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Determination of Optimal Core Fissile Loadings in the TREAT Upgrade Reactor*

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The TREAT Upgrade (TU) reactor[1] design is presently nearing completion. The reactor will be used to test LMFBR fuel under simulated accident conditions. Such simulations are done by operating the reactor in complex power time histories of several second duration which include, adiabatic super-prompt critical bursts. The physics of the TU core is complicated by a number of factors related to the planned application of the facility. In this paper we report on the design approach used to produce the core fissile loading spatial distribution needed to satisfy the requirement to test a number of different test clusters in various test loops in a transient operating mode.

The TU core consists of a driver comprised of the pre Upgrade Zircaloy-clad TREAT fuel assemblies and a central converter region of new Inconel-clad fuel assemblies surrounding the test loop location. Separating the two concentric zones is a buffer zone that is made up of new Inconel clad assemblies operating at lower temperatures than the converter region (see Fig. 1, inset). The fuel in all regions is a dilute dispersion of enriched UO_2 in a graphite matrix. In order to meet the demanding functional requirements, it was necessary to maximize the energy deposited in each fuel rod in the converter during the transients within a maximum allowable clad temperature constraint, i.e.,

$$\int_{o}^{t} pP(t)\,dt \rightarrow \text{maximum for each rod with } T_{clad} < T_{clad}^{Limit} \ldots \qquad (1)$$

where P(t) is the time dependent power in each rod. The degree of freedom in the design is the fissile loading in each fuel rod (characterized by a C/U atom ratio) within the fabrication cost constraint of a limited maximum number of different C/U loadings permitted.

Such a maximization condition can be satisfied exactly for a single spatial core configuration and a single transient shape. The need to accommodate an ensemble of test loops with different degrees of absorptiveness makes it necessary to adjust the fissile loadings in the vicinity of the loops to compensate for the specific perturbations caused by the individual loops. Since there is a coupling of neutron source throughout the core, the effect propagates throughout the core, decreasing with increasing distance from core center.

The transient shape effects arise due to power shifts that occur during the transient because of control rod movements and spatially non-uniform effects of core heating (dynamic α/v absorption effects and temperature feedback effects). The time integral in Eq. 1 makes the optimization specific to individual transient shapes. In order to determine the fissile loadings in each fuel rod, the design calculations had to be performed in considerable spatial detail. Thus, detailed time-dependent optimizations were ruled out from computing cost considerations. The following approach was devised to perform general design optimizations using static neutronic analyses that most closely approximated the condition of Eq. 1.

The need to accommodate an ensemble of test loops was treated by the use of three specific core "inserts" consisting of a 5x5 array of fuel assemblies designed to match the major classes of test loops geometrically and neutronically. The balance of the modified core was designed to operate with any insert. This was a cost-effective solution that maintained a high performance level for the ensemble of planned test clusters.

The dynamic effects were treated by performing the static optimizations at conditions corresponding to the peak of the burst. At this time the inverse period, α, is 0 and the α/v poisoning term vanishes. Additionally, all of the transient-producing control rods are out of the core, allowing an essentially unrodded optimization. Space-time calculations using the FX-2[2] code established the fact that the several dynamic effects are compensatory and therefore that approximate methods are acceptable. Scoping quasistatic analyses (using simple core models) showed that this approach gave the best approximation to the more detailed time-dependent analyses.

For the design analyses, an 18 energy group cross-section set was used. The cross-sections were generated from basic ENDF/B IV nuclear data using the AMPX cross-section generation package[3] at the temperature corresponding to the peak of the burst. The $S(\alpha,\beta)$ scattering matrix representation was used for graphite. From kinetics analyses of all of the desired classes of transients, it was established that the temperature at the peak of the burst was within 100°C for all of the different transients planned. The optimization was done for the most demanding case since the performance margins over requirements were the smallest for this case. The design calculations were performed by a modified DIF3D[4] diffusion theory code in which a transport slot treatment was imbedded[5]. An auxiliary code was written to perform the fissile loading optimizations such that the condition of Eq. 1 was satisfied for each fuel rod to within the tolerance which could be achieved given the constraint of a limited number of possible C/U loadings. The optimization and diffusion calculations were performed iteratively.

The primary result of the analyses was the specification of the optimal fissile loading in each fuel rod within the modified core and the insert configurations. An indicator of how well the design approach worked is the distribution of relative energy depositions in each fuel assembly in the modified core. As shown in Fig. 1 for one of the core inserts, the energy distribution in the converter is very flat. Similar flat distributions were obtained for the other core inserts. This is important from considerations of clad life degradation over the period of reactor operation. Inset in Fig. 1 are cross-sectional views of the entire core and of a typical fuel assembly showing the range of C/U loadings within a single assembly. Overall, in the core there are 46 different C/U loadings ranging in value from 511 to 5284. The intra assembly energy deposition distribution within each assembly (characterized by a max/average or max/min ratios) is also acceptably flat.

Fuel fabrication is in progress at Los Alamos National Laboratory. It is anticipated that fabrication of all fuel rods will be completed in 1984. Initial criticality of the TU reactor is expected in early 1985.

References

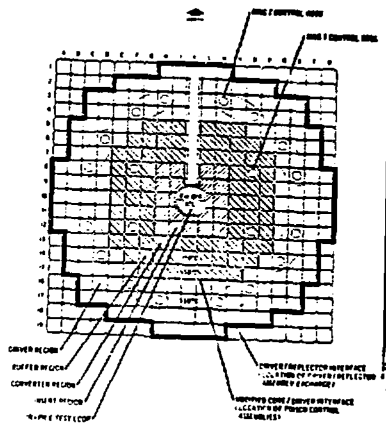
1. D. C. Wade, S. K. Bhattacharyya, W. C. Lipinski and C. C. Stone, "Core Design of the Upgraded TREAT Reactor," Proc. Conference on Fast, Thermal and Fusion Reactor Experiments, Salt Lake City, April 1982.

2. R. A. Shober, T. A. Daly and D. R. Ferguson, "FX2-TH: A Two-Dimensional Nuclear Reactor Kinetics Code with Thermal Hydraulic Feedback," ANL-78-97, October 1978.

3. N. M. Greene, et.al., "AMPX: A Modular Code System for Generating Coupled Multigroup Neutron-Gamma Libraries from ENDF/B," ORNL/TM 3706, March 1976.

4. D. R. Ferguson and K. L. Derstine, "Optimized Iteration Strategies and Data Management Considerations for Fast Reactor Finite-Difference Diffusion Theory Codes," Nucl. Sci. Eng., Vol. 64, p. 593, 1977.

5. R. M. Lell, R. B. Pond, and S. K. Bhattacharyya, "Development and Validation of a Method to Treat Large Internal Voids in Reactor Design Calculations," NUREG/CP-0034, Vol. 2, p. 657, September 22-24, 1982.



Row	A	B	C	D	E	F	G	H	J	K	L	M	N	O	P	R	S	T	U
1	0.000	0.000	0.000	0.000	0.250	0.246	0.249	0.244	0.231		0.231	0.244	0.249						
2	0.000	0.000	0.000	0.260	0.249	0.251	0.251	0.244	0.229		0.229	0.244	0.251						
3	0.000	0.000	0.271	0.262	0.267	0.269	0.265	0.254	0.231	Ho DOSCOPE SLOT	0.231	0.254	0.265						
4	0.000	0.279	0.270	0.278	0.283	0.279	0.255	0.232	0.207		0.207	0.232	0.255						
5	0.000	0.281	0.283	0.292	0.287	0.255	0.581	C.575	0.561		0.561	0.575	0.581						
6	0.293	0.285	0.295	0.297	0.264	0.575	0.569	0.994	0.989		0.989	0.994	0.569	0.575	0.264	0.297	0.295	0.285	0.293
7	0.296	0.294	0.305	0.287	0.580	0.563	0.997	0.978	0.969		0.969	0.978	0.997	0.563	0.580	0.287	0.305	0.294	0.296
8	0.308	0.306	0.317	0.287	0.573	0.000	0.990	0.982	0.978		0.978	0.982	0.990	0.000	0.573	0.287	0.317	0.306	0.308
9	0.318	0.316	0.323	0.294	0.570	0.998	0.987	0.984	0.000	MARK III TEST LOOP	0.000	0.984	0.987	0.998	0.570	0.294	0.323	0.316	0.318
10	0.324	0.321	0.329	0.300	0.572	0.995	0.989	0.978	0.000		0.000	0.978	0.989	0.995	0.572	0.300	0.329	0.321	0.324
11	0.323	0.321	0.329	0.301	0.572	0.996	0.985	0.971	0.000	0.000	0.000	0.971	0.985	0.996	0.572	0.301	0.329	0.321	0.323
12	0.318	0.316	0.329	0.301	0.577	0.000	1.000	0.980	0.988	0.993	0.988	0.980	1.000	0.000	0.577	0.301	0.329	0.316	0.318
13	0.310	0.310	0.323	0.309	0.580	0.572	0.996	0.998	0.972	0.983	0.972	0.998	0.996	0.572	0.580	0.309	0.323	0.310	0.310
14	0.312	0.305	0.318	0.326	0.299	0.581	0.571	0.998	0.993	0.993	0.993	0.998	0.571	0.581	0.299	0.326	0.318	0.305	0.312
15							0.582	0.576	0.572	0.576	0.572	0.576	0.582	0.310	0.330	0.326	0.311	0.306	0.000
16							0.334	0.336	0.342	0.345	0.342	0.336	0.334	0.340	0.332	0.317	0.302	0.309	0.000
17							0.345	0.359	0.364	0.368	0.364	0.359	0.345	0.333	0.318	0.305	0.309	0.000	0.000
18							0.330	0.343	0.352	0.356	0.352	0.343	0.330	0.314	0.301	0.308	0.000	0.000	0.000
19							0.330	0.344	0.354	0.357	0.354	0.344	0.330	0.313	0.307	0.000	0.000	0.000	0.000

1022	1022	1078	1138	CLAD
1078	1078	1138	1201	INSULATION
1078	1138	1138	1269	FUEL
1138	1138	1201	1339	

ASSEMBLY H-9