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Abstract. Simulation of a variety of transient conditions has been successfully achieved in the Transient Reactor Test (TREAT) facility during operation between 1959 and 1994 to support characterization and safety analysis of nuclear fuels and materials. A majority of previously conducted tests were focused on supporting sodium-cooled fast reactor (SFR) designs. Experiments evolved in complexity. Simulation of thermal-hydraulic conditions expected to be encountered by fuels and materials in a reactor environment was realized in the development of TREAT sodium loop experiment vehicles. These loops accommodated up to 7-pin fuel bundles and served to simulate more closely the reactor environment while safely delivering large quantities of energy into the test specimen. Immediate TREAT restart operations will be focused on testing light water reactor (LWR) accident tolerant fuels (ATF). Similar to the sodium loop objectives, a water loop concept, developed and analyzed in the 1990's, aimed at achieving thermal-hydraulic conditions encountered in commercial power reactors. The historic water loop concept has been analyzed in the context of a reactivity insertion accident (RIA) simulation for high burnup LWR 2-pin and 3-pin fuel bundles. Findings showed sufficient energy could be deposited into the specimens for evaluation. Similar results of experimental feasibility for the water loop concept (past and present) have recently been obtained using MCNP6.1 with ENDF/B-VII.1 nuclear data libraries. The old water loop concept required only two central TREAT core grid spaces. Preparation for future experiments has resulted in a modified water loop conceptual design designated the TREAT water environment recirculating loop (TWERL). The current TWERL design requires nine TREAT core grid spaces in order to place the water recirculating pump under the TREAT core. Due to the effectiveness of water moderation, neutronics analysis shows that removal of seven additional TREAT fuel elements to facilitate the experiment will not inhibit the ability to successfully simulate a RIA for the 2-pin or 3-pin bundle. This new water loop design leaves room for accommodating a larger fuel pin bundle than previously analyzed. The 7-pin fuel bundle in a hexagonal array with similar spacing of fuel pins in a SFR fuel assembly was considered the minimum needed for one central fuel pin to encounter the most correct thermal conditions. The 9-rod fuel bundle in a square array similar in spacing to pins in a LWR fuel assembly would be considered the LWR equivalent. MCNP analysis conducted on a preliminary LWR 9-rod bundle design shows that sufficient energy deposition into the central pin can be achieved well within range to investigate fuel and cladding performance in a simulated RIA. This is achieved by surrounding the flow channel with an additional annulus of water. Findings also show that a highly significant increase in TREAT to specimen power coupling factor (PCF) within the central pin can be achieved by surrounding the experiment with one to two rings of TREAT upgrade fuel assemblies. The experiment design holds promise for the performance evaluation of PWR fuel at extremely high burnup under similar reactor environment conditions.

Keywords: TWERL, Accident Tolerant Fuels, TREAT.

INTRODUCTION

The Transient Reactor Test (TREAT) facility housed at Idaho National Lab (INL) within the Materials and Fuels Complex (MFC) near Idaho Falls, Idaho has successfully conducted a variety of experiments in support of nuclear reactor fuel development and qualification during operation between initial start-up in 1959 and relegation to operational stand-by in October of 1994. A little more than two decades later, the need for transient testing to qualify advanced light water reactor (LWR) accident tolerant fuel (ATF) for use in commercial nuclear power reactors with the intention of mitigating consequences similar to those encountered in the Fukushima, Japan incident has sparked Department of Energy (DOE-NE) interest in TREAT operational restart [1]. TREAT is a highly versatile testing reactor and once on-line it is expected to eventually assume a role in providing irradiation services to a variety of experimental programs. Preparation efforts for TREAT re-start, including necessary maintenance and refurbishments (e.g. control systems) and training of operations personnel, are under way. TREAT full start-up is expected as soon as 2018 [2].

TREAT is uniquely designed to simulate a wide variety of nuclear excursions encountered in off-normal operating conditions or in various accident scenarios. Historically, transient testing had been performed at various levels of complexity depending on experimenter needs. Phenomenological experiments were conducted to analyze physical fuel response to rapid overheating. These tests could be administered in a simple dry capsule inserted into the center of the reactor core. More complex integral experiments were conducted to analyze fuel response in a more prototypic reactor operational environment [3]. Such an environment included excess heat removal from the fuel pin(s) via forced convection through the coolant medium. Like reactor environment thermal-hydraulic conditions were realized by the MARK series sodium recirculation loops. Fuel specimens were surrounded by flowing liquid sodium that was recirculated by induction pumps carefully contained within a structure that could be easily inserted into the center grid spaces of the TREAT core without disrupting the nuclear power excursion simulation capability of the reactor (see Figure 1). Mark-III was designed to safely accommodate 7-pin bundle testing in simulated liquid metal fast breeder reactor (LMFBR) hypothetical accidents (see Figure 2). Several of these tests have been safely conducted in the 1980’s to evaluate fuel performance in prototypic fuel-pins [4].

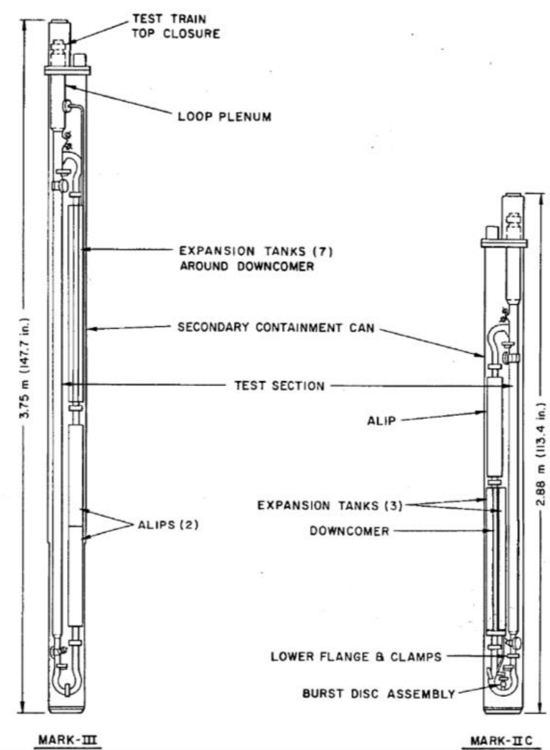

FIGURE 1. MARK-III Loop (left) and MARK-IIC Loop (right) [4].

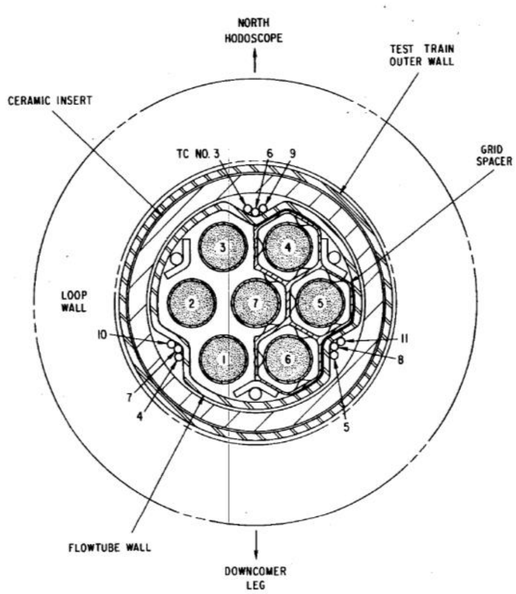

FIGURE 2. Cross-sectional diagram of MARK-III 7-pin bundle recirculation sodium flow loop [4].

Although a large portion of TREAT historical integral transient testing has been conducted on sodium cooled fuels, LWR fuels have also been a subject of such testing. A serious testing campaign, designated the Source Term Experiments Project (STEP), conducted in a once-through flowing steam environment provided data for characterizing radiological source term in simulated LWR specified accident scenarios. High pressure steam was released at the bottom of the fuel rods and flowed upward through a series of instruments to measure isotopic source term while TREAT simulated decay heat in a slow descending power 20 minute transient. The STEP experiment vehicle accommodated a 4-pin bundle [5]. Subsequently, a conceptual design of a high pressure, circulating water loop developed to investigate the impact of a reactivity insertion accident (RIA) in a LWR reactor thermal-hydraulic environment on prototypic fuel rods was given serious consideration in the 1990's [6]. Neutronics analysis at the time predicted TREAT capability to meet the energy deposition needs for such an investigation. Unfortunately, further progress beyond the conceptual design stage was curtailed given the operational status of the reactor.

Immediate TREAT restart operations will focus on phenomenological testing of ATFs [2]; however, anticipation of future integral testing needs has renewed interest in the circulating water loop. Current neutronics analysis using MCNP6.1with ENDF/B-VII.1 nuclear data [7] of the old water-loop concept accommodating a BWR 2-pin bundle or a PWR 3-pin bundle [8] yielded similar findings to that conducted in the 1990's. The old water-loop concept has recently been developed and the design has matured to what is referred to as the TREAT Water Environment Recirculation Loop (TWERL) as shown in Figure 3 alongside the 1997 concept. The new design shifts to emphasis on the contained package loop form where, similar to the Mark-III sodium loop, the pump is contained within the test vehicle insert rather than placing circulation supporting equipment outside of the loop on top of the reactor. Unlike the Mark-III, wherein the addition of a slender induction pump required no more room than the typical two core grid space slot needed for the test vehicle, the centrifugal water pump in TWERL will require a 9-slot resting place for the test loop vehicle insert. Consequently, more room is available to accommodate larger fuel rod bundles. The TWERL design was primarily intended to test individual rods in flow tubes, and possibly a four-rod mini-bundle [9]. Although the TWERL apparatus is currently only developed to the pre-conceptual design phase, it was used as a starting point for PWR 9-rod capability studies since existing Monte Carlo N-Particle (MCNP) models of it already existed (see Figures 3 and 4). The TWERL model was modified to include larger internal piping to support the 9-rod bundle (hereafter referred to as "Super-TWERL"). The 9-pin bundle in a square array at PWR assembly pitch is considered similar in data gathering value to the 7-pin LWFBR bundle in a hexagonal array at SFR assembly pitch given that the center rod/pin will more closely approach prototypic boundary conditions. No other efforts were made to design the Super-TWERL's support equipment (e.g. water pump) as these scoping studies were intended to simply investigate the nuclear capabilities of the core and these support items will likely lie outside of the active core.

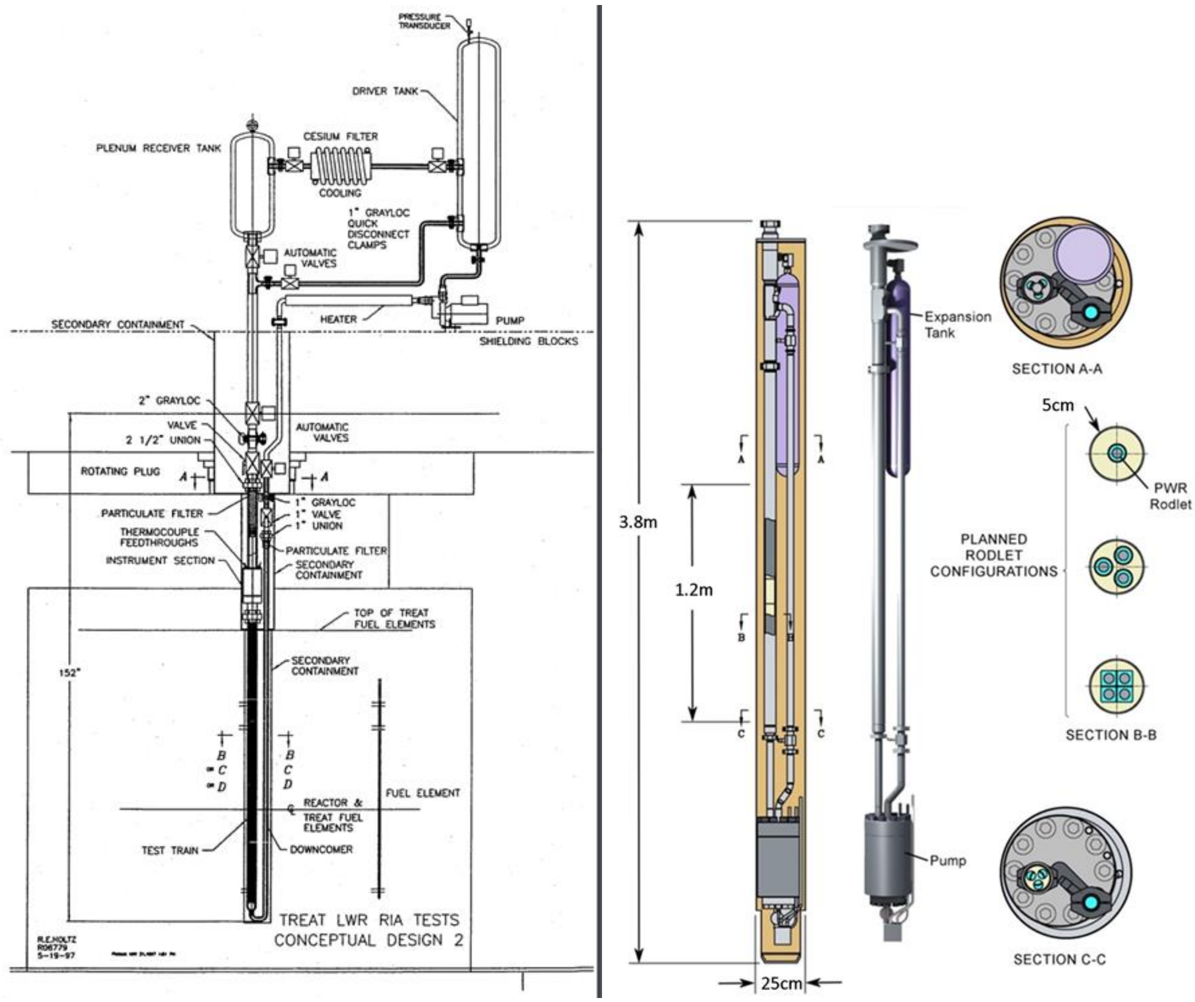

FIGURE 3. TREAT circulating water-loop concept developed in 1997 [8] (left) and current concept, TWERL (right).

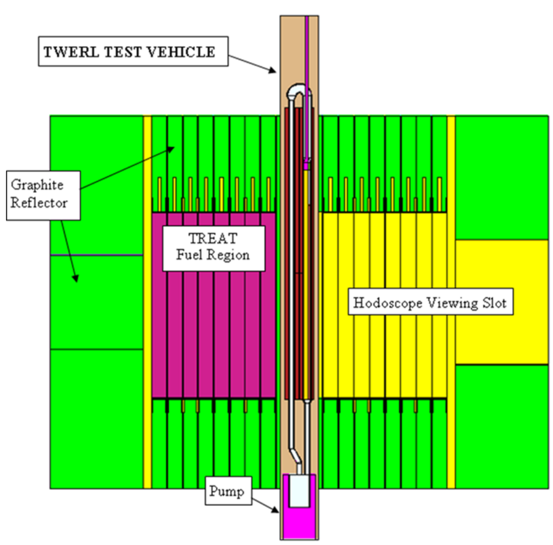

FIGURE 4. MCNP vertical profile plot of TWERL inserted into TREAT half-slotted core.

TREAT CAPABILITY UPGRADE

In the mid 1980's, the facility had been upgraded in many ways to expand capability for testing a much larger experiment consisting of a LMFBR 37-pin bundle. The 37-pin bundle was determined the minimal size needed to substantively move forward in the LMFBR licensing process. This capability was "not possible at any other experiment facility [10]." Most necessary upgrades had been completed including design and manufacture of TREAT upgrade (TU) fuel assemblies. The new assemblies were designed with the intent to boost neutron energy transfer to the test load in a more flattened radial distribution among the test pins. In order to accomplish this, the TREAT core was divided into four zones with various fuel loadings. A radial gradient of uranium to carbon (U/C) atom ratio fuel loading was introduced to flatten the radial power profile in each zone, and likewise, an azimuthal gradient was introduced to offset the negative effects of the open hodoscope viewing slot [11]. TREAT fuel consists of highly enriched uranium (HEU) microspheres, ~93wt% U-235 enriched, dispersed in a carbon/graphite matrix. Original TREAT fuel element U/C atom ratio is 1:10,000. TU fuel element canisters house a 4x4 fuel block array. Each of the 16 columns within one fuel element could consist of a different U/C atom ratio loading in the range shown in Table 1; however, axial fuel loading in the individual columns remains unchanged. [11] The TU fuel elements did change in fuel block height. Rather than the 1.22 m height of the fuel columns in the original TREAT fuel elements, the TU element fuel height was increased to 1.52 m, pulling 0.15m from the graphite reflector region of the original fuel element dimensions at each end. The outer dimensions of the TU fuel assemblies remained essentially the same as the original assemblies [11].

TABLE 1. TU Designed Fuel Loadings Compared to Simplified Average Core Loading used in Scoping Studies.

Region	TU Designed U/C atom ratio	MCNP 9-Rod Scoping Study U/C atom ratio
Insert	1:700-1:1200 [11]	1:950
Converter	1:1000-1:1500 [11]	1:1,250
Buffer	1:2500-1:5000 [11]	1:10,000
Driver	1:10,000[12]	1:10,000

The segmentation of core zones was primarily implemented to allow a region of fuel elements near the test specimen to operate at a much higher power at a much higher temperature compared to the original core. The first zone, a 5 x 5 core element array, was termed the insert zone designed to accommodate a specific experiment. The two inserts of consideration were the Mark-III (7-pin bundle) test loop, and the 37-pin LMFBR advanced test loop (ATL). Figure 5 shows a cross section of the upgrade core with the Mark-III insert depicting separate loading zones along with a smaller cross section of the ATL version in the upper right corner. Note how 9 central elements in the Mark-III version have no fuel loading, similar to the TWERL configuration. Outside the insert region is the converter region. The assemblies in this zone are clad in Inconel-625 which allows the fuel elements to reach higher temperatures than the Zircaloy cladding would allow for the original TREAT fuel elements. The converter region lies within the 11x11 buffer region that is populated with fuel elements at a less dense fuel loading to protect the Zircaloy clad driver elements in the outside driver region. Fuel elements in the driver region are the original TREAT fuel elements. Wade, et. al. mentioned a reference core of 17x17 elements across as a way to modify excess reactivity for various experiments and claimed the TU core was predicted to generate twice the number of neutrons of that generated in the original TREAT core [11]. This extraordinary capability is explored to a small degree in this scoping study. Two simple rings of assemblies surrounding the test vehicle were introduced with fuel loadings per ring determined by the average TREAT upgrade insert and converter region loadings as shown in Table 1.

9-ROD MCNP SCOPING STUDY SET-UP

The Original Core

Two Super-TWERL concepts were evaluated, one each for Inconel and Zircaloy loop piping containment structures, with approximate wall thicknesses needed for pressure containment, as shown Figure 6. All specimens were assumed UO_2 in Zircaloy-4 PWR rods at initial 4.95% enrichment with 1.22 m fuel length. Two specimen burnup levels were evaluated including fresh and relatively high burnup of 70GWD/kgU. The isotopic inventory of the high burnup rod was computed using the ORIGEN code. Although the Super-TWERL is intended to be a liquid-water

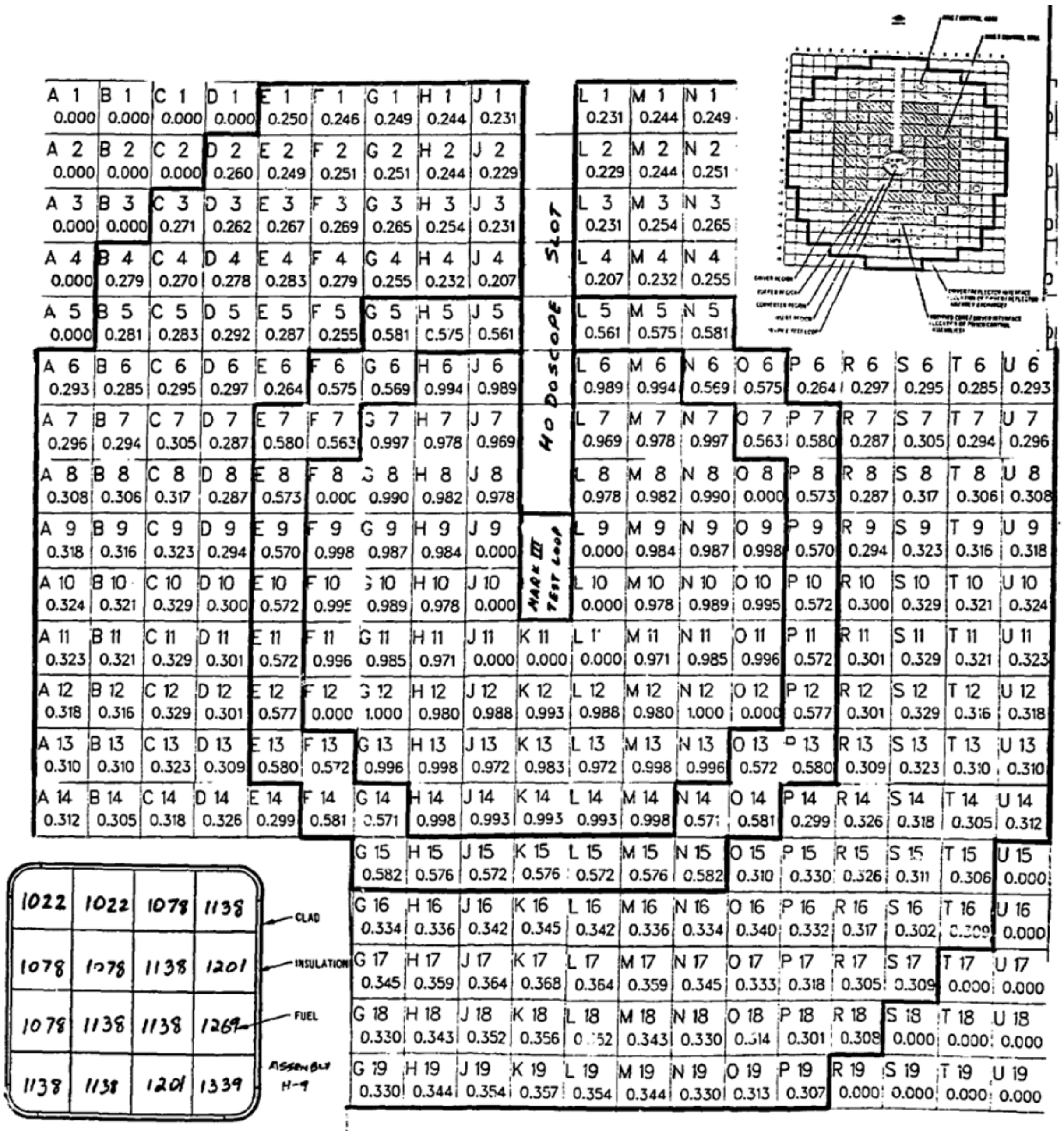

FIGURE 5. TREAT Upgrade fuel element loading with corresponding avg./peak power profile [13].

loop, both PWR water and steam environments were evaluated to also give some rough indication of TREAT's capability for steam-based testing. The core excess reactivity was calculated to show whether a suitably large transient can be performed, the target transient being akin to the 1400 MJ 72ms Full Width Half Maximum (FWHM) clipped transient as described in a historic TREAT water loop proposal [4]. The steady state Power Coupling Factor (PCF) was calculated to show whether adequate specimen energy could be deposited, the target PCF being ≥0.88 J/g per MJ of reactor energy so that ≥1200 kJ/kg (290 cal/g) could be deposited as described in reference [4]. These calculations using MCNP are considered to be for scoping analysis and are therefore only suitable for that purpose. MCNP model renderings and the rod numbering scheme can be seen in Figure 6.

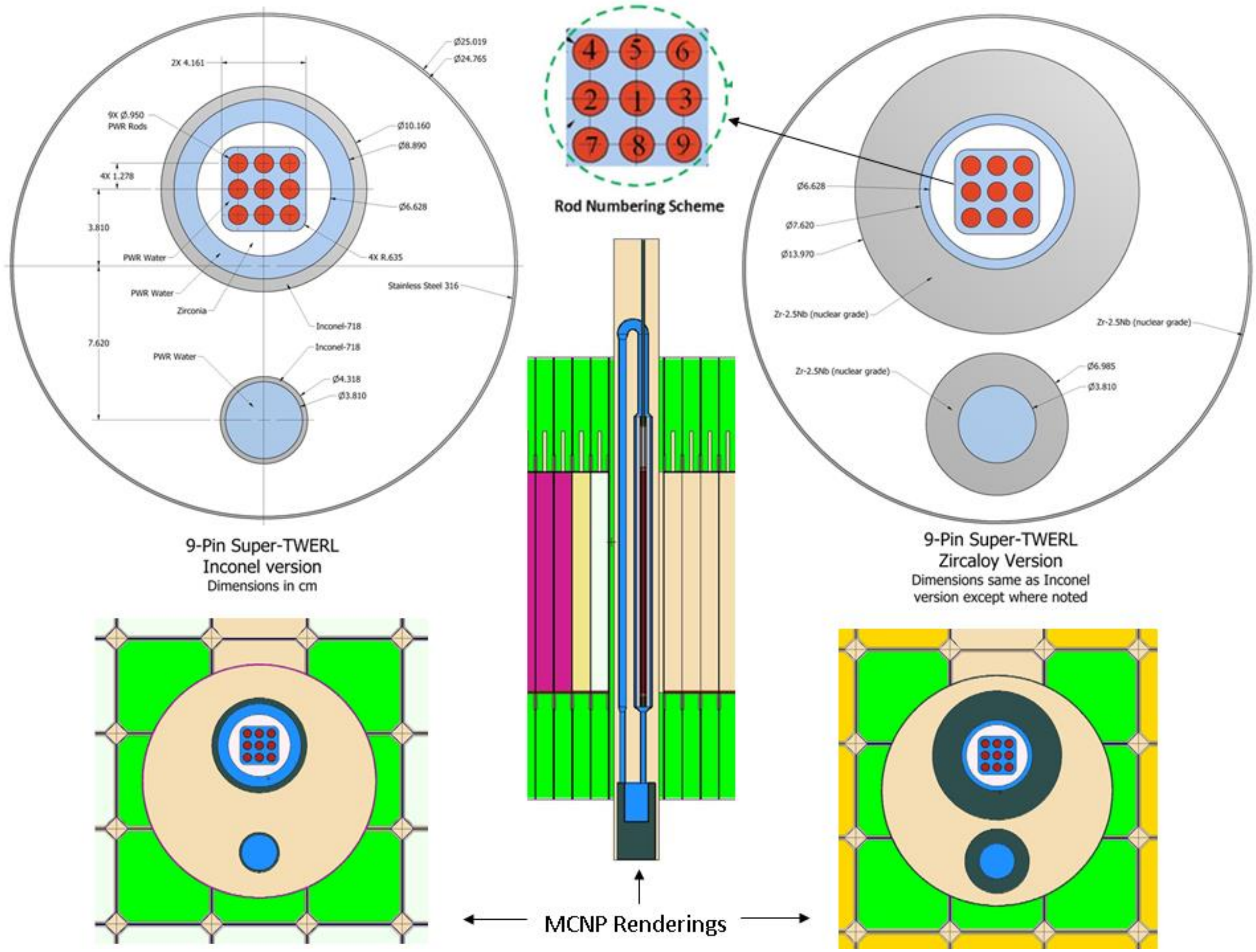

FIGURE 6. Super-TWERL cross section sketches for PWR 9-rod scoping studies

TREAT Upgrade Core

All scoping study calculations done for the original core are done for a simplified, partial version of the TU core. The TU fuel elements were modeled with the same fuel dimensions as that for the original core; however, the cladding material was changed from Zircaloy-3 to Inconel-625. The core configuration was changed from the original TREAT core to include a single ring of the insert region fuel elements around the test vehicle and a single converter region ring immediately surrounding the insert ring. Fuel elements at the outer edge of the core were removed to reduce core excess reactivity. See Figure 7. Fuel loadings for the two rings immediately surrounding the test vehicle region are given in Table 1.

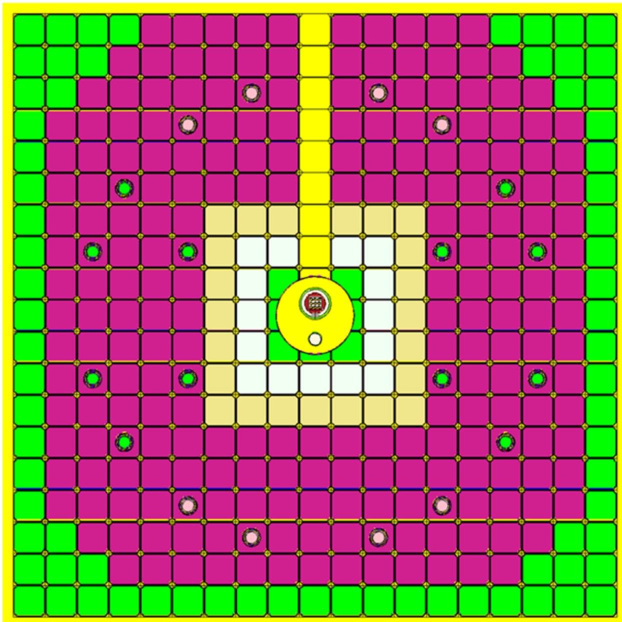

FIGURE 7. TU core configurations used in MCNP 9-rod scoping study

RESULTS AND DISCUSSION

The predicted peak PCF's for the center rods are shown in Table 2. The product of the 1400 MJ reactor energy transient and case-specific PCF is also tabulated as a quick check to see whether at least 1200 J/g-UO_2 can be deposited in the specimens. These results demonstrate that the TREAT original core, coupled with the Super-TWERL design, will allow for adequate energy deposition in cases with fresh specimens. These results also show that reduced absorption of neutrons through a Zircaloy loop pipe wall can boost PCF to meet targets. However, it was found that the original core can approach, but not achieve the 1200 J/gUO_2 target with high burnup specimens in an Inconel water loop for this design. However, the increased PCF shown in the partial TU core indicated that the specimen energy target can be met for high burnup rods in the Inconel loop.

The Inconel loop is considered the baseline option due to its substantial strength and successful use in PWR-environment transient testing at the Power Burst Facility (alloy -718) [14]. At this point the Zircaloy loop can only be considered a potential opportunity without further detailed mechanical design. While Zircaloy's inferior mechanical properties can be compensated by increased wall thickness this approach may encounter an upper limit due to thermal gradient stresses through the wall, inferior mechanical properties at elevated temperature, and potential for exothermic reaction during hypothetical scenarios. Inconel is also the preferred option due to availability, cost, weldability, and other engineering considerations. These studies indicate that the increased PCF observed in the partial-TU core is likely to be an important capability in meeting experiment requirements for high burnup bundles.

TABLE 2. Predicted PCF for Center Rod, Water Environment.

Case			Super-TWERL in **Original** Core		Super-TWERL in **Partial TU** Core	
Environment	Super-TWERL Design	Specimen Burnup	Peak PCF (J/gUO_2-MJ)	Peak Energy Injected, 1400 MJ transient, (J/gUO_2)	Peak PCF (J/gUO_2-MJ)	Peak Energy Injected, 1400 MJ transient, (J/gUO_2)
PWR Water	Inconel Loop	Fresh	1.339	1874	1.655	2317
		70 GWD/kgU	0.755	**1058**	0.926	**1296**
	Ziracaloy Loop	Fresh	1.811	2535	1.716	2403
		70 GWD/kgU	1.058	1481	0.969	1356

Similar cases were computed for the Super-TWERL with a steam environment and results are presented in Table 3. These results indicate that TREAT likely has some excellent capabilities for simulating steam-environment accidents such as PWR Loss of Coolant Accidents (LOCA). LOCA simulations would likely be much longer shaped transients without clipping. This would enable planned reactor energy depositions of up to ~2900 MJ [15] with substantial energy deposition in specimens.

TABLE 3. Predicted PCF for Center Rod, Steam Environment.

Case			Super-TWERL in **Original** Core		Super-TWERL in **Partial TU** Core	
Environment	Super-TWERL Design	Specimen Burnup	Peak PCF (J/gUO_2-MJ)	Peak Energy Injected, 2900 MJ transient, (J/gUO_2)	Peak PCF (J/gUO_2-MJ)	Peak Energy Injected, 2900 MJ transient, (J/gUO_2)
Steam	Inconel Loop	Fresh	1.151	3337	0.689	1999
		70 GWD/kgU	0.634	1837	0.380	1103
	Ziracaloy Loop	Fresh	1.971	5715	0.938	2721
		70 GWD/kgU	1.059	3071	0.518	1502

Comparison of water and steam cases in the Inconel loop demonstrates that the presence of liquid water is instrumental in boosting PCF. This result is consistent with observations from recent modelling of static-water ATF test designs [16]. The effect is further enhanced in the partial-TU core likely due to the spectral hardening combined with the water-moderator effect. The PCF's are reduced in both the steam environment and Zircaloy contained partial-TU cases most like due to lack of water reflection and hardening of the spectrum. These results also show that PCF is slightly greater in the steam cases, when compared to similar water cases, for the Zircaloy loop design. This would indicate that the thin water annulus may be shielding incoming core neutrons. In this case the effect is small, but demonstrates the interplay between light-water as both moderator and parasitic absorber with the sensitivity to spectral shifts in influencing PCF. These studies do not show the same PCF-boosting effects from TU elements as shown in PWR water testing. However, it may be possible to design a test vehicle which provides steam environment for the specimens with a surrounding moderator region such as a liquid water cooling jacket or beryllium ring to maximize the PCF-boosting interaction between the TU elements' and moderator surrounding the test region. Overall these results continue to emphasize the complex relationship between light-water as both moderator and parasitic absorber, neutron absorption through loop structural materials, and spectral effects from driver fuel in influencing the energy deposited in the specimens. No attempts were made to optimize these relationships in this quick scoping study, but PCF's higher than those shown in Table 2 could possibly be obtained from a future optimization effort.

Scoping studies indicated core reactivity worth of ~5.5 (k-excess %). Apart from fueling the hodoscope slot, which likely represents an unacceptable sacrifice in experiment data, the core assumed in these models is the largest that can practically be achieved. A 1400 MJ pulse will essentially require initiation of a ~2500 MJ naturally shaped pulse (4.5% Δk/k insertion needed, 23ms period [15]) followed by clipping. Noting that significant model biases could be present, especially since this experiment configuration is quite unlike any potential benchmark cases in TREAT's history, it appears that the Super-TWERL original core configuration has just enough excess reactivity available. Considering that future design evolutions of the Super-TWERL could call for thickening of the containment structures for added safety margin, which is especially impactful in the Inconel case, or that additional neutron absorbing materials could be needed in the active core region for specimen power shaping, there is a risk that the final core excess reactivity would be inadequate to support 9-rod testing. Shortly after the first calculations, the MCNP model of the Super-TWERL partial-TU core had to be modified so that the peripheral ring of original core fuel assemblies were replaced with graphite dummies in order to yield a subcritical core with control rods fully inserted. While the fidelity of this model was certainly insufficient to produce critical rod predictions for such a core, this effect quickly led to the realization that TU elements could probably be used to recover core excess reactivity when needed. This capability would be particularly useful for large experiment configurations which displace several fuel assemblies and/or for vehicles with large negative reactivity worth. Hence, the capability to install TU elements could become a critical aspect in meeting future experimental needs.

Another interesting comparison from the original core and partial-TU core is the relative power distribution between rods. Since the TU core was originally designed with the intent of spectral hardening to reduce peaking in exterior pins, it is not surprising to see similar effects in 9-rod bundles. This aspect of using a partial-TU core configuration could be very valuable in enabling adequate energy deposition in the central rod without depositing too much energy in the surrounding rods. See Table 4.

TABLE 4. Power Distribution Comparisons.

Original Core			Partial-TU Core		
Inconel, Fresh, PWR Water			**Inconel, Fresh, PWR Water**		
Rod to Bundle Average Ratio			Rod to Bundle Average Ratio		
Rod 4	Rod 5	Rod 6	Rod 4	Rod 5	Rod 6
1.03	0.96	1.03	1.01	0.97	1.01
Rod 2	Rod 1	Rod 3	Rod 2	Rod 1	Rod 3
0.98	0.89	0.98	0.99	0.93	0.99
Rod 7	Rod 8	Rod 9	Rod 7	Rod 8	Rod 9
1.06	0.99	1.07	1.04	1.00	1.05
Inconel, 70 GWD, PWR Water			**Inconel, 70 GWD, PWR Water**		
Rod to Bundle Average Ratio			Rod to Bundle Average Ratio		
Rod 4	Rod 5	Rod 6	Rod 4	Rod 5	Rod 6
1.03	0.97	1.03	1.01	0.97	1.01
Rod 2	Rod 1	Rod 3	Rod 2	Rod 1	Rod 3
0.98	0.91	0.99	0.99	0.94	0.99
Rod 7	Rod 8	Rod 9	Rod 7	Rod 8	Rod 9
1.04	0.99	1.06	1.04	1.01	1.04
Zircaloy, Fresh, PWR Water			**Zircaloy, Fresh, PWR Water**		
Rod to Bundle Average Ratio			Rod to Bundle Average Ratio		
Rod 4	Rod 5	Rod 6	Rod 4	Rod 5	Rod 6
1.04	0.94	1.03	1.00	0.97	1.00
Rod 2	Rod 1	Rod 3	Rod 2	Rod 1	Rod 3
0.97	0.87	0.97	0.99	0.95	0.99
Rod 7	Rod 8	Rod 9	Rod 7	Rod 8	Rod 9
1.09	1.00	1.09	1.05	1.01	1.05
Zircaloy, 70 GWD, PWR Water			**Zircaloy, 70 GWD, PWR Water**		
Rod to Bundle Average Ratio			Rod to Bundle Average Ratio		
Rod 4	Rod 5	Rod 6	Rod 4	Rod 5	Rod 6
1.02	0.95	1.02	1.00	0.97	1.00
Rod 2	Rod 1	Rod 3	Rod 2	Rod 1	Rod 3
0.98	0.90	0.98	0.99	0.95	0.99
Rod 7	Rod 8	Rod 9	Rod 7	Rod 8	Rod 9
1.07	1.00	1.07	1.05	1.02	1.05

CONCLUSIONS

The neutronic studies presented in this report show that the TREAT facility is capable of performing transient tests on fresh 9-rod PWR bundles in a water loop design constructed from Inconel piping using only fuel assemblies from the original core. High burnup rods in the same proposed design configuration approach, but do not meet, energy deposition targets for narrow-pulse RIA testing. The option to use a Zircaloy-based water loop was evaluated and found to increase PCF adequately, but several engineering considerations show that the Inconel loop should be considered the baseline design option until detailed design work is performed. TREAT's capabilities for steam-based LOCA testing were also evaluated and found to allow for higher specimen energy deposition than RIA tests primarily because the reactor can release more energy over longer shaped transients. The same test configurations were evaluated in the presence of a partial TU core with the notable observation that TU elements can enable high burnup rods to achieve RIA energy targets with the Inconel water loop. Further observations show that the partial TU core can help recover excess reactivity as needed and flatten pin-to-pin power profiles for more prototypic test conditions. These studies demonstrate that a simple U/C atom ratio gradient implementation results in more energy deposition into the test specimen and works to flatten the rod-to-rod power profile. It is expected that the actual, specially designed TREAT upgrade core will be much more effective than this study of a modified version demonstrates and deserves a comprehensive investigation to identify true capability. Overall, the complex interplay between fuel, geometry, neutron absorbers, and moderators emphasize that the facility's capabilities are flexible and further enabled by the ability to use TU elements.

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