INL/RPT-24-77958-Revision-0

Boiling Water Reactor Testing Capability in the Advanced Test Reactor

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U.S. Department of Energy
Under DOE Idaho Operations Office
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Irradiation Testing in the I-Loop

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ABSTRACT

I-Loop is an irradiation facility that is currently being installed at the Advanced Test Reactor. It is a two-loop test facility capable of performing Light Water Reactor (LWR) irradiations in prototypic coolant conditions. The two loops are being installed to be capable of both Boiling Water Reactor (BWR) and Pressurized Water Reactor (PWR) pressure, temperature, and chemistry environments. Each loop is nominally dedicated as a BWR or PWR for simplicity of operations. In-reactor water loop testing that an I-Loop provides is key to the deployment of new accident tolerant fuel technologies and other advanced LWR fuel concepts. Currently, pressurized water loops are the only testing facilities available to test BWR fuel concepts. Their test environments are non-prototypic at higher pressure/temperature and at single-phase fluid flow conditions. This void in the LWR test bed capabilities is one that the I-Loop is uniquely situated to provide. This report discusses the mechanical design, thermal hydraulic calculations, and neutronic calculations of a proposed standard experiment of accident tolerant BWR fuel concepts. Mechanical design examines the geometry and features of the main components. Thermal hydraulic calculations examine the modeling and results of the two-phase flow options available. Lastly, neutronic calculations examine the Monte Carlo analysis of enrichment, heat rates, and flux spectrum.

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Calculations in this study were made using the Common Monte Carlo Design Tool (CMCDT) system, which is developed and maintained by the Naval Nuclear Laboratory. The CMCDT system includes the Monte Carlo code MC21 and the Physics Unified Modeling and Analysis system.

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ACRONYMS

ATR	Advanced Test Reactor
BWR	Boiling Water Reactor
CMCDT	Common Monte Carlo Design Tool
HTSS	High Temperature Steady State
ILT	I-Loop Tube
INL	Idaho National Laboratory
LWR	Light Water Reactor
LHGR	Linear Heat Generation Rate
Nb	Niobium
PALM	Powered Axial Locating Mechanism
PICT	Peak Inner Cladding Temperature
SS347	Stainless Steel 347
SW	Southwest
THCP-Mk II	Top Head Closure Plate – Mark II
Zr-2.5Nb	Zirconium – 2.5% Niobium

UNITS

cm	centimeter
gpm	gallons per minute
in	inches
m	meter
MPa	Megapascal
MW	Megawatt
ppb	parts per billion
ppm	parts per million
psig	pounds per square inch gauge
W/cm	Watts per centimeter

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Boiling-Water Reactor Testing Capability in the Advanced Test Reactor

1. INTRODUCTION

Idaho National Laboratory (INL) is deploying loop experiment facilities, known as I-Loops, that will be located in medium I-positions (I-13 and I-14) in the Advanced Test Reactor (ATR), as shown in Figure 1. Unlike flux-trap-based loops, the I-Loops will be placed in ATR's outer beryllium reflector where specimens will be subjected to neutronic, thermal hydraulic, and chemical boundary conditions representative of commercial light-water reactor (LWR) nuclear power plants. These facilities will fill the gap for LWR fuel irradiation testing capabilities left by the 2018 closure of the Halden test reactor.[1] Presently, two I-Loop facilities are planned: one employing a pressurized-water reactor environment and the other a boiling water reactor (BWR) environment. General Electric–Hitachi is particularly interested in fuel testing in BWR conditions as accommodating this testing environment in current U.S. test reactors proves challenging.

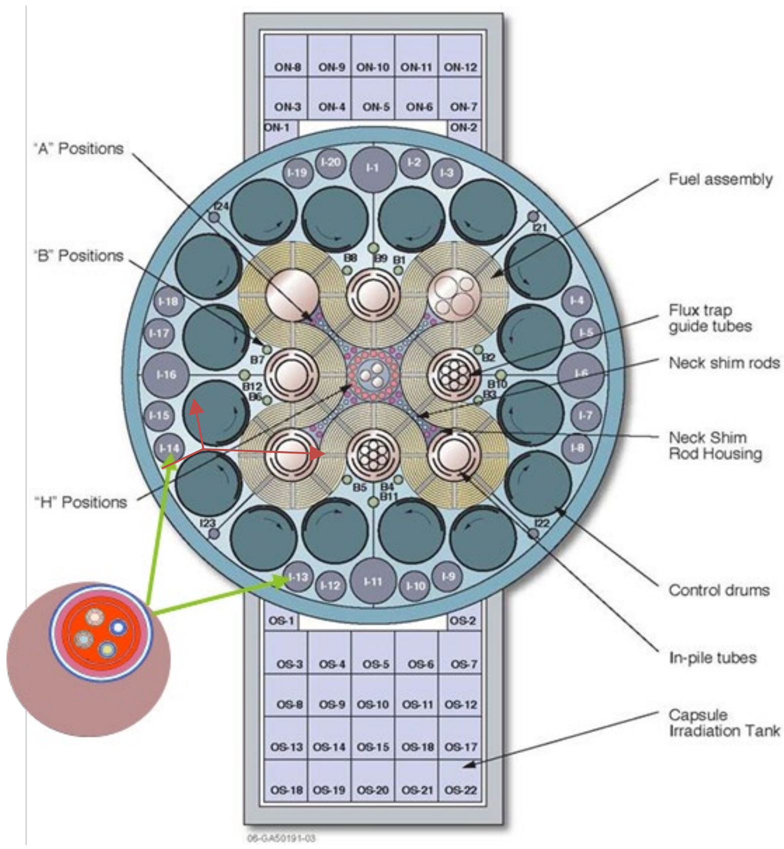

Figure 1. Location of I-Loop positions in ATR.

The I-Loop facilities are located in the outer reflector region which decouples their operations from the ATR core, making them distinguished from existing flux-trap-based loops. This distinction permits using BWR conditions in the isolated I-Loop without impacting the overall reactor operation. Because of restrictions in the flux-trap region, two-phase flow has yet to be demonstrated in an ATR experiment. Therefore, the first irradiation commissioning test will validate a new capability for BWR fuel vendors to investigate failure mechanisms and return-to-service operations. Extensive out-of-core testing will precede the commissioning test to confirm design feasibility and safety.

The proposed BWR loop experiment will explore the effects of bundle dryout on industry fuel pin designs. Bundle dryout occurs when the critical heat flux is exceeded and bubbles form on the cladding surface (or around the rod bundle), insulating the fuel pin. This insulation significantly reduces the heat transfer coefficient and drastically increases the fuel surface temperature. Bundle dryout conditions in this experiment are defined by the target void fraction (gas-to-liquid volume ratio) and peak inner cladding temperature (PICT). The respective target values are a void fraction greater than 0.990 and a PICT of around 1100 °F (600 °C)–1300 °F (700 °C). The experiment plan is to irradiate fuel pins under normal operating conditions for 3–6 ATR cycles to meet a target fuel burnup. These cycles will be performed during typical ATR cycles (60 days at a Southwest SW lobe power of 23 MW). With the target burnup met, the fuel pins will then undergo bundle dryout conditions, achieved through manipulating the loop flow. This off-normal operation will be performed in what is known as a high-temperature steady state (HTSS) cycle (40–50 days at an increased SW lobe power of 32 MW). After the simulated transient, fuel pins will continue normal irradiation to simulate the return to service of impacted fuel pins. Experiment outcomes and post-irradiation examination data will be used to investigate performance of the fuel and cladding technologies when subjected to routine and transient BWR operating conditions.

2. BOILING-WATER REACTOR LOOP FLUID

A commercial BWR plant uses demineralized water as the primary coolant and as a moderator. The I-Loop BWR system is designed to mimic the commercial BWR by providing a prototypic loop fluid environment. The main bulk fluid parameters of focus here are pressure, temperature, chemistry, and two-phase flow (boiling).

The fluid pressure and temperature of 7.1 MPa (1,030 psig) and 260°C (500°F) ensure that the fluid can stay subcooled. The saturation temperature at this pressure is 288°C (550 °F) in the ex-core piping system[2]. The fluid is delivered subcooled to the nuclear fuel specimens where nuclear heating will provide the energy to turn the fluid into a saturated mixture.

The loop coolant chemistry is primarily deionized water with small amounts of iron and zinc additives. The exact specifications are:

- pH 5.6–8.6 at 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).

3. MECHANICAL DESIGN

3.1 I-Loop Facility Design

INL recently retrofitted the ATR top head closure plate (THCP-Mk II) to include new penetrations for the I-positions and other experiment facilities. The THCP-Mk II is located on top of the reactor vessel as part of the pressure boundary and allows operator access to insert and remove experiments via a shielded cask. Since the top-head closure plate I-penetrations are not centered directly over the medium I-position, an offset pipe integrated into the upper portion of the I-Loop experiment housing, known as an I-Loop tube (ILT), to align the desired I-position with the THCP-Mk II penetration. This axial offset is approximately 7 in. (18 cm) and approximately 60 in. (1.5 m) in length where it meets a service junction flange. This junction connects the upper portion of the ILT, ex-reactor support facilities like the coolant supply, and the in-core ILT portion. The ILT then continues down from this junction approximately 16 ft

(5 m) where it terminates at the bottom of the ATR active core region. Figure 2 shows the main ILT components.

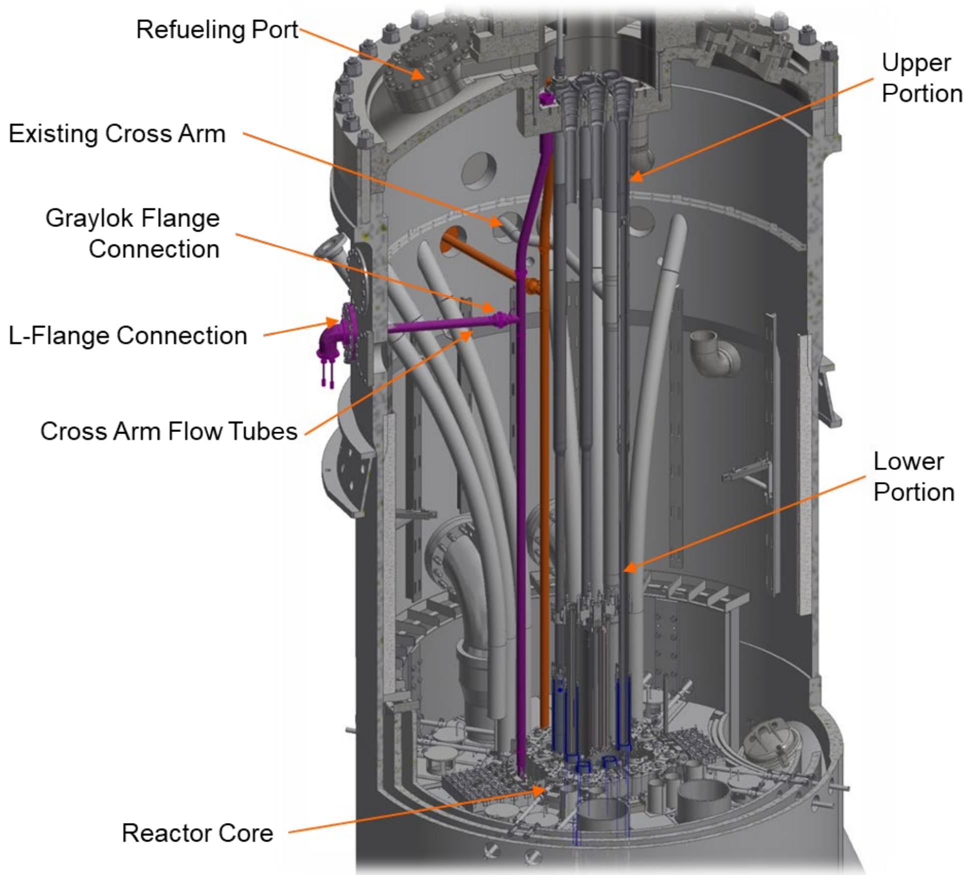

Figure 2. ILT with support cross arms installed in the ATR vessel.

A main difference between the ATR center loop and the I-Loop is the direction in which flow is supplied in the BWR flow tube. To induce the effects of boiling, flow is supplied at the service junction and travels through an annular downcomer. When the flow reaches the internal termination point of the ILT at the bottom of the active core, it reverses direction and flows upwards through a central flow tube that houses the experiment specimens. Once it passes the experiment housing and its support column, it returns to the service junction and the ex-reactor loop piping system. Figure 3 depicts this flow path.

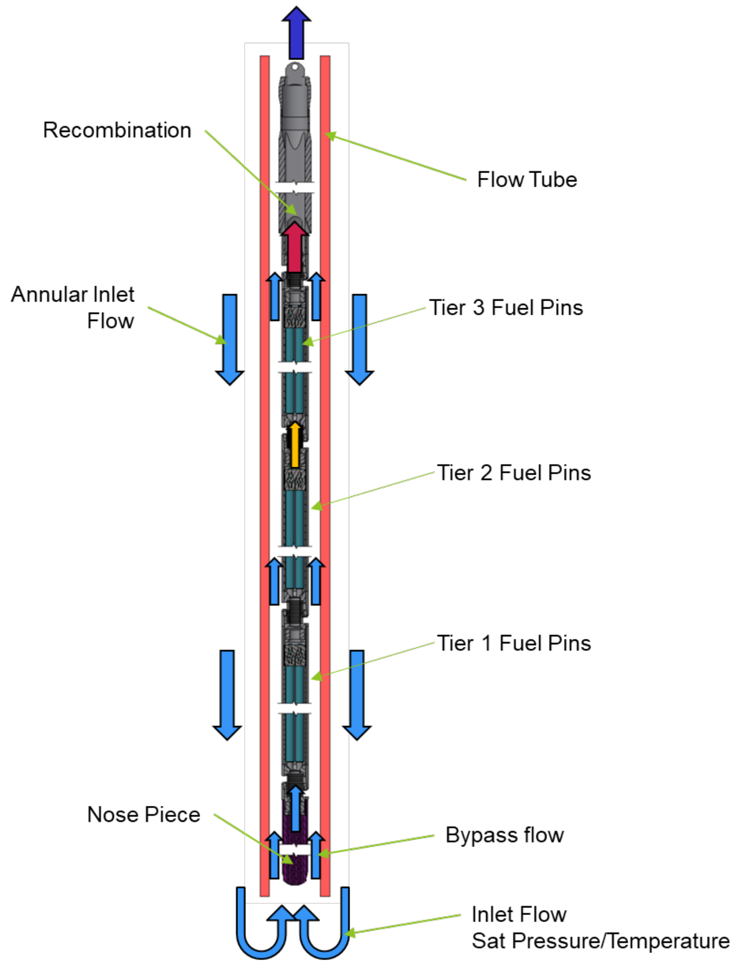

Figure 3. Flow path within an ILT (ILT not shown for clarity).

3.2 Above Core Test Train Design

A test train is a series of segments, "train cars," joined together to create the experiment assembly "train" that is inserted into the reactor. The test train, see Figure 4, contains the following components in top-down order: a bayonet lift adapter, instrument mast, pressure boundary seal and nut, hanger rod tubes, and specimen holders. The bayonet lift adapter is a small feature that interfaces with a below-the-hook device that connects to a crane or hoist hood for lifting and handling the experiment. The instrument mast is an extension pipe that contains soft instrumentation cabling and an Amphenol instrument plug for connecting to the data acquisition system. The pressure boundary seal and nut contain O-rings that will seal the ILT and lock the entire test train assembly in place. The extension tubes serve as a spacer from the reactor top head closure plate to the core region, 14 ft (4.3 m). And finally, the specimen holders are just that, components that hold the fuel specimens and pins.

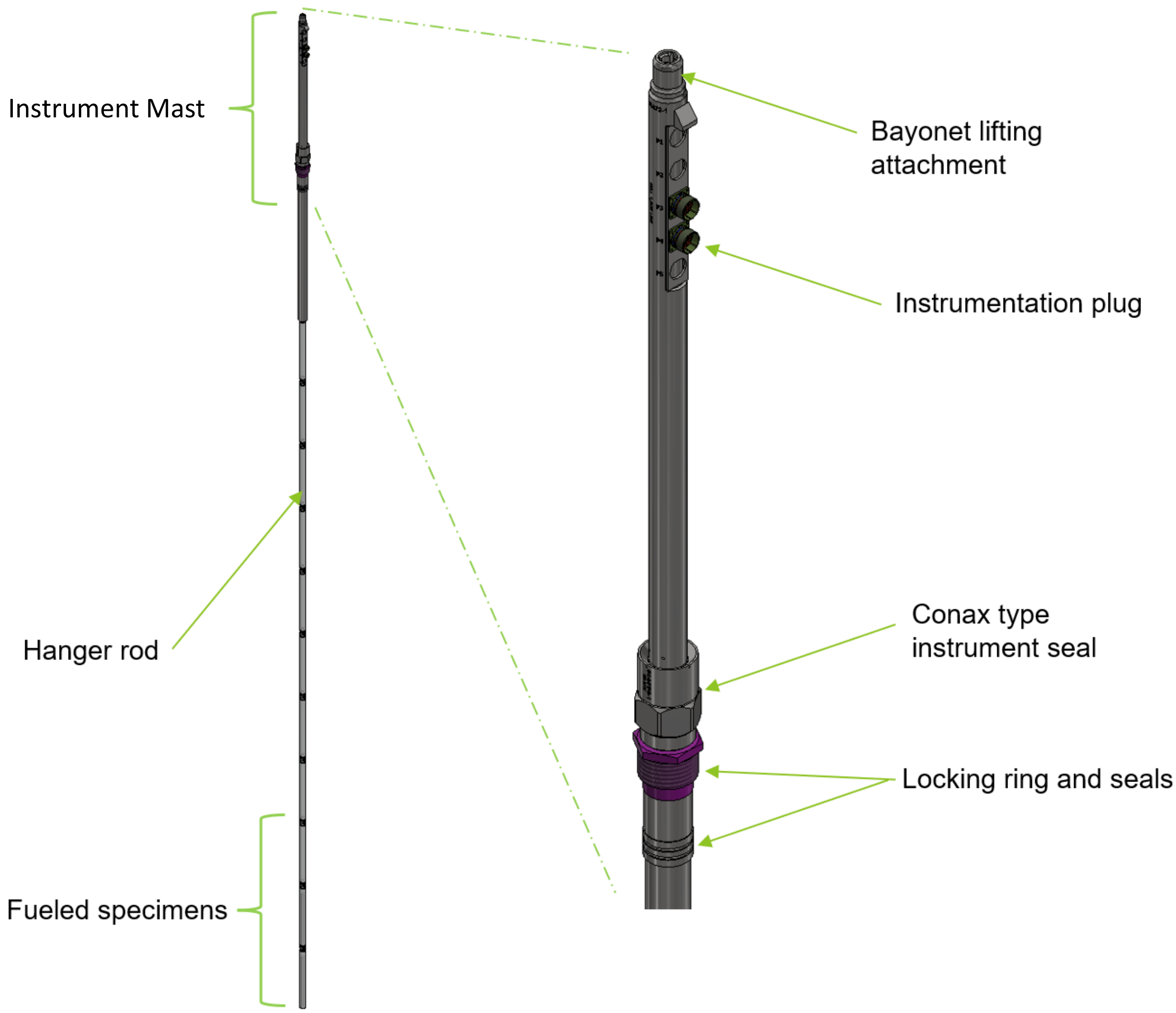

Figure 4. Typical I-Loop test train components.

The ILT is a reactor retrofit that requires the offset upper pipe section. This forces the test train to be flexible enough to articulate ~15 degrees to navigate the offset. Test train segments are separated by u-joints, shown in Figure 5, to achieve the articulation. Each section is limited in length to 18 in. (40.6 cm).

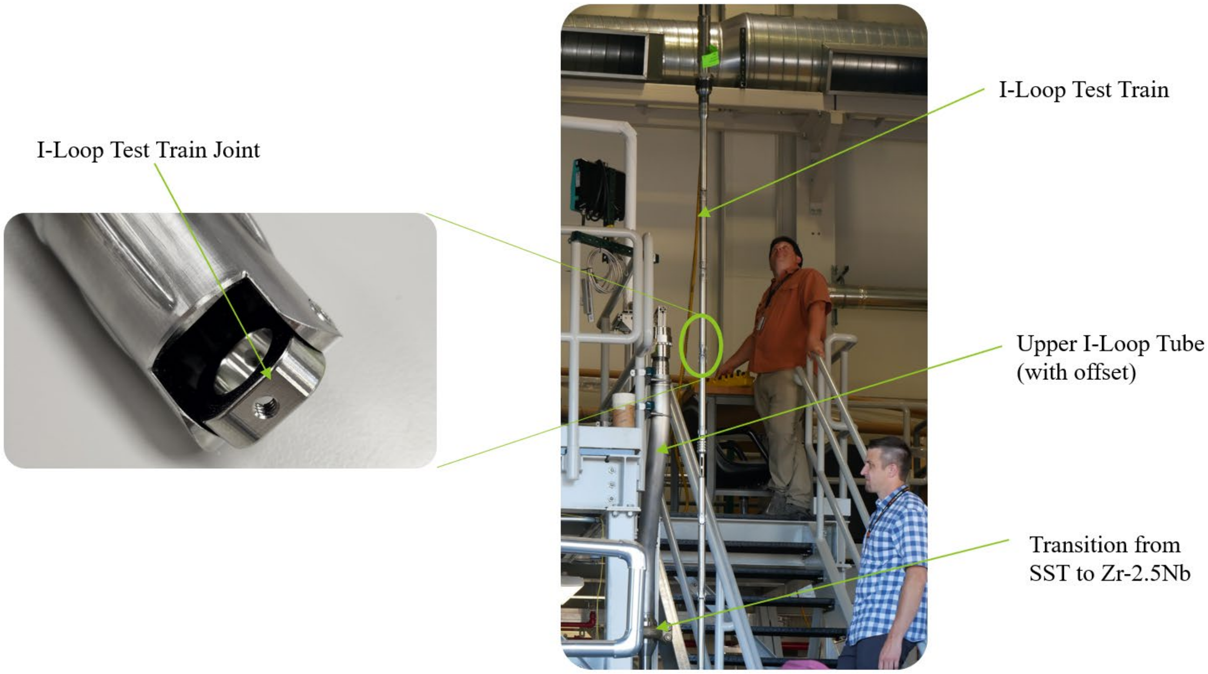

Figure 5. I-Loop test train joint.

3.3 In-Core Test Train Specific to Boiling-Water Reactor Fuel Specimens

The in-core portion of the test train is comprised of three tiers of prototypic BWR fuel pins and an inlet nozzle, see Figure 6. The BWR fuel specimens are designed to be at a height such that the three tiers are centered at ATR's core mid-plane. The inlet nozzle allows a specified reduced flow through the center of the test holder to cool the pins, and the pin coolant is isolated from the bypass flow around the outside of the hold through all three tiers. The flow enters the inlet nozzle subcooled and becomes a saturated mixture as it is heated from the lower tier to the upper tier. The saturated mixture will then recombine with the bypass flow above the core and the fluid will condense back into a subcooled state to circulate back through the loop system.

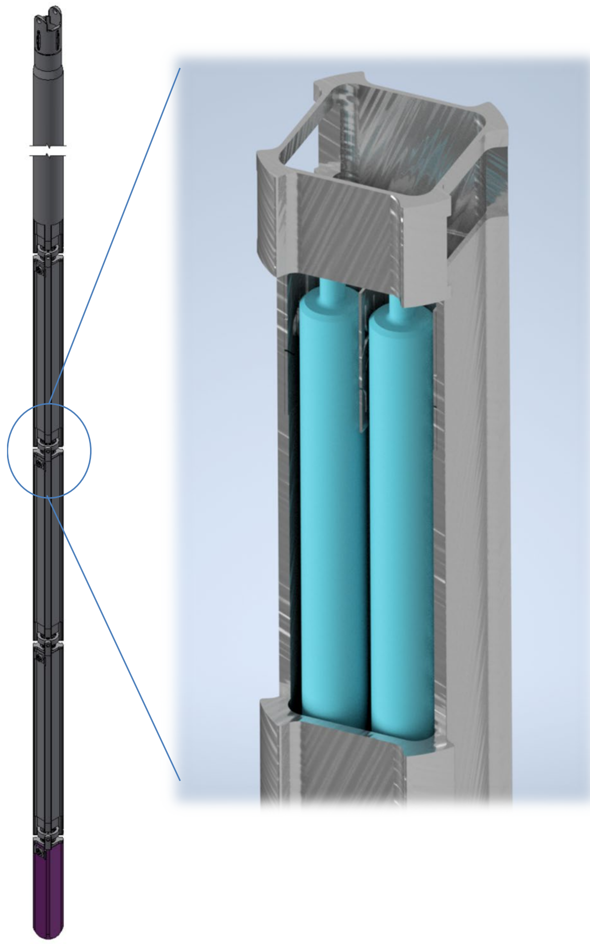

Figure 6. BWR fuel pin tiers and inlet nozzle.

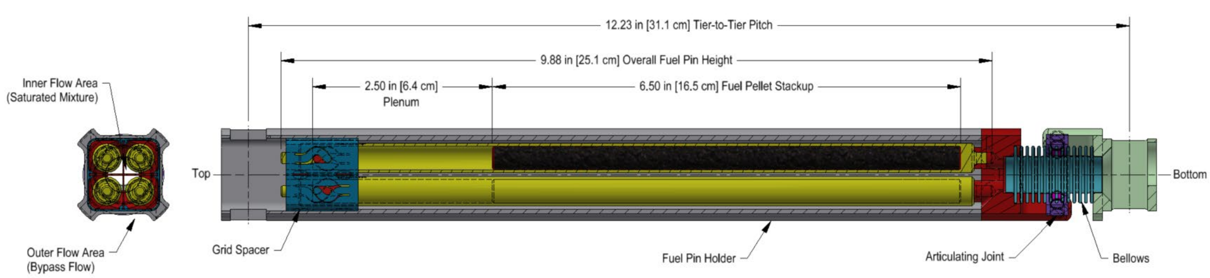

Figure 7. BWR fuel pin tier.

Each tier contains a 2 × 2 array of fuel pins, see Figure 7, for a total of 12 fuel pins. The fuel pins are the diameter of a typical BWR pin and have typical BWR pitch spacing. Each pin is 9.88 in. (25.1 cm) long with a fueled length of 6.50 in. (16.5 cm). The pitch distance from tier to tier is 12.23 in. (31.1 cm). Similarly to the hanger rod joints above, the tiers must be able to slightly articulate at the joints. The joints must also ensure the inner fluid cannot mix with the bypass flow. This is accomplished with a flexible bellows, see Figure 8.

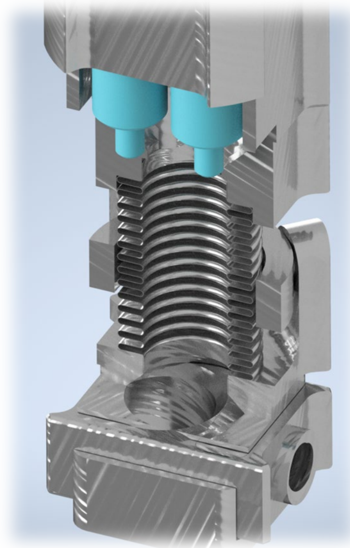

Figure 8. BWR bellows joint.

4. NEUTRONICS ANALYSIS

This section outlines the neutronic analyses performed to support the conceptual design for a BWR fuel test in an ATR I-Loop. Calculations in this assessment are made using the Common Monte Carlo Design Tool (CMCDT) code system, which is developed and maintained by the Naval Nuclear Laboratory and includes the MC21 Monte Carlo code[3] and the Physics Unified Modeling and Analysis system.

4.1 Model Description

An MC21 model of the test train was created based on the geometry described in Section 3.2. Figure 9 shows a radial cross section of the experiment and Figure 10 shows the position of the experiment relative to the rest of the ATR core. Axially, the experiment is modeled such that it spans the entire 48 in. (1.22 m) of the ATR active core height and is broken up into 1 in. calculational segments. Although this configuration does not represent how an actual experiment would look, it allows for quick assessments of heating rates at different axial locations, which is appropriate for the conceptual design stage. More axial detail will be added for future design stages. Finally, despite this being a BWR test where the coolant is expected to boil at some point along the axial length of the test train, the water inside the holder was modeled as 100% dense (void fraction = 0). Due to the large amount of beryllium and water between the ATR core and I-Loop location, boiling in the experiment is not expected to drastically alter heating rates.

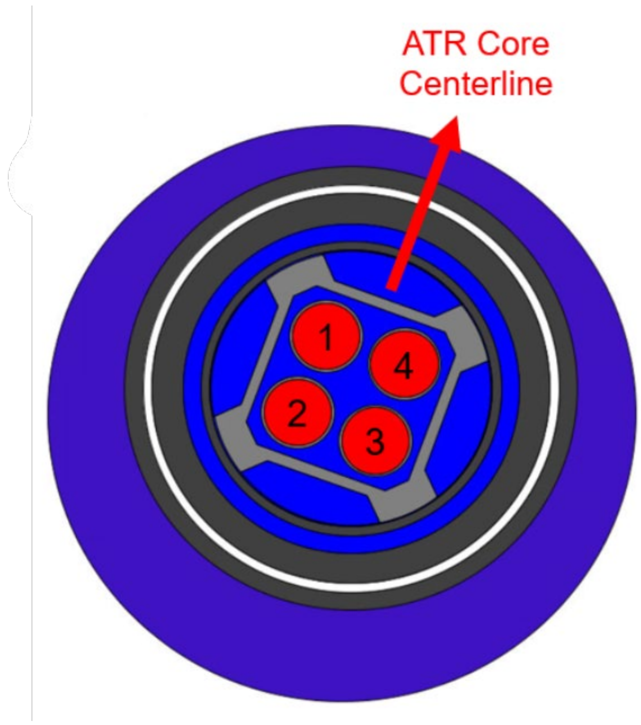

Figure 9. Radial cross section of a conceptual BWR irradiation experiment in an ATR I-Loop.

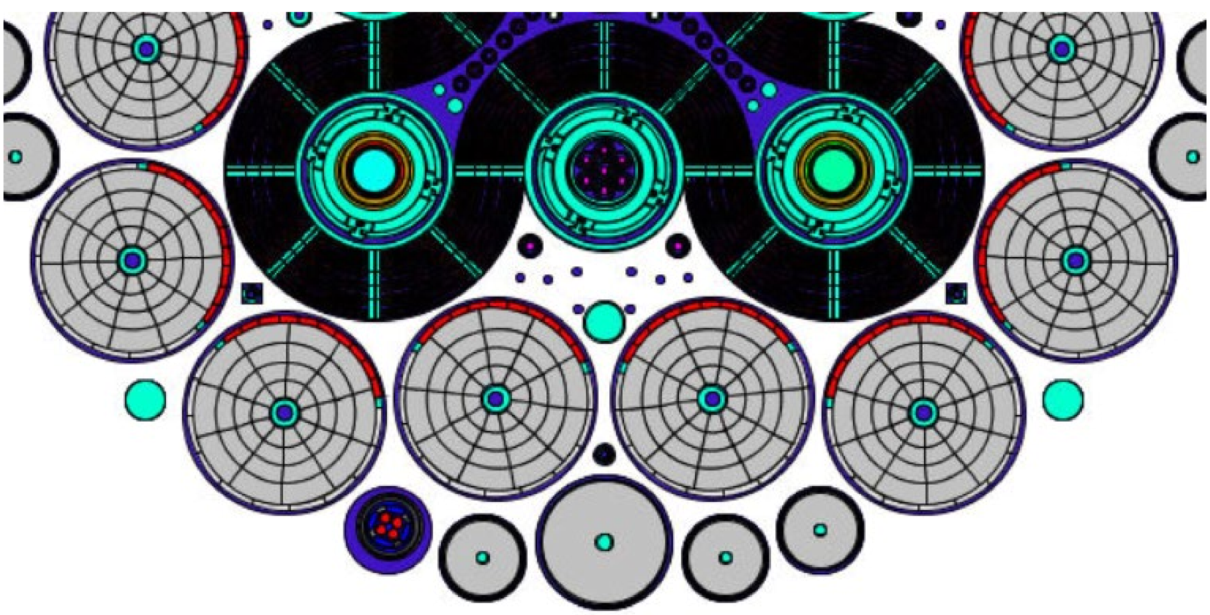

Figure 10. Neutronics model core cross section with ILT and BWR test train in ATR's I-13 position

4.2 Heating Generation Rate

At this stage in the design process, the main neutronic parameter of interest is the heat generation rates within the experiment. The fuel rods in this assessment are modeled as uranium dioxide clad in zircaloy, but since the experiment parameters are still being developed, a range of uranium enrichments are considered. Additionally, at the time of the analysis, it was unknown whether the I-Loop itself would be made out of Zr-2.5Nb or SS347, so heating rates were calculated with both materials. Table 1 shows the peak linear heat generation rate (LHGR) for each of the four fuel pins when the ILT material is Zr-2.5Nb, and Table 2 shows the data for SS347. The lower enrichments are not shown for the SS347 data because it was judged to be too low for any potential experiment. Figure 9 shows the numbering and orientation of the fuel pins. All data is normalized to a 23 MW SW lobe power, which is representative of a standard 60 day ATR cycle.

Table 1. Peak heat generation rates for fuel pins of varying enrichment when the ILT material is Zr-2.5Nb. Data is normalized to a 23 MW SW lobe power.

Enrichment (wt% ^{235}U)	Pin 1 (W/cm)	Pin 2 (W/cm)	Pin 3 (W/cm)	Pin 4 (W/cm)
4%	192.7	166.8	179.7	216.0
7%	250.4	212.6	233.6	286.8
10%	291.1	241.0	267.8	334.2
12%	308.0	253.3	283.7	355.6
15%	330.4	272.5	300.4	380.7

Table 2. Peak heat generation rates for fuel pins of varying enrichment when the ILT material is SS347. Data is normalized to a 23 MW SW lobe power.

Enrichment (wt % ^{235}U)	Pin 1 (W/cm)	Pin 2 (W/cm)	Pin 3 (W/cm)	Pin 4 (W/cm)
10%	200.1	154.1	168.8	225.2
12%	210.4	163.2	180.8	239.6
15%	226.6	174.7	189.6	262.6

Based on the data presented in Table 1 and Table 2, using SS347 for the I-Loop material equates to a roughly 35% reduction in heating rate compared to Zr-2.5Nb due to parasitic neutron absorption in the SS347 material. The fuel pins were situated such that the heating rates were similar across all the pins, but Pins 1 and 4 ultimately have higher heating rates because they are positioned closer to the ATR driver fuel.

Furthermore, the heating rates in Table 1 and Table 2 are calculated based on a standard 60 day irradiation cycle, but heating rates for other ATR operating tempos (e.g., HTSS and PALM cycles) can be estimated via scaling factors based on the SW lobe powers. Table 3 shows a scaling example using the data for fuel pins enriched to 10% in a SS347 I-Loop. Similar scaling can be done for the other enrichments and I-Loop materials.

Table 3. Peak heat generation rates for different ATR operating tempos. Data is shown for a fuel pin enrichment of 10% with a SS347 ILT material.

	SW Lobe Power (MW)	Scaling Factor	Pin 1 (W/cm)	Pin 2 (W/cm)	Pin 3 (W/cm)	Pin 4 (W/cm)
Standard Cycle	23	1.00	200.1	154.1	168.8	225.2
HTSS Cycle	32	1.39	278.4	214.4	234.9	313.3
PALM Cycle	55	2.39	478.5	368.5	403.7	538.5

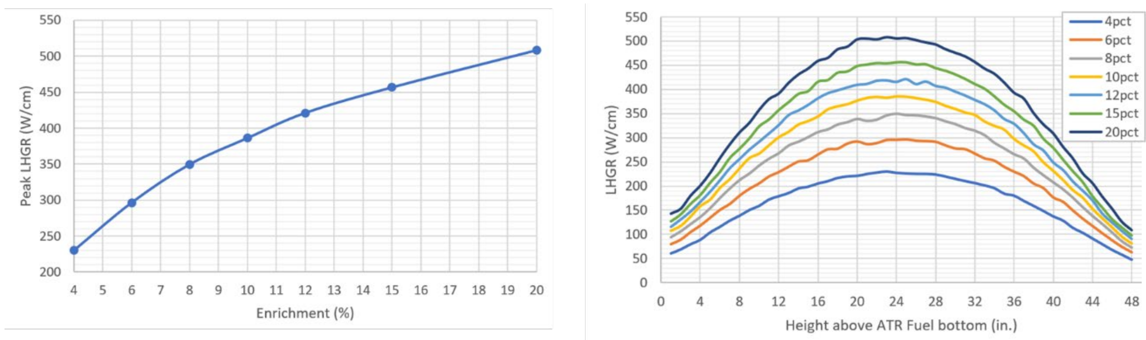

Figure 11. Peak heat generation rates for varying enrichments in a standard ATR operating cycle.

5. THERMAL HYDRAULICS ANALYSIS

To validate the operability and feasibility of the BWR test train, the Sawtooth computing cluster was used to run RELAP5-3D v4.5.0 (RELAP). RELAP is a lumped parameter system code used for course predictive modeling of thermal-hydraulic systems and their environments. The I-Loop test train model was created using the built-in hydrodynamic components in RELAP, such as branch, pipe, single and time-dependent junctions, and single and time-dependent volumes. These components, while built into the code, require manual input from the user to address the physical environment being modeled.

Using computer-aided design models, geometric values were extracted, and spatial discretization was generated for the RELAP input. Heating input values were provided by predictive Monte Carlo methods and input into RELAP as discretized Watt-per-inch values. Figure 12 shows the visual nodal diagram of the RELAP input, and Table 4 defines the pertinent components of the diagram. The nodal diagram components account for the flow direction and heating effects; however, friction losses through the bellows junctions are addressed using values derived from Idelchik's efforts[4]. ATR engineering will use auxiliary components modeled in RELAP to understand pressure drop effects over the length of the test train and over the support column, informing ex-core system requirements.

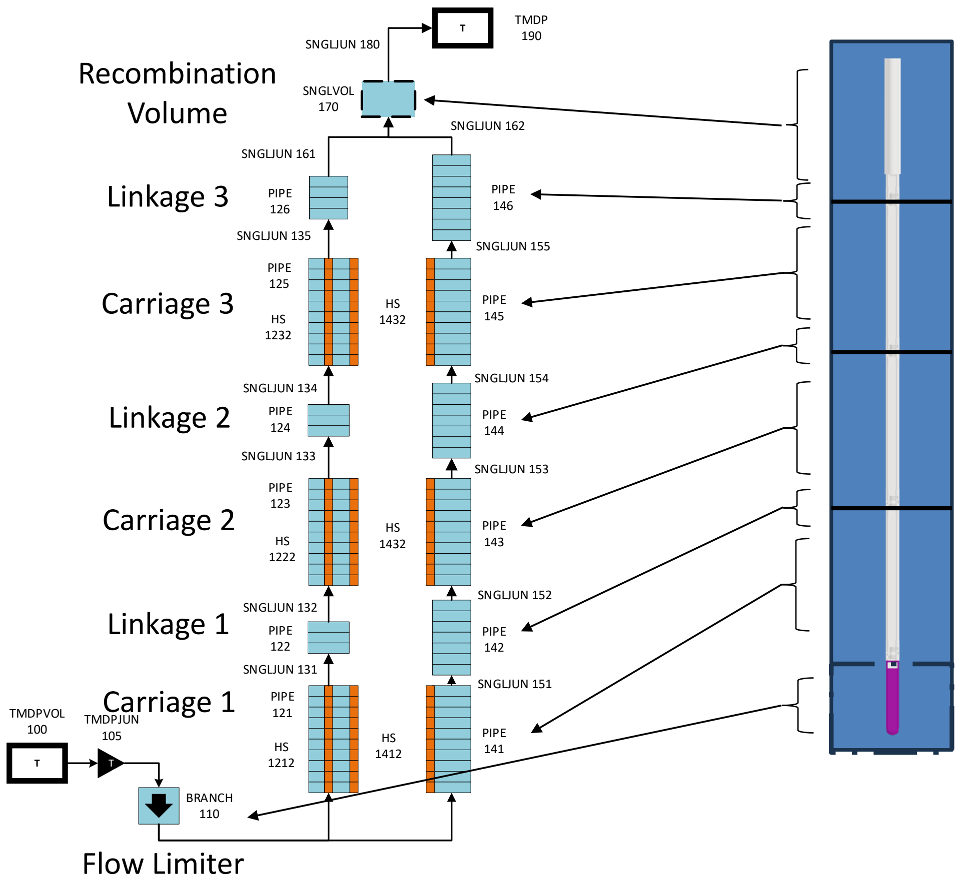

Figure 12. Nodal diagram of a BWR I-Loop test train.

Table 4. Pertinent components of the RELAP nodal diagram.

Component Number	Type	Description
110	Branch	Termination point of the ILT and fluid turnaround point. (captures the frictional resistance imposed by the flow splitter)
121, 123, 125	Pipe	Internal test train flow
122, 124, 126	Pipe	Test train flow through linkages
141, 143, 145	Pipe	Bulk coolant bypass flow
142, 144, 146	Pipe	Bulk coolant bypass flow through linkages
1212, 1222, 1232	Heat structures	Heat input from the fuel experiments
1412,1432	Heat structures	Representative walls—heat transfer from test train to bulk coolant
170	Pipe	Recombination volume—where internal test train flow and bypass bulk flow recombine

Employing the RELAP model described in this section, sensitivity cases explored the inlet temperature, pressure, volumetric flow rate, and fuel pin heating variables in the test train. This investigation helped to determine loop operating conditions necessary to simulate bundle dryout in the uppermost fueled test train. Adjusting the inlet temperature and loop pressure had marginal effects on resultant void fraction and cladding temperatures. As such, these parameters will not serve as variables to prescribe transient conditions in the loop and were held constant for further sensitivity studies. The test train environment, however, was significantly affected by perturbations in fuel pin heating and loop flow. Loop flow is bounded by anticipated pump performance with the low end being 5 gpm and the upper end being 20 gpm. Flow rates as low as 2 gpm can be achieved in the I-Loop experiment facility through a bypass operation but are not recommended.

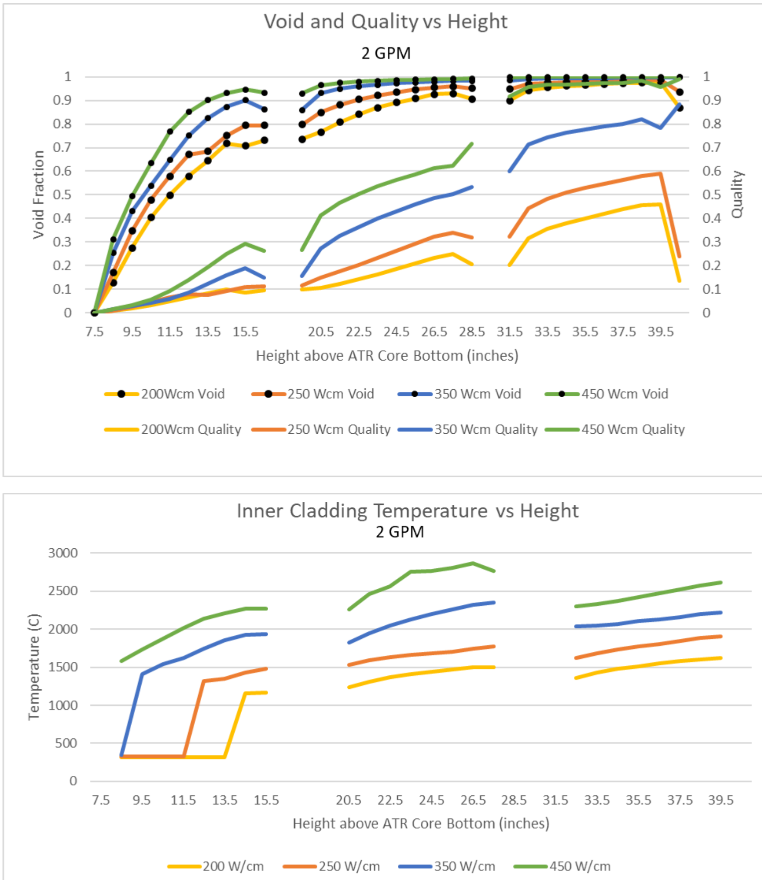

Figure 13. Void fraction, quality values, and inner cladding temperatures over the length of the three fueled test train tiers (2 gpm).

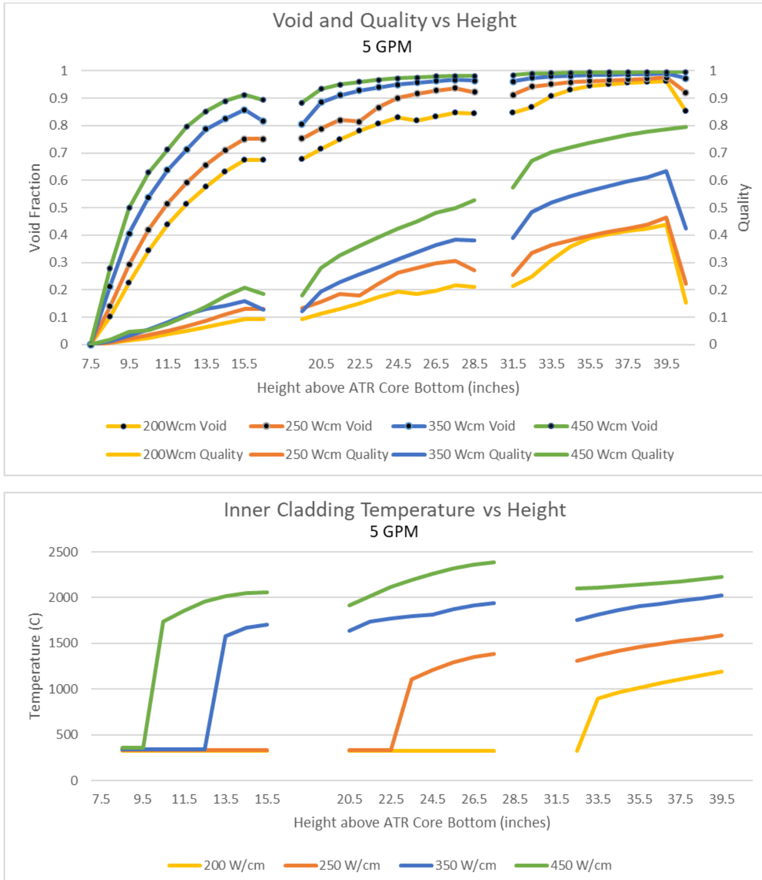

Figure 14. Void fraction, quality values, and inner cladding temperatures over the length of the three fueled test train tiers (5 gpm).

Figure 15. Void fraction, quality values, and inner cladding temperatures over the length of the three fueled test train tiers (20 gpm).

The plotted horizontal axes in Figure 13, Figure 14, and Figure 15 indicate the suspended height of the fuel along the ATR core total height. The flow limiter nose piece spans the first 7 in. Discontinuity on the plot is indicative of the test train joints where there is no fuel heat being added to the coolant. Additionally, while the fuel pin is 10 in. long, the heated region is only 9 in., with a 1 in. buffer at the bottom and top of each pin where no heat is generated within the pin. Frictional losses are captured in the single joint inputs. Experiment pressure and inlet temperature were held constant at 1,025 psi and 525ºF (274°C), respectively.

As expected, results identified that lower flow rates and higher fuel pin heating rates resulted in higher overall void fractions and cladding temperatures. At a flow rate of 2 gpm, the void fraction in all three fueled tiers exceeded 0.90 with the uppermost fueled region achieving 100% voiding with a mass vapor content of 99.1%. These conditions may be desirable for investigating dryout; however, they raise concerns around effectively collapsing bubbles in the recombination tube and result in cladding temperatures exceeding the melting temperatures of zircaloy. The PICTs did however remain just below melting points of some advanced claddings (see Figure 16) [17].

Advanced materials	Melting point (°C)	Density (g/cm^3)	Thermal conductivity (W/m-K)	Heat capacity (J/Kg-K)	Linear expansion coefficient	Young's Modulus (GPa)
Accident tolerant fuels						
UO_2	2850	10.96	8.68	235	9.76	200
U_3Si_2	1665	12.2	16.3	202	15.2	77.9
UN	2365	14.33	13.0	190	7.52	199
UC	2507	13.5	25.3	200	10.0	172
Advanced cladding						
Zircaloy	1850	6.56	21.5	285	6.0	99.3
FeCrAl	2732	7.25	11.0	480	12.4	220
H-Nicalon	2800	2.65	1.5	670	4.66	250
CVD-SIC	2700	3.21	9.5	640	2.2	466

Figure 16. Accident-tolerant fuel and advanced cladding thermophysical properties.

Increasing the flow rate to 5 gpm, the lower bound of pump operation, still achieved a maximum void fraction of 0.995 with a max quality factor of 0.796. Operation at 20 gpm significantly reduced the void fraction in the uppermost test train tier to a max of 0.95 with only a 38% mass vapor content. Inner cladding temperatures at this flow rate remained below the temperature limits of advanced fuels and claddings, but only fuel pins with an LHGR of 450 W/cm reached the desired temperature range to investigate bundle dryout effects on the cladding. This makes the 20 gpm operation ideal for normal operating irradiation cycles to reach target burnup and for the return-to-service run. Bundle dryout conditions will be achieved through a reduction in flow within the pump operation range of 5–20 gpm. The exact flow rate prescribed will be determined according to the fuel pin LHGRs at the point of the "transient" run.

While the preliminary results represent the void fractions achieved in the fueled test train tiers, the fluid-to-vapor profile above the fueled region is still under scrutiny. This region is considered the recombination volume and is intended to collapse the void steam bubbles, ideally restoring the fluid to a single liquid phase before it returns to the ex-core support systems. If a total collapse of the steam bubbles is not achieved within the reactor region of the ILT, the support system heat exchangers will be responsible for completing the conversion back to a liquid coolant. Depending on heat exchanger capacity, a threshold void fraction will be targeted in the recombination volume. Further analysis is needed to determine this threshold and validate assumptions.

6. SUMMARY

U.S. test reactors have been pursuing opportunities to close gaps in irradiation testing capabilities introduced with the shutdown of the Halden test reactor in 2018. The ATR I-Loop project is a two-loop test facility capable of performing BWR and PWR irradiations is prototypic coolant conditions, as part of DOE's effort to establish a modern day LWR test bed. Particularly, the new I-Loop experiment facilities are of interest to the Advanced Fuels Campaign for use in qualifying new accident-tolerant fuels and other advanced LWR fuel technologies. The I-Loops will be capable of ramp testing, bundle dryout investigation, and general irradiation in a prototypic LWR environment. Two-phase flow and prototypic BWR test environment has yet to be demonstrated in ATR, and therefore, this commissioning test will validate modeling and establish a new capability.

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