INL/RPT-22-66802
Revision 0

Conceptual Design of a Pumped Water Capsule for TREAT

April 2022

Nick Woolstenhulme, Charles Folsom, Klint Anderson

Idaho National Laboratory

INL is a U.S. Department of Energy National Laboratory
operated by Battelle Energy Alliance, LLC

INL/RPT-22-66802
Revision 0

Conceptual Design of a Pumped Water Capsule for TREAT

Nick Woolstenhulme, Charles Folsom, Klint Anderson

April 2022

Idaho National Laboratory
Idaho Falls, Idaho 83415

http://www.inl.gov

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

Page intentionally left blank

ABSTRACT

This study performed mechanical design, thermal-hydraulic, and fuel performance investigations to determine whether an existing large water capsule design could be modified to create forced convection boundary conditions on nuclear fuel specimens in the Transient Reactor Test facility. This concept is viable and offers several advantages compared to a static water environment for a variety of transient power shapes. We do not foresee the pumped capsule being superior to the past concept for a larger water loop but recommended further engineering and deployment based on its high cost-to-value ratio primarily as a stop gap capability until a larger water loop can be realized.

Page intentionally left blank

CONTENTS

FIGURES

Page intentionally left blank

ACRONYMS

BUSTER	Broad Use Specimen Transient Experiment Rig
BWR	boiling-water reactor
DNB	departure from nucleate boiling
JMTR	Japan Material Test Reactor
LHGR	linear heat generation rates
LOCA	loss-of-coolant accident
LWR	light-water reactor
PCF	power coupling factor
PCI	pellet-cladding interactions
PCMI	pellet-cladding mechanical interaction
PWR	pressurized-water reactor
RIA	reactivity-initiated accident
TREAT	Transient Reactor Test facility
TWERL	Treat Water Environment Recirculating Loop
TWIST	Transient Water Irradiation System for TREAT

Page intentionally left blank

Conceptual Design of a Pumped Water Capsule for TREAT

1. INTRODUCTION

The Transient Reactor Test facility (TREAT) is a multimission material test reactor with a unique power transient capability. From a nuclear physics viewpoint, TREAT's core is essentially a large graphite block with a small amount of uranium oxide dispersed throughout. TREAT's core absorbs the fission energy produced in power excursions that, when paired with an automatically controlled control rod system, enables it to safely produce a variety of extreme power maneuvers. This capability is typically used to expose fuel specimens in the core center to simulate accident conditions postulated in other types of nuclear reactors. TREAT operated from its initial construction in the late 1950s until the mid-1990s when reductions in fuel safety research funding put the reactor into an operational standby. Years later, the facility was refurbished and resumed operation in 2017 to resume fuel safety research addressing long-standing data gaps, advanced fuels, and new reactor designs.

TREAT performed numerous tests on water-cooled reactor specimens throughout its first era of historic operation using static water capsules and once-through steam systems [1] but never deployed a forced convection liquid water loop since this capability was realized in a contemporary test reactor (the Power Burst Facility). By the late 1990s, both TREAT and the Power Burst Facility were not operating, and efforts to resume operations at TREAT clearly pointed to the need for a water loop to address emergent needs in the light-water reactor (LWR) community [2]. These efforts were not realized but showed the value of providing forced convection boundary conditions on test specimens. Not long after, the CABRI facility in France began a decades-long project to retrofit a water loop into their reactor for similar reasons. The CABRI retrofit was a tremendous undertaking, and its new water loop only recently began supporting transient experiments.

CABRI is well subscribed for the foreseeable future to address data needs for "standard" LWR fuel designs (e.g., UO_2 pellets in zirconium-alloy cladding tubes). Water-cooled fuel testing needs in the United States (U.S.) include standard LWR fuels as well as a variety of advanced accident-tolerant fuel designs. U.S. transient testing interests may also find value in assessing novel fuel designs for potential application in small modular reactors or fuel assemblies for material test reactors (e.g., aluminum-clad, plate-type, high-density fuel systems). All these fuel technology areas could benefit from more prototypic forced convection boundary conditions in TREAT. Based on these diverse needs, it is unlikely that CABRI alone could be a workable solution to support U.S. needs in water-cooled reactor fuel research.

Due to its innate design, the CABRI reactor's main competencies are limited to pulse-type power transients, simulating reactivity-initiated accident (RIA) conditions. CABRI pulses tend toward shorter durations (~10–30 ms full width half max). TREAT is also capable of pulsed operations buts tends toward longer durations (~90 ms currently [3] and as low as ~45 ms with planned facility upgrades [4]). The pulse width effect can be an important parameter influencing the timing of cladding heat up and ductility as thermal expansion drives the pellet-cladding mechanical interaction (PCMI). Working together, TREAT and CABRI could encompass the pulse widths typically postulated for LWR RIAs (~30–60 ms) to create a comprehensive data set, but a comparison of these data sets would be more valuable and straightforward if TREAT had comparable forced convection water capabilities.

In addition to RIA testing, TREAT has unique competencies in simulating other types of power maneuvers, including cyclic power oscillations, rising power ramps for pellet-cladding interactions (PCIs), and loss-of-coolant accident (LOCA) simulations. The ability to provide forced convection water conditions in TREAT could enhance these types of tests by manipulating the surface heat transfer to better simulate radial temperature gradient transient evolutions. A forced convection capability could also accurately capture hydrodynamic effects, fuel-coolant interactions, fuel relocation and sweep out, and rewet behaviors following a boiling crisis. Simply put, the long-recognized need for a forced convection water capability continues to this day with far reaching impacts in fuel safety research.

2. BACKGROUND

The water loop concept envisioned for TREAT in the late 1990s had significant hydraulic support equipment outside the core and plumbing routed outside the reactor concrete shielding. If this device would have been installed, the approach would have differed from most historic TREAT experiments, which favored self-contained devices that could be hoisted entirely into shielded casks to simplify installation, removal, contamination control, and personnel radiation protection. This design approach is exemplified by the workhorse Mk-III sodium loop in TREAT where a simple piping structure and compact electromagnetic pumps [1] provide forced convection liquid sodium environments. Such an approach could perhaps be thought of more accurately as a "pumped capsule" than a "loop" when compared to typical system-scale loops.

Currently, TREAT is highly utilized by a diverse user community who require a rapid transition between test devices with various coolant types (e.g., gases, water, liquid metal). Accordingly, previous efforts worked to devise a compact water loop with the same type of self-contained mechanical layout. This preferred approach would aid in the loop's rapid installation and removal and to preclude the risk of radioactive coolant plumbed outside the reactor's shielding. This device was termed the TREAT Water Environment Recirculating Loop (TWERL). The TWERL design matured in some detail during past efforts [5] and showed viability as the TREAT seminal water loop. TWERL, however, requires the development of a custom compact centrifugal water pump and has little in common other TREAT water capsule design features. Thus, budget constraints and other priorities have continued to force TWERL into the "far future" phase of TREAT's capability schedule, see Figure 1.

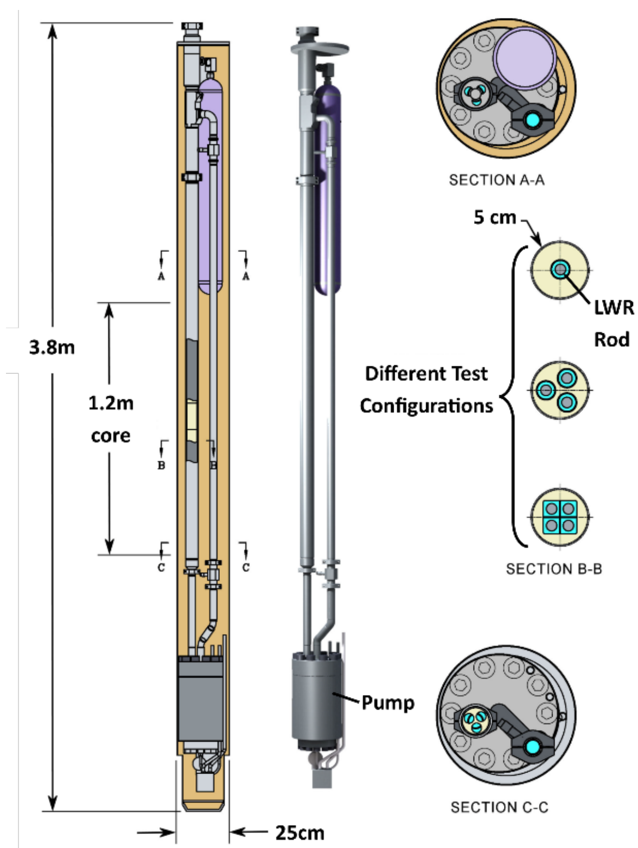


Figure 1. TWERL concept [5].

An alternate approach investigated whether a tall capsule and the deliberate placement of both heat sources and sinks could be used to create a natural circulation water environment for TREAT. The approach was workable but later abandoned due to difficulties in achieving the full range of heat transfer conditions needed relative to the deployment cost [6].

More recently, researchers developed a new larger diameter static water capsule, with an integrated blowdown tank, to support near-term LOCA testing in TREAT. This concept, the Transient Water Irradiation System for TREAT (TWIST), is now completing final engineering and prototype testing to support LOCA tests beginning in 2023. The present work emerged as a conceptual study to determine whether the TWIST design could be adapted with a small motor and impeller integrated directly into the capsule top. The so-called pumped-TWIST design (or P-TWIST) could be a cost-effective stopgap strategy to provide forced convection water conditions until the full TWERL design deployment (presumably years from the present day). Another inspiration for this concept came from a similar pumped capsule approach, developed to some degree for use in the Japan Material Test Reactor (JMTR), as shown in Figure 2 [7]. JMTR was a water-cooled steady-state-type test reactor with a markedly different mission than TREAT, and it is unclear from the archival literature whether this concept was ever sufficiently matured before JMTR was shut down. Still, the existence of this pumped water capsule concept gives further credibility to support a more detailed study about deploying such a device in TREAT.

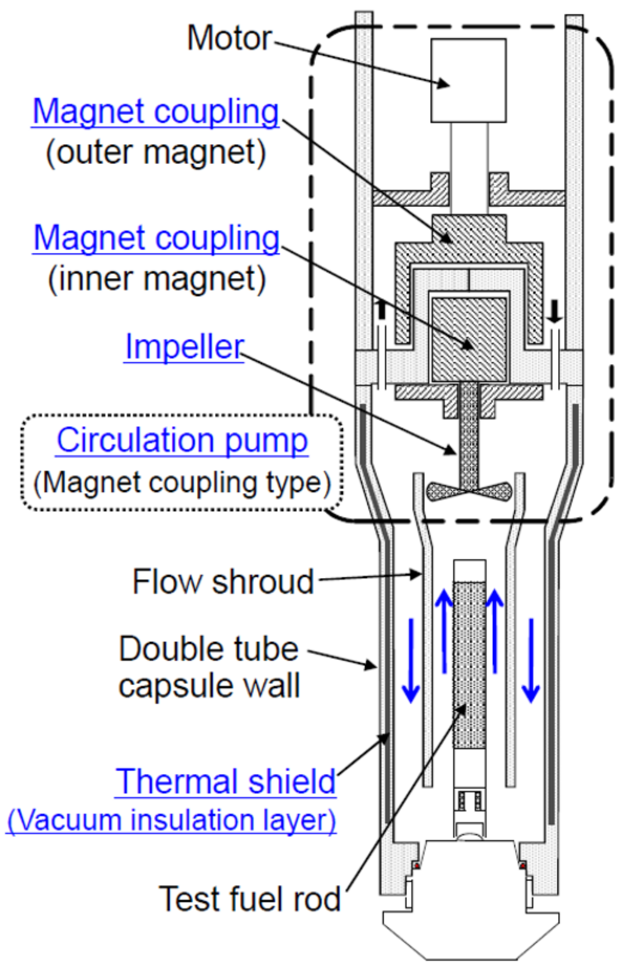


Figure 2. Pumped capsule concept for JMTR [7].

2.1 Motivation

This report describes the analytic and prototyping studies to assess the viability of the mechanical design, thermal-hydraulic behavior, and fuel performance considerations for the P-TWIST concept. This work was supported by the Advanced Low-Enriched Uranium project to determine the P-TWIST's suitability for testing advanced low-enriched uranium fuel concepts. One such fuel concept is essentially a modification of the standard UO_2 pellet design where the pellets are shorter "wafers" and sandwiched between thin metallic molybdenum discs [8]. This conductive insert approach could transfer heat more efficiently to the cladding surface, resulting in reduced fuel temperatures, which, in turn, can favorably affect steady-state fuel performance phenomena such as fission gas retention. This sandwich fuel design would also behave differently in transient conditions, especially with regard to thermal transport and resulting hydraulic and cladding mechanical behaviors. Fast overpower RIA events, for example, would see a brief moment of increased peak surface-cladding heat flux due to a more rapid heat conduction from the fuel stack. Slower events, such as power ramps and power cycles, would exhibit reduced radial average temperatures in the fuel and thus less severe PCIs. These examples illustrate the importance of manipulating surface heat transfer conditions to simulate reactor environments for advanced fuel designs. The fuel system modeled in our feasibility studies was standard LWR fuel because model inputs and comparison cases were readily available to assess P-TWIST's performance. Future studies should investigate P-TWIST's performance with the sandwich fuel concept and other advanced and alternate fuel designs to support more detailed experiment designs.

3. MECHANICAL DESIGN

3.1 Experiment Design Overview

3.1.1 Experiment Containment

Previous TREAT experiments have utilized the Minimal Activation Retrievable Capsule Holder system. This innovative approach irradiates capsules in a reusable containment structure, the Broad Use Specimen Transient Experiment Rig (BUSTER), so capsules can be easily installed and extracted to lower costs and accelerate logistics.

Experiments irradiated in BUSTER are limited in size due to its geometric constraints. To accommodate larger experiments, an enlarged containment structure, known as Big-BUSTER, was developed. Due to the size requirements of a flowing water capsule, P-TWIST will utilize the Big-BUSTER containment system for irradiation in TREAT. See Figure 3 for a schematic of the P-TWIST module and Big-BUSTER containment system.

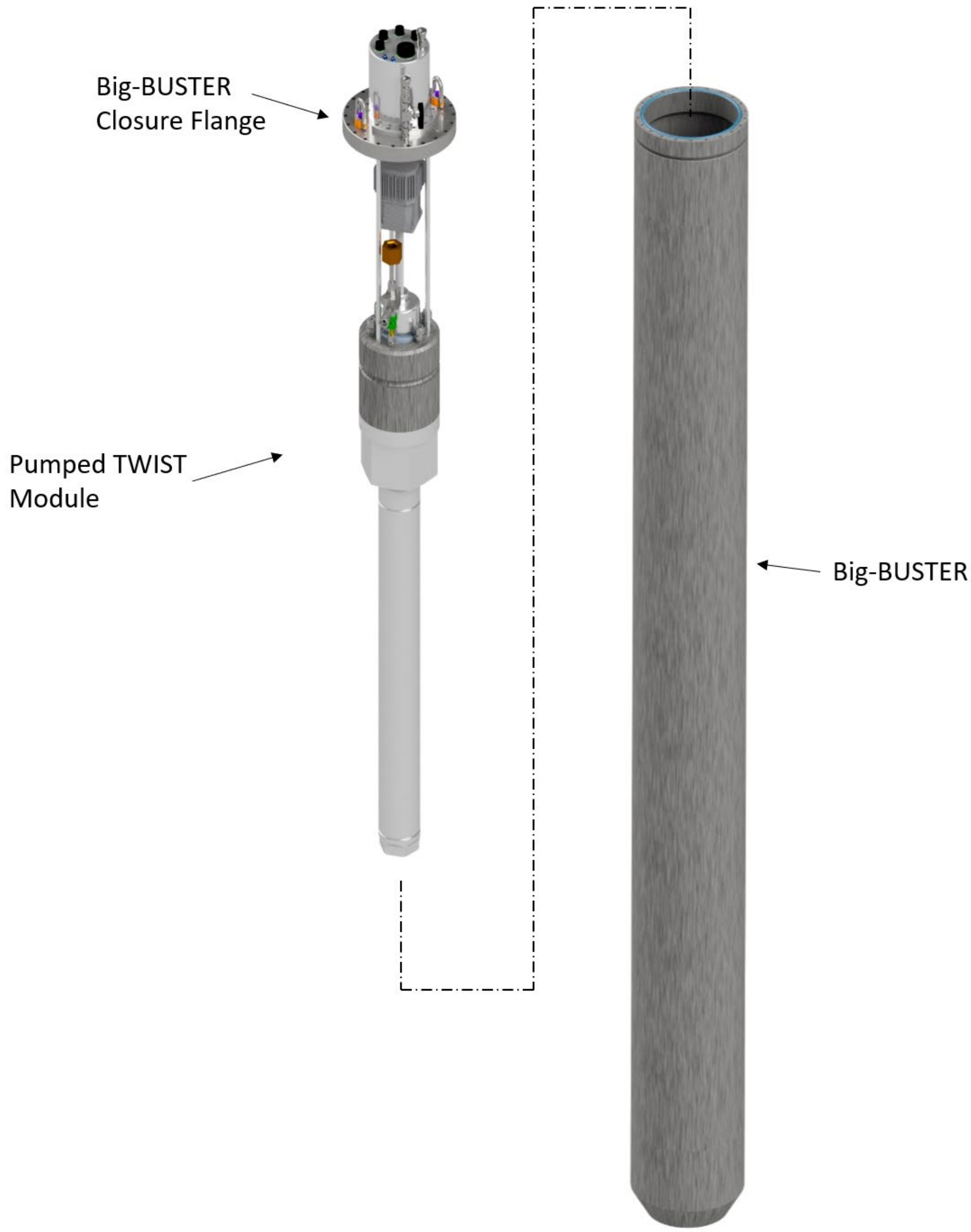


Figure 3. Schematic of the P-TWIST experiment module and Big-BUSTER containment system.

3.1.2 Experiment Module

The P-TWIST experiment module consists of a top closure flange, which forms the Big-BUSTER primary containment and is credited with serving an pressure containment function. The closure flange also features compression seal fittings, which allows instrumentation leads to pass through for further routing near the reactor core.

The capsule is secured to and supported by the closure flange and is not considered a code-compliant pressure vessel. An electric motor is located between the closure flange and capsule and provides torque to a co-axial magnetic coupler on the top of the capsule.

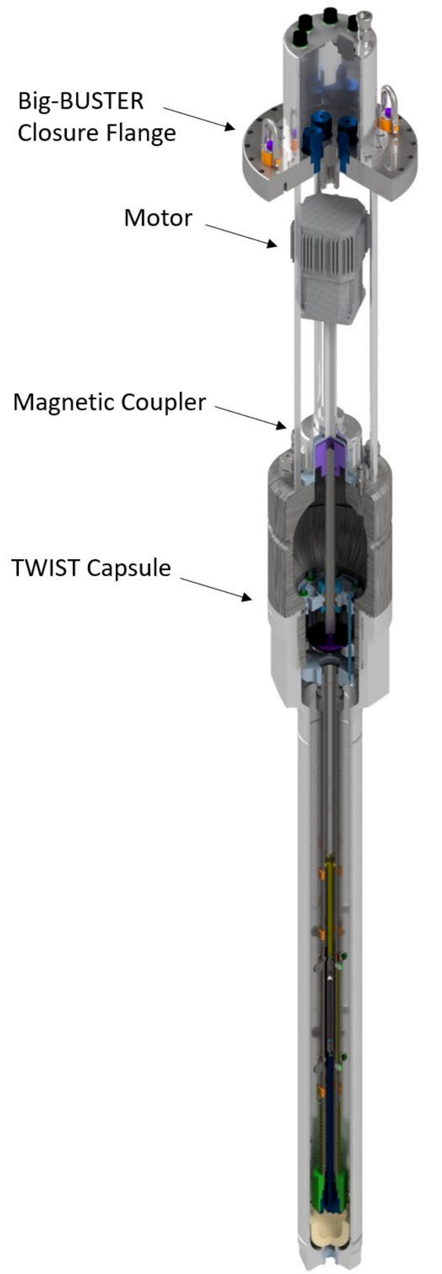


Figure 4. Schematic of the P-TWIST experiment module.

3.1.3 Capsule Assembly

The capsule assembly consists of two 316 stainless-steel weldments that thread together to make the main body of the capsule. The capsule top weldment features a co-axial magnetic coupler, which transfers torque from the motor to the impeller shaft inside the sealed capsule. The capsule top also contains compression seal fittings, which allows instrumentation leads from thermocouples, boiling detector plates, and other instrumentation to pass through the capsule top. The capsule bottom contains features O-ring glands that seal the capsule assembly when the two weldments are threaded together.

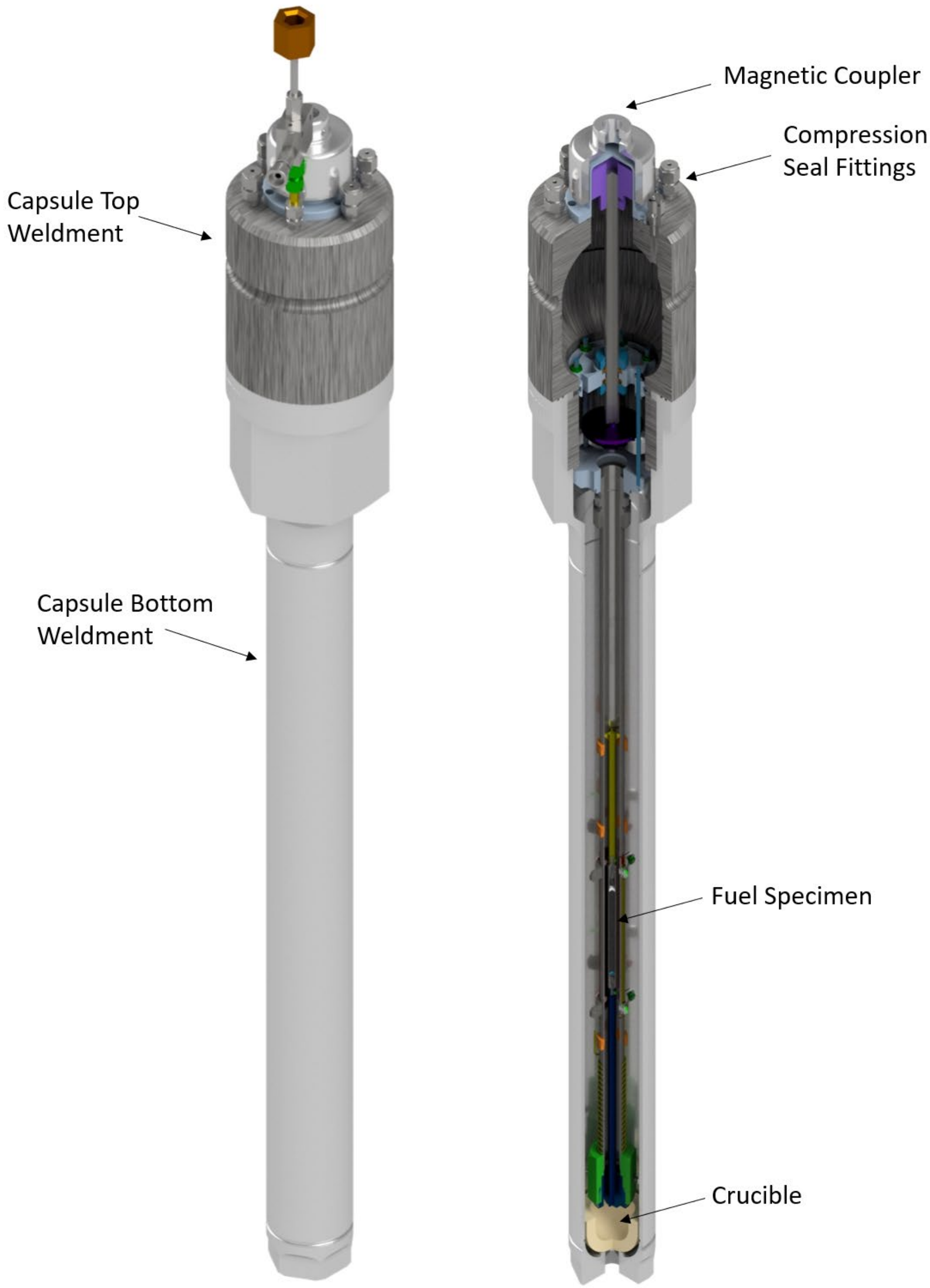


Figure 5. Schematic of the P-TWIST capsule assembly.

3.1.4 Capsule Insert Assembly

The capsule insert assembly is attached to the capsule top weldment and contains the internal components of the pumped capsule. An impeller shaft extends from the magnetic coupler on the top of the capsule down to the impeller, which sits directly above a flow tube. The flow tube houses the fuel specimen, instrumentation, and associated hardware. Other hardware included in the capsule insert assembly provides support for the internal components while preserving flow paths and instrumentation clearance. During assembly, the capsule insert is secured to the capsule top and inserted into the capsule bottom to complete the capsule assembly.

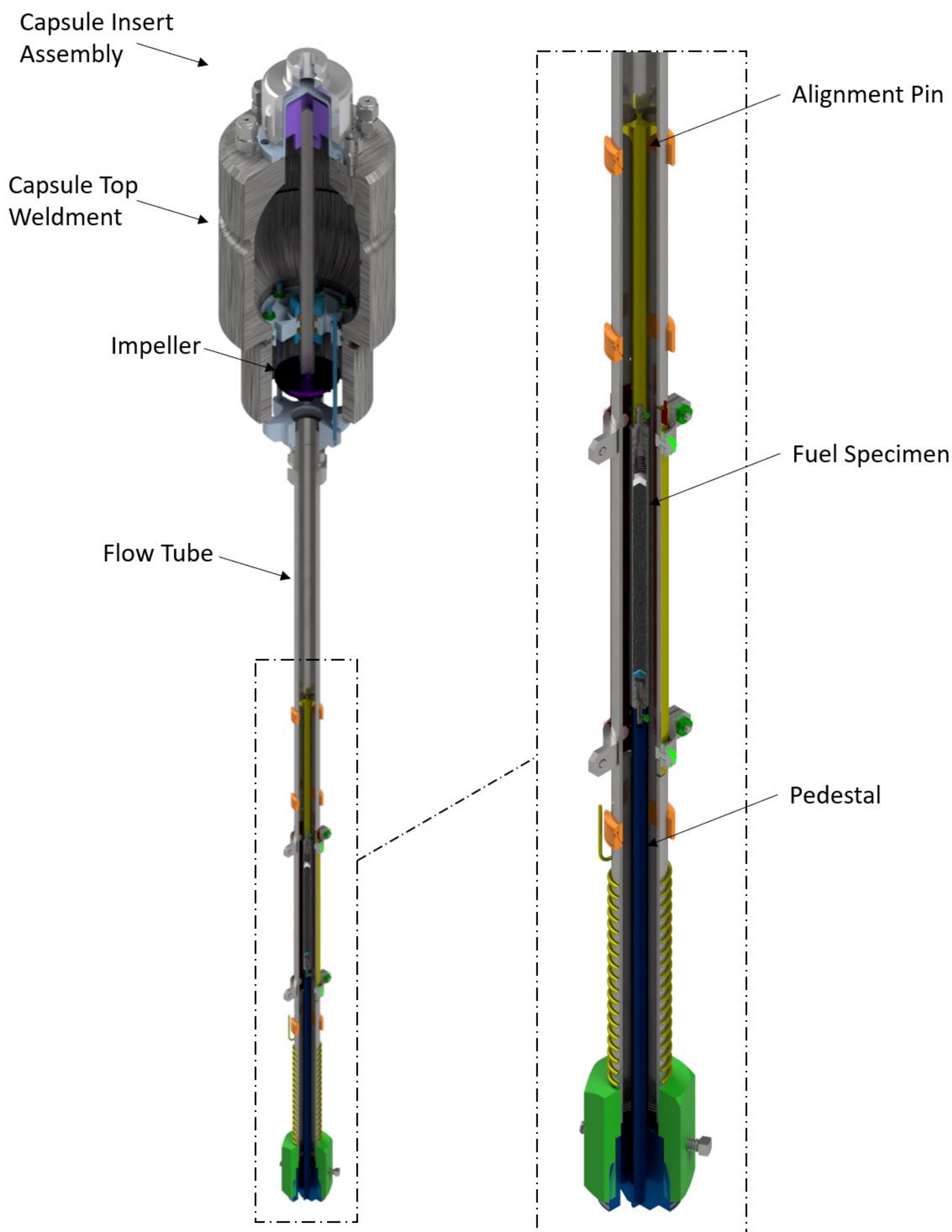


Figure 6. Schematic of the P-TWIST capsule insert.

3.1.5 Specimen Assembly

The P-TWIST experiment is designed to accommodate various fuel specimen types and geometries, including LWR, boiling-water reactor (BWR), and small modular reactor fuel pins. Specimens are assembled to an alignment pin on top to center the rodlet in the flow tube and a pedestal on bottom to provide the rodlet with the correct axial location during irradiation. The specimen assembly is inserted in the bottom of the flow tube where it threads in place and is secured. Figure 7 shows the basic components of a conceptual specimen assembly that can be modified to accommodate multiple different fuel specimen types and lengths.

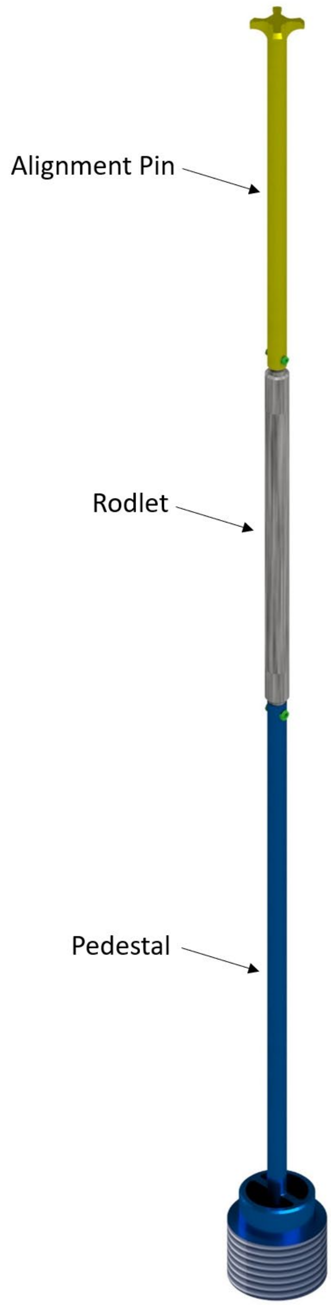


Figure 7. Schematic of a conceptual specimen assembly.

3.2 Flow Paths

As discussed above, the P-TWIST module contains a motor that provides torque to a co-axial magnetic coupler during irradiation. This coupler transfers the torque through the sealed capsule to the impeller located below the water level above a flow tube containing the fuel specimen. The impeller creates an upward flow in the flow tube surrounding the fuel specimen. The water recirculates around the outside of the flow tube with a downward flow in the outer annulus to create a constant flow around the specimen. Figure 8 shows a schematic of the P-TWIST flow paths.

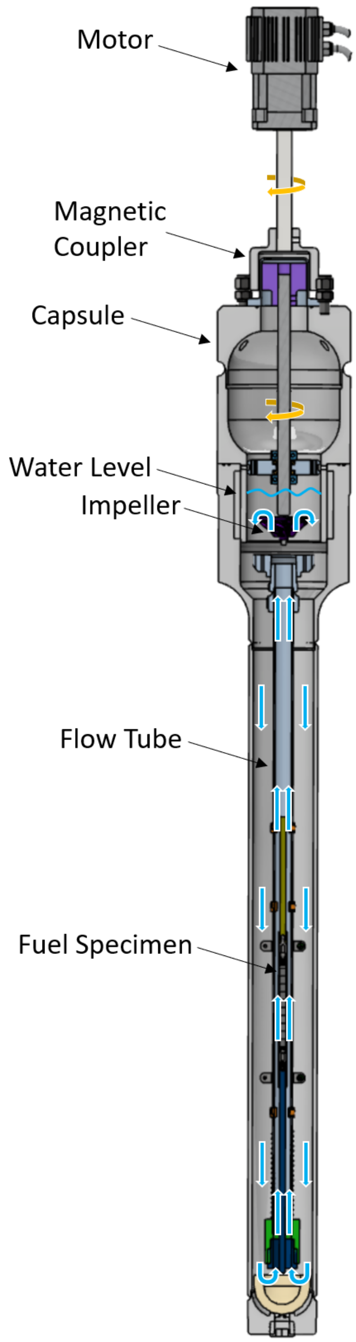


Figure 8. Schematic of the flow paths through the P-TWIST capsule.

3.3 Pressurized-Water Reactor Conditions

Modifications made to the P-TWIST conceptual design have been considered to reach prototypic pressurized-water reactor (PWR) conditions. Minor modifications made to the capsule and included components would allow the starting temperature and pressure inside the capsule to be increased prior to irradiation.

The original TWIST capsule design featured capsule blowdown internal notches on the bottom end of the lower capsule weldment. These notches increased the hydraulic area between the capsule and the internal crucible to facilitate the faster blowdown of the TWIST capsule. These internal notches decrease the wall thickness of the capsule pipe, decreasing the allowable pressure inside the capsule. These notches will be removed from the P-TWIST capsule design to allow a higher pressure inside the capsule.

As discussed and pictured above, the TWIST capsule weldments attach to each other via large threads. Modifications to the capsule design, including changing the capsule mating components to a blind flange, may also be implemented in the P-TWIST capsule design moving forward.

Internal capsule temperature can be increased prior to irradiation with the addition of electrical heaters in the capsule insert design. Cable heaters can be placed around the flow tube to bring the starting temperature to prototypic PWR operation conditions.

3.4 Concept Testing

To test the feasibility of the pumped capsule concept, Idaho National Laboratory sponsored a Brigham Young University-Idaho engineering senior undergraduate design capstone project to develop a working prototype and achieve a proof of principle that a small impeller could be oriented directly above a flow tube and achieve a meaningful flow rate.

The prototype utilized a 3D-printed impeller coupled to a motor fixed on top of the prototype assembly. Flow was tracked optically using neutrally buoyant beads and a high-speed camera. Results showed that the prototype, which had nominal dimensions consistent with the P-TWIST capsule concept, could reach and surpass suction through the flow tube of 2 m/s flow velocity around the fuel pin.

The prototype also revealed several items to address in the design moving forward. First, the prototype showed that the impeller could cause vortices on the free water surface and draw air bubbles into the water resulting in unstable fluidic conditions. The prototype also showed the importance of rotational alignment of the components in achieving the maximum coolant flow rate. Thrust forces from the impeller were also found to have a significant effect on the prototype.

Additional capstone projects will provide a pump curve characterization and design a baffle to mitigate the vortex effect. The students will design and build a test apparatus to capture critical hydraulic features in the coolant path and measure flowrate, pressure change, and motor rpm to create a pump curve for the pumped capsule. Students will also design and build a baffle with features, such as bearings, to mitigate the vortex effect and thrust forces and to determine the maximum achievable flowrate in the pumped capsule design. Additional designing and testing will be completed on the co-axial magnetic coupler to determine a containment barrier that works under PWR conditions.

The university capstone projects provide valuable information to help drive decisions and progress designs quickly. Ultimately, these projects lead to a fully functioning out-of-pile instrumented prototype designed and built by INL. This prototype built by INL will be fabricated with materials and tolerances prototypic to the pumped capsule and will be instrumented to gain performance data on the pumped capsule prior to beginning irradiations.

4. THERMAL-HYDRAULIC SCOPING

The forced convective coolant conditions inside the P-TWIST capsule may lend to more prototypic conditions for a variety of different transient testing, but as with all testing, there are trade-offs to consider. Are the benefits for forced convective cooling worth a more expensive experiment or a lack of instrument capabilities? To help provide insight into some of these trade-offs, we performed thermal-hydraulic modeling of the P-TWIST capsule under various transient scenarios. The thermal-hydraulic studies used RELAP5-3D because it provides the best combination of thermal-hydraulic modeling capabilities and providing temperature predictions for the fuel rod. If more detailed thermo-mechanical predictions of the fuel rod are of interest for the different transient studies, the boundary conditions predicted by RELAP5-3D can be incorporated into a Bison fuel performance model.

The P-TWIST capsule design described above in Figure 3–Figure 8 was converted into a RELAP5-3D model depicted by the nodalization diagram in Figure 9. The RELAP5-3D model incorporates a 10-pellet LWR fuel rod (Heat Structure 210) in a flow channel (Volume 111) sized so the hydraulic diameter was the same as a PWR fuel rod in a 12.6-mm rod pitch bundle. We chose the 10-pellet fuel rod for the modeling to reduce the total energy that goes into the capsule. The current capsule design has a finite volume of water that will be repeatedly circulated, and some transients of interest may require testing for a few minutes. A long fuel rod would increase the temperature of the water, which could change the thermal-hydraulic conditions throughout the transient.

4.1 Steady-State

Typical PWR parameters for a (as defined in Appendix K of Reference [9]) specify the average coolant velocity at the interior of a fuel bundle is 5.2 m/s. We performed a steady-state study with the RELAP5-3D model to see the predicted impact to fuel rod heat flux and fuel and cladding temperatures at varying coolant velocities. At this point, the pump and impeller design for the capsule were not defined, so the velocity in the RELAP5-3D model was directly specified at the exit of the flow tube at the desired velocity.

The steady-state study looked at three different coolant initial temperature and pressure conditions. Nominal PWR conditions (280°C at 15.5 MPa), room temperature and pressure (25°C and 1 atmosphere), and similar subcooling as PWR conditions (200°C at 3.45 MPa). The latter two conditions have been used as initial conditions for recent water-based tests performed at TREAT. The predictions are shown in Figure 10. The fuel and cladding temperatures show little change with decreasing coolant velocity, but the largest changes are seen in the coolant heat transfer coefficient and heat flux plots. Figure 10c shows the heat transfer coefficient decreases slightly until 2 m/s and then it gradually increases, and in some cases increases significantly. The increase below 2 m/s is because the mode of heat transfer changes from forced convection to nucleate boiling. Based on the results from the steady-state models, a coolant velocity greater than 2 m/s gives similar temperatures and thermal-hydraulic performance to nominal PWRs at 5 m/s and avoids a change in the heat transfer mode for this length of a fuel rod.

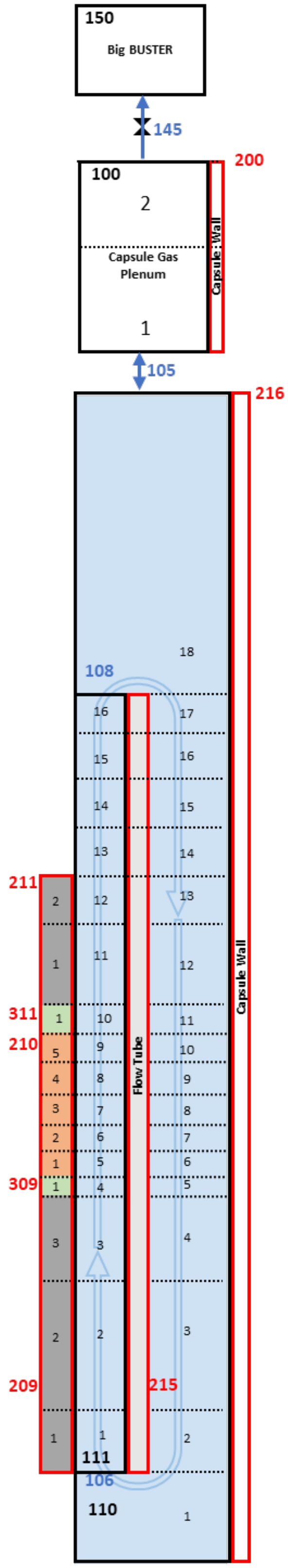
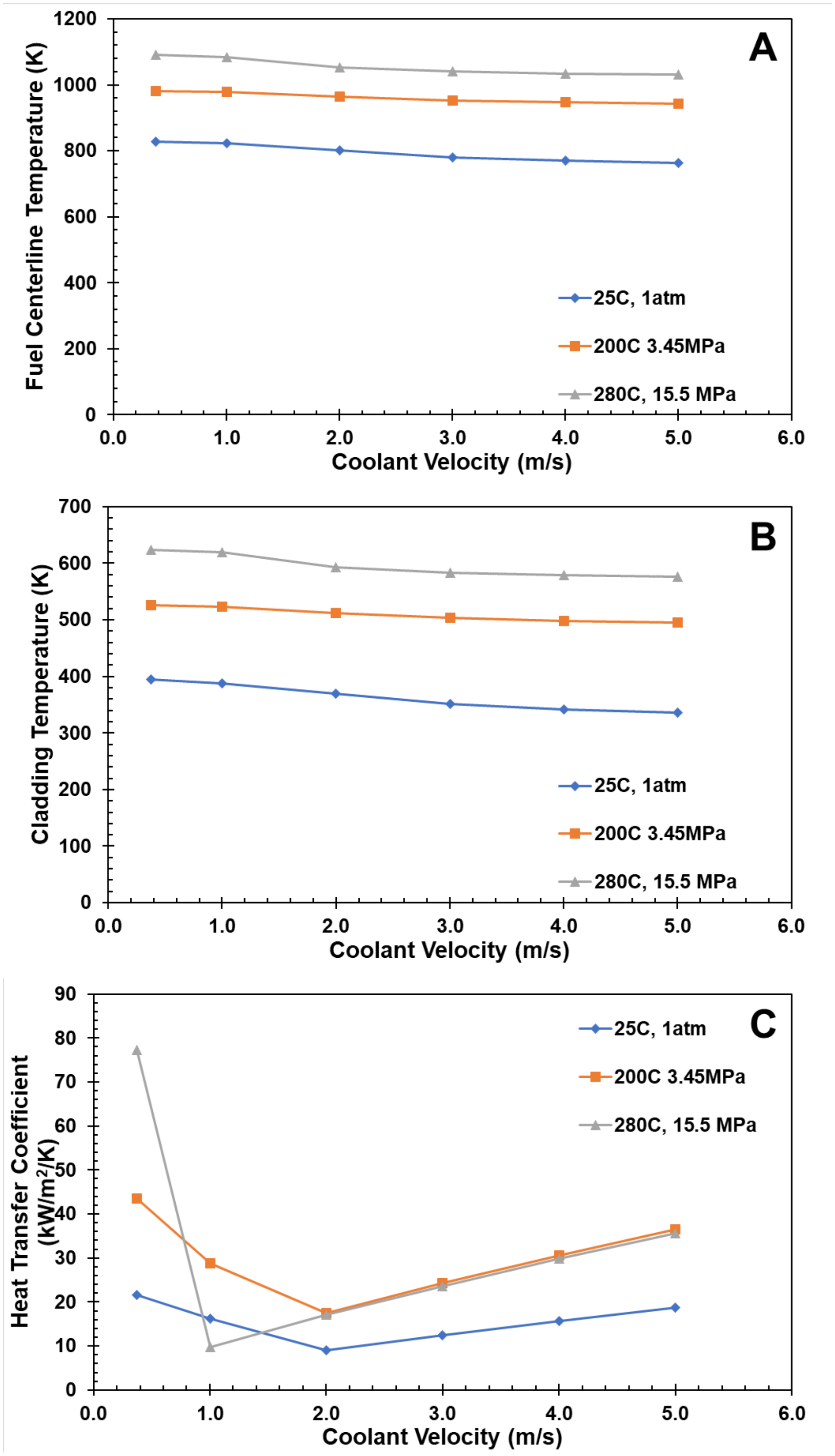


Figure 9. RELAP5-3D nodalization diagram for the P-TWIST capsule design.


A
Fuel Centerline Temperature (K)
Coolant Velocity (m/s)
25C, 1atm
200C 3.45MPa
280C, 15.5 MPa
B
Cladding Temperature (K)
Coolant Velocity (m/s)
25C, 1atm
200C 3.45MPa
280C, 15.5 MPa
C
Heat Transfer Coefficient (kW/m2/K)
Coolant Velocity (m/s)
25C, 1atm
200C 3.45MPa
280C, 15.5 MPa

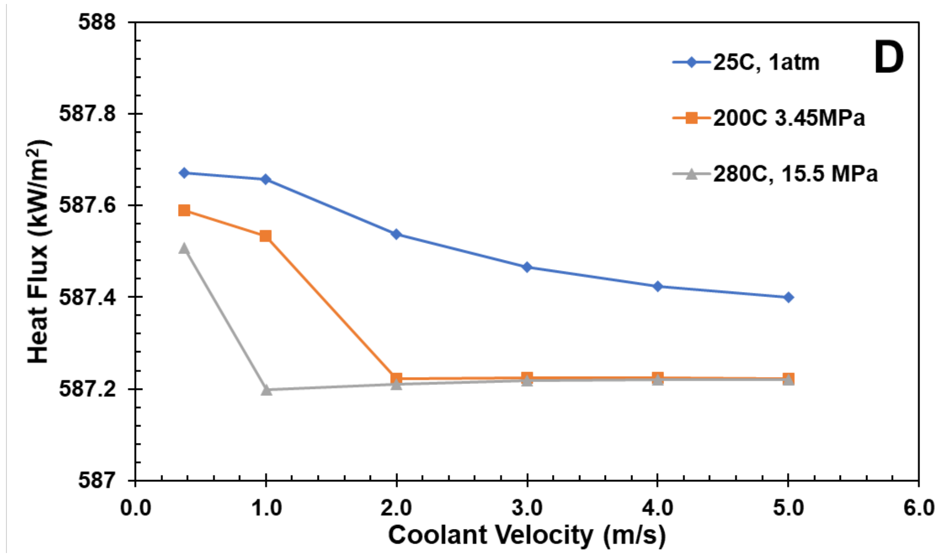


Figure 10. Steady-state predictions of fuel centerline temperature (a), cladding temperature (b), coolant heat transfer coefficient (c), and fuel rod heat flux (d).

4.2 Boiling-Water Reactor Anticipated Operational Occurrence

One scenario of interest is the ability to simulate an anticipated operational occurrence in a BWR, such as a turbine trip. These accidents typically result in a power excursion due to pressure increases in the core affecting the coolant void feedback coefficient. The increase in power and decrease in flow during a turbine trip can cause the fuel and cladding temperatures to increase significantly, and dryout events may occur.

The path to dryout in a BWR is different than a departure from nucleate boiling (DNB) event in a PWR. Since the water is close to the saturation point, the transition to dryout is an integral effect that takes place over the length of the fuel rod whereas DNB can be a locally driven effect. Achieving dryout conditions on a short rodlet in P-TWIST is not possible, therefore achieving a mild cladding temperature excursion (less than is seen in an RIA discussed in Section 4.4) requires being able to control the reactor power and coolant flow simultaneously to achieve DNB for a short period of time.

The difficulty with trying to achieve cladding temperature excursions at ~800°C is that DNB events usually overshoot to above 1200°C due to the high temperatures and stored energy in the fuel (see Figure 11), or in cases where DNB does not occur, the cladding temperatures remains ~300°C. By controlling the reactor power and flow rate simultaneously, the power could be reduced while the flow rate was decreased. This allows the fuel temperature and stored energy to be reduced while allowing the coolant conditions to transient into DNB.

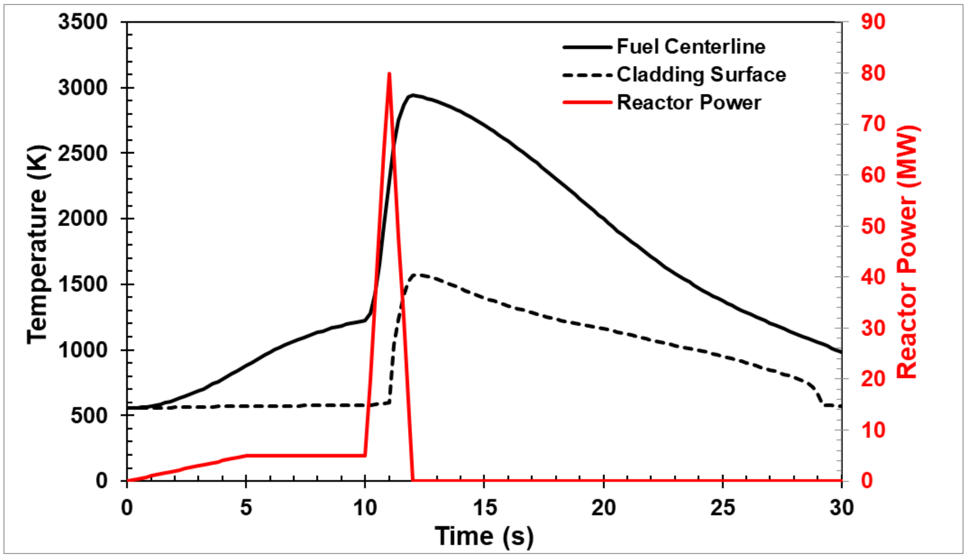


Figure 11. RELAP5-3D predictions of fuel and cladding temperatures when DNB is reached.

The following simulation is an example where the reactor power and coolant velocity are controlled together to drive a DNB event without a significant increase in cladding temperature (Figure 12). In this case, the initial coolant conditions inside the capsule were 510 K (237°C) at 3.45 MPa (4.7 K subcooled) flowing at 2 m/s. The cladding reached a peak temperature of 756°C and boiled for ~15 seconds. This case results in very high fuel temperatures, so Figure 13 shows another example where the reactor power reaches 10 MW and the coolant velocity is ramped down faster. This reduces the peak fuel temperature while still achieving peak cladding temperatures of 800°C with the temperature excursion lasting for 17 seconds. If longer durations under DNB are required, the reactor power can be modified so the cladding stays at elevated temperatures longer. An example of modification is shown in Figure 14, where the boiling duration is extended to just under 40 seconds.

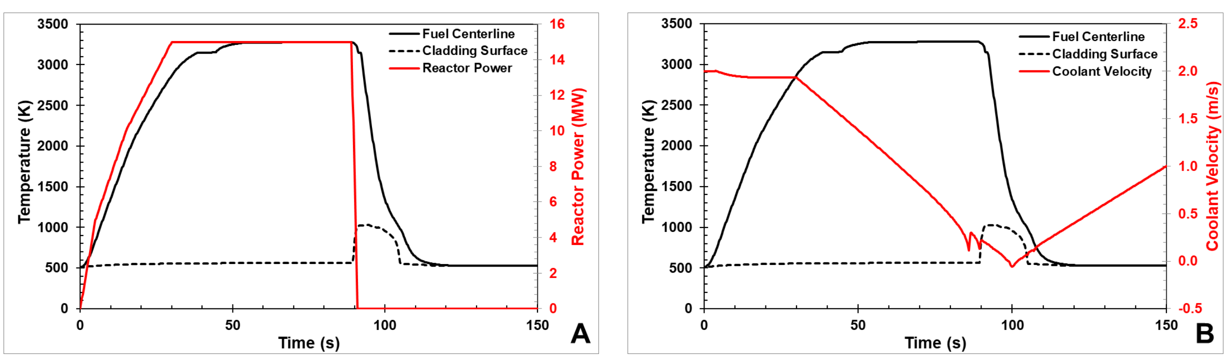


Figure 12. RELAP5-3D predictions of fuel and cladding temperatures with reactor power operating up to 15 MW (a) and coolant velocity initially at 2 m/s (b) when DNB is reached when controlling both reactor power and coolant flow.

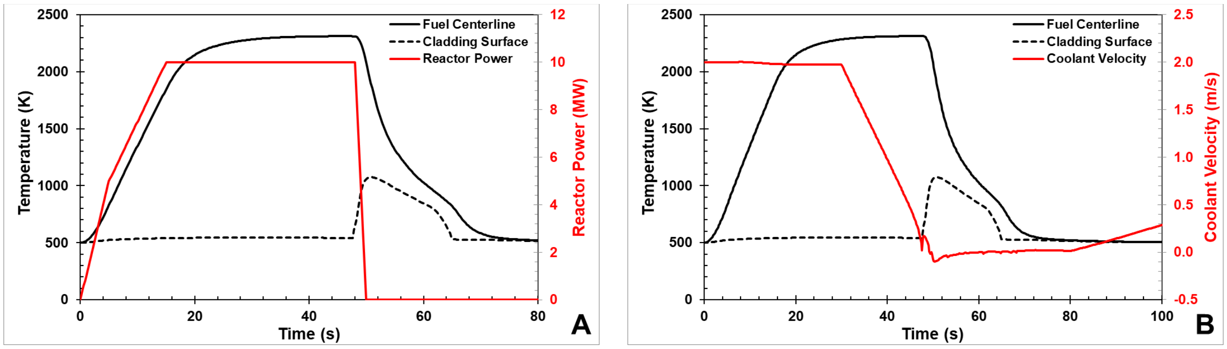


Figure 13. RELAP5-3D predictions of fuel and cladding temperatures with reactor power operating up to 10 MW (a) and coolant velocity initially at 2 m/s (b) when DNB is reached when controlling both reactor power and coolant flow.

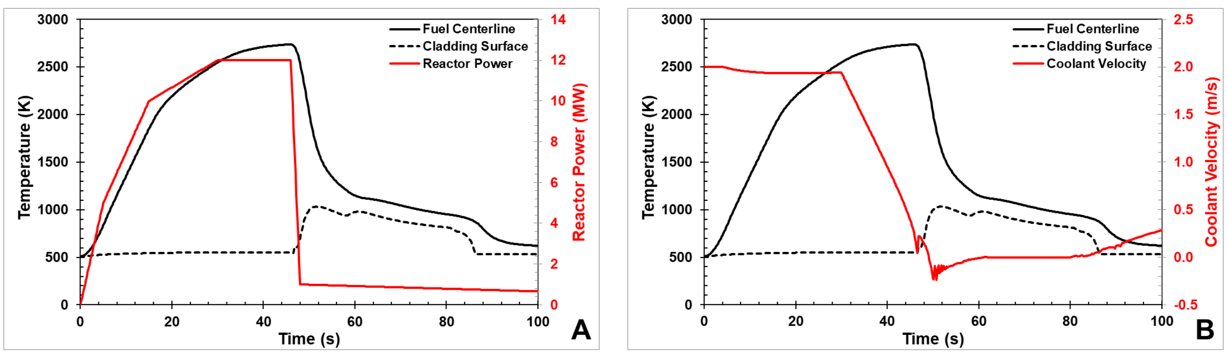


Figure 14. RELAP5-3D predictions where the reactor power is extended to lengthen out the time under DNB.

An alternative method to creating the desired conditions is to start the coolant with no flow and increase the velocity throughout the transient to end DNB. This example is shown in Figure 15, where the reactor power is kept much lower (5 MW) but still high enough to experience DNB. During this transient, the flow around the rodlet is increased to a point that allows the rodlet to be quenched. This example results in cladding temperatures of 810°C and the boiling duration just under 10 seconds.

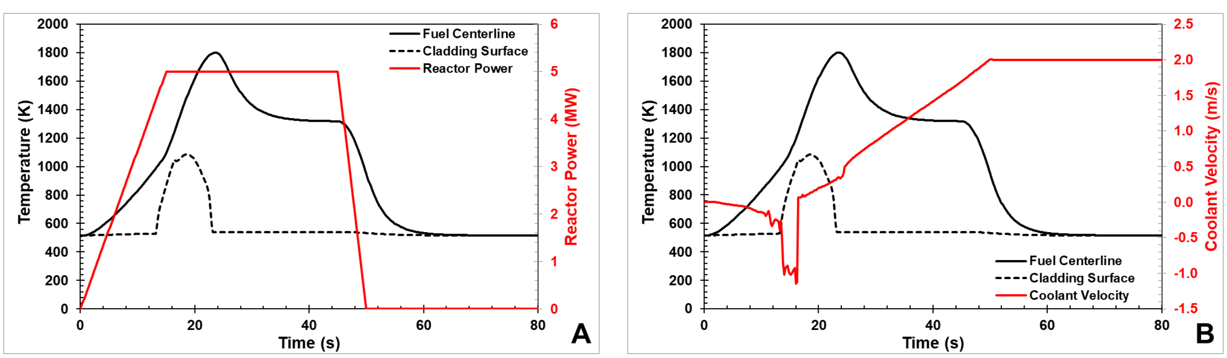


Figure 15. RELAP5-3D predictions of fuel and cladding temperatures using an alternative method of increasing the flow during the transient to end DNB.

These models show that, with careful control of the reactor power and coolant flow around the rodlet, mild temperature excursions can be achieved, which is not possible with the current static water capsule.

This method will require a proper understanding of the pump characteristics and how fast the flow can be developed or changed during a transient. The performance of the pump when voids occur will also need to be better understood. In all the examples, the coolant velocity becomes unstable or has a short period of downward velocity when the flow is slow and DNB occurs. Current understanding of how the pump will perform during these events is unknown.

4.3 Light-Water Reactor Power Cycling

Another transient testing scenario of interest for LWR and other advanced fuel concepts is the ability to do multiple power cycles. An LWR example would be load following, where the fuel rod power is cycled up and down changing the stress state of the cladding which may lead to PCMI failures. Effectively performing power cycling transients in TREAT is a compromise between a number of competing parameters. TREAT is capable of performing almost any variety of transient shapes until the reactor runs out of reactivity, which typically occurs once the reactor core has generated ~2,400 MJ. The reactivity in TREAT is limited by the temperature of the reactor core, and the reactor heats up adiabatically. Once the transient has completed, the reactor core must be cooled down to room temperature before another transient can be run. In the past, the reactor has shown it is capable of performing two transients in a single day. Therefore, the number of cycles will be dictated by the magnitude and duration of the power cycle.

There is an infinite number of transient shapes to choose from, and in this scoping study, we investigated a number of different combinations of transient shapes including triangle shaped, symmetric fast ramp to a flat top, asymmetric slow ramp to a flat top (examples of these shapes are shown in Figure 16). The two flat top transients achieved very similar fuel temperature profiles (see Figure 17), but the slower ramp case resulted in more reactor energy per cycle and the triangle shaped transients required ~40% more power to achieve a similar fuel temperature. For the remainder of this discussion, our study used the symmetric flat top transient for further detail investigations and comparisons in the P-TWIST capsule. All the simulations shown moving forward use coolant at 200°C at 3.45 MPa. Results at PWR conditions were very similar to the 200°C cases except with higher temperatures due to the higher initial temperature.

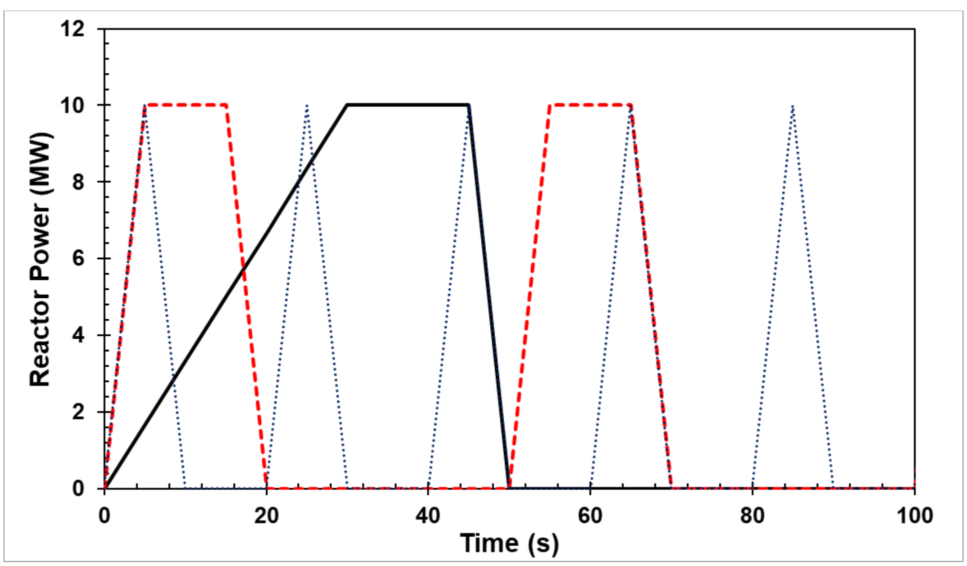


Figure 16. Examples of power cycling transients.

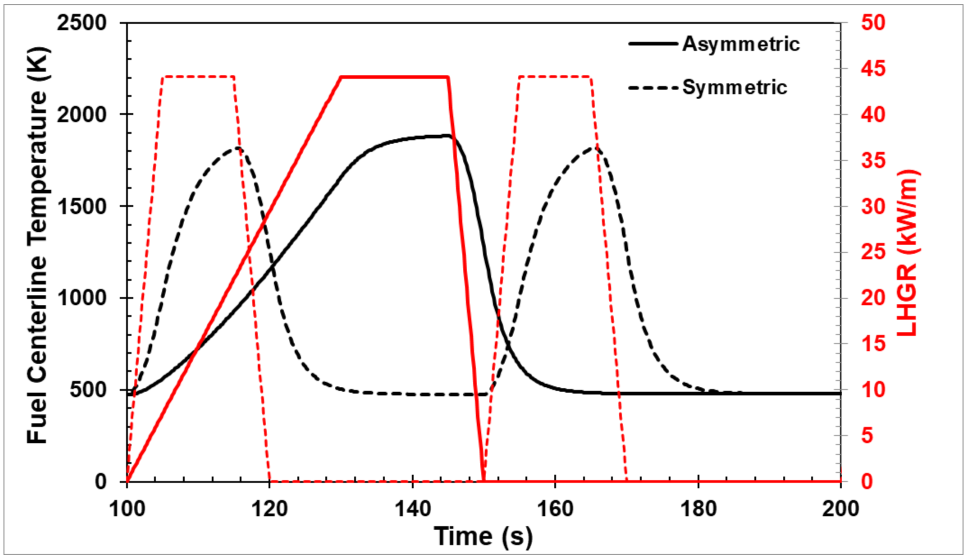


Figure 17. Fuel temperature predictions comparing different transient shapes.

The P-TWIST capsule in TREAT will provide capabilities not possible with a static water capsule. Without convective cooling and the larger volume of water available in the P-TWIST capsule, compared to the smaller Static Environment Rodlet Transient Test Apparatus (SERTTA) capsule, the fuel rod quickly goes into DNB after a few cycles due to the heat up of the water in the capsule. Comparisons were made between different coolant velocities (2 m/s vs. 4 m/s) in the P-TWIST capsule, and the difference between the peak fuel temperature was within one degree between the two velocities and between the cladding temperatures. Therefore, we used a 2 m/s coolant velocity in the flow tube around the fuel rod for the remaining simulations.

The symmetric transient shaped compared the fuel and cladding temperatures for increasing peak linear heat generation rates (LHGR), shown in Figure 18. Due to the short nature of the transients, larger LHGRs are needed to achieve representative fuel temperatures. For these simulations, the fuel has an enrichment of >10% U^{235}, which allows a more favorable coupling between the fuel rod power and the TREAT core power. The power coupling factor (PCF) was assumed to be 10 W/g/MW. Under these conditions, TREAT can operate up to 8 MW to achieve an LHGR of 44 kW/m, which allows ~19 cycles per transient. Assuming three cycles could run in a 24-hour period, ~575 total cycles could be executed in 10 days. The number of cycles reduces to 460 for 10 MW peak TREAT power and 380 for 12 MW. If 4.9% U^{235} fuel were used, the PCF would be approximately 3.5 W/g/MW, and to achieve the same temperatures shown Figure 18, TREAT would have to operate at 23 MW, 29 MW, and 34 MW, which would reduce the total number of transients to 200, 155, and 135, respectively.

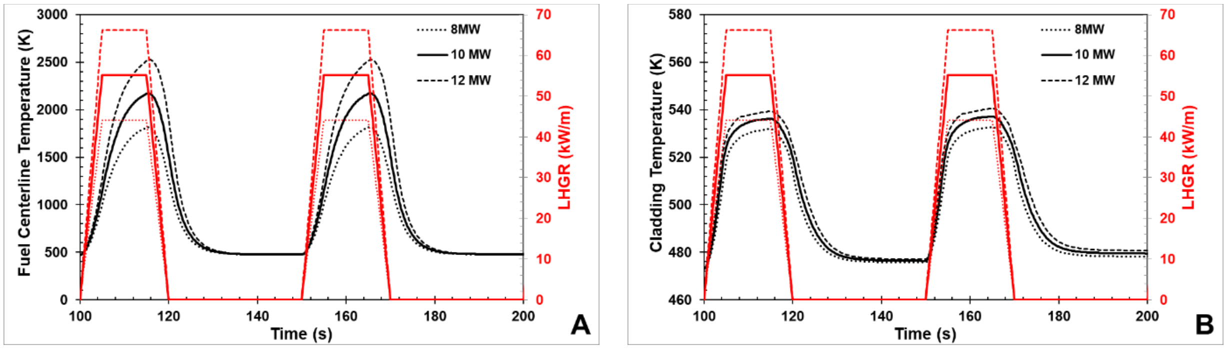


Figure 18. Fuel centerline (a) and cladding surface (b) temperature predictions with varying peak LHGR transients.

The large volume of water in the P-TWIST capsule allows for longer operations (allows more cycles) before the coolant temperature increases to the saturation temperature. The coolant temperature and pressure for the 10 MW case in Figure 18 is shown in Figure 19. The coolant temperature increases ~45 K and the pressure increases 1.1 MPa, but the coolant still remains subcooled by 15 K (initially at 42 K). These simulations are assuming a fuel rod with 10 pellets; if a longer fuel rod is used, the number of cycles may have to be reduced to prevent water approaching the saturation temperature or as a means to cool the water during the transient.

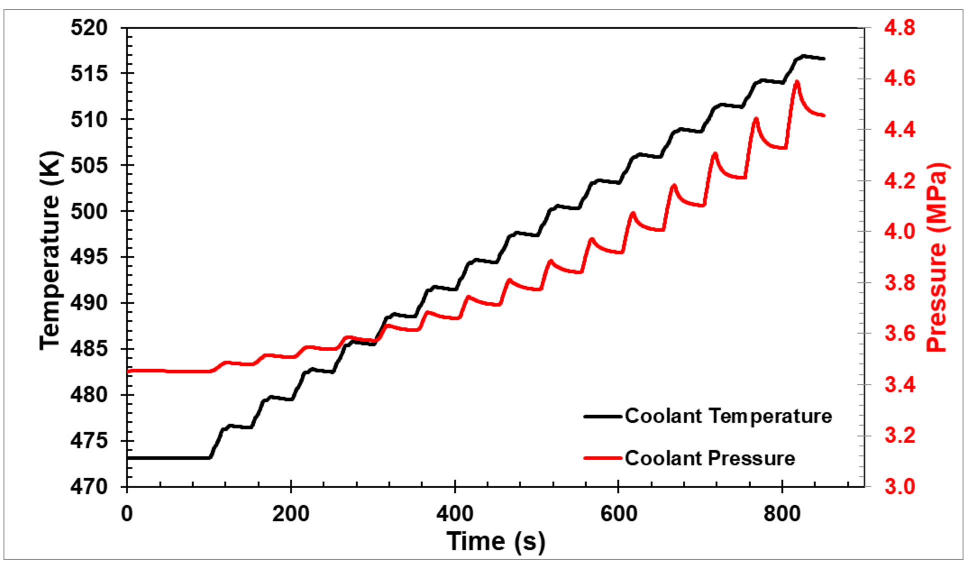


Figure 19. Coolant temperature and pressure increase over multiple cycles.

The fuel and cladding temperature predictions provide a starting assessment of the capsule performance during power cycles but do not provide any information on the stresses seen in the fuel rod throughout the transient. The thermal-hydraulic predictions from RELAP5-3D were supplied as boundary conditions to a Bison fuel performance model to provide more detailed information on the thermo-mechanical performance of the fuel rod. The Bison models can provide more detailed information on the cladding stress state and whether PCMI failure occurs.

The more detailed fuel performance model shows that very little hoop stress occurs in the cladding in the 8 and 10 MW cases (see Figure 20a). The 12 MW power level (~66 kW/m) was needed before higher levels of stress in the cladding was apparent. This was due to the fact that, in the 8 MW case, the fuel-cladding gap never closed (Figure 21a), and thus no significant strains were applied to the cladding from the fuel thermal expansion. Therefore, an understanding of the desired fuel and cladding stress state during the cycles is a necessary part of the cycle shape and duration design.

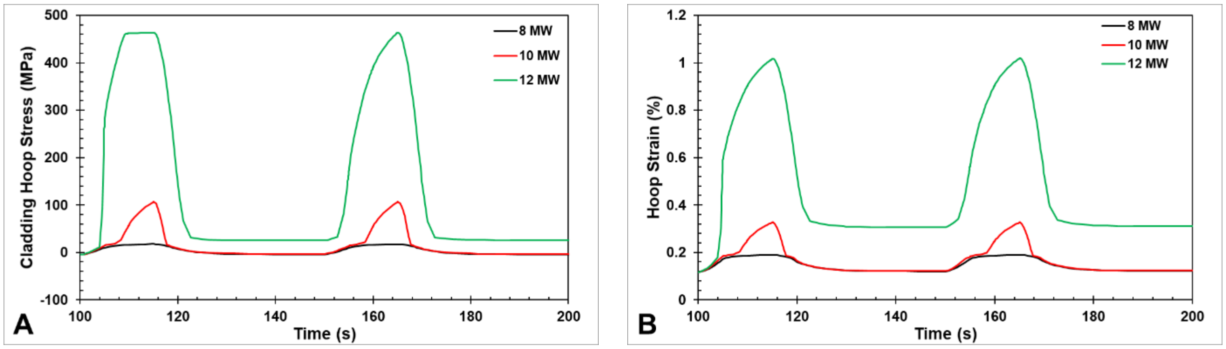


Figure 20. Bison fuel performance predictions of cladding hoop stress (a) and hoop strain (b).

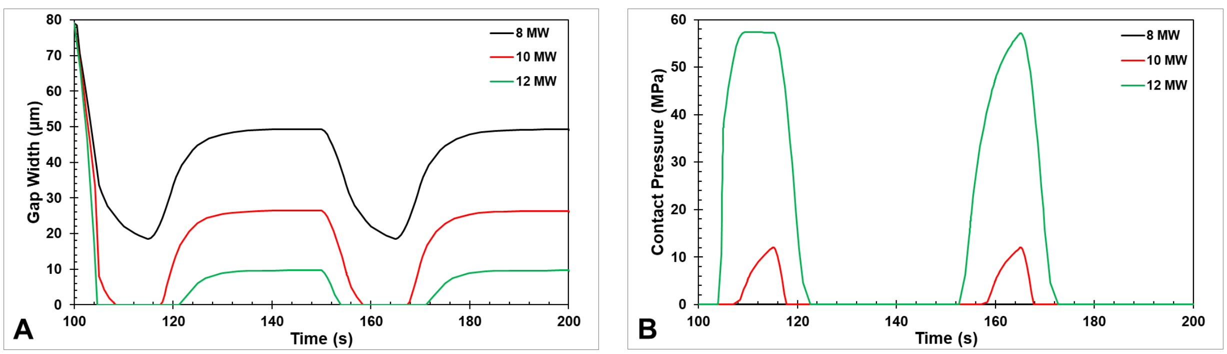


Figure 21. Bison fuel performance predictions of fuel-cladding gap (a) and fuel-cladding contact pressure (b).

4.4 Light-Water Reactor Reactivity-Initiated Accident

RIA is an active area of transient testing at TREAT. The current testing is being performed in a static water capsule (SERTTA capsule). The ability to perform RIA tests with flowing coolant may provide results that are more representative of the conditions in a commercial reactor. RELAP5-3D was used to model an RIA in the P-TWIST capsule at different flow rates compared to a RELAP5-3D model of the static water SERTTA capsule. In all models, a 90-ms full-width-at-half-max Gaussian shaped transient deposited 650 J/gUO_2 into the fuel. Hot zero-power simulations began with initial conditions of 25°C at 1 atmosphere of pressure, 200°C at 3.45 MPa, and PWR conditions of 280°C at 15.5 MPa. In the P-TWIST capsule coolant velocities of 0–5 m/s were used.

The results of the predicted fuel centerline and cladding surface temperatures for the different initial coolant conditions are shown in Figure 22–Figure 24. In all cases, the P-TWIST capsule running with no velocity showed similar results to the SERTTA capsule. All stagnant water cases experienced longer periods of DNB than the flowing water cases. Surprisingly, the 1 m/s case shows the shortest DNB duration for all cases and increasing the coolant velocity increases the boiling duration. This is contrary to expectations, and further investigations into the RELAP5-3D thermal-hydraulic correlations are needed to understand this better. Even though the 1 m/s case showed the shorted boiling duration, the cladding temperature trends near the peak cladding temperature are different than seen in the 2–5 m/s cases, which all show very good agreement until rewet. This is another example that achieving a coolant velocity around the fuel rod of at least 2 m/s gives very similar results to the velocities up to 5 m/s.

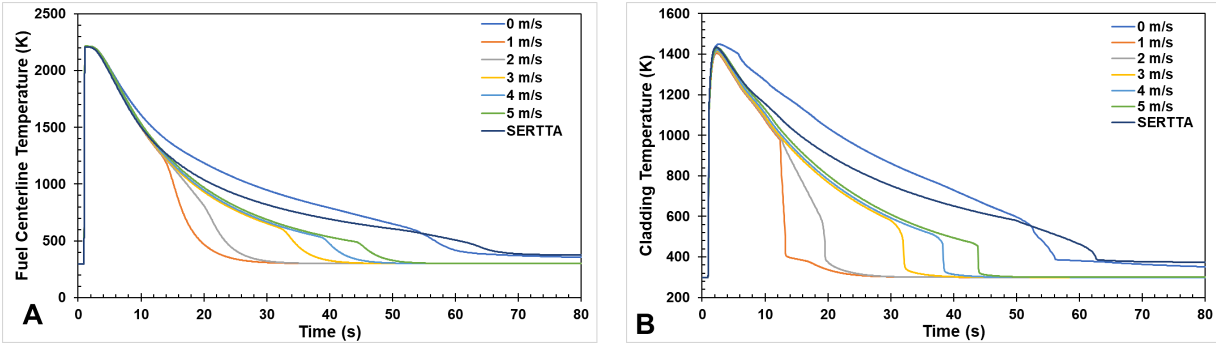


Figure 22. RELAP5-3D predictions of the fuel centerline temperature (a) and cladding surface temperature (b) comparing the P-TWIST capsule and SERTTA capsule at 25°C and 1 atmosphere.

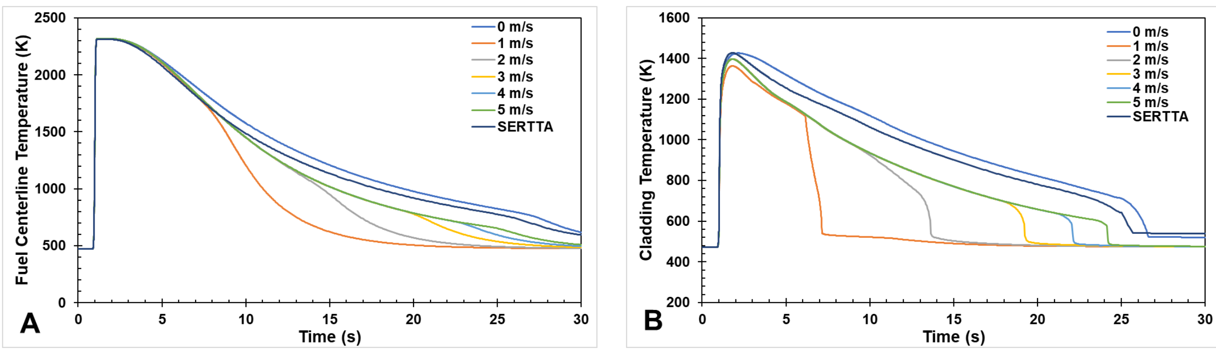


Figure 23. RELAP5-3D predictions of the fuel centerline temperature (a) and cladding surface temperature (b) comparing the P-TWIST and SERTTA capsules at 200°C and 3.45 MPa.

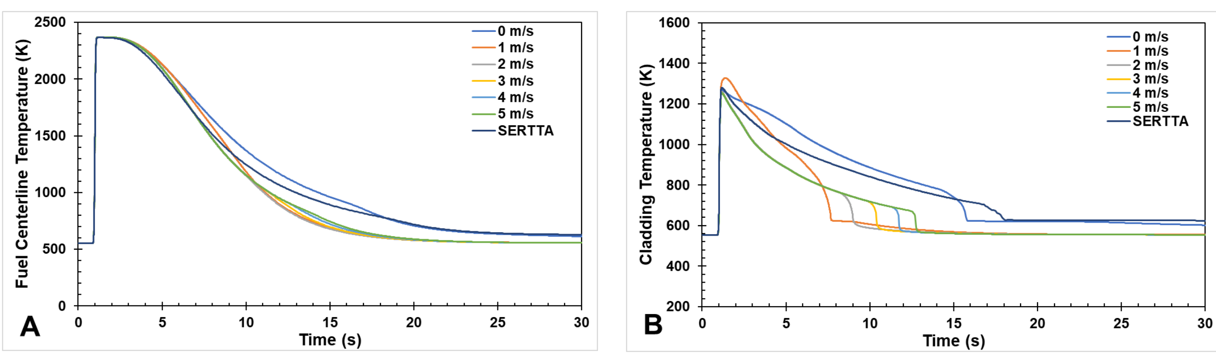


Figure 24. RELAP5-3D predictions of the fuel centerline temperature (a) and cladding surface temperature (b) comparing the P-TWIST and SERTTA capsules at 280°C and 15.5 MPa.

Figure 25 compares the different coolant conditions in the SERTTA and P-TWIST capsules (2 m/s coolant velocity). In all cases, the PWR conditions provide better cooling, keeping cladding temperatures ~150 K cooler than the other cases even though it starts at a higher temperature. The PWR coolant conditions also promote shorter DNB durations.

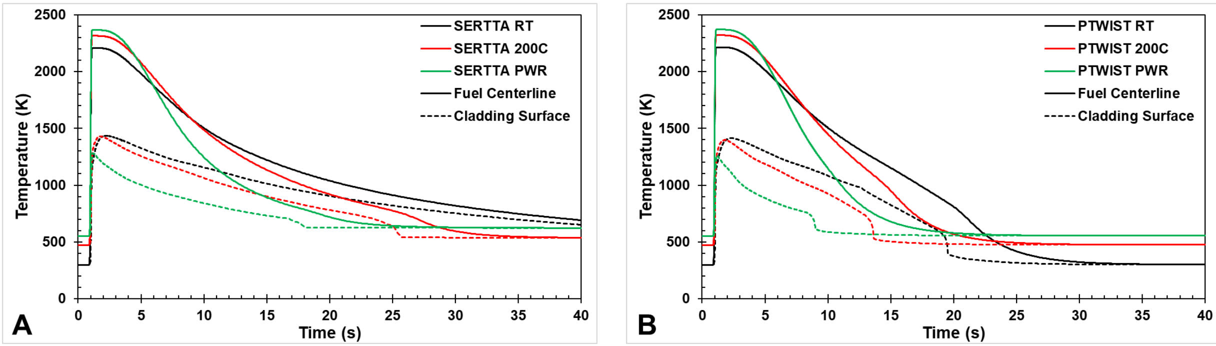


Figure 25. RELAP5-3D predictions of the fuel centerline temperature and cladding surface temperature under differing coolant initial conditions in the SERTTA capsule (a) and the P-TWIST capsule (b) with coolant at 2 m/s.

The P-TWIST capsule provides the capability to perform hot full-power RIA tests with prototypic radial temperature profiles and stored energy in the fuel rod. The hot zero-power RIA condition is the most limiting as it has narrower and more energetic power transients [10] that are more susceptible to early PCMI-type failures, especially with high-burnup fuel that has higher hydride concentrations and larger oxide thicknesses. With accident-tolerant fuels and advanced cladding concepts, this early-phase PCMI failure may not be the limiting case. The late-phase balloon or high-temperature failure during rewet may be a more limiting case that might require hot full-power tests.

4.5 Light-Water Reactor Pellet-Cladding Interaction Ramping

PCI is a broad area of transient fuel performance research. Unlike the fast-cladding strain phenomena, such as those discussed for PCMI above, PCI research is typically focused on slower ramps cycles, which cause modest cladding strain. Acting in concert with chemically aggressive fission products released from the fuel (e.g. iodine), these cladding strain cycles can cause cracks to initiate on the cladding inner diameter that penetrate to the surface, causing cladding breaches. PCI research is associated less with postulated scenarios and more with transients expected in operating reactors (e.g., startup, shutdown, power increase and decrease for load following).

Due to limitations on its core heat capacity and cooling system, TREAT is not capable of performing the hours-long transients, such as ramp and hold tests where the fuel rod is brought to a lower LHGR slowly (10 W/cm/min) and held for 12–24 hours to precondition the fuel rod and followed by a faster ramp (100 W/cm/min) to the maximum power held for an extended period of time. Tests in TREAT will have to forego the precondition phase and the ramps will be limited to minutes instead of hours. Assuming a 4.9% U^{235} fuel with some burnup accumulation would have a PCF of ~3.5 W/g/MW, a 10 W/cm/min ramp could last for 200 seconds before TREAT runs out of reactivity. A ramp of 50 W/cm/min could last for 145 seconds, 100 W/cm/min for 120 W/cm/min, and 500 W/cm/min for 75 seconds. All the transients will begin with a 10-second ramp to 10 kW/m that is held for 10 seconds prior to the final ramp. Depictions of these ramps are shown in Figure 26.

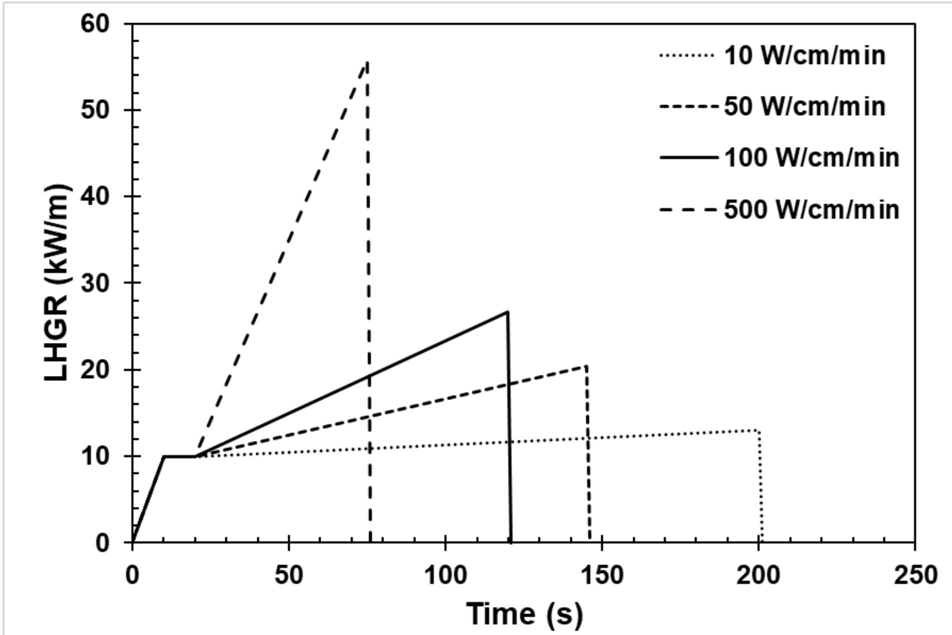


Figure 26. Ramp transients.

Simulations of the ramps in Figure 26 were run in RELAP5-3D with initial conditions starting from 25°C at 1 atmosphere, 200°C at 3.45 MPa, and 280°C at 15.5 MPa and coolant velocities from 0–4 m/s. All the fuel and cladding temperature results are very similar among the different coolant velocities and differences in the peak temperatures are due to the difference in initial starting temperature. Therefore, we used the starting conditions of 280°C with a coolant velocity of 2 m/s for the figures and discussion below. Fuel centerline and cladding surface temperature predictions are shown in Figure 27 for the different ramp rates.

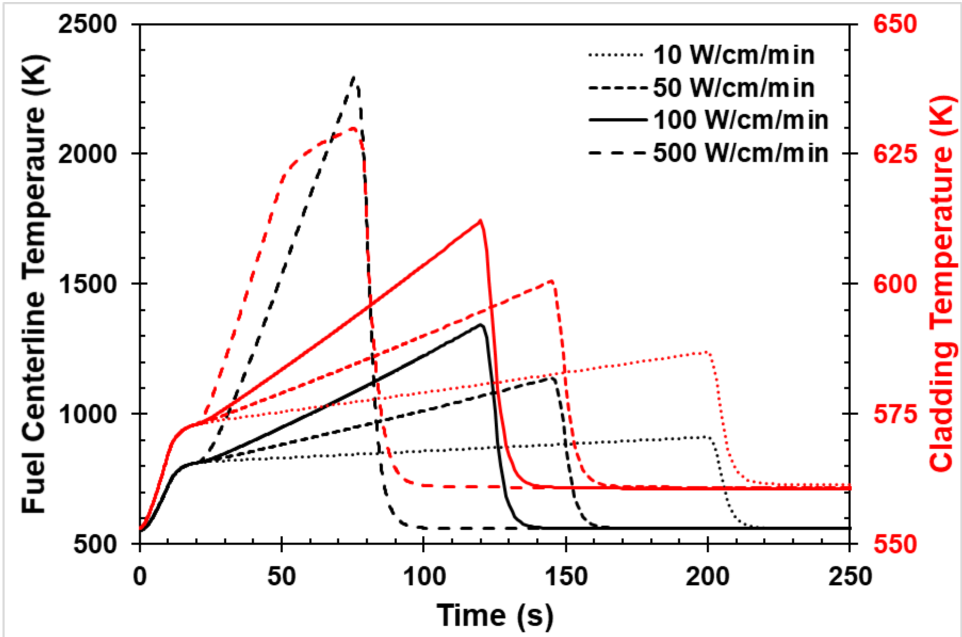


Figure 27. Fuel and cladding temperatures for the various ramp rates. PWR initial conditions with a coolant velocity of 2 m/s shown for all cases.

The TREAT reactor design limits its ability to perform slow ramps over a long period of time due to the negative temperature feedback of the core. Each of these transients are limited by the reactivity available in TREAT, and as such, the slower ramps run out of time before the fuel rod reaches temperatures much above normal operating conditions. Therefore, large ramp rates of >100 W/cm/min are required to achieve desirable fuel temperatures before TREAT runs out of reactivity.

The fuel and cladding temperatures are not the only indication of parameters of interest. For example, the ramp cases are mainly interested in the PCI and therefore need a more detailed Bison fuel performance model. The thermal-hydraulic boundary conditions from the RELAP5-3D simulations were applied to a Bison fuel performance model to investigate further detail in the stresses imposed to the cladding during these different ramps. Figure 28 shows the predicted cladding hoop stress and strain. Only the 500 W/cm/min ramp case resulted in any tensile hoop stress in the cladding. This is because only the 500 W/cm/min generated high enough fuel temperatures to close the fuel-cladding gap (Figure 29) and generate enough strain in the cladding to overcome the compressive stresses due to the coolant pressure. These results are for a "fresh" fuel rod geometry with a large initial fuel-cladding gap; if fuel rod "pre-conditioning" took place or a mid- to high-burnup fuel rod with a smaller gap was studied, the lower ramp rates could result in the desired stresses.

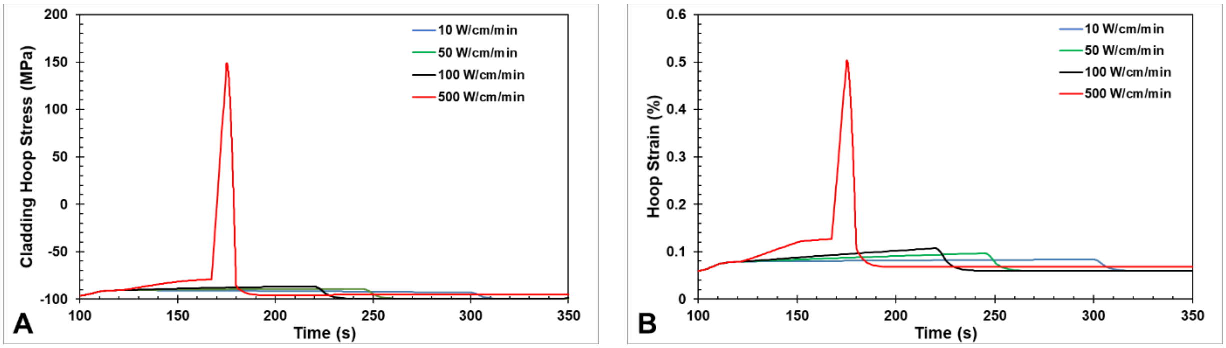


Figure 28. Bison fuel performance predictions of cladding hoop stress (a) and hoop strain (b) for the ramp cases

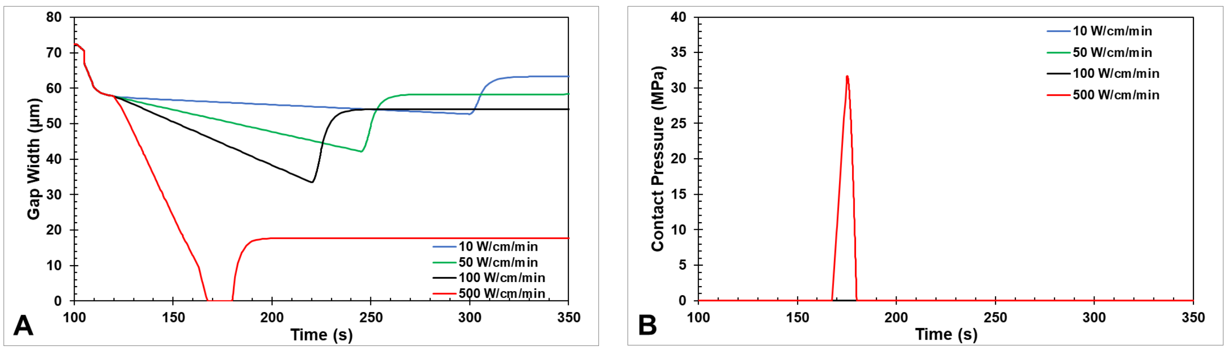


Figure 29. Bison fuel performance predictions of fuel-cladding gap (a) and fuel-cladding contact pressure (b) for the ramp cases.

TREAT is capable of performing ramp studies in the P-TWIST capsule that are faster than classical ramp experiments but still much slower than accident category ramp rates. TREAT is not capable of holding power plateaus long enough for phenomena like creep and iodine-assisted stress corrosion cracking to be dominant. However, truncated ramp tests like this have been suggested in order to "freeze the state of the fuel rod at the top…of the ramp" as purposeful test series meant to isolate physics and posttest observations for model development and scientific studies [11]. These types of separate effect tests could potential augment understanding and corroborate certain observations made using classical ramp tests, such as those planned in the Advanced Test Reactor.

5. SUMMARY

We investigated the pumped capsule modification of the TWIST capsule by mechanical design, prototyping, thermal-hydraulic modeling, and fuel performance predictions for a variety of transient shapes within TREAT's capabilities. The concept's ability to simulate heat transfer conditions found in water-cooled reactors naturally provides advantages for the transient types investigated here. Multiphase heat transfer phenomena in particular (e.g., dryout, rewet) are better represented in a flowing environment. P-TWIST's ability to recirculate water helps disperse heat generated by the test specimen throughout the water volume and capsule structure. As a result, P-TWIST maximizes its own mass and heat capacity as a suitable transient heat sink enabling more stable water temperature through longer transients such as power cycles and power ramps. These advantages support the conclusion that the P-TWIST design should undergo detailed engineering and deployment to enable refinements in transient fuel performance research.

The P-TWIST concept is a cost effect stopgap capability because it builds largely upon an existing capsule design and requires significantly less hardware fabrication and assembly, enabling deployment on an accelerated schedule as needed. This approach, however, will not be the seminal water capability for TREAT. A full water loop, such as the TWERL concept, should be retained in long term plans. A detailed assessment of the TWERL concept is not provided herein, but a transient water loop with features like heat exchangers and a larger pump capacity would be needed for longer test rods and multi-rod bundles.

6. REFERENCES

1. N. Woolstenhulme, C. Baker, C. Jensen, D. Chapman, D. Imholte, N. Oldham, C. Hill, and S. Snow. 2019. "Development of Irradiation Test Devices for Transient Testing," Nuclear Technology 205: 1251–1265. https://doi.org/10.1080/00295450.2019.1590072.

2. D. C. Crawford, A. E. Wright, R. W. Swanson, and R. E. Holtz. 1998. "RIA Testing Capability of the Transient Reactor Test Facility," in the proceedings of the IAEA Technical Committee Meeting on Fuel Cycle Options for LWRs and HWRs, Victoria, Canada, May 1998, IAEA-TECDOC-1122, pp. 99–109.

3. T. Holschuh, N. Woolstenhulme, B. Baker, J. Bess, C. Davis, and J. Parry. 2019. "Transient Reactor Test Facility Advanced Transient Shapes," Nuclear Technology 205: 1346–1353. https://doi.org/10.1080/00295450.2018.1559712.

4. J. D. Bess, N. E. Woolstenhulme, C. B. Davis, L. M. Dusanter, C. P. Folsom, J. R. Parry, T. H. Shorthill, and H. Zhao. 2018. "Narrowing Transient Testing Pulse Widths to Enhance LWR RIA Experiment Design in the TREAT Facility," Annals of Nuclear Energy 124: 548–571. https://doi.org/10.1016/j.anucene.2018.10.030.

5. N. Woolstenhulme and A. Epiney. 2019. "Status Report on Development of TREAT Water Loop," INL/EXT-19-55730, Idaho National Laboratory. https://doi.org/10.2172/1572404.

6. N. Woolstenhulme, C. Jensen, C. Folsom, R. Armstrong, J. Yoo, and D. Wachs. 2020. "Thermal-Hydraulic and Engineering Evaluations of New LOCA Testing Methods in TREAT," Nuclear Technology 207: 637–652. https://doi.org/10.1080/00295450.2020.1807280.

7. J. Ogiyanagi, S. Hanawa, and F. Nagase. 2011. "Status of Power Transient Test Program on LWR Fuels using JMTR," presented at the IAEA Technical Meeting, October 18–21, 2011, Mito, Japan.

8. P. G. Medvedev and R. D. Mariani. 2020. "Conductive inserts to reduce nuclear fuel temperature," Journal of Nuclear Materials 531: 151966. https://doi.org/10.1016/j.jnucmat.2019.151966.

9. N. E. Todreas and M. S. Kazimi. 2012. Nuclear Systems Volume I: Thermal Hydraulic Fundamentals. CRC press.

10. L. O. Jernkvist and A. R. Massih. 2010. "Nuclear fuel behaviour under reactivity-initiated accident (RIA) condition: State-of-the-art report." Nuclear Energy Agency, Organisation for Economic Cooperation and Development.

11. International Atomic Energy Agency. 2021. "Progress on Pellet-Cladding Interaction and Stress Corrosion Crack," IAEA-TECDOC-1960.