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Changing the World's Energy Future

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ABSTRACT

One of the most crucial performance areas for fuel rods in water cooled nuclear power plants is interaction between cladding tubes and fuel pellets. Experimental programs in test reactors have provided key data to help fuel developers and plant operators understand Pellet Cladding Interaction (PCI) phenomena and optimize their strategies for reliable fuel performance. Approximately 50 years of "ramp" testing programs, where the fission heating rate is deliberately manipulated in test rods, have been performed in a handful of test reactors to reveal and understand PCI behaviors such as iodine-assisted stress corrosion cracking. Unfortunately, the test reactors most engaged in this type of work have all been retired over the years up to the recent and unexpected closure of the Halden Boiling Water Reactor (HBWR) which effectively caused a hiatus in PCI ramp testing programs. The need for ramp testing is crucial at this time to enable refined understanding as more plants consider implementing flexible operations, increased fuel rod burnup limits, and new fuel technologies with enhanced accident tolerance. This paper reviews some of the key PCI phenomena that must be addressed with in-pile testing and surveys past test reactor's methods for achieving the needed conditions. A strategic approach is then presented using test reactors and complimentary facilities which are still available today at the Idaho National Laboratory (INL). Near term data opportunities are put forth along with capability development strategies that will ensure future longevity in this field of research.

BACKGROUND

Some of the earliest developers of nuclear power technologies rapidly identified zirconium alloy claddings that could tolerate the hydrothermal environment of Light Water Reactors (LWR). Improvements and refinements were made to these alloys in the decades that followed, but the modern nuclear fleet is still principally supported by derivatives of these early alloys. Having largely overcome the fundamental challenges on the coolant side of the cladding, the next obstacle in achieving increased fuel cladding reliability focused on phenomena inside the cladding tube. Decades of research followed to understand the complicated physics and improve behaviors between zirconium alloy cladding tubes and cylindrical pellets of uranium dioxide (UO_2). This paper includes an abbreviated discussion of PCI phenomena for context. Other papers and reports have been written on PCI subjects with greater detail on these research programs and their synthesis in predictive codes. [1, 2, 3, 4]

UO_2 pellets are manufactured with an outer diameter slightly smaller than the inner diameter of cladding tubes. These resulting mechanical clearances or "gaps" are important in optimizing trades offs between total fuel loading, early life pellet temperature, and late life accommodation for pellet swelling. The first rise to power creates pellet thermal expansion and thermal gradients which cause pellet fracture primarily in the radial direction. Operation at elevated temperatures also causes some densification in the pellets and increases the fuel-to-cladding gap. Elevated temperatures in the cladding, compressive hoop stress from external coolant pressure, and atom displacement from neutron collision all cause cladding tubes to "creep down" while accumulation of fission products starts to cause pellet swelling outward. These phenomena eventually close the fuel-to-cladding gap at approximately 20-30 GWD/MTU burnup in most fuel designs. Pellet-cladding contact conditions set the stage for PCI phenomena to begin.

Fuel pellets continue to swell as they accumulate burnup, principally creating hoop stress in the cladding. Both thermal and irradiation creep mechanisms help relieve stresses in the cladding during steady state operation. However, rapid increase in power can cause additional mechanical stress in the cladding due to thermal expansion of the pellet. These phenomena are usually referred to as Pellet-Cladding Mechanical Interactions (PCMI). Extreme power increases, such as postulated overpower accidents, can even cause cladding rupture by PCMI, especially if the cladding has become embrittled by hydrogen uptake. PCMI behaviors are not typically catastrophic in planned power increase maneuvers because the imposed cladding strains are more modest and there is adequate time for cladding stress relaxation. These power ramps, however, still cause a temporary increase in cladding tensile hoop stress. The magnitude of the power increase directly influences the thermal expansion difference from the "cold state" to the "hot state". An extended period of lower power operation will cause the cladding to creep down onto the pellet which, if followed by a relatively fast power ramp to a higher power state, can cause a higher tensile hoop stress state in the cladding.

In power ramps, pellet thermal expansion drives cladding hoop stress which is compounded by local stress concentrations at areas with pellet surface fractures. At the microstructural level these stresses tend to "pry apart" the cladding's metallurgical grains while cracks and thermal gradients in the pellets provide a pathway for migration and transport of ever-accumulating fission products. Some of these fission products, particularly isotopes of iodine, are chemically aggressive with zirconium alloys. Chemical reactions with the cladding at these concentration sites can develop small cracks which penetrate the cladding from within. These troublesome behaviors are often referred to as iodine-assisted stress corrosion cracking or often just Stress Corrosion Cracking (SCC). SCC is often the most prominent behavior first thought of in the broader category of PCI. See Figure 1 for basic representation of SCC-PCI.

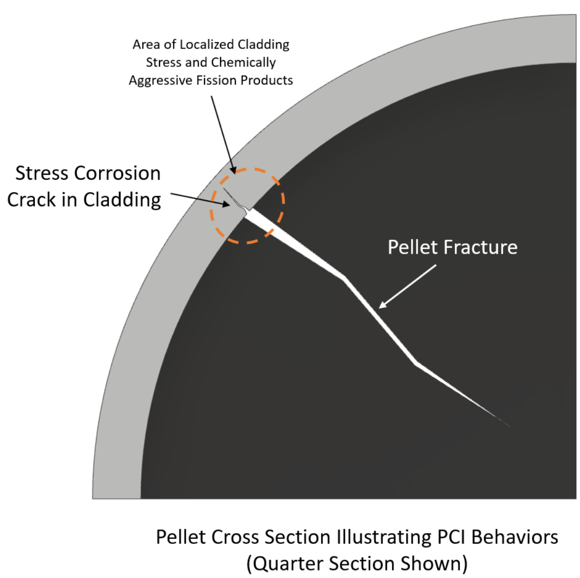

Figure 1: Simplified Representation of PCI Behavior

The progression of PCI phenomena is affected by late life fuel performance in important ways, some of which actually work to mitigate PCI. Of course, further embrittlement of cladding through hydrogen pickup, accumulation of cladding fatigue through power cycles, continual production of iodine, and further pellet swelling all continue to affect PCI behavior in deleterious ways. However, fission gas pressure continues to accumulate inside the cladding, reducing pressure difference across the cladding, and slowing cladding creep inward during lower power conditions. Other chemical interactions inside of the cladding, notably oxidation of its inner surface, can also create layers which tend to protect from SCC attack. Some fuel designs, notably those used in BWRs, have found success in using pure zirconium coatings or "Zr liners" on the cladding inner diameter to help harvest this beneficial characteristic combined with enhanced local ductility at the sharp contact points with pellet fracture edges.

Cladding breaches can be caused by SCC, which usually occur as localized pinholes in "leaker rods". These events typically result in only small amounts of noble fission gas release and, depending on how long operation persists after such a leak, water ingress and hydrothermal corrosion of UO_2 cause further fission product transport in reactor cooling water. In rare cases, fuel material itself can be washed out into the primary coolant. The presence of a small number of leaking fuel rods in a reactor does not typically affect safety regulation thresholds, but it is operationally undesirable especially in Boiling Water Reactors (BWR) where contaminants from the primary system can collect in turbine piping paths. As a result, great effort has gone into reducing frequency of rod leakage occurrences. Since these leaks are stochastic in nature (thus affecting only a limited number of rods), and since they are relatively manageable from an operations perspective, SCC is not usually the most prominent concern from a public protection and regulatory standpoint. Thus, much of the research on SCC has focused around achieving a realistic understanding of the phenomena and their frequency rather than simulating hypothetical conditions in a conservative fashion. Among the diverse event categories addressed by in-reactor transient testing, these phenomena are somewhat unique as they are not always approached from the viewpoint of "safety testing".

While much has been learned about PCI behaviors, current programs seeking to extend fuel burnup licenses to 72 GDW/MTU or beyond, corresponding with an increase in fresh enrichment and end of life fuel power levels, create unexplored conditions relevant to PCI phenomena. Simultaneously, an increasing number of nuclear power plants are finding new needs and opportunities to integrate energy production with the renewable power sources. This situation creates demand for power cycles essentially on a daily basis and highlights the need for PCI data focused on fatigue behaviors. New Accident Tolerant Fuels (ATF) designs are under development to help increase resilience in off-normal conditions. ATF designs may also provide benefit in facilitating extended burnup and/or load following conditions. ATF developments such as chromium external cladding coating will affect cladding mechanical behaviors and may form a last line of defense against SCC penetration from within. ATF pursuits of large-grain "doped" UO_2 pellets will exhibit differences in early life densification behaviors and late-life fission production retention performance. There is a firm need for continued PCI research in the future.

MOTIVATION FOR IN-REACTOR TESTING

As shown in the previous discussion, PCI behaviors are influenced by complicated and interrelated physics which require thermomechanical states only achievable with nuclear heating and active cooling. Other important PCI behaviors also require reactor environments to drive fuel swelling, cladding atom displacement damage/creep, and production of chemically aggressive fission products. Out-of-reactor PCI testing can offer beneficial opportunities to isolate physics of interest and develop a refined research focus but cannot offer a complete understanding without complimentary in-reactor testing capabilities.

Some PCI behaviors naturally occur in the course of regular operation at nuclear power plants. Occasional leaker rods from commercial plants have created useful material for Post Irradiation Examinations (PIE) to help research PCI. Deliberate irradiations for PCI research, however, create conflicts for commercial nuclear plant operators since such events can be detrimental in managing power production cycles and radiologic dose limits to plant workers. PCI research is also closely related with the ability to actively manipulate power transients in test rods, often to challenging conditions, which is not reasonable to perform in commercial power plants. Thus, material test reactors have been used to perform many of the seminal tests in historic PCI research. A handful of historic test reactors have participated in this type of work. A detailed review of all these programs is not presented in this paper for the sake of brevity, but programs from two material test reactors are presented as they primarily form the story arc to present day efforts.

The R2 reactor in Sweden was a 50 MW tank-in-pool type material test reactor operated by Studsvik Nuclear starting in 1960 for approximately 40 years until it was decommissioned. R2 was cooled by light water and driven by aluminum-clad plate-type fuel. Ramp testing began in R2 in 1969 first by ramping the entire reactor in order to test rods within its pressurized water loops. This mode of ramp testing was only used for a few years as it was operationally challenging and caused other R2 experiments, which did not desire transient conditions, to be temporarily extracted from the reactor. In 1973 a new approach was deployed where test rods were surrounded by a volume of the helium isotope ^{3}He (a strong neutron absorber). The pressure of ^{3}He gas in this volume was varied to adjust the neutron flux on test specimens while the reactor maintained a constant power level. The ^{3}He screen was combined with a test rig able to extract/insert test rods from the active core so that multiple rods could be tested, one at a time, in a ramp test series during a single reactor cycle. This test rig included instrumentation to measure test rod power by water enthalpy rise, a linear variable differential transform sensor to measure rod elongation, and dosimetry to measure neutron flux. [5,6]

After closure of the R2 reactor, HBWR in Norway became the new venue for ramp testing program in the manner used at R2. While the HBWR was constructed in the same era in a neighboring country as R2, it was a markedly different reactor design not originally intended to focus on high flux material test reactor missions. The HBWR operated up to 25 MW using rod-type fuel in boiling water thermal hydraulic regimes. While the Halden flux was modest compared to other plate-fuel-driven material test reactors, its unique use of heavy water coolant enabled large lattice spacing in the core to afford several in-reactor test loops with high usable volume. As a result, the HBWR became a centerpiece for PCI ramp tests and other irradiation missions for approximately two decades. Apart from some necessary mechanical modifications, the HBWR approach to ramp testing was essentially the same as R2 where ^{3}He screen pressure was used to adjust test rod flux and resulting specimen fission power. A similar insertion/extraction mechanism was used to test rods, one at a time, in a single cycle test series. HBWR's instrumentation package was similar to that used in R2 ramp tests. A simplified illustration of the HBWR ramp test rig is show in Figure 2.

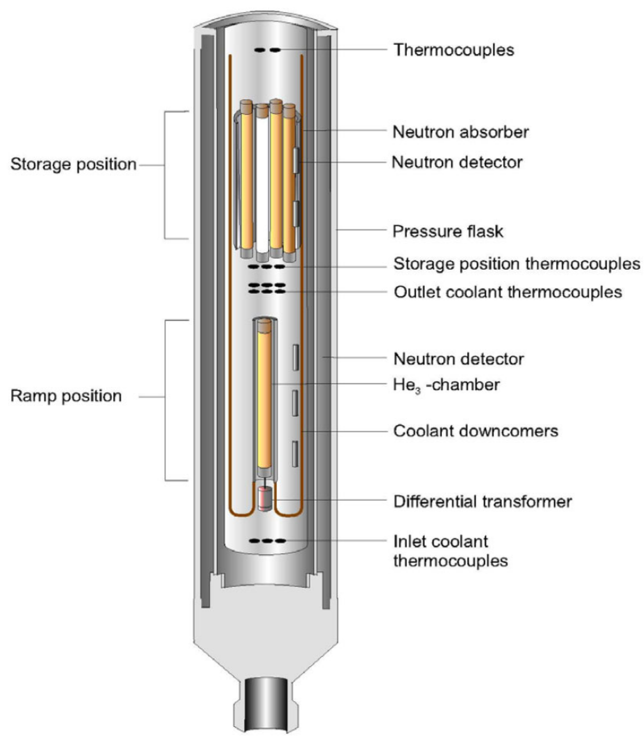

Figure 2: HBWR Ramp Test Rig [7]

In 2018 an unexpected determination was made to shutdown HBWR permanently. At the time no other material test reactors were actively involved in ramp testing programs. In response, the United States Department of Energy Advanced Fuels Campaign (AFC) assessed capability needs to fill the gaps left by HBWR's closure, especially to enable further development of ATF technologies. Among other recommendations, this assessment concluded that additional water loop and water capsule capabilities should be developed for the Advanced Test Reactor (ATR) and the Transient Reactor Test facility (TREAT), respectively, both currently in operation at INL [8]. The remainder of this paper will discuss how these new capability projects, which are currently underway, can be used to address PCI ramp testing in a world without the HBWR.

RAMP TESTING PLANS IN ATR

ATR was constructed and began operating in the late 1960's to provide irradiation testing capabilities for numerous fuels and materials specimens. ATR's iconic serpentine shape of driver fuel wraps around nine flux traps, most of which have pressurized water loops installed. ATR is cooled and moderated by light water. The core is surrounded by metallic beryllium reflector blocks. ATR's plate type driver fuel, pressurized vessel, and high coolant flow rates enable it to operate at very high-power relative to its core size (up to 250 MW, although 100-150 MW total core power is more typical). At the time HBWR's closure was announced, Loop 2A in ATR's center flux trap was the only water loop available for fuels irradiations in the western world, and at that time was already well utilized for steady state testing of ATF specimens. Loop 2A operates with Pressurized Water Reactor (PWR) conditions but with a total flux and average neutron energy spectrum higher than commercial nuclear plants. Neutron absorbers such as hafnium are often employed in Loop 2A irradiation hardware to reduce fission rate in test rods and obtain conditions typical of LWRs. ATR's 1.2m long active core exhibits a "chopped cosine" axial flux profile which allows test rods (~30 cm length) to be placed at different axial locations as another means to adjust rod power. These types of test parameters can be used to achieve the desired test rod power levels during steady state conditioning irradiations prior to ramp testing.

The HBWR gap study showed that missions displaced by the closure of HBWR could not be consolidated into Loop 2A alone (especially noting that Loop 2A exclusively operates in PWR conditions) and that availability of other ATR flux trap loops for civilian programs was not realistic. Since ATR's underutilized I-positions in the beryllium reflector offered a flux similar to HBWR's core, the "I-Loop" project was initiated to install at least two new water loops in ATR. These new loops will enable PCI ramp testing and a general increase in specimen irradiation capacity for both BWR and PWR environments. This project is presently underway with the objective to commission the first I-Loop in 2025 followed shortly by the second. Owing to a well thermalized spectrum and weak neutronic interactions with the core, I-Loops will be an ideal venue where ^{3}He neutron screens will be highly effective in adjusting rod power without undue complications to reactor operations. ATR's peripheral reflector positions can experience flux gradients across test specimens. The I-Loop ramp testing rig is specifically designed to counteract this effect by surrounding the in-pile tube with extra primary coolant water to help scatter/reflect neutrons "behind" the rod and by using one test rod (in cross section) to reduce rod-to-rod shielding.

At least one of these I-Loops will have a coolant ion exchange system with increased capacity and shielding features to facilitate decontamination in PCI ramp tests which routinely cause cladding breaches. Figure 3 displays a representation of the ATR core where Loop 2A and I-Loop positions are indicated. Figure 4 shows a design rendering of the I-Loops in ATR's upper vessel area.

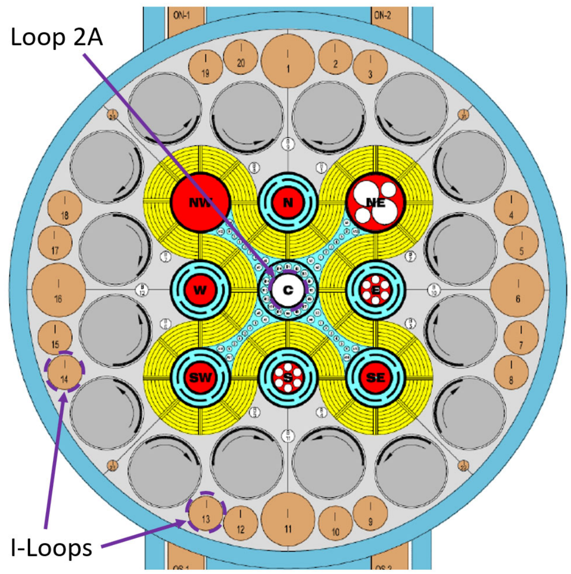

Figure 3: ATR Core Map with Loop Locations

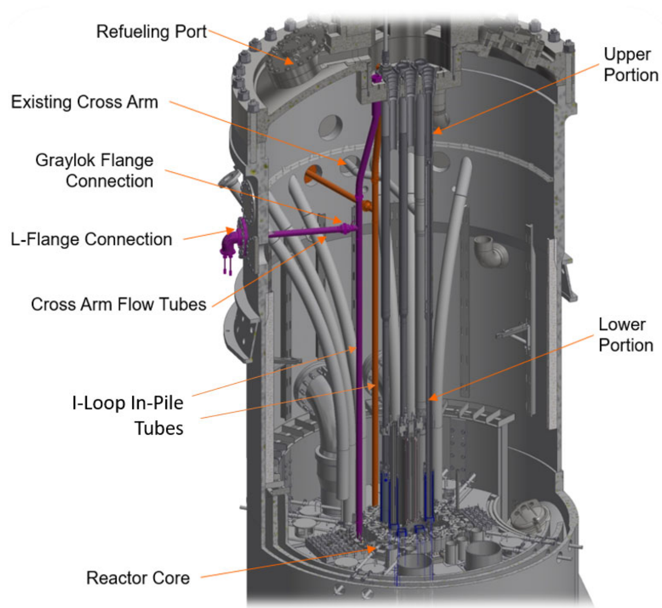

Figure 4: I-Loop Design Overview in ATR Upper Vessel

Neutronic studies were performed by Monte Carlo calculations to determine the effectiveness of ^{3}He in adjusting test rod power in I-Loop and Loop 2A ramp test concepts. Test rig cross sections were conceptualized akin to that used in HBWR with a ^{3}He screen. The Loop 2A concept assumed two test rods in the cross section with stainless steel holder hardware and an outer hafnium sleeve consistent with similar Loop 2A designs. Owing to the lower flux in reflector positions, the I-Loop concept assumed a single test rod with zirconium alloy hardware and no hafnium sleeve to reduce self-shielding and neutron absorption in the hardware. Both cases assumed a ^{3}He screen that was essentially an annulus surrounding the rods' coolant channels. See Figure 5 for illustrations of the test rig cross sections. An additional scenario was investigated for comparison in the Loop 2A concept where hafnium replaced the ^{3}He volume.

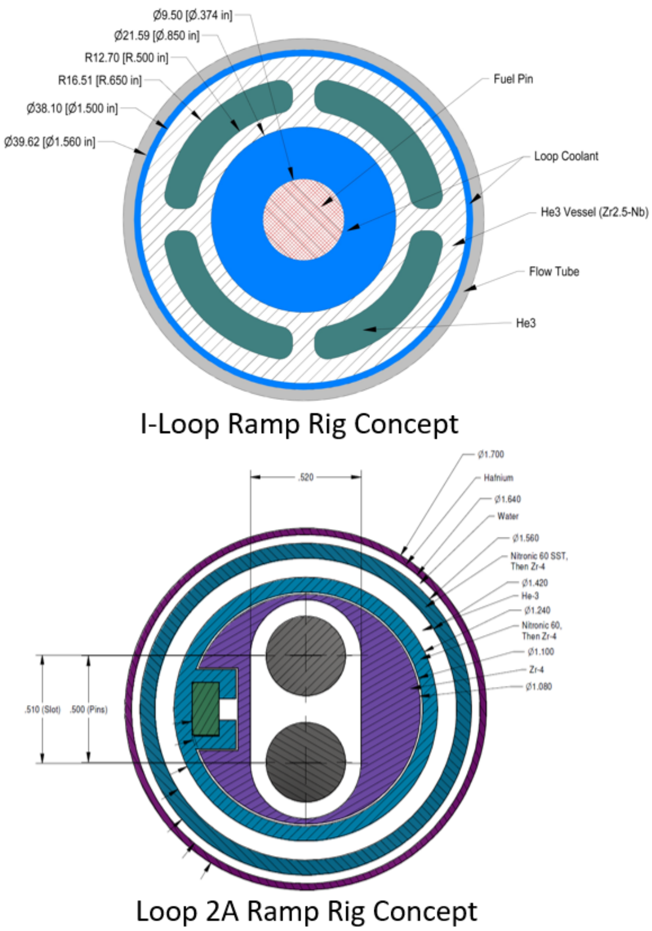

Figure 5: Cross Section of Ramp Rig Concepts used in Neutronic Studies

These studies varied ^{3}He atom densities correlating to practical pressure ranges achievable for a pressurized gas system. These predictions showed capability for more than 10X rod power adjustment range in the I-Loop case, but at most 2X power adjustment range in Loop 2A (see Figure 6). A high range of adjustability is a major advantage both in performing the power ramp transient and in providing the ability to "turn off" the test rod after a leak is detected to reduce its temperature and slow hydrothermal corrosion. This action is useful for minimizing fission product release into the loop and preserving the post breached state of the test rod for PIE.

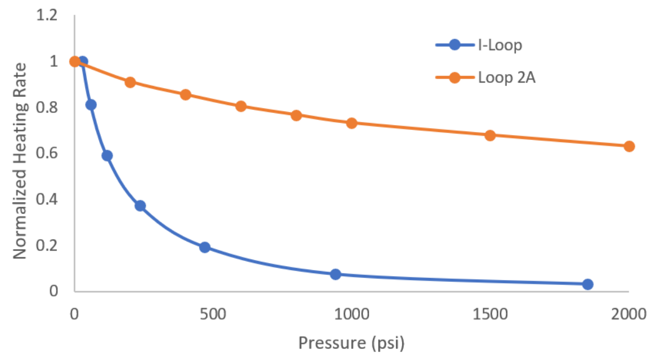

Figure 6: Test Rod Normalized Heating Rate as a Function of ^{3}He Pressure in I-Loop and Loop 2A Cases

The principal reason for this difference in ^{3}He effectiveness is that Loop 2A's neutron flux spectrum is higher energy than the I-Loop case and ^{3}He is more effective at absorbing low energy neutrons. This effect is displayed in Figure 7 where increasing ^{3}He pressure in the Loop 2A case is shown to reduce thermal neutron population most and only the hafnium cases is effective at reducing epithermal and fast neutron flux. The less pronounced ^{3}He pressure effect in Loop 2A, however, is predicted to only create a ~\$0.05 change in reactivity worth which should make this approach manageable for reactor operation considerations.

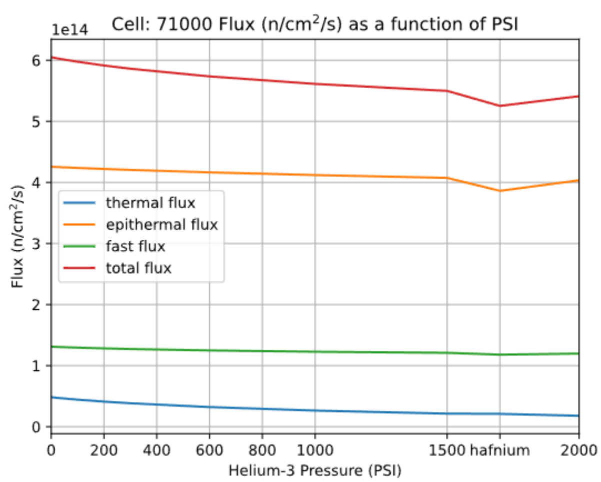

Figure 7: Effect of ^{3}He Pressure on Neutron Energy in Loop 2A

In practice the power histories and ramp rates that fuel rods could experience in commercial power plants encompass myriad conditions, especially as more plants consider load integration with intermittent renewables in different regions and markets. The "standard" historic ramp test rate of 100 W/cm/min will be achievable in this new system. Based on successful experience with other feedback-controlled gas systems used in ATR, it is also foreseen that this new ^3He pressure control system will have the capability to increase/decrease ^3He pressure, resulting in different rod power rate changes, in order to investigate these effects.

These concepts represent simplified features of neutronic significance only. Thus, instrumentation was not included in the nuclear modeling studies. However, future detailed mechanical design of these test devices is planned to accomodate key instrumentation options available in HBWR tests including water inlet/outlet thermocouples (TC), cladding elongation measurement by Linear Variable Differential Transformer (LVDT), rod plenum pressure via metallic bellows with LVDT, and fuel pellet centerline TC.

ATR power distribution can be varied among the five lobes where each of the four corner lobes' power level can be individually controlled by the angular position of beryllium control drums and the fifth center lobe effectively experiences the average power of corner lobes. I-positions are different as they experience a flux level more proportional to the power of the nearest corner lobe. ATR cycles typically operate in one of two modes. Usually three "normal cycles" are performed per year, each lasting ~60 days with reactor lobe powers at ~25 MW. One or two special high-power cycles are performed per year, each lasting ~10 days, where lobe powers can be as high as 60 MW. These cycles are often referred to as "PALM cycles" named for the Power Axial Locator Mechanism device which is often used in such cycles to insert/remove specimens in flux trap loops for power cycle testing.

Typical unmodified thermal neutron fluxes for normal cycle power levels are approximately 4.4E14 and 3.4E13 n/cm^{2}s for flux trap and Medium-I positions, respectively [9]. Cycle-specific lobe power distributions and experiment materials/configurations can have significant effects on the flux experienced by test specimens. Neutron absorption in Loop-2A's stainless steel in-pile tube markedly reduces flux to test specimens. Conversely, use of Zr-2Nb alloy for I-Loop in-pile tubes and use of ATR driver fuel assemblies in nearby Large-I positions to boost neutron population will increase flux in I-Loop tests. For these reasons, the neutronic models described above were also needed for a more accurate comparison of the test rod power capabilities in these different loop capabilities.

Assuming normal cycle powers and commercial UO_2 pellet enrichment (4.9%) the peak LHGR achievable (lowest ^3He pressure case) was predicted to be approximately 180 W/cm for I-Loop and 350 W/cm for Loop 2A, respectively. Since ramp tests can be accomplished in a few days it is likely more practical to use normal cycles for burnup accumulation and then perform ramp tests in PALM cycles when fluxes would approximately double. Since pellets with increased enrichment are under consideration for enabling high burnup in LWRs, the effect of increased pellet enrichment (8%) was also investigated and found to increase LHGR by approximately 40%. The effect was not expressly predicted in these neutronic studies, but other design experiences at ATR suggest that a neutronically transparent version of the Loop 2A rig (e.g., zirconium alloy hardware without hafnium sleeve) should have a similar effect as the increased enrichment case, thus extremely high terminal LHGRs should be achievable in Loop 2A even with current commercial enrichment levels. These effects are summarized in Table 1 and a simplified representation of the overall approach to ramp testing in ATR is illustrated in Figure 8.

Table 1: Summary of Peak LHGR Capabilities

Peak LHGR Capability (W/cm at lowest ^3He Pressure)			
	Normal Cycle	PALM Cycle	PALM w/ 8% Enr. Pellets
Loop 2A	350	700	970
I-Loop	180	360	500

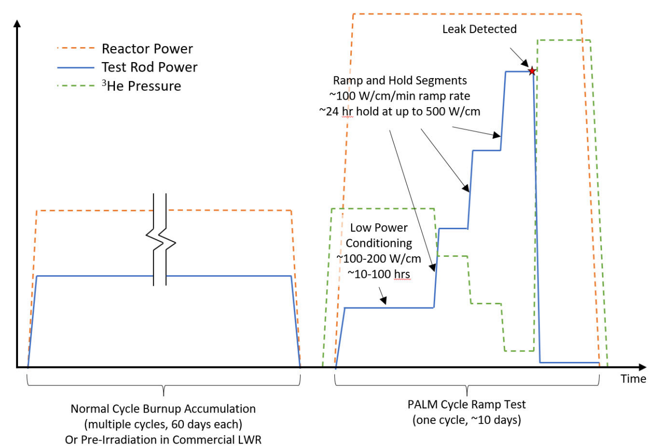

Figure 8: Simplified Representation of I-Loop Based Ramp Testing Strategy

As can be seen, the I-Loop concept will offer a ramp capability consistent with historic PCI programs using ^3He and is thus viewed as the priority in recovering this HBWR gap. The effectiveness of ^3He in adjusting test rod power in the I-Loop concept will offer high flexibility in achieving desired power histories.

The prospect of ramp testing in Loop 2A shows capability in achieving very high LHGRs which are comparable to and some cases exceeding capabilities planned for future European ramp testing programs (620 W/cm and 750 W/cm in the Joules Horowitz Reactor ADELINE test device [10] and Belgian Reactor -2 pressurized water capsule [11], respectively). Use of ^3He alone, however, does not appear to offer the necessary adjustment test rod power range in Loop-2A. Thus, a key conclusion of the work presented here is that a mechanical device capable of manipulating test rod axial position within ATR's core will be needed to access this unparalleled capability. Loop 2A's hydraulic systems are already in service and not sized/shielded for routine fission product release experiments. Thus, truly accessing the long-term potential of high LHGR ramp testing in Loop 2A would also require some adaptations in this regard. Realization of this development is also viewed as a high priority to be deployed after completion of the I-Loop project (e.g., 2027).

While Loop 2A's full potential for PCI ramp testing will be realized in the further future, this loop can still be used in the near term to commission systems and build understanding in the years prior to I-Loop commissioning. Loop 2A is currently occupied with steady-state tests of ATF fuel designs and is scheduled to be occupied by these experiments for several years to come; however, these steady-state ATF experiments are not currently installed during high power PALM cycles. Hence, PALM cycles can be utilized without displacing tests already underway. Some of these PALM cycles also perform reactor lobe power manipulations that would result in an effective ~3X power ramp in Loop 2A. Plans are being formulated for special Loop 2A tests in such a cycle to commission and calibrate the test rig design and instrumentation package over a transient domain. Efforts are also underway to assess the viability of manipulating specimen axial location to achieve ramp conditions in near-future Loop 2A tests. The first transient condition tests in Loop 2A should be achievable in ~2023-2024 and will mark the beginning of INL's in-reactor campaign toward full ramp testing capability.

Similarly, the I-Loop ^{3}He pressure control system is currently under development and planned to be used with a neutronically-equivalent I-Loop mockup in the ATR Critical facility (ATR-C, a low power reactor which mimics ATR). Use of dosimetry and various ^{3}He pressures in ATR-C runs should be achievable in ~2024 in advance of I-Loop commissioning. These tests will develop/validate neutronic models, commission systems, and incrementally achieve progress for ^{3}He competencies to be used in newly installed I-Loops starting in ~2025.

RAMP TESTING VIABILITY IN TREAT

TREAT is a graphite-based test reactor built in the late 1950's for testing nuclear fuels under extreme conditions. TREAT did not operate for a several years starting in the late 1990's but was recently refurbished and resumed reactor operation for fuel safety testing in 2018. During transient operations TREAT's fuel assemblies heat up quickly and provide strong negative temperature feedback which permits extreme power maneuvers to be performed safely. Fast-moving transient control rods are controlled automatically by a feedback system capable of providing virtually any power transient shape desired within the core's energy capacity of ~2500 MJ. After transient runs are complete TREAT's core is cooled back to room temperature by an air-cooling system to enable approximately one transient operation per day. TREAT's core is shielded by concrete blocks with slots for experiment installation and instrumentation leads. Capsules and loops are used to contain test samples, support instrumentation, and provide desired coolant boundary conditions for test specimens. See Figure 9 for an overview of the TREAT reactor.

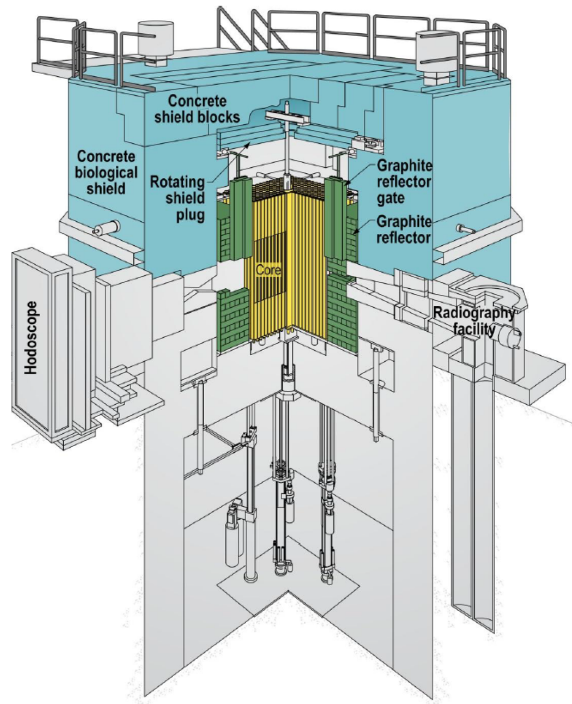

Figure 9: TREAT Reactor Overview [10]

TREAT's power shaping abilities are superb but transient length is limited to a few minutes at most due to the core's energy capacity and temperature limitations. Hence TREAT is not a strong candidate for burnup accumulation or classical "ramp-and-hold" type PCI testing. Thus, test rods to be ramp tested in TREAT would likely undergo a preconditioning irradiation at the desired power levels in another reactor (e.g., commercial LWR or ATR). Existing hot cells, casks, and equipment at INL would then be used to outfit and install rods into TREAT for ramp testing. This is the same approach used to accomplish LWR fuel safety research tests in TREAT such as reactivity initiated or loss of coolant accident testing.

Experts in the PCI research community have recognized that the typical 12 hour hold after the ramp tends to "erase" some of the incipient behaviors needed for model refinement and validation. It has been recommended that some special ramp tests be performed where rods are ramped up to the desired power and immediately ramped down to "freeze" the state of the rod for PIE [1]. A design project was already underway to investigate the capabilities of a recirculating water capsule for various purposes in TREAT. Based on this design, thermal hydraulic calculations were performed to determine whether this capsule could be used for "ramp-and-stop" transient tests. Design renderings of this recirculating capsule design are shown in Figure 10. A more detailed description of this capsule design and its thermal hydraulic performance are available in the proceedings of this conference [13]. Thermal hydraulic studies were performed for TREAT ramp testing assumed nuclear heating rates consistent with neutronic predictions previously for a similar test arrangement.

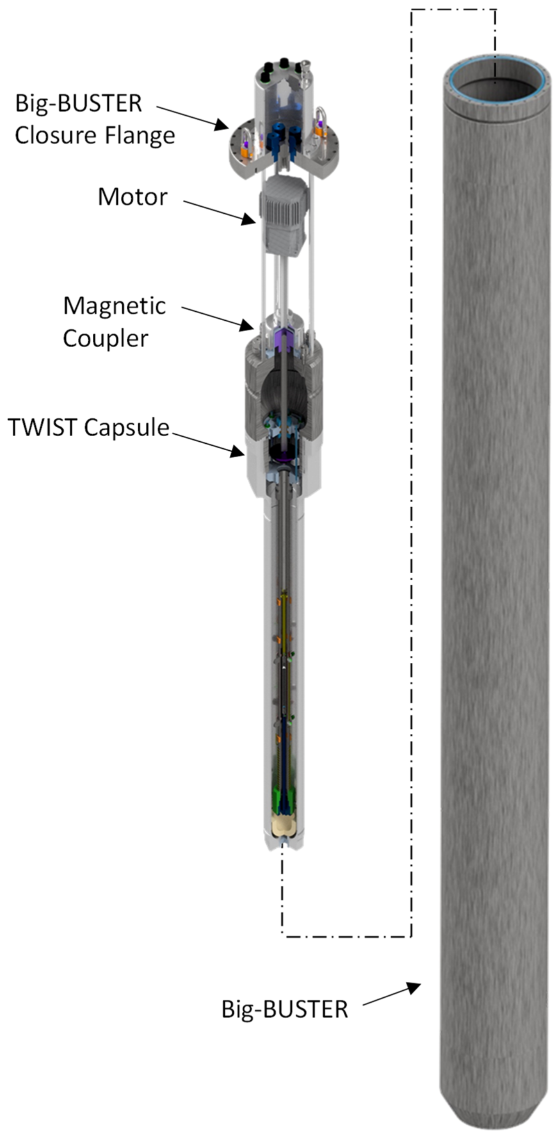

Figure 10: Design Renderings of a Recirculating Water Capsule for TREAT

These investigations showed that the typical LHGR ramp rate used in classical ramp tests (100 W/cm/min) was essentially too slow since TREAT's energy capacity would be exhausted before the desired state was reached. It was shown, however, that with a somewhat faster ramp rate (500 W/cm/min) and moderate cooling flow velocity (2 m/s), this approach could achieve the desired range of thermomechanical stresses in the fuel cladding. Figure 11 shows the cladding and fuel temperature response under various ramp rates. Cladding stress calculations were also performed and are reported elsewhere in the proceedings of this conference [13]. Plans for deployment of this recirculating water capsule are under development, and it could be available as soon as 2024 depending on whether this unconventional option for ramp testing is viewed as highly valuable to the user community.

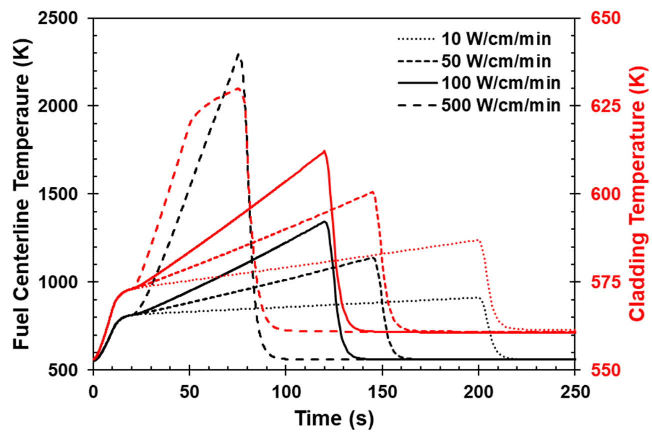

Figure 11:Fuel and Cladding Temperature Responses Resulting for Various LHGR ramp rates in TREAT

CONCLUSIONS

The fuel performance phenomena occurring as a result of interactions between nuclear fuel pellets and cladding tubes continues to hold value as an important area for enhancing fuel reliability, especially considering new needs for ATF designs, increased burnups, and load following power cycles. PCI ramp testing in material test reactors is a key capability to enable this type of research but is at risk of becoming dormant following the closure of HBWR. The I-Loop project is underway to install new water loops in the reflector of ATR which will reclaim the HBWR gap of PCI ramp testing via ^{3}He screen. Near term plans to use Loop 2A and ATR-C will be useful to develop and commission test rigs, instruments, and control systems in order to pave the way for I-Loop capabilities. In the further future, expansion of Loop 2A competencies will include active manipulation of test rod axial location will provide an unparalleled capability for ramp testing with high terminal LHGR. Finally, an alternative ramp testing approach using the TREAT facility shows promise as a unique way to produce data for phenomena isolation and model validation. Realization of the test devices presented here will support future research programs with capabilities that recover and advance the state-of-the-art in PCI ramp testing.

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