 screen for flux manipulation independent of reactor
  - BWR testing – no positive reactivity to two-phase flow ~$0.01
  - Test trains extraction permits transport to ATR's sizable storage pool, dry transfer cubicle hot cell, or hot cell for examination





ATR Storage Pool

BWR Coolant / Chemistry

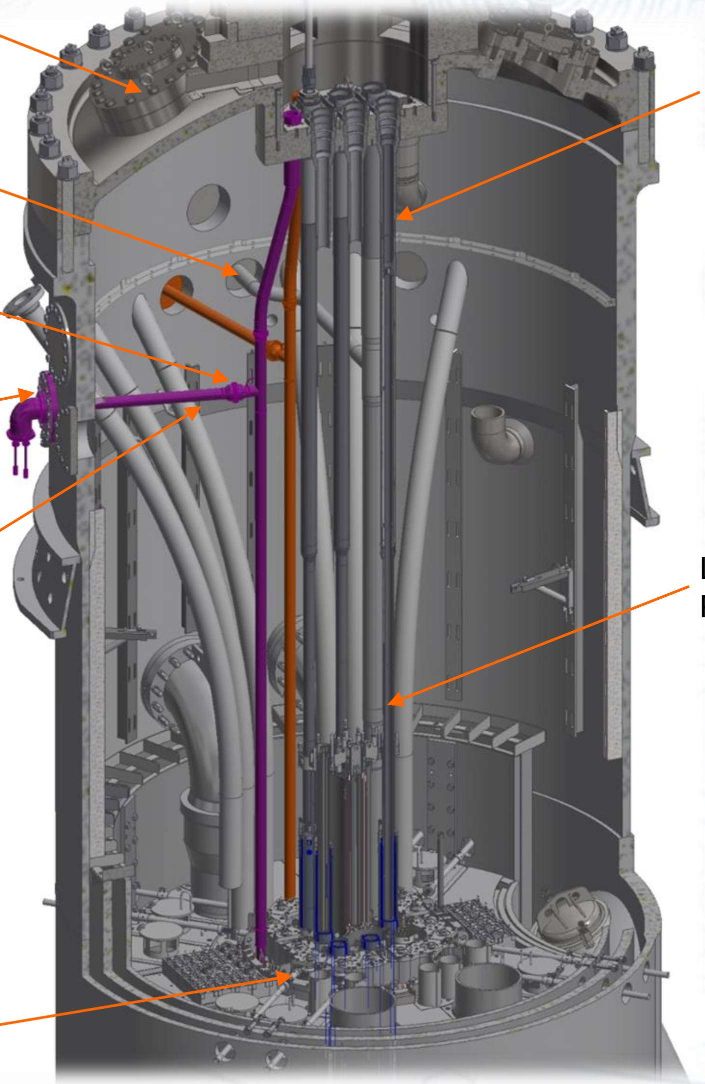
The I-Loop BWR system shall be capable of circulating prototypic BWR coolant conditions as follows:

- Liquid water from 250°C (482°F) to 288°C (550.4°F) and steam at 288°C (550.4°F) at a pressure of 7.1 MPa (1,030 psig)
- Subcooled liquid water conditions of a maximum temperature of 260°C (500°F) at the experiment inlet portion of the I-Loop tube.

The I-Loop BWR system coolant shall be capable of providing prototypic BWR coolant chemistry including the following conditions:

- pH 5.6-8.6 @ 25°C (77°F)
- excess Hydrogen inventory to control Oxygen
- < 0.3 ppm Iron
- 10-20 ppb Zinc
- <100 Chloride (ppb)
- <200 dissolved Oxygen (ppb)

I-Loop Tubes



I-Loop Tubes in ATR

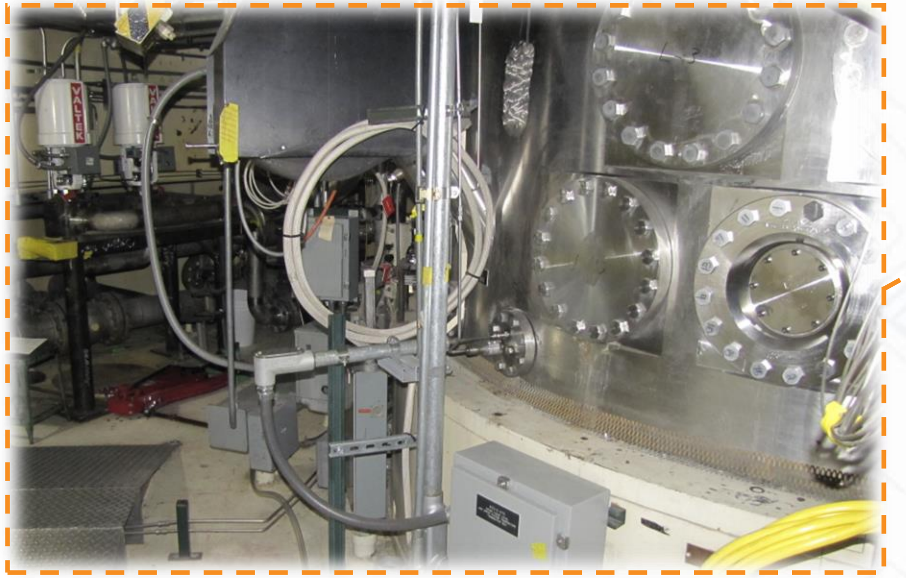
Upper Portion

- Material is stainless steel
- Offset tube
- Allows for test train from the reactor top

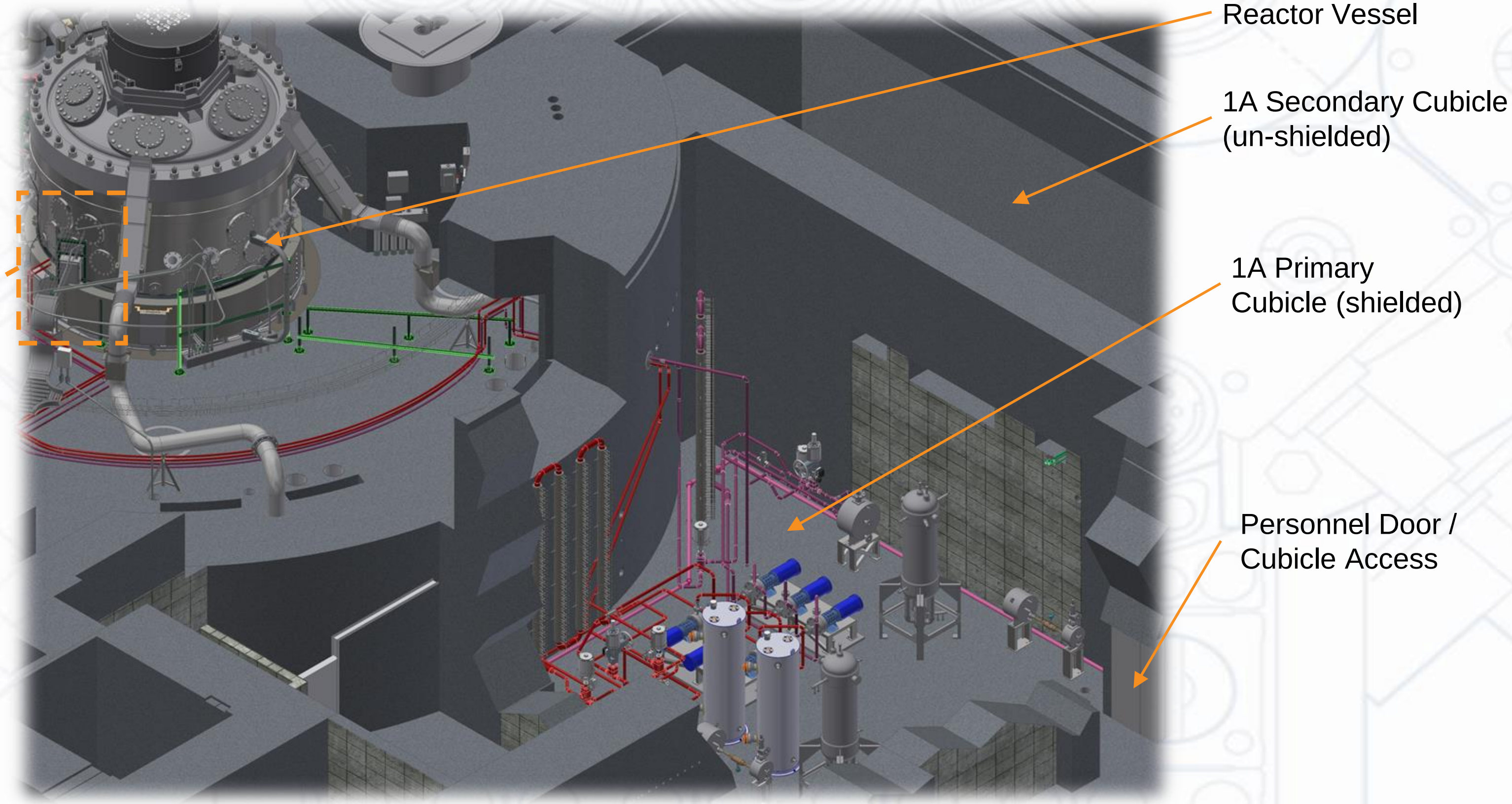
Lower Portion

- Material is Zr-2.5Nb (strongest available nuclear grade zirconium alloy) for neutron economy
- Annular flow
- Bolted hub connections
- ~1.56” (40 mm) inside diameter test section
- Installed in Medium-I position

Loop X-Core System



Nozzle Trench



1A X-Core Equipment
(equipment layout for two loops)

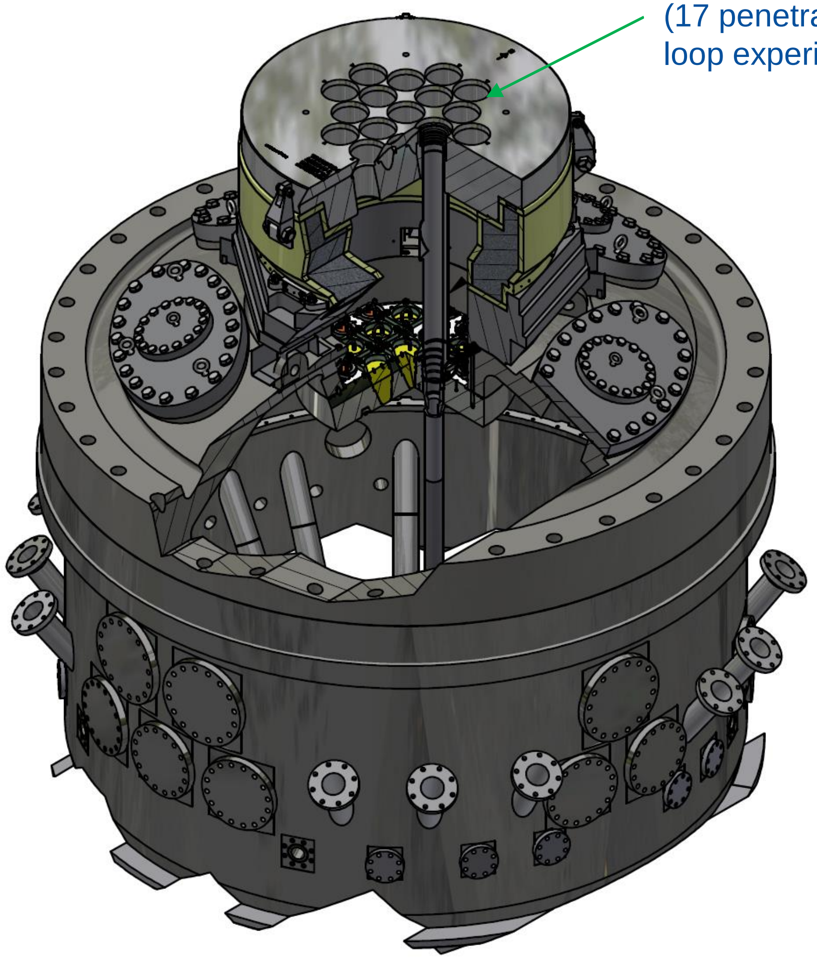
- **Loop X-Core System**
  - Piping System
  - Pumps
  - Pressurizers
  - Ion Exchanges
  - Sampling

ATR Facility Modifications

Transfer Shield Plate
(nine penetrations for loop experiments)

Shield Cylinder
Aka Donut

Current Reactor Top Shielding for Experiment Handling

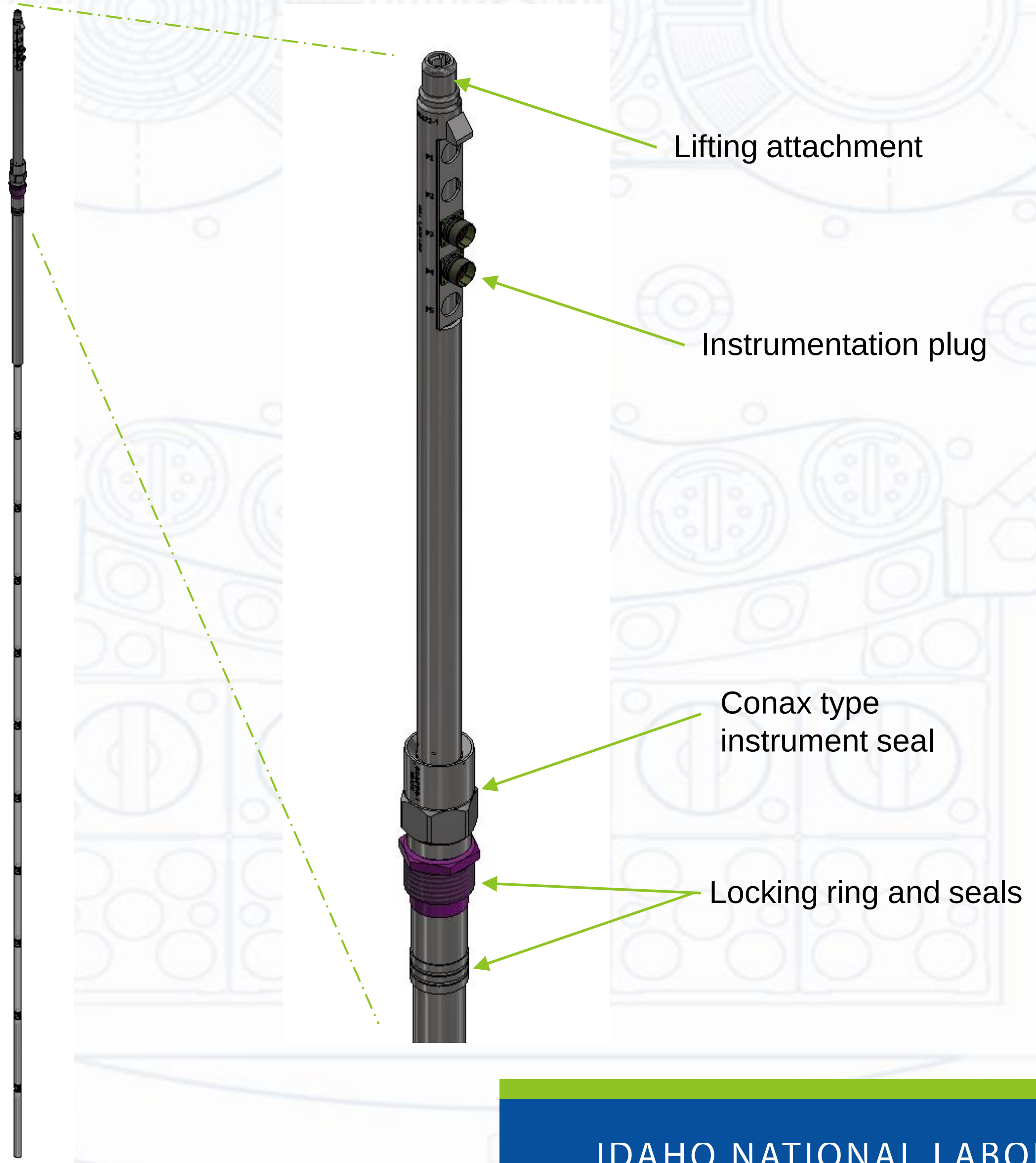


Transfer Shield Plate
(17 penetrations for loop experiments)

New Transfer Shield Plate Design

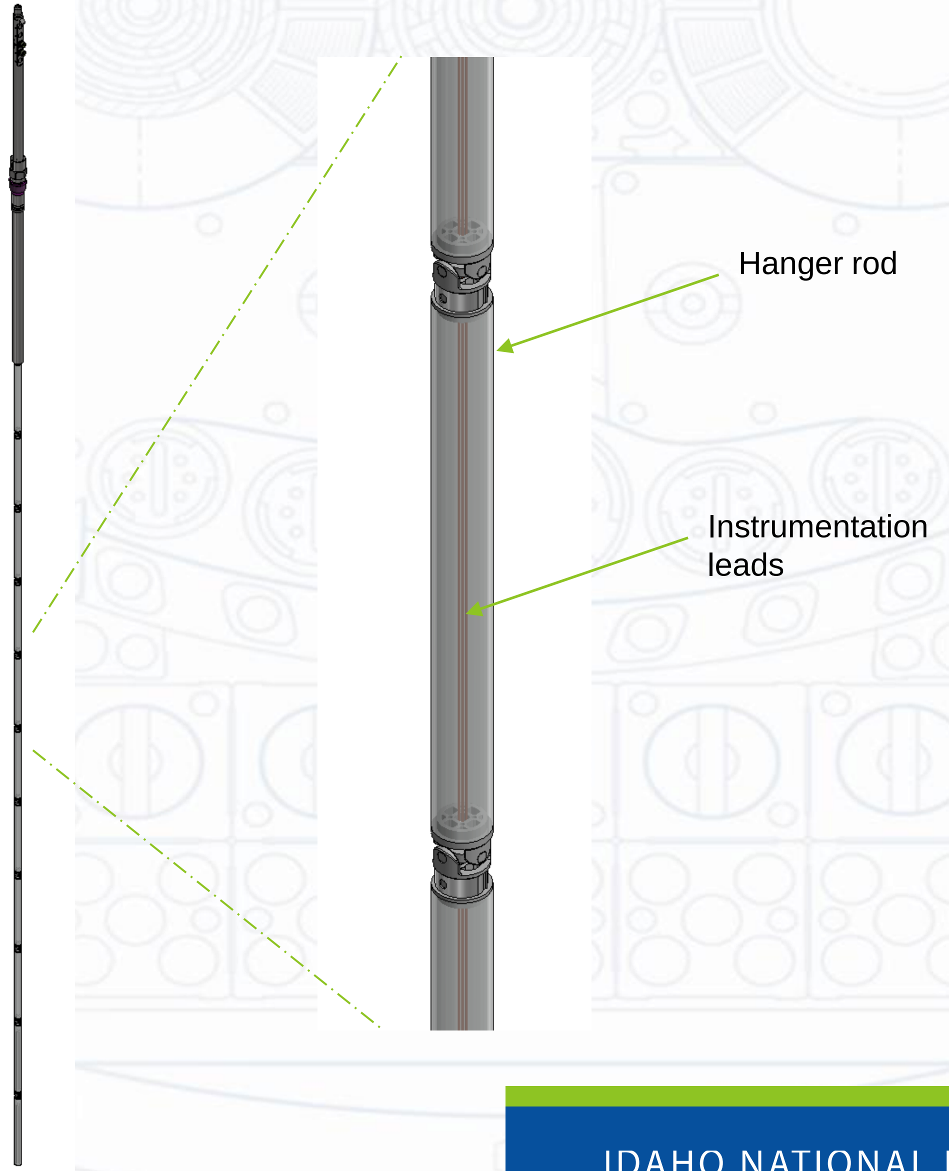
I-Loop Experiment Typical Design

- Experiment test train
  - ~20 feet long
  - Lifting attachment
  - Instrumentation plug
  - Conax type instrument seal
  - Locking ring and seals to I-Loop Tube
  - Hanger rods housing leads
  - Fueled test sections



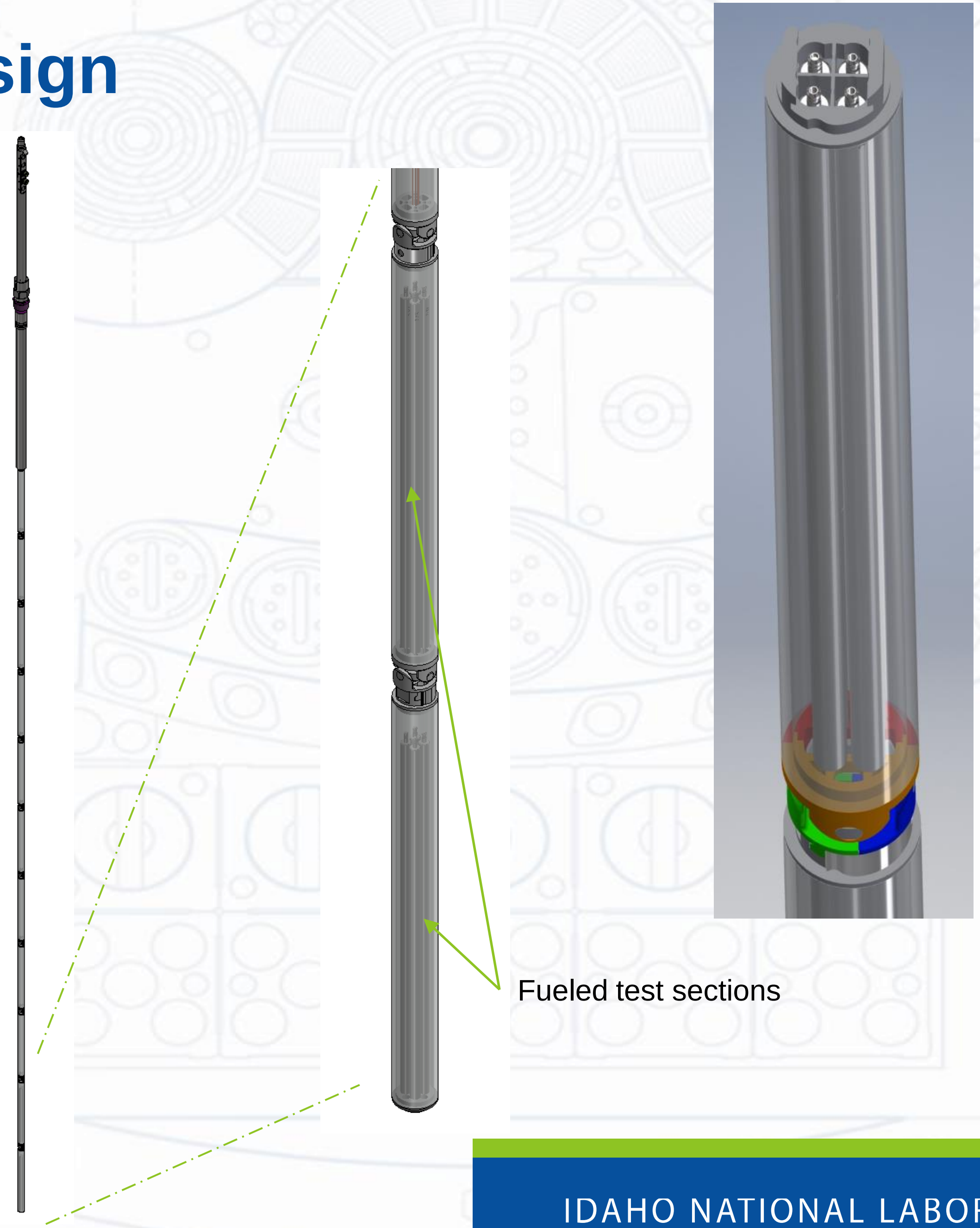
I-Loop Experiment Typical Design

- Experiment test train
  - ~20 feet long
  - Lifting attachment
  - Instrumentation plug
  - Conax type instrument seal
  - Locking ring and seals to I-Loop Tube
  - Hanger rods housing leads
  - Fueled test sections



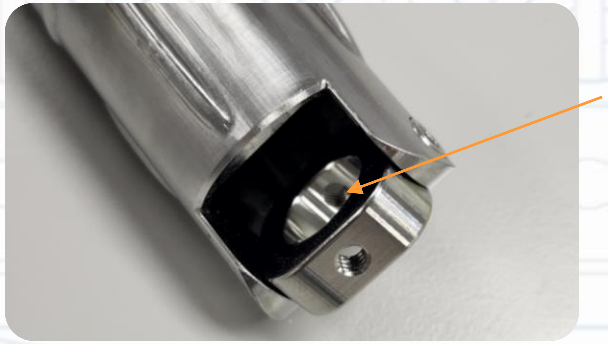
I-Loop Experiment Typical Design

- Experiment test train
  - ~20 feet long
  - Lifting attachment
  - Instrumentation plug
  - Conax type instrument seal
  - Locking ring and seals to I-Loop Tube
  - Hanger rods housing leads
  - Fueled test sections

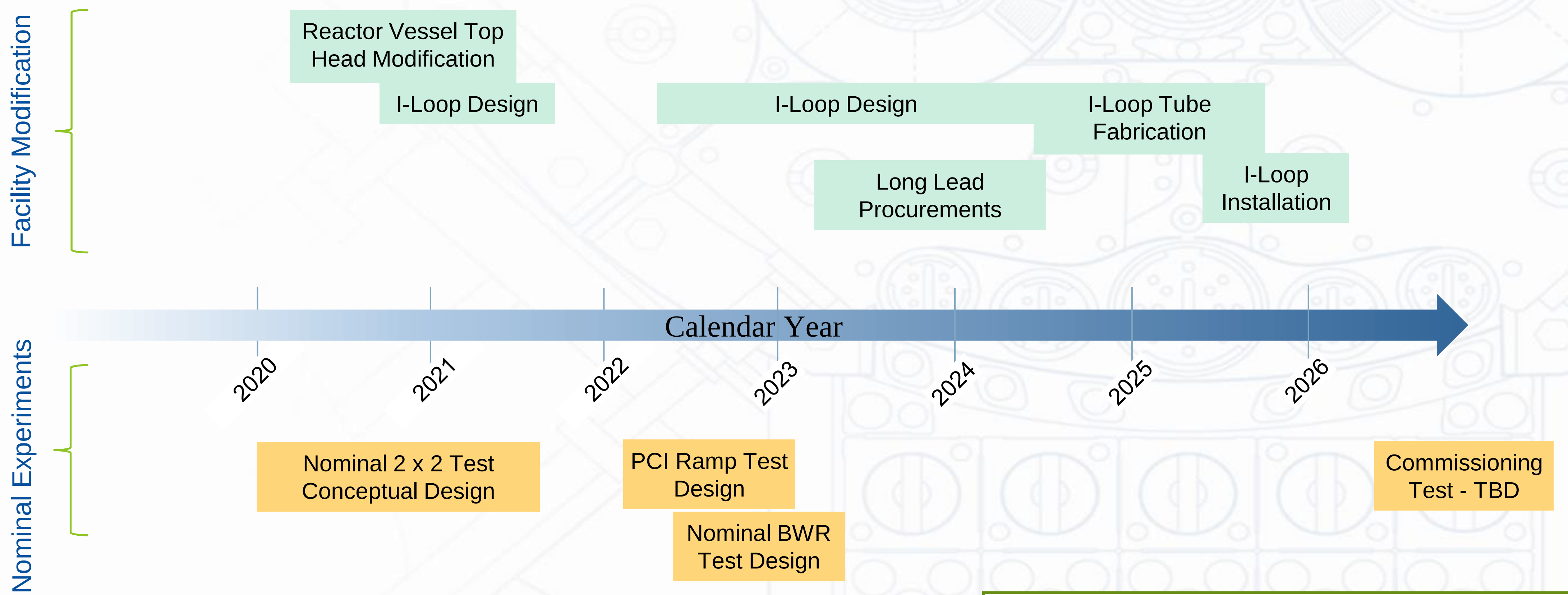


I-Loop Test Train Mockup

- Demonstration of the mechanical design of the I-Loop Tube and experiment test train.



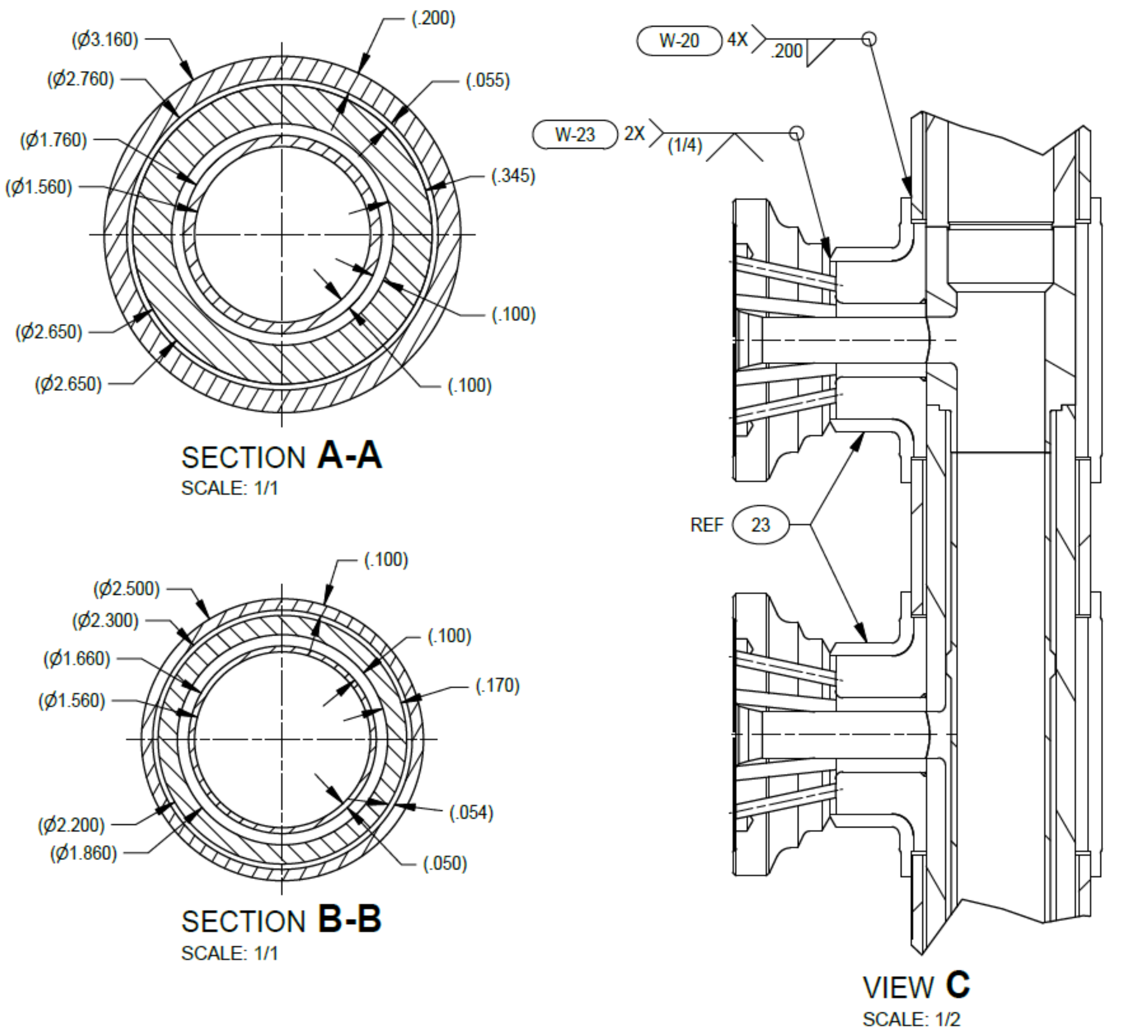
I-Loop Project Schedule



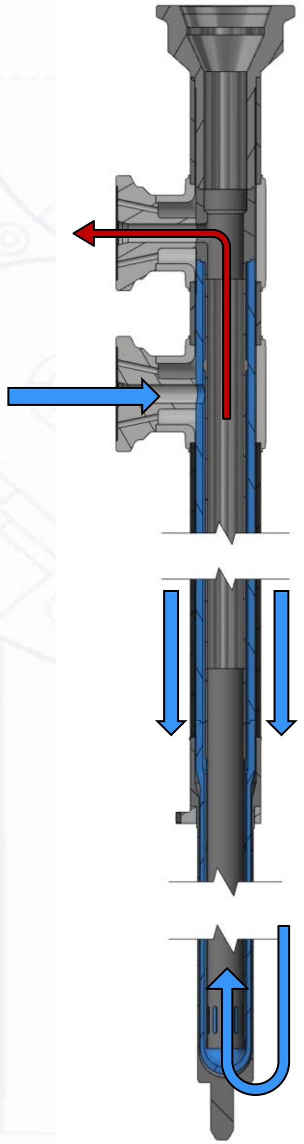
- Estimated completion date of March 2026
- Commissioning testing June 2026

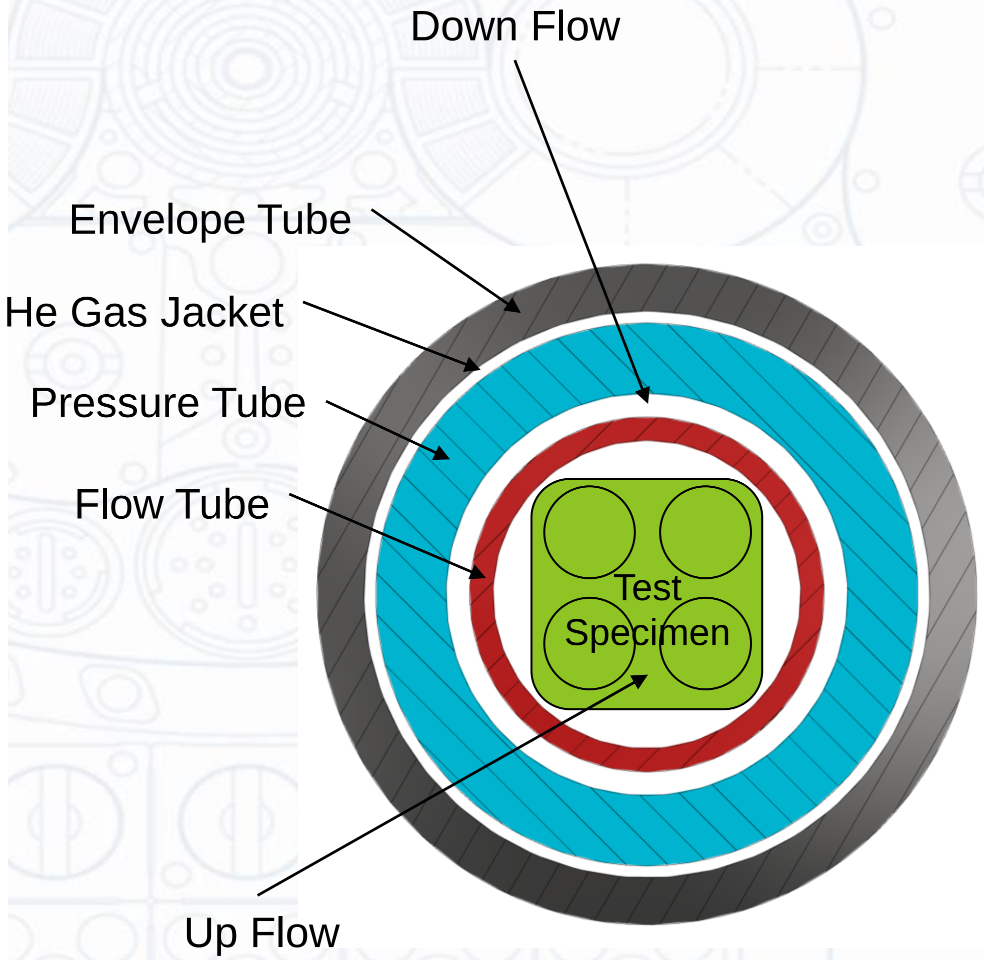
Contents

ILT Flow



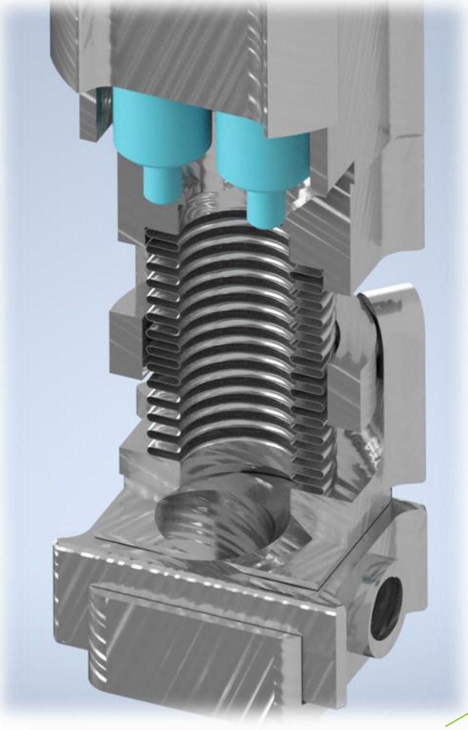
(Ø3.160)
(Ø2.760)
(Ø1.760)
(Ø1.560)
(Ø2.650)
(Ø2.650)
(.200)
(.055)
(.345)
(.100)
(.100)
SECTION A-A
SCALE: 1/1
(Ø2.500)
(Ø2.300)
(Ø1.660)
(Ø1.560)
(Ø2.200)
(Ø1.860)
(.100)
(.100)
(.170)
(.054)
(.050)
SECTION B-B
SCALE: 1/1
W-20
4X
.200
W-23
2X
(1/4)
REF
23
VIEW C
SCALE: 1/2



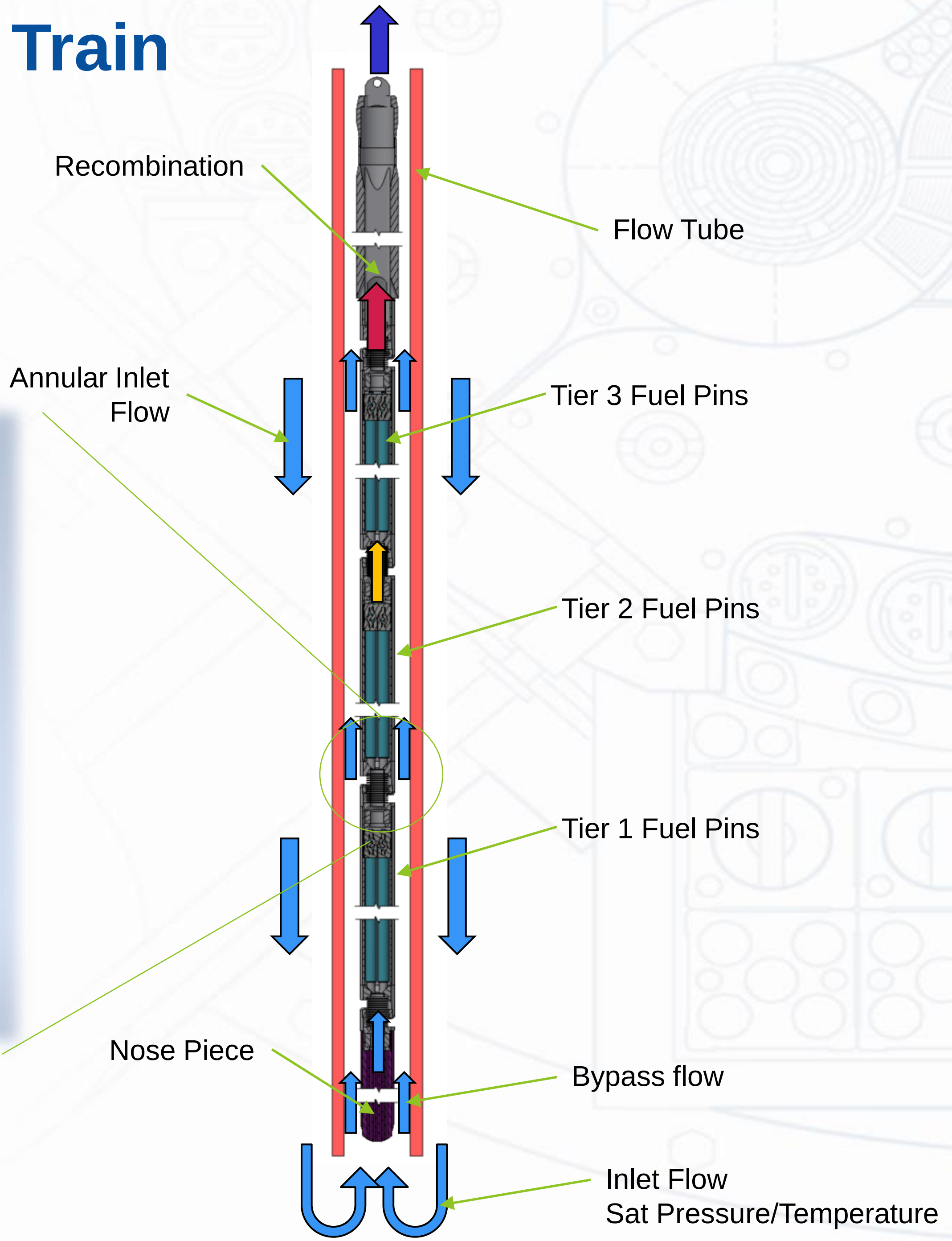


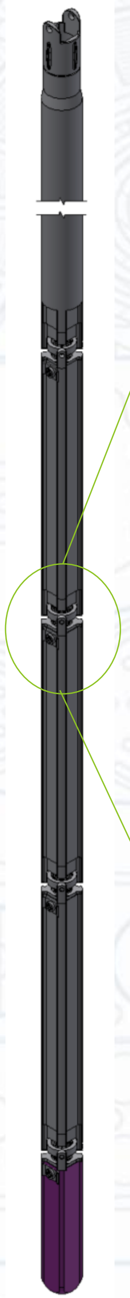
Down Flow
Envelope Tube
He Gas Jacket
Pressure Tube
Flow Tube
Test Specimen
Up Flow

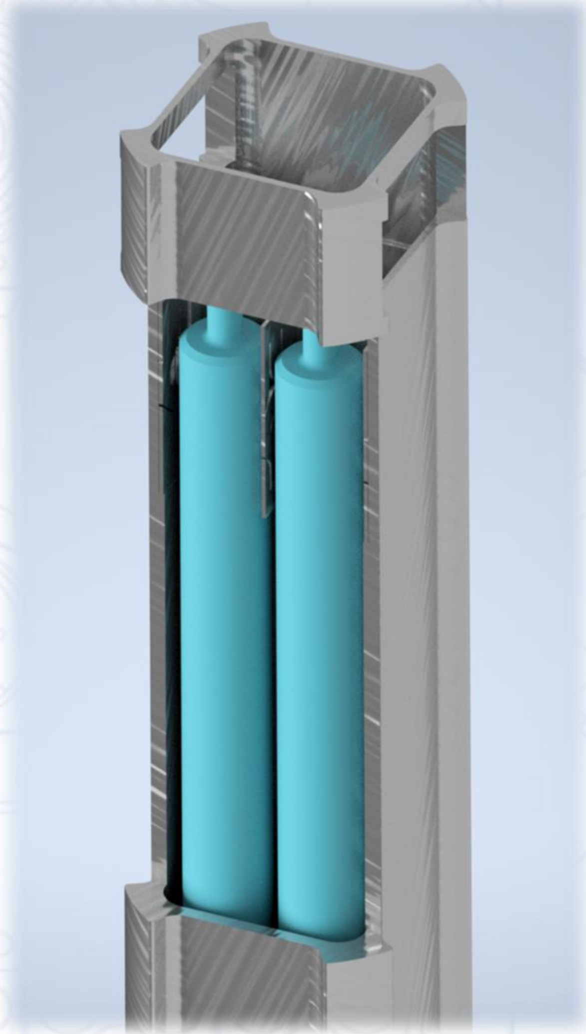
BWR Test Train



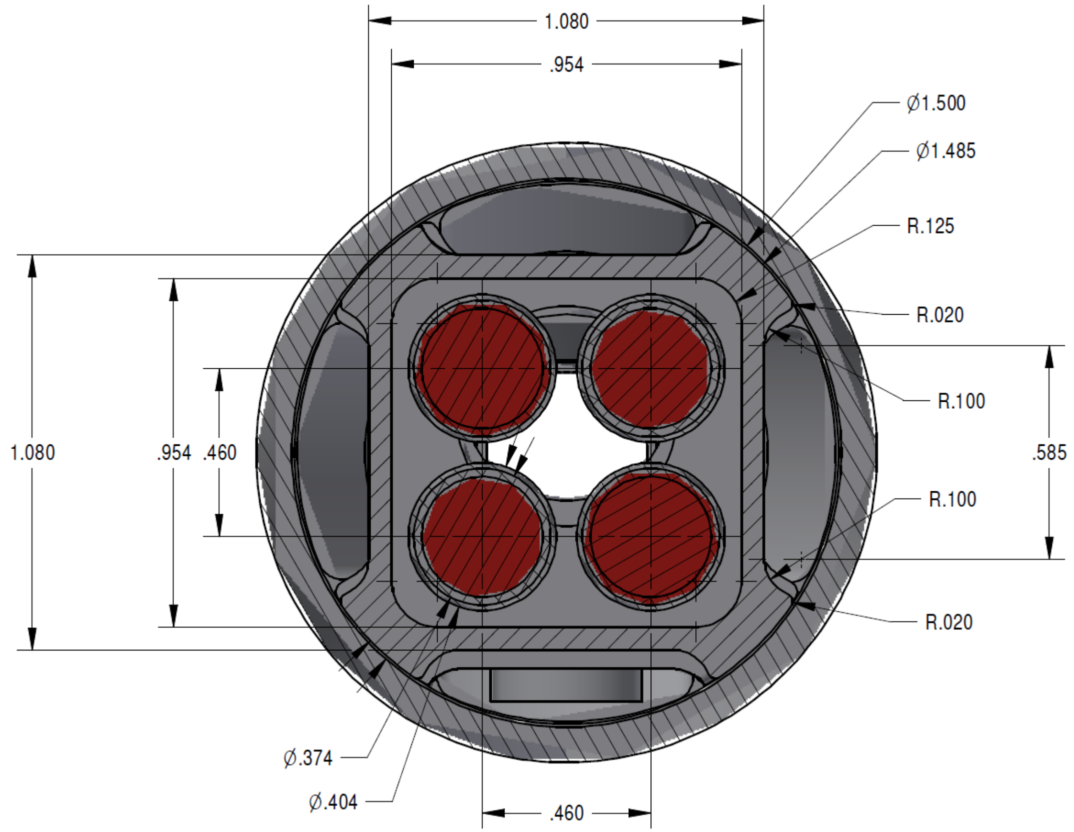
Linkage (bellows)

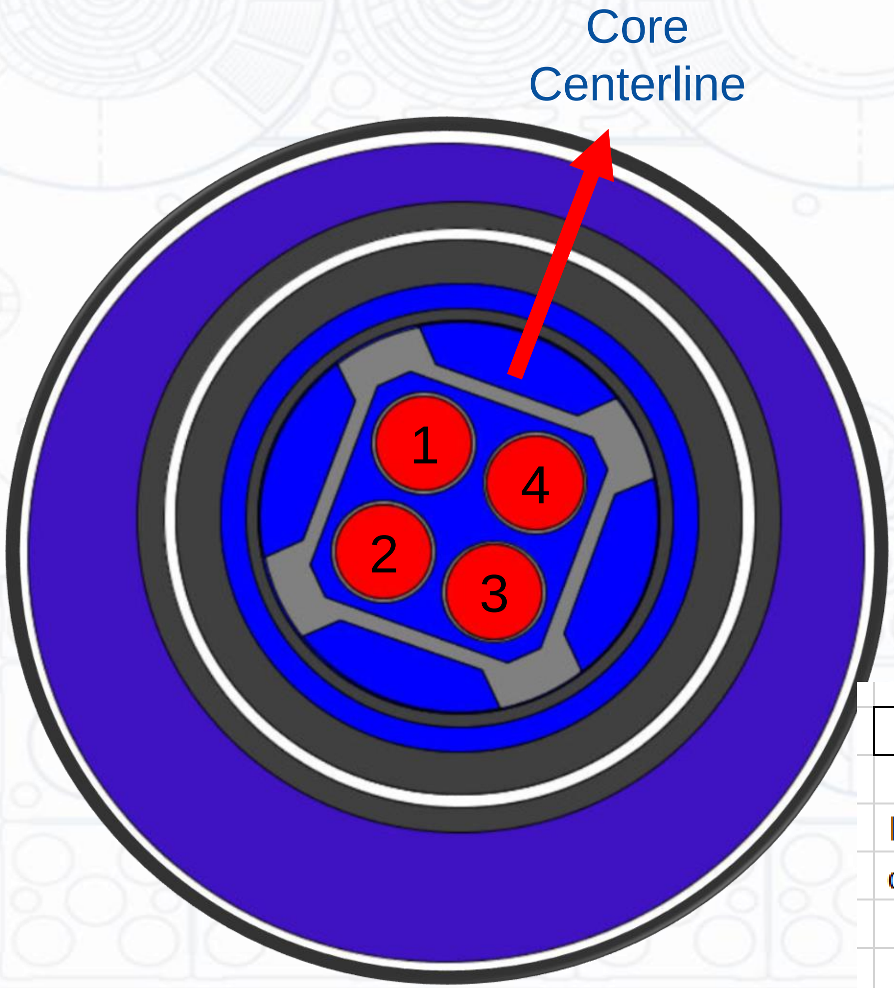






BWR Test Tier / Holder

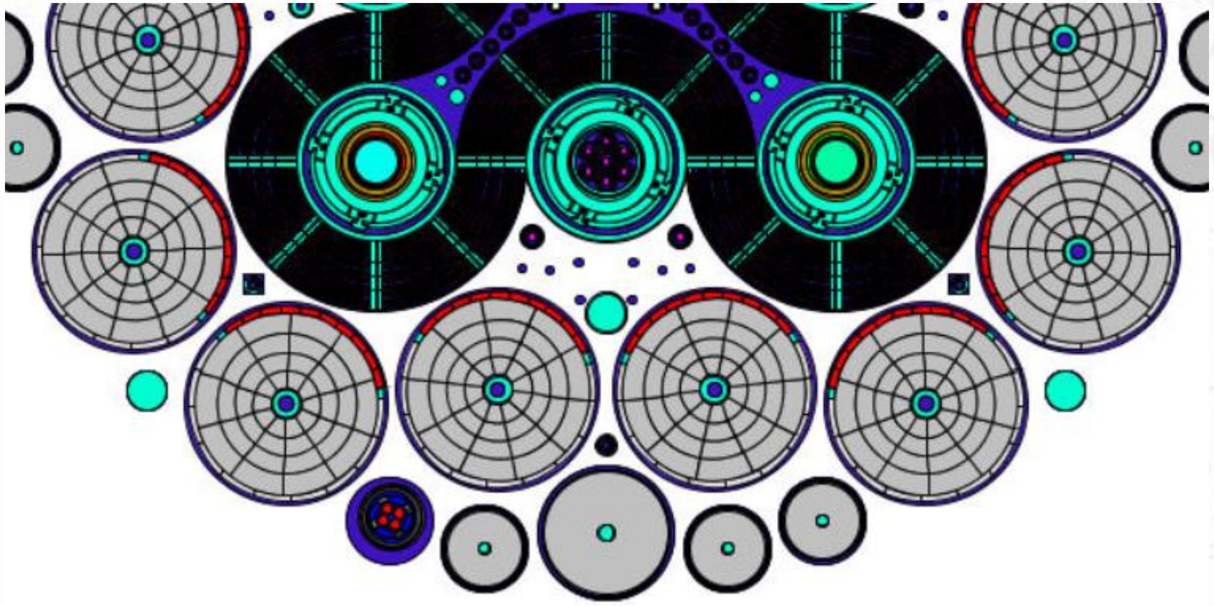


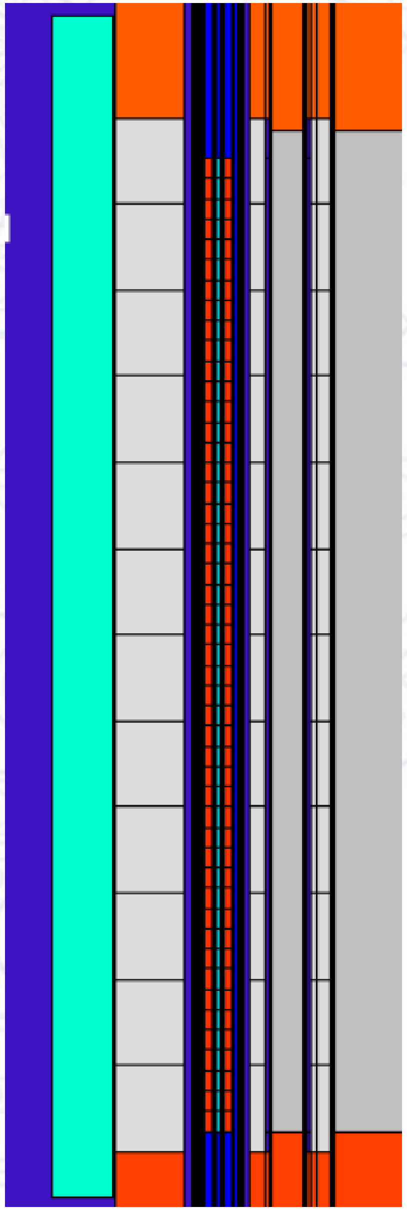


Color Legend	
blue	Water
light grey	Zr-4
dark grey	Zr-2.5Nb
red	UO2
white	He-4

Neutronics model of BWR Test Train

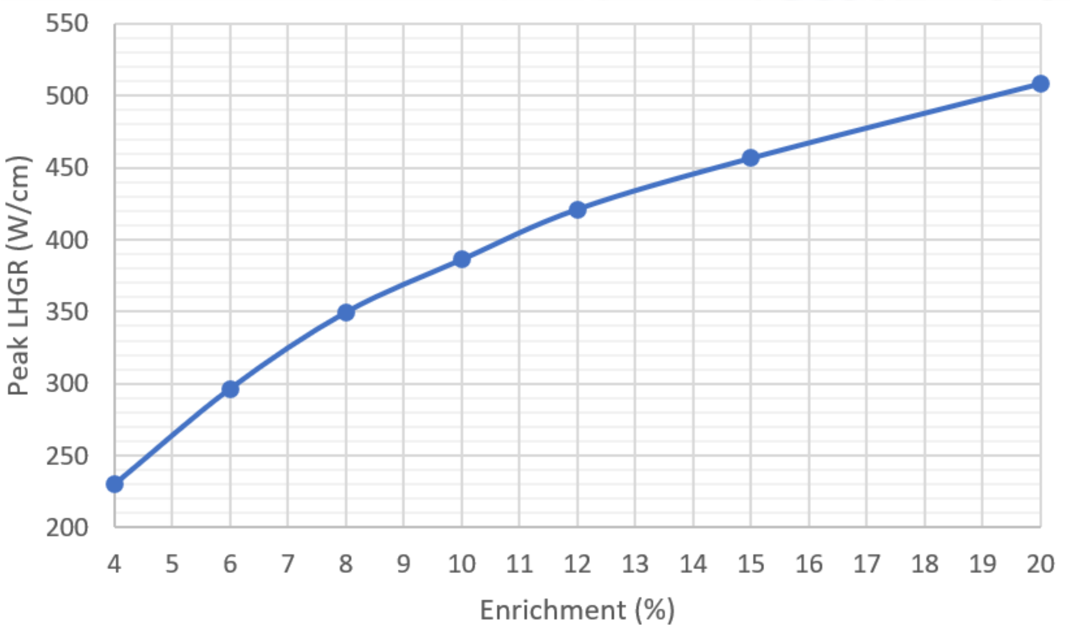
- I-13 position
- 1” axial segments from top to bottom of ATR core
- One continuous tier of 4x4 fuel array
- Cycle 167A loading, but use BOC 169A shim positions

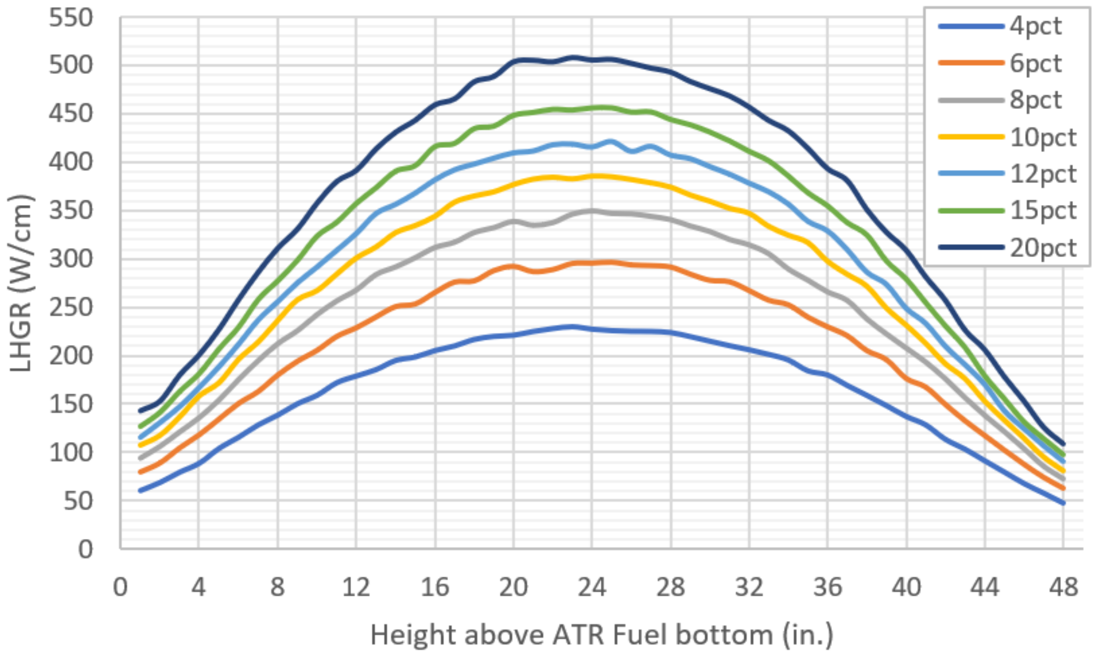




I-Loop BWR – Enrichment Sweep – 23 MW Lobe

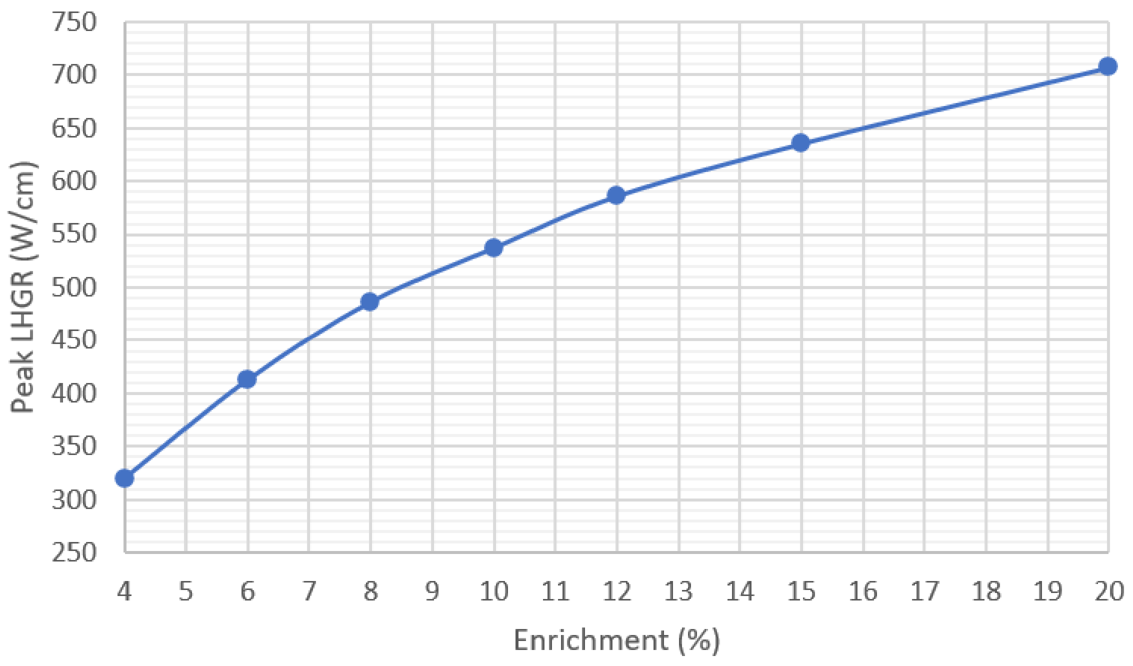
- Single pin, (no boiling)
- Uranium enrichments range from 4-20% U235
- Data scaled to 23 MW lobe power (Standard ATR Cycle)

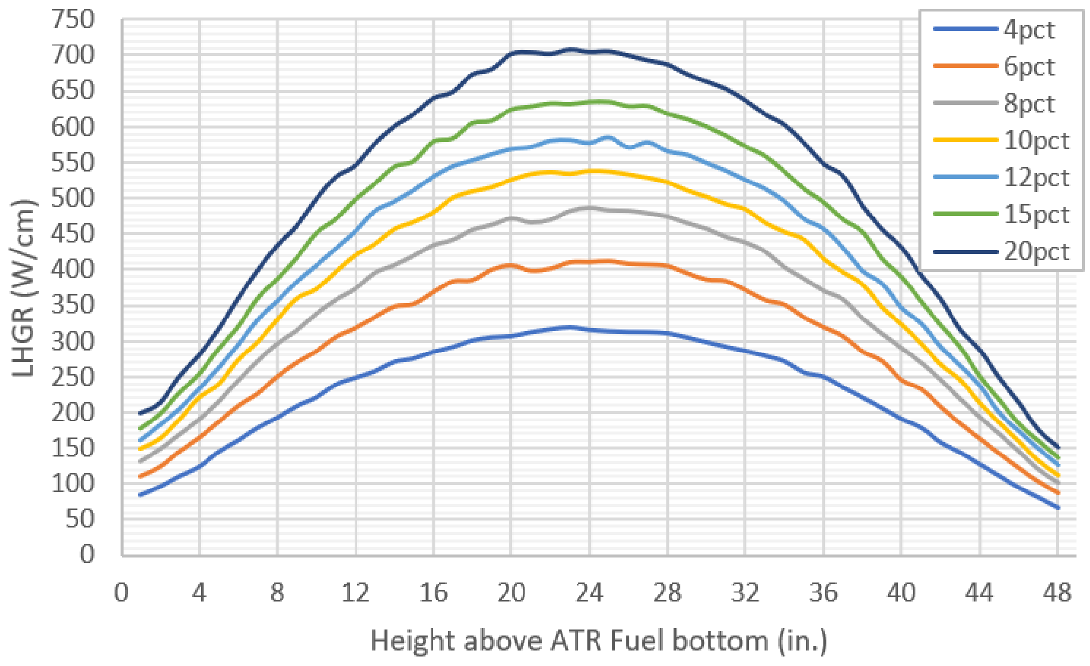




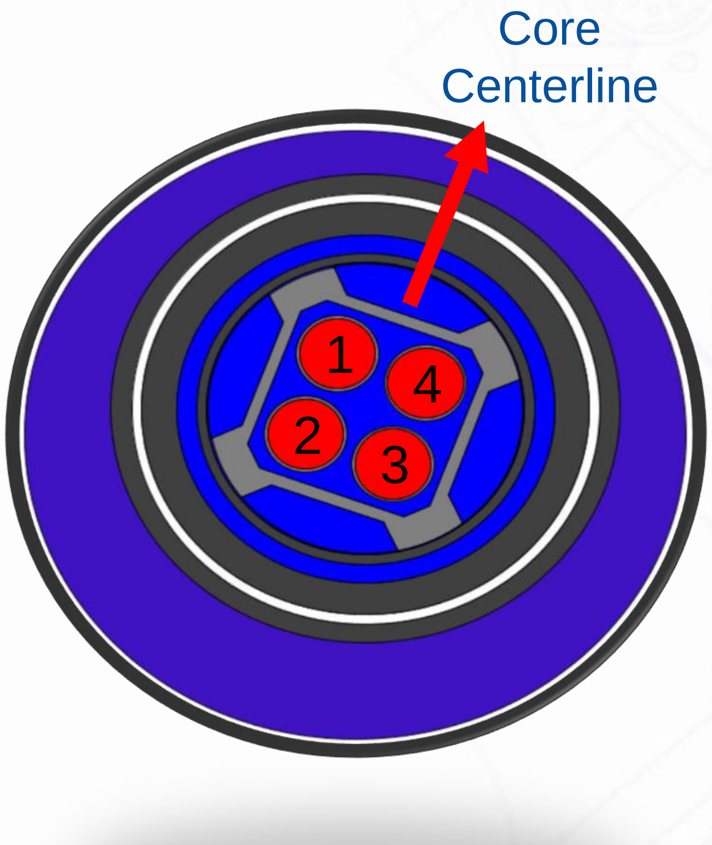
I-Loop BWR – Enrichment Sweep – 32 MW Lobe

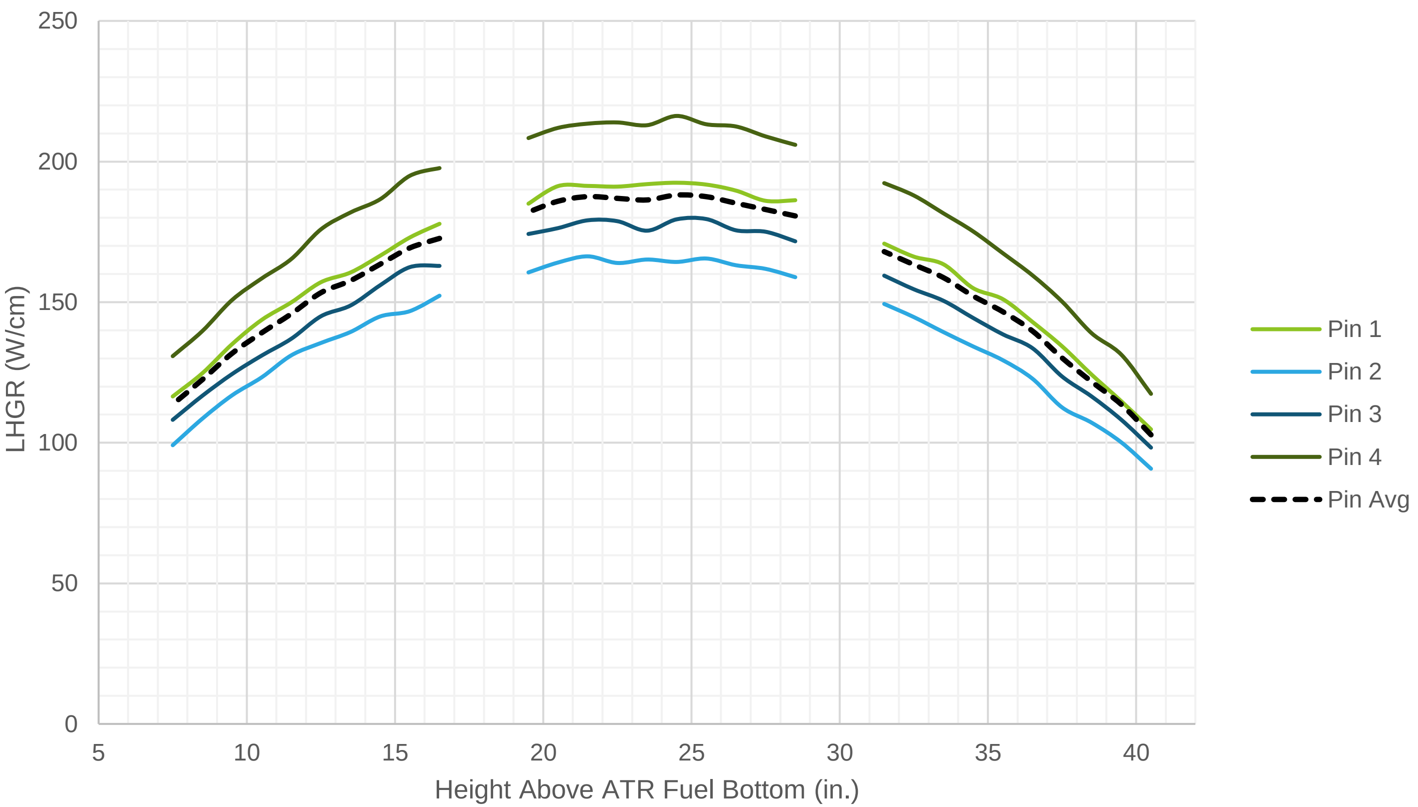
- Single pin, (no boiling)
- Uranium enrichments range from 4-20% U235
- Data scaled to 32 MW lobe power (HTSS ATR Cycle, ~2027)





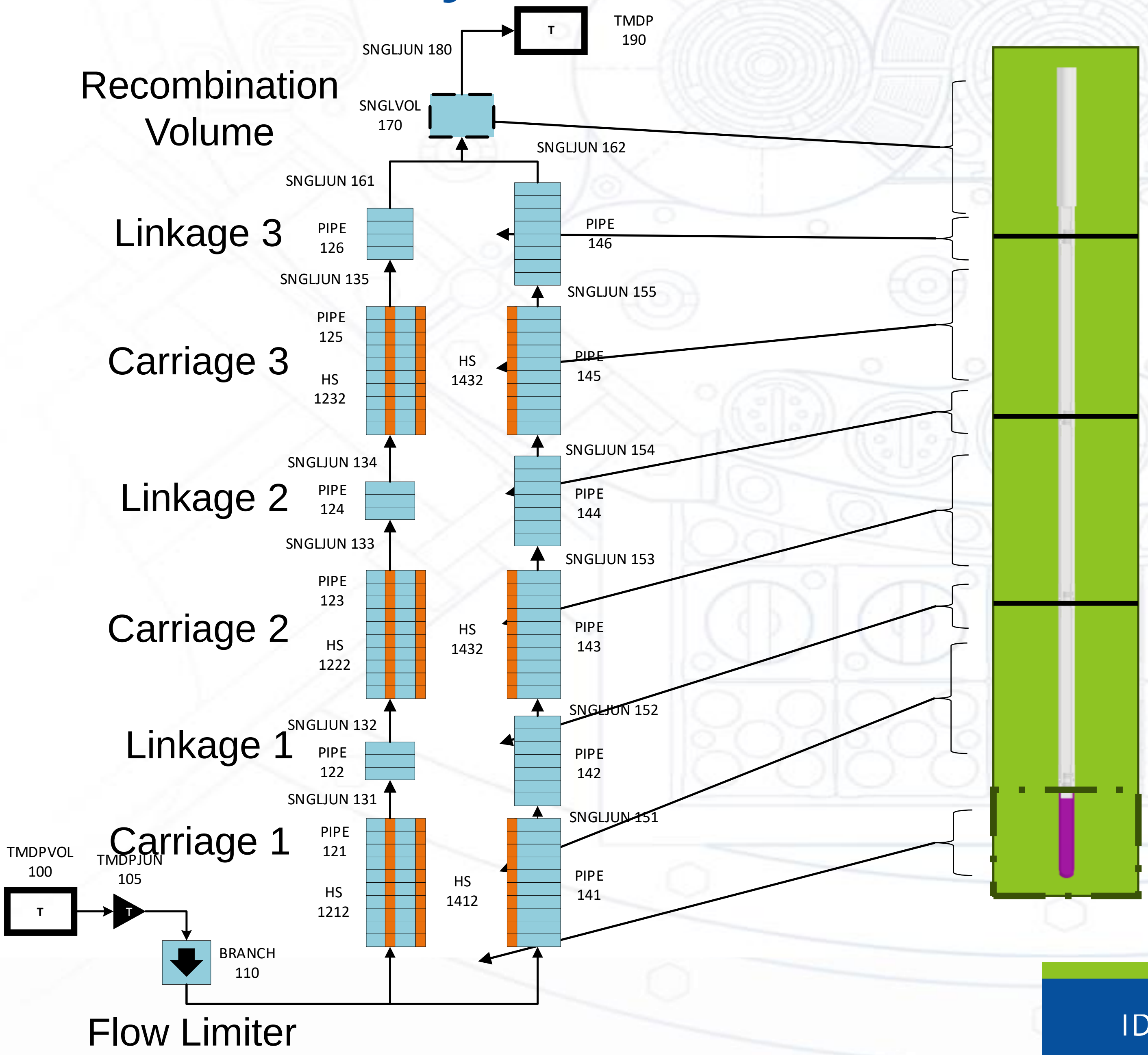
I-Loop BWR Pin Power Variation



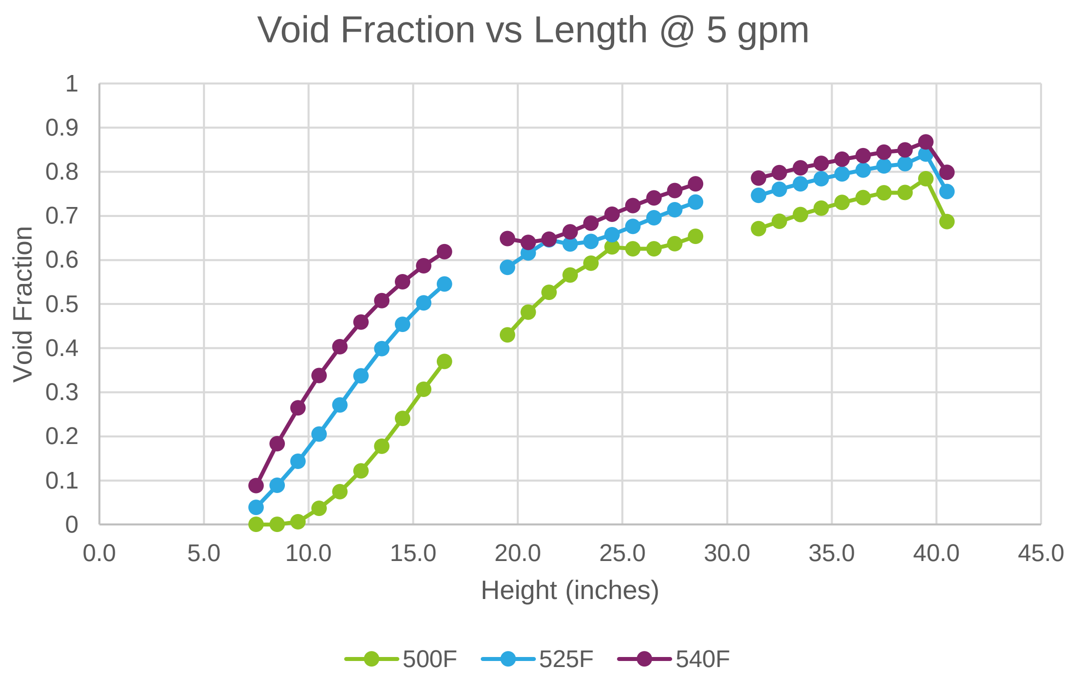


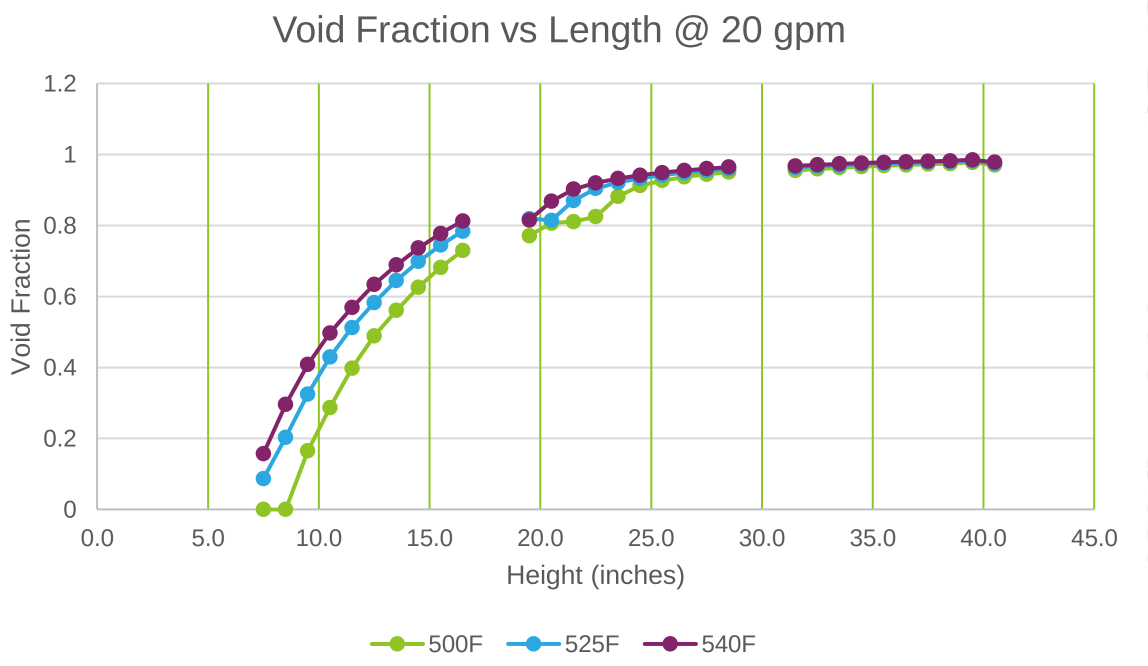
- Pins 3 and 4 are lower heat rate
- Pins 1 and 2 are higher heat rate
- Tiers 1 and 3 have uneven axial heat rate

I-Loop BWR Thermal Hydraulics RELAP



I-Loop BWR Void Fraction - Preliminary





- TH results are preliminary but show that two-phase flow can develop across axial length of ~1 meter core height

Contents

Background: What is “PCI ramp testing”

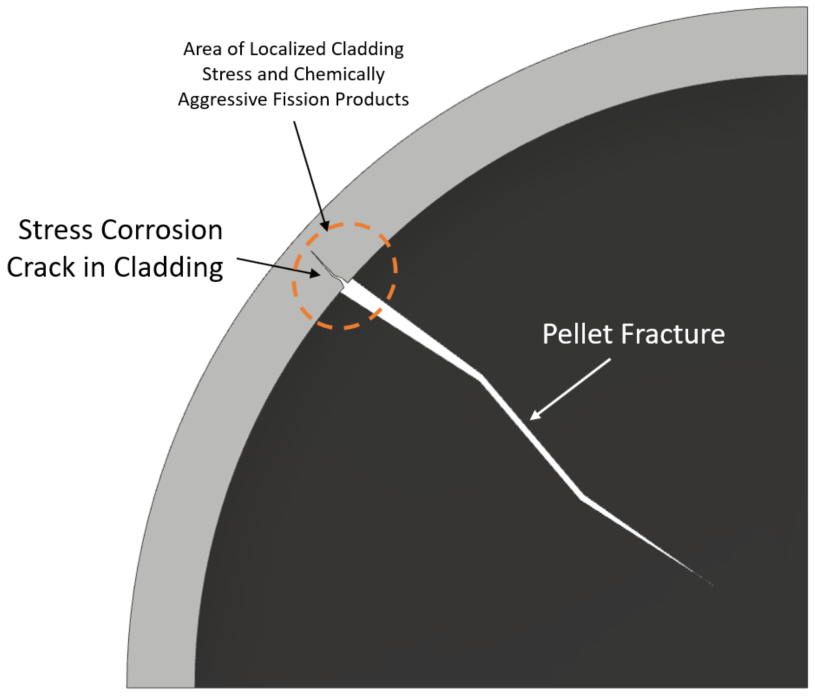
- Interaction between cladding tubes and fuel pellets, Pellet Cladding Interaction (PCI)
  - One of the most crucial performance areas for fuel rods in water cooled nuclear power plants
- ~50 years of “ramp” testing programs in test reactors, where the fission heating rate is deliberately manipulated in test rods
  - Test reactors engaged in this work have all retired over the years
  - Recent closure of the Halden Boiling Water Reactor (HBWR) effectively caused hiatus in PCI ramp testing
- Need for ramp testing is still crucial
  - Enable refined understanding as more plants consider flexible operations
  - Increased fuel rod burnup limits
  - New fuel technologies with enhanced accident tolerance

FR2 (1961-1981)

HBWR (1958-2018)

Summary of PCI

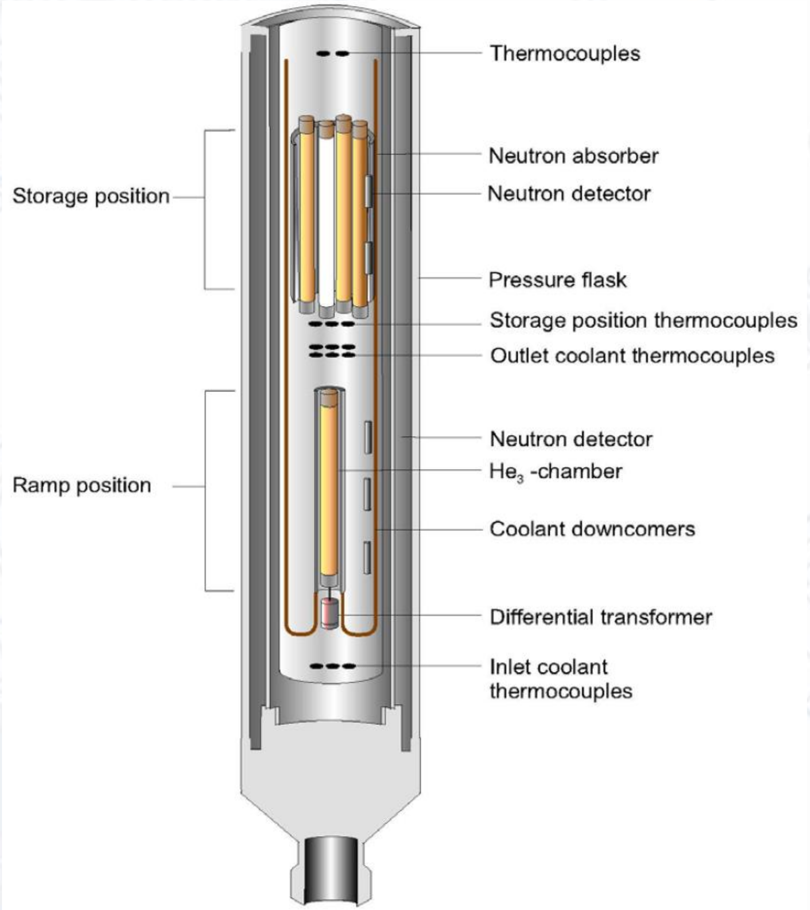
- Thermal gradients causes radial fractures in UO_2 on first rise to power
- External coolant pressure cladding “creep down” closes pellet-cladding gap at mid-life burnup
- From this state, modest ramp rates (planned power maneuvers) create a stress state susceptible to Stress Corrosion Cracking (SCC)
  - Chemically aggressive fission products (e.g., iodine) concentrates SCC phenomena near pellet fracture-cladding interfaces
  - Can penetrate through cladding to create pin hole “leaker rod”
- The general scheme for PCI ramp testing:
  - Irradiation at low power long enough for cladding creep down
  - Followed by power ramp and hold, repeat until leak is detected
  - Terminate specimen fission heating to conclude test shortly thereafter



Pellet Cross Section Illustrating PCI Behaviors
(Quarter Section Shown)

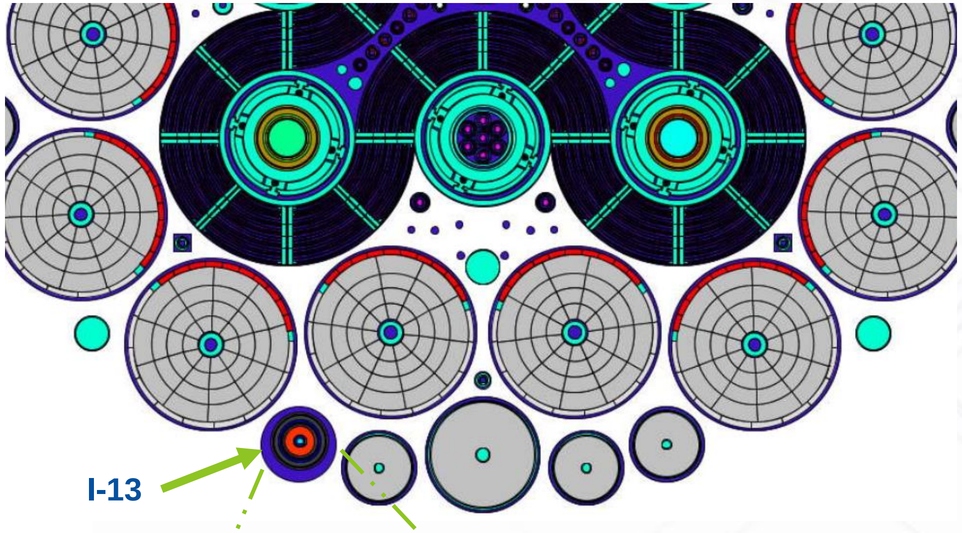
Methods for Manipulating Specimen Power

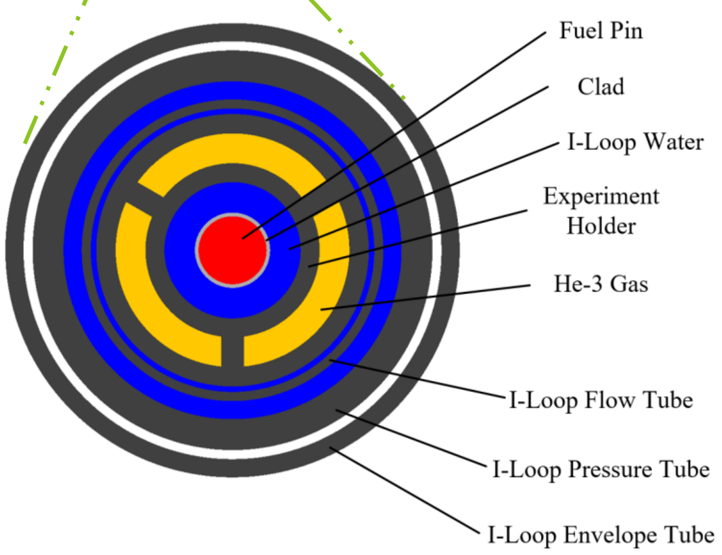
- Three ways to manipulate test rod power
- Ramp power level of entire test reactor
  - Employed by R2 ramp testing program for a short time period
  - Least popular method, affects all experiments in a reactor cycle
- Moving specimen through flux gradient (in/out of core, or near/far from core)
  - Plunging experiments vertically in/out of central core (Powered Axial Locator Mechanism, PALM), used for decades at Advanced Test Reactor for power cycle testing
  - Translating experiments toward/away from core (planned method for Joules Horowitz Reactor)
- Vary flux suppression locally in experiment rig
  - Vary pressure in 3He chamber surrounding test rod, most popular PCI ramp testing method used at R2 and HBWR



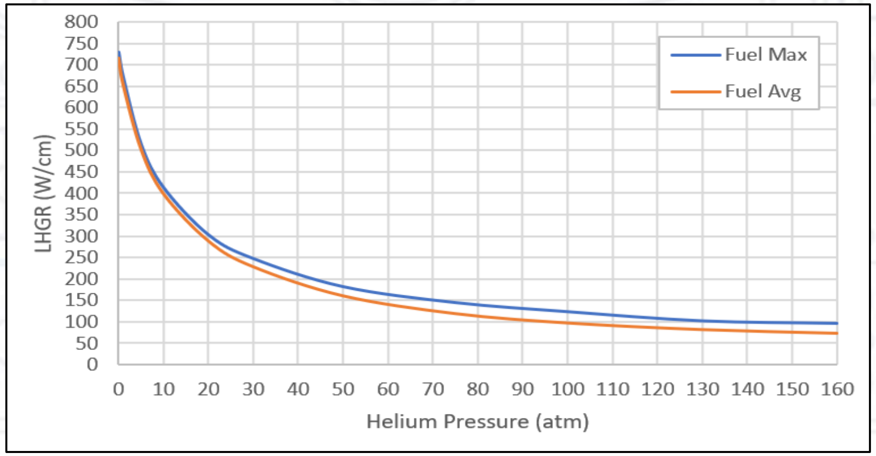
HBWR Ramp Test Rig

I-Loop Ramp Testing with Helium-3



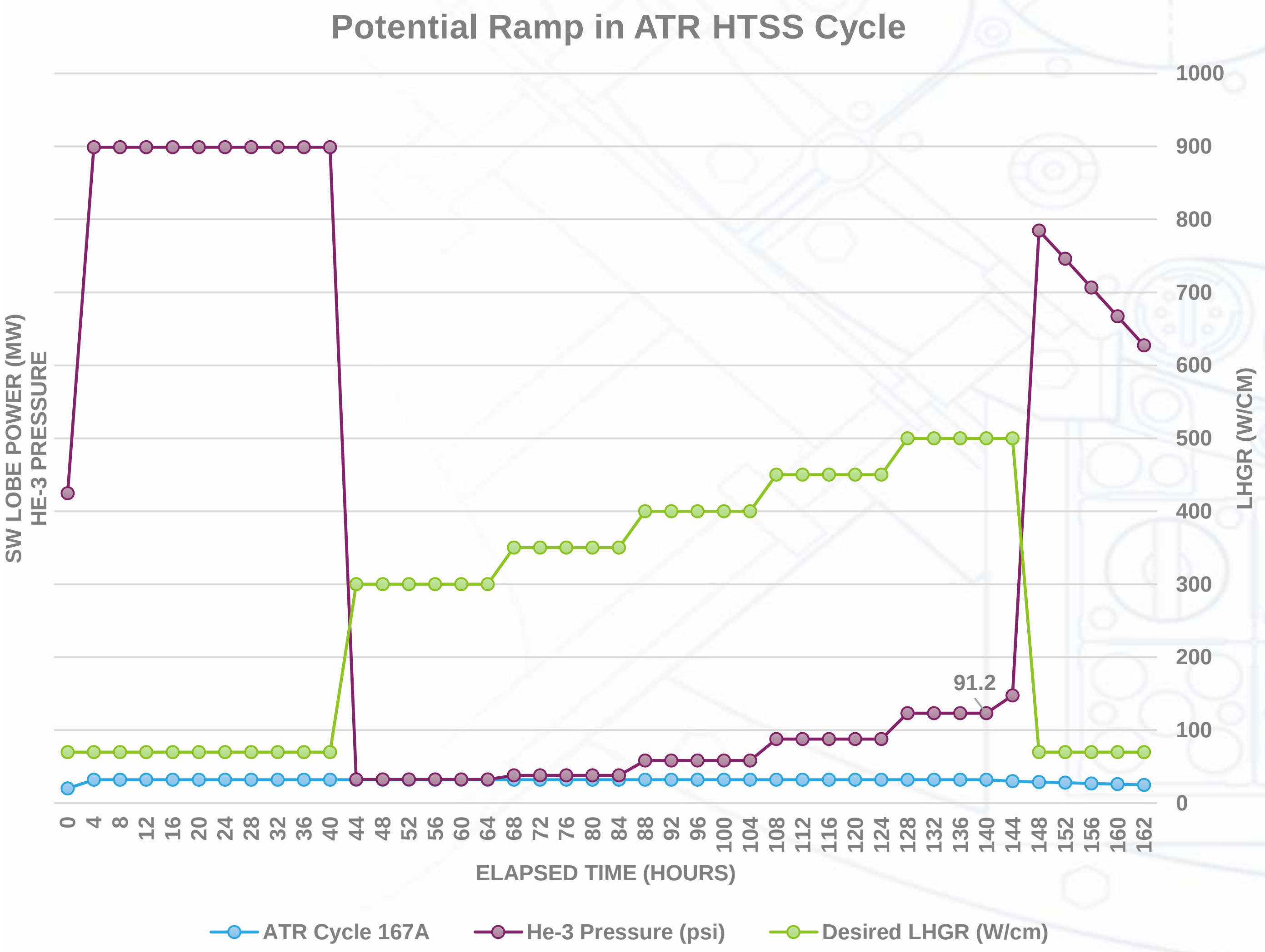


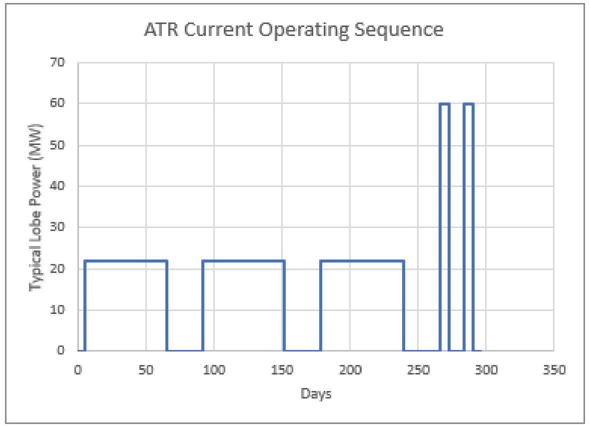
- Used “burned” fuel inventory
  - Starting enrich: 7%
  - Burnup: 19.5 MWd/kgU
  - Ending enrich: 5.1%
  - Decay time: ~6 months
- 55 MW SW Lobe Power
- Average OSCC position from Cycle 167A
- Varied the He-3 pressure to see effect on fuel pin linear heat generation rate (LHGR)

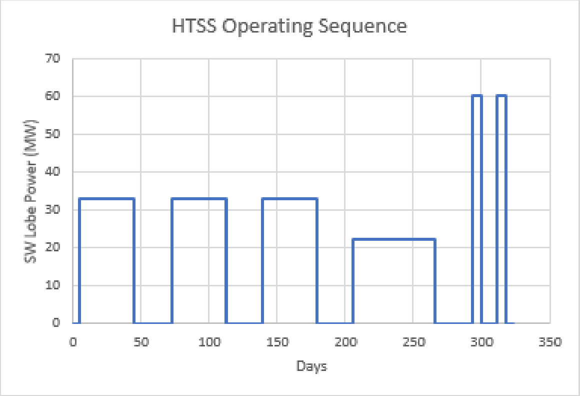


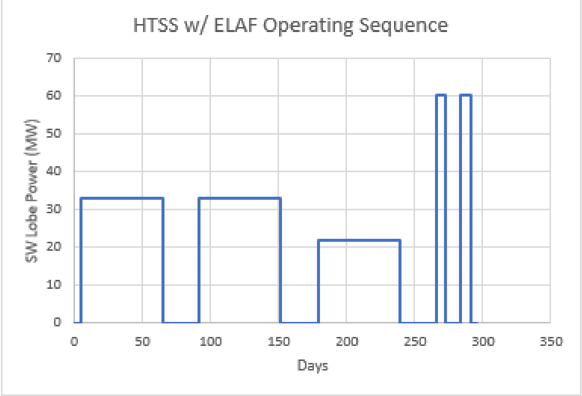
Fuel Pin LHGR as a Function of He-3 Pressure

PCI Ramp Test – Proposed Ramp Cycle



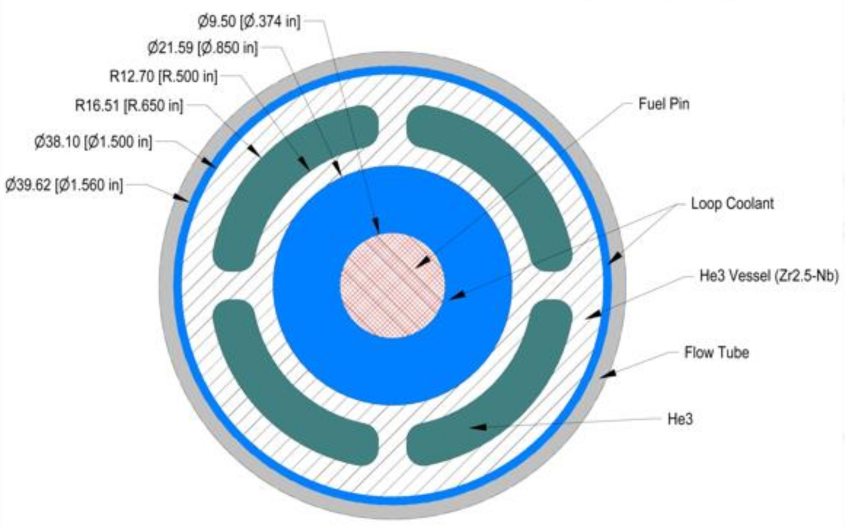


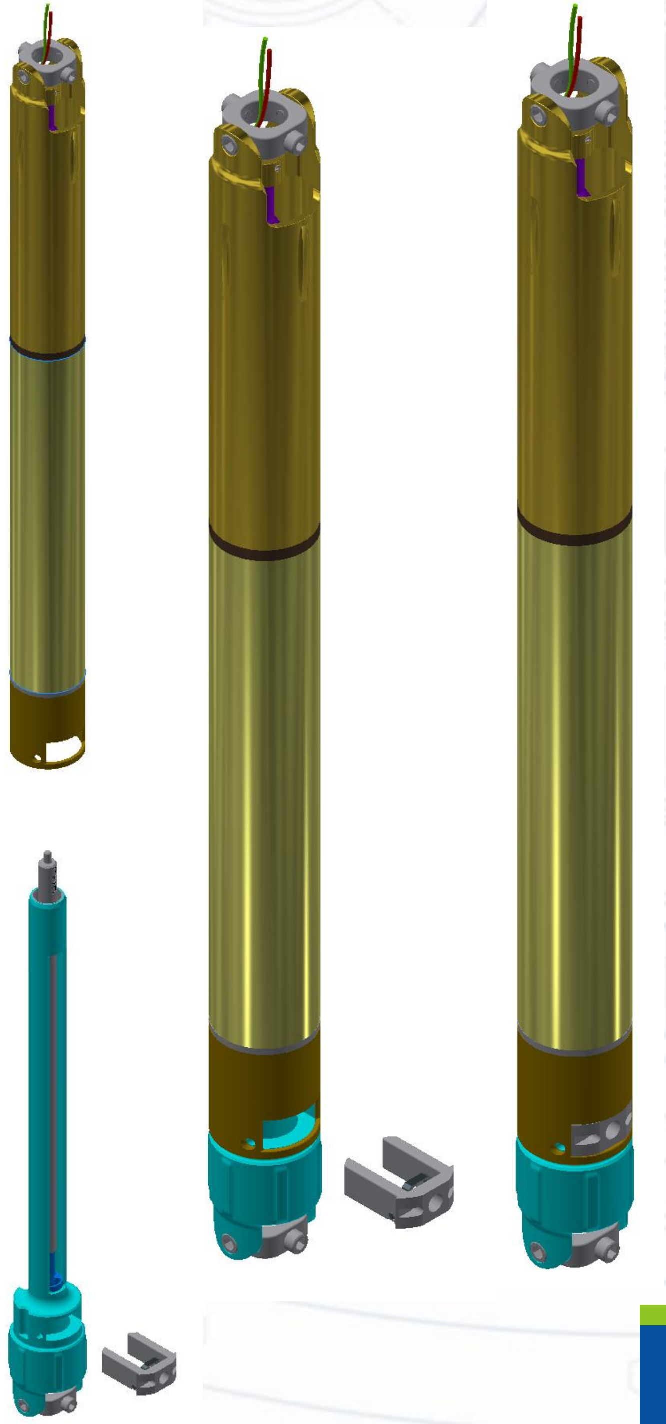


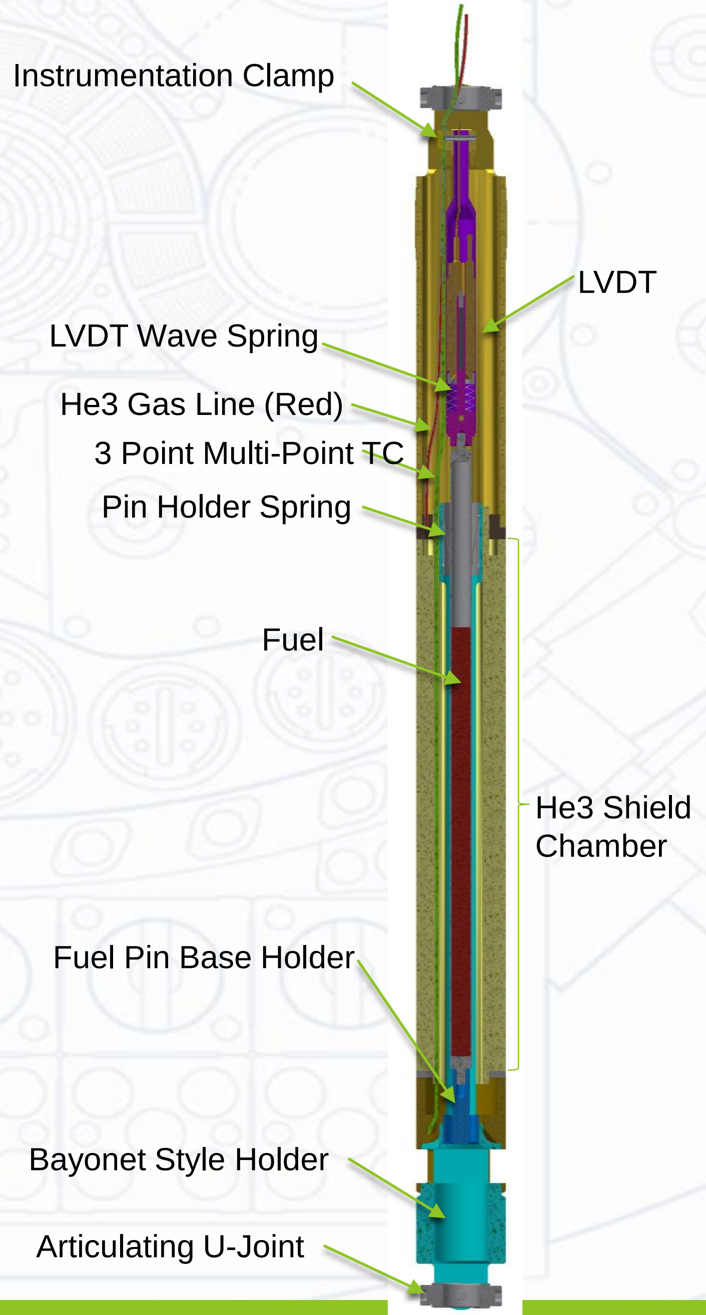


I-Loop Ramp Test Rig

- Significant Design Challenges
  - Pre-irradiated fuel – underwater handling
  - Instrumentation – LVDT and thermocouples
  - Low-pressure helium-3 annulus in high-pressure loop







Idaho National Laboratory

Battelle Energy Alliance manages INL for the U.S. Department of Energy's Office of Nuclear Energy. INL is the nation's center for nuclear energy research and development, and also performs research in each of DOE's strategic goal areas: energy, national security, science and the environment.