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SULI Intern Report

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Enhancing TREAT's Pulsing Abilities Using TREAT Upgrade Fuel

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The Transient Reactor Test Facility (TREAT) at Idaho National Laboratory is an air-cooled, thermal, heterogeneous facility used to test reactor materials in simulated accident conditions by inducing fission heating with intense neutron pulses[1]. TREAT operated from 1959 until 1994 with the primary goal of testing fast reactor fuels[2] and was brought back online in 2017 to re-establish DOE's nuclear fuels transient testing capabilities[3]. Beginning in the late 1970s, Idaho National Laboratory worked in conjunction with other organizations to increase TREAT's capability for in-pile testing[1]. While these upgrades had not been implemented by the time the reactor was put into standby mode, new assemblies and graphite-urania fuel blocks with increased uranium concentrations had already been designed and fabricated. This project, as part of Idaho National Laboratory's Department of Reactor Physics, modeled the TREAT Upgrade (TU) fuel assemblies in MCNP and implemented them in an existing TREAT model. Calculations were performed using INL's high performance computer to find a critical combination of TU and standard fuel and this geometry's excess reactivity. Future work will include calculating a power coupling factor in the experiment, determining power peaking factors throughout the core, and assessing how much reactivity would be added by replacing the Inconel-625 cladding with silicon carbide.

I. INTRODUCTION

TREAT is an air-cooled, graphite moderated, thermal, heterogeneous test facility designed to evaluate reactor fuels and structural materials under conditions that simulate various types of nuclear accident situations and excursions from normal operation. Among the things studied are behavior in transient conditions, fuel melt-down, metal-water reactions, and thermal interactions between overheated fuel and coolant[1]. Transient testing involves placing the test material into the core and subjecting it to short bursts of intense, high power radiation. This contributes to reactor safety knowledge by providing basic data to predict the safety margin of fuel designs and the severity of potential accidents, and as a method of proving fuel design concepts meant to reduce or prevent hazards[4]. TREAT was was also designed to be used as a large neutron radiography facility to examine assemblies up to 15 ft in length, providing nondestructive test data of fuel samples irradiated in other test reactors[4].

TREAT operated from 1959 to 1994 with the primary purpose of testing fast-reactor fuels by simulating accident conditions that lead to fuel damage, including melting or vaporization of test specimens, while not damaging the reactor's fuel. In that time, TREAT generated over 720 MWh of energy in 6604 reactor startups and 2884 transient irradiations. It was brought back online in late 2017, resuming operations in 2018[5]. In alignment with more current DOE missions, TREAT is now used to test light water reactor fuels and space reactor fuels. Its current basic capabilities include producing pulses up to 20 GW for up to 80 milliseconds for severe accident testing, producing flexible power shapes for up to one minute, steady state operation at 120 kW, testing static capsules, sodium loops, and water loops, and acting as a neutron radiography facility[5].

The TREAT Upgrade project began in the late 1970's in conjunction with Argonne National Laboratory, the NUS Corporation, and the University of Chicago. It aimed to increase TREAT's in-pile testing capabilities. The upgrade involved the design and testing of a new test vehicle called the Advanced TREAT Loop (ATL), its associated systems, and modifications of the TREAT reactor and associated systems to accommodate the increased energy deposition requirements. In 1988, new instrumentation and control systems were installed and the rod drive systems refurbished in order to improve energy delivery to the core by adjusting the shape of the spatial flux. Upgraded fuel, the intended layout of which is seen in Figure 1, was also designed and fabricated, but not implemented before the reactor was decommissioned. The new fuel layout utilized some of the standard TREAT fuel already in use, referred to in the image as driver fuel, and also implemented upgrade fuel with higher concentrations of uranium than the standard fuel.

Now that the reactor is back online, one focus of improving operations is shortening the transient pulse and hardening its neutron spectrum in order to simulate more severe accident conditions. This SULI project, in conjunction with INL's Department of Reactor Physics, aims to implement modeled TREAT Upgrade (TU) fuel in a model of TREAT's current layout, find a critical geometry, the excess reactivity, power peaking factors (PPFs), power coupling factors (PCFs), and model silicon carbide as the cladding instead of the designed Inconel-625. Implementing TU fuel could shorten the pulse and harden the spectrum because of its higher uranium concentration. PPFs will be used to confirm that any additional heating from the added reactivity would not exceed de-

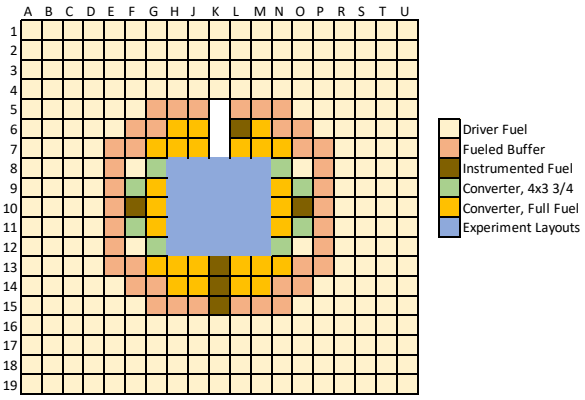

FIG. 1. A map of the intended Treat Upgrade core layout, where driver fuel is standard TREAT fuel. Fueled buffers, instrumented fuel, full and 4x3 converter fuel are different assembly types created for the TREAT Upgrade geometry.

sign thermal stresses, and excess reactivity and PCF will be used to determine if there is enough negative reactivity in the control rods to keep the reactor from going supercritical, but enough excess reactivity to create a strong enough transient. General Atomics's new silicon carbide design process means that silicon carbide could be used as a cladding material, which would also add reactivity as it would be replacing Inconel, the nickel in which absorbs neutrons.

II. STANDARD FUEL VS. TREAT UPGRADE FUEL

The TREAT Upgrade project's main goal was to accommodate new test fuel conditions typical of a fast breeder reactor design. To do so, it designed a new larger experiment test vehicle, the Advanced TREAT Loop (ATL). A fuel redesign was required because operation of the ATL and its associated systems required higher energy deposition than previous experiments, causing high thermal stresses on the existing fuel design. The main differences between the standard assemblies and the upgraded ones are materials of construction, the method of attaching the fuel to the top and bottom fittings, increased fuel length, and a continuous outer cladding can[1]. The geometric dimensions of the upgrade fuel are essentially the same as a standard treat assembly, but with higher design temperature. The design temperature of the TU fuel for normal operation is 950 °C, compared to the standard fuel's design temperature of 650 °C. The center 11x11 portion of the 19x19 fuel arrangement, highlighted in Figure 1, was upgraded in order to provide greater energy deposition to the ATL[1] and includes buffer assemblies, converter assemblies, and instrumented assemblies.

From a modeling and neutronics perspective for the scope of this project, only some of the differences between the standard and upgrade fuel are of importance and therefore focused on. These include changes in construction materials, increase in fuel length and resulting decrease in both upper and lower reflector length, continuous outer cladding, increase in uranium concentration in fuel, and introduction of radial uranium concentration gradation. Table I highlights the differences in dimension and material for the major components of a fuel assembly.

TABLE I. A table comparing the materials and dimensions for the standard fuel (SF) and TREAT Upgrade fuel (TU)[14].

Component	SF Material	TU Material	SF Length	TU Length
Top Fitting	Al-1100	Inconel-600	3.75”	4.25”
Top Reflector	CP-2 Graphite	CP-2 Graphite	25.5”	18.6”
Fuel	Graphite-Urania Fuel, C/U 10,000	Graphite-Urania Fuel, C/U 5300 - 500	48.125”	60”
Bottom Reflector	CP-2 Graphite	CP-2 Graphite	23.875”	17.25”
Bottom Fitting	Al-1100	Inconel-600	1”	1.5”
Alignment & Support Pin	Al-1100	Inconel-600	6”	5.65”
Fuel Cladding	Zircaloy	Inconel-625	0.025” thick	0.025” thick

The most notable difference between standard and TU fuel is the change in carbon to uranium ratio, referred to as C/U ratio. The C/U ratio in the standard fuel is approximately 10,000, but ranges between 5,300 and 500 in the TU fuel. The standard fuel is one solid 4”×4”×48” block of uranium dispersed in a graphite matrix. The TU fuel, while also using graphite-urania blocks, is comprised of 16 1”×1”×60” fuel rods with varying C/U ratios arranged in a 4×4 to make a 4×4”×60” fuel block. These rods were sectioned into shorter pieces for machining purposes[6], but are modeled as one full length rod as there is no difference neutronically. Figure 2 shows this 4×4 grid of different C/U ratios in each TU assembly. The C/U ratios were outlined in an Argonne National Laboratory specification report[6], and Equation 1 was used to convert from C/U ratio to uranium weight percent for MCNP isotopics.

$$\frac{C}{U} = \left(\frac{99.94}{\text{wt \% U}} - 1.136\right)\frac{235.2}{12.0111} \tag{1}$$

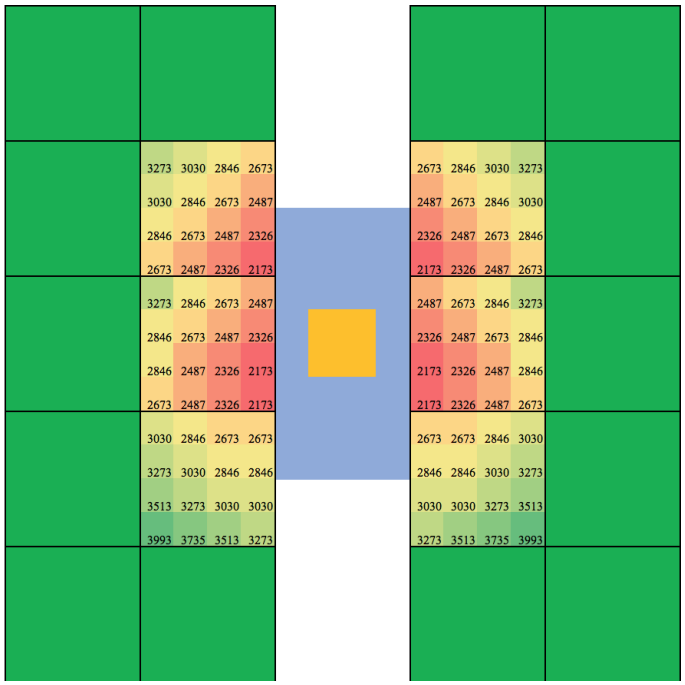

FIG. 2. The concentrations of the TU fuel included in the arrangement. The solid green blocks are the standard fuel, with the experiment in blue and yellow.

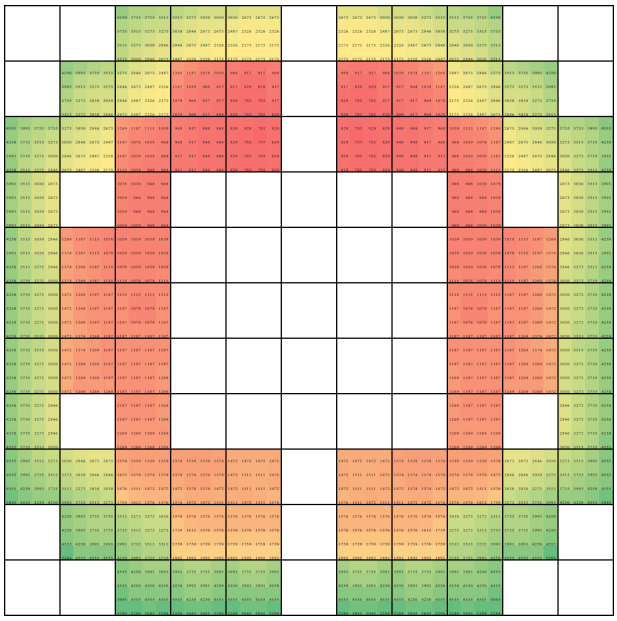

FIG. 3. The intended concentration gradient of TU fuel, with green at the lowest uranium concentration of 1:5300 and red at the highest of 1:500.

III. RESULTS

A. Modeling TREAT Upgrade Fuel

Figure 1 shows the intended TREAT Upgrade core layout for the ATL experiments. The full fuel assemblies in the converter region have a pyrolytic graphite layer that surrounds the fuel and partially surrounds both the top and bottom reflectors. The buffer fuel, while also referred to as full fuel, does not have this pyrolytic graphite layer as the uranium concentration is much lower. Figure 3 shows the uranium concentration gradient of this intended layout. In the outer ring of buffer assemblies, the carbon-uranium ratio ranges from approximately 5300 to approximately 2200; in the inner ring of converter assemblies, it ranges from approximately 1900 to approximately 750. The lower ratio TU fuel, down to approximately 500, was intended for use in the inner 5x5 square to surround the experiments. Two experiment layouts were designed for use with the TU fuel, the ATF experiment and the Mark III experiment.

B. Criticality

Various combinations of TU and standard fuel were used to find a to find a critical geometry that combined standard and TU fuel. The buffer TU fuel assemblies were used, as they have the lowest uranium concentrations. Using the higher concentration assemblies would cause power peaking factors to be too high in the TU assemblies.

One way criticality is controlled and maintained is using control rods, which in TREAT use boron carbide powder to absorb neutrons and can be inserted and removed into the reactor to control reactivity. A critical arrangement was determined by calculating k_{eff} using MCNP with the control rods fully inserted and fully removed to confirm that a k_{eff} of 1 would be achieved somewhere in that window. Criticality was found to occur using assemblies F7, O7, G6, N6, F13, and O13 from the intended TU arrangement, placed at the six fuel locations closest to the experiment as shown in Figure 2.

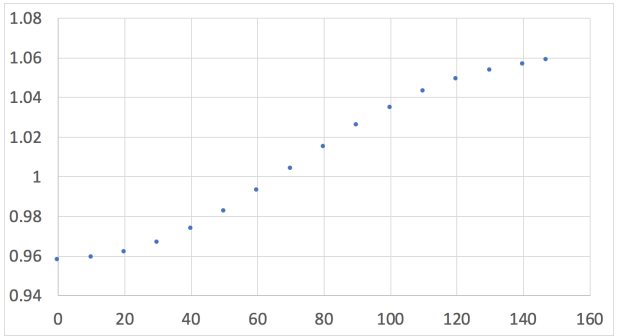

FIG. 4. The progression of k_{eff} as the control rod is removed in increments of 10 cm from 0 to 147.32. The error on each of these results is ±0.0005.

Once it was established that k_{eff} would equal 1 somewhere in this window, the control rod position was increased from 0 in increments of 10 cm until fully removed,

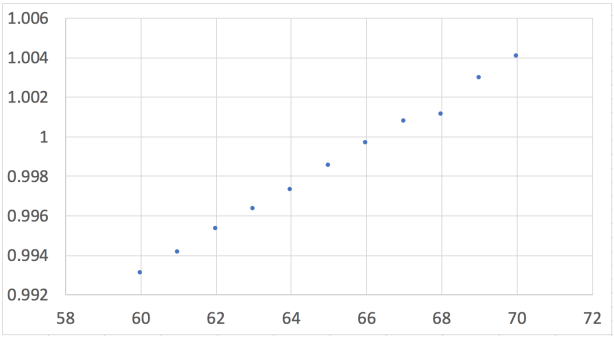

FIG. 5. The progression of k_{eff} as the control rod is removed in increments of 1 cm from 66 to 67. The error on each of these results is ±0.0005.

the progression of which can be seen in Figure 4. Once a 10 cm window in which k_{eff} became 1 was found, in this case between 60 and 70 cm, the control rod position was shifted in increments of 1 cm. This can be seen in Figure 5. Criticality with this arrangement was occurs at a control rod insertion of approximately 66.3 cm, where 0 cm is fully inserted and 147.32 is fully removed. K_{eff} with the rod fully removed is 1.05926 ±0.0005, making the excess reactivity in the reactor 5.926%.

C. Power Peaking Factors

Power peaking factor is a ratio of an assembly's power to the core average power density to determine where in the reactor the peak thermal energy occurs. This is useful in determining which assemblies will be the limiting assemblies in terms of maximum temperature and thermal stress. Figure 6 shows a power peaking factor map of the core, showing that the TU fuel assemblies have much higher power density compared to the rest of the core.

	A	B	C	D	E	F	G	H	J	K	L	M	N	O	P	R	S	T	U		
1	0.016	0.023	0.709	0.714	0.727	0.731	0.728	0.725	0.729	0.010	0.737	0.748	0.767	0.784	0.795	0.797	0.806	0.027	0.018	0	Minimum
2	0.024	0.729	0.727	0.739	0.743	0.735	0.715	0.698	0.733	0.012	0.741	0.721	0.755	0.792	0.818	0.830	0.833	0.854	0.028	0.337	10%
3	0.745	0.753	0.785	0.807	0.793	0.754	0.737	0.570	0.772	0.013	0.781	0.590	0.781	0.817	0.877	0.913	0.907	0.894	0.916	0.673	20%
4	0.778	0.810	0.867	0.901	0.869	0.646	0.847	0.869	0.885	0.015	0.896	0.901	0.898	0.702	0.965	1.022	1.007	0.972	0.977	1.010	30%
5	0.839	0.883	0.957	1.012	1.024	1.011	1.038	1.044	1.003	0.017	1.016	1.085	1.104	1.098	1.138	1.150	1.115	1.063	1.056	1.346	40%
6	0.904	0.956	1.044	0.835	1.176	1.202	1.204	1.172	1.083	0.019	1.099	1.218	1.284	1.307	1.305	0.949	1.219	1.152	1.134	1.683	50%
7	0.950	1.013	1.117	1.215	1.288	1.321	1.309	1.240	1.107	0.022	1.125	1.293	1.396	1.439	1.432	1.384	1.310	1.229	1.202	2.019	60%
8	0.972	1.051	0.871	1.283	1.365	1.035	1.359	1.237	1.048	0.027	1.069	1.296	1.453	1.129	1.521	1.468	1.026	1.288	1.255	2.356	70%
9	0.964	1.068	1.206	1.331	1.415	1.436	1.374	1.184	3.294	0.035	3.365	1.244	1.475	1.572	1.580	1.524	1.429	1.328	1.290	2.692	80%
10	0.951	1.076	1.223	1.350	1.437	1.457	1.382	1.161	3.043	ms	3.119	1.222	1.483	1.593	1.604	1.548	1.450	1.345	1.303	3.029	90%
11	0.972	1.078	1.215	1.340	1.425	1.448	1.389	1.206	2.959	0.082	3.029	1.266	1.488	1.581	1.589	1.532	1.433	1.333	1.295	3.365	Maximum
12	0.993	1.069	0.885	1.300	1.383	1.050	1.386	1.278	1.127	0.027	1.146	1.334	1.478	1.142	1.537	1.482	1.034	1.297	1.265		
13	0.984	1.042	1.143	1.239	1.313	1.349	1.343	1.288	1.184	0.022	1.201	1.342	1.429	1.467	1.457	1.404	1.325	1.243	1.216		
14	0.951	0.994	1.078	0.858	1.205	1.235	1.245	1.223	1.156	0.020	1.173	1.271	1.322	1.340	1.334	0.967	1.239	1.170	1.150		
15	0.901	0.929	0.996	1.047	1.057	1.045	1.080	1.097	1.076	0.018	1.090	1.140	1.147	1.134	1.168	1.178	1.140	1.085	1.075		
16	0.843	0.856	0.904	0.934	0.902	0.673	0.888	0.924	0.960	0.016	0.972	0.957	0.943	0.730	0.999	1.053	1.034	0.995	0.997		
17	0.799	0.792	0.817	0.836	0.823	0.790	0.781	0.614	0.853	0.014	0.864	0.637	0.831	0.859	0.915	0.947	0.937	0.921	0.942		
18	0.025	0.759	0.750	0.760	0.769	0.772	0.766	0.767	0.829	0.013	0.839	0.797	0.818	0.845	0.863	0.868	0.867	0.885	0.029		
19	0.016	0.024	0.722	0.722	0.745	0.771	0.791	0.814	0.849	0.011	0.861	0.849	0.848	0.850	0.849	0.842	0.845	0.028	0.019		

FIG. 6. A PPF map of the core with TU fuel, where red is hotter and green is cooler.

IV. CONCLUSIONS AND FUTURE

Criticality has been established with TREAT Upgrade fuel assemblies implemented in a model of the standard TREAT layout with an excess reactivity of 5.926%. Now that use of TU fuel has been established as a viable option to shorten the transient pulse and harden the neutron spectrum, further work will include replacing the Inconel-625 cladding with silicon carbide to determine how much more reactivity is added by removing the neutron-absorbing nickel. Power coupling factors will also be calculated to determine how much energy is being deposited in the experiment.

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