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FINAL REPORT
on Government Support
to Cover Analyses of
University of Florida Training Reactor
Conversion from HEU to LEU Fuel

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1.0 INTRODUCTION

The University of Florida Training Reactor (UFTR), operated under NRC License R-56, is a light-water-cooled, graphite-and-light-water moderated, modified Argonaut-type reactor licensed to operate at steady-state power levels up to 100 kW. The UFTR is now tentatively scheduled to have its high-enriched (93%) uranium (HEU) materials test reactor plate-type fuel exchanged for low-enriched uranium (LEU) fuel in late 2001 based upon commitments from the United States Department of Energy (DOE) (Lockheed Martin Idaho) University Reactor Fuel Assistance Program. This HEU to LEU conversion effort is intended to meet the regulatory requirements of the Nuclear Regulatory Commission (NRC) per Title 10, Part 50.64 of the Code of Federal Regulations (10 CFR 50.64).

The present report addresses the analyses and technical work undertaken with DOE support under Grant Number DE-FG07-98ID13697, UFTR University Reactor Program. It should be noted that specific activities were to differ depending on the fuel option selected—SPERT pin or silicide plate. Neutronics and thermal hydraulics analyses for the SPERT existing fuel option had been performed previously to the point of demonstrating feasibility^{[1] [2]} but license considerations would still require documentation of this work to be performed and a full set of supporting calculations to be produced for submittal to NRC. In contrast to the silicide plate fuel conversion option, the physical facility changes required for this option could require significant changes due to fuel pin length (nearly a foot longer than HEU plate fuel) and additional mass to be supported (nearly a metric ton of fuel pins versus only about 20 kilograms of HEU fuel plates currently in place). The key would be to determine the cheapest option while optimizing UFTR performance capabilities. With silicide plate fuel, no neutronics or thermal hydraulics analyses had been performed for the UFTR; therefore, the safety analysis for the

silicide fuel option would start from zero bases. However, it was expected that essentially no physical facility changes would be required for the silicide LEU fuel option; hence this option promised the simplest post-analysis effort to implement conversion from HEU to LEU fuel.

This report addresses R-56 licensee fuel conversion activities associated with the following areas:

1. Selection of LEU fuel option—pin versus plate—to include input into qualification of the PERT fuel if selected.
2. Performance of conversion analysis (reactor physics, thermal hydraulics, shielding, etc.), preparation of analyses units and presentation of documentation required by the final NRC conversion rule as necessary to obtain NRC approval for UFTR conversion process.

Initial efforts in the process to convert the UFTR from use of HEU fuel to LEU fuel were to consist of preliminary tests and an evaluation to determine whether the SPERT-type fuel available to the R-56 licensee but stored under License SNM-1050 could likely be qualified for use in the UFTR and whether that qualification would be warranted on a cost-benefit basis. Efforts were then to continue with activities to select a fuel, either SPERT or silicide, based upon prequalification testing of existing SPERT fuel and identification of modifications in existing reactor systems necessitated by use of the new fuel. This effort was to address maintaining and/or improving the UFTR neutronics characteristics while minimizing the overall cost of UFTR conversion. The only two fuels planned to be considered were the existing SPERT UO_2 , stainless steel clad pin-type fuel held under the SNM-1050 license and the silicide fuel to be available through the Reduced Enrichment for Research and Test Reactors (RERTR) program at Argonne National Laboratory. The first choice would be to use the already existing

SPERT fuel for which a number of neutronics and thermal hydraulics analyses are in existence.^{[1][2]} Clearly, this would be the cheaper fuel if acceptable since it is already manufactured. However, a prequalification program would have to be developed and, after approval, the qualification tests used to assure the SPERT fuel could meet UFTR requirements without compromising safety and without requiring costly modifications which could outweigh the low SPERT fuel cost (no manufacturing costs) and positive impact on core neutronics per earlier analyses.^[2] Because of the lead times involved in performing the analyses and then preparing a full submittal for NRC review, the decision on which fuel to use was to be made within the first six (6) months of receiving DOE support for conversion. Key points to be addressed in the work to select the better of the two fuels are listed as follows:

1. Does the mass of SPERT fuel at 2,000 pounds impact the support structure of the UFTR considering the existing fuel weighs only about 50 pounds while the silicide LEU option is about the same?
2. Is the SPERT fuel likely to be able to be qualified based on Rensselaer Polytechnic Institute's experience with its zero power facility and the differing usage to which the University of Florida SPERT fuel has been subjected?
3. Is there likely to be a significant advantage neutronically (based primarily on maximum thermal flux for irradiations) to one option or the other?
4. Based on the previous analyses in items 1, 2 and 3, is one LEU option expected to be significantly cheaper?

The bottom line in the evaluation of the SPERT versus silicide LEU options is maintenance of facility capabilities (benefit) relative to cost for the conversion. Here the ability to project future requirements would be a key selection factor.

After selection of an LEU fuel and identification of key reactor systems changes, a safety analysis was to be performed (beginning toward the middle of the first year of the proposed work) for the selected LEU fuel conversion and associated system changes. For the SPERT fuel option, all pins would likely be required to be radiographed for acceptance prior to assembly into bundles for use in the core. In addition to neutronic and thermal hydraulic analyses (reduced for SPERT fuel option), shielding and effluent analyses would require documentation to identify any procedural, security plan, technical specification and other license changes that must be considered as part of conversion. The NRC submittal would also contain documents detailing the various tests and surveillances planned as part of the conversion. At this point, a complete set of licensing documents for the conversion would be submitted to NRC along with a conversion application for review and approval. Assuming eventual resolution of all questions, this phase of HEU to LEU conversion work would conclude with the issuance by the NRC of the specific Order to Convert the UFTR.

2.0 Low Enriched Fuel Option Selection

With receipt of funding, the first work undertaken was to perform analyses directed to selecting the low-enriched fuel option for conversion of the UFTR. The first option was the SPERT F-1 stainless steel clad pin-type fuel stored at the University of Florida under License SNM-1050; the second option was the DOE-developed aluminum clad uranium silicide plate-type fuel.

Initially, extensive efforts were applied to decontaminate and remodel a room in which to store the SPERT LEU fuel, to change the license description of the SPERT storage facility, to move the fuel to the new facility, to release the previous storage room to unrestricted usage, to revise the facility security plan (SNM-1050) and then to perform a detailed pin-by-pin visual inspection and verification of serial numbers. The required visual inspection and identification of SPERT fuel pins was completed on September 19, 1988. As committed, a sufficient number of SPERT fuel pins was then radiographed utilizing an industrial X-ray machine to provide an LEU core and replacement pins for the UFTR by March 31, 1989, when the SPERT usage license was to expire.

As for the SNM-1050 license, a significant effort was involved as the renewal license application for renewal under "storage only" conditions was submitted and approved by the NRC. In June 1989, an engineering-based decision was finally made not to select the SPERT LEU fuel option, but to use the alternate low-enriched silicide plate-type fuel. The basis for the decision was two-fold involving the need for core and structural modifications as well as changes in auxiliary equipment and facilities if the SPERT fuel were to be used. First, considerable core and shield structural modifications would be required for SPERT fuel. The mass of the silicide plate-type fuel is nearly the same as the existing fuel loading versus the

alternative SPERT pin-type fuel which would necessitate loading nearly a ton of fuel into the UFTR fuel boxes. In addition, the SPERT fuel shown in Figure 2.1 is nearly fifty percent longer than the silicide fuel plates necessitating consideration of changing out the UFTR fuel boxes, as shown in the isometric in Figure 2.2, for longer ones. These SPERT fuel induced changes would also require extensive shielding changes above the core to maintain radiation levels at current levels during normal operations. The overall implications with the fuel shape, length and mass changes were that considerable core changes would be required necessitating extensive analyses. Of course, the subsequent redesign and following reactor modifications to accommodate using the SPERT fuel would have been considerable, to say nothing of SPERT fuel qualification considerations that were raised and addressed within the RERTR program at Argonne National Laboratory.

Second, auxiliary equipment and facilities would be severely affected. The existing lead shield fuel transfer cask and fuel handling equipment would have had to be replaced since they would not accommodate the increased fuel length. The fuel assembly handling tool would certainly have been pushing design limits in handling a SPERT fuel assembly weighing as much as 100 kilograms or more. In addition, the irradiated fuel storage pits used for off-loading fuel during core maintenance and for storing irradiated fuel when no longer of use was designed for the existing 25-inch long fuel bundles and would not accommodate the meter long SPERT fuel, a length of the fuel pin meaning a bundle would have to be somewhat longer. Because of limited restricted area space in the storage pit area, changes in the fuel storage pit height would have severely restricted normal operations in accessing the equipment pit. Again, the conclusion was that a major design change with economic penalties would be required. Depending on the new cask design, even the overhead crane, rated at three (3) tons, might have had to be replaced

to accommodate the larger cask needed to move SPERT fuel bundles. Again, the commitment of resources was considerable.

The only significant reason for using the pin-type SPERT fuel versus the plate-type silicide fuel had been the financial savings to be realized by not needing to manufacture fuel since the SPERT fuel already existed. The DOE Fuel Assistance Program manager considered this savings important enough that it was required to be investigated. However, the economic disadvantages resulting from reactor redesign due to significant changes in fuel mass, shape and length as well as the replacement of auxiliary equipment and facilities made the decision clear. The uranium silicide plate-type low-enriched fuel option was selected for the UFTR HEU to LEU conversion as the clear choice both for overall economics as well as for general availability of the UFTR facility based upon no need to make structural or other design changes in the UFTR itself. This decision is particularly appropriate since the UFTR is also one of the more heavily used mid-sized university reactors, accommodating a wide variety of educational, training, research, and service projects.

SPERT FUEL ROD CHARACTERISTICS

Length of F-1 Fuel rod	106.05	cm
Cladding Outside Diameter	1.184	cm
Cladding Thickness	0.051	cm
Active Fuel Length	91.44	cm
Fuel Pellet Outside Diameter	1.067	cm
Weight of U-235 Per Rod	35.2	gm
Uranium-235 Enrichment	4.81	wt%
UO ₂ Fuel Pellet Density	10.08	g/cc

Figure 2.1 Isometric of SPERT Fuel Rod

Figure 2.2 Isometric of UFTR Fuel Boxes

3.0 Neutronics Analysis

3.1 Overview and Geometry Considerations

The University of Florida Training Reactor (UFTR) is a 100 kW light-water-cooled, graphite-and-light-water-moderated, Argonaut-type reactor which utilizes high-enriched (93%) materials test reactor-type (uranium-aluminum alloy) fuel plates in a two-slab core configuration consisting of 11-plate fuel bundles loaded into six (6) fuel boxes. The purpose of this work was to perform the neutronics analysis necessary to support a licensing submittal to the NRC for conversion of the UFTR from high-enriched uranium (HEU) to low enriched uranium (LEU) fuel as required by 1986 revisions to the Code of Federal Regulations (10 CFR 50.64). After elimination of pin-type SPERT fuel as an option, the fuel selected for the conversion analysis was the uranium silicide-aluminum dispersion standard plate-type fuel which has been well evaluated for use in non-power reactor fuel conversions.^[3]

As shown in the isometric cutaway sketch of Figure 3.1 as well as the horizontal core section diagram in Figure 3.2 and the isometric component diagram of Figure 3.3, the unique, extremely heterogeneous arrangement of this small compact reactor core system represents a challenge for neutronics analysis. The longitudinal and transverse sections through the UFTR core center illustrating fuel and fuel box arrangement in Figure 3.4 and Figure 3.5 further emphasize heterogeneities involved. On this basis and to facilitate student ability to use the necessary computer code schemes, the graduate student performing these calculations received preliminary training with the RERTR personnel at Argonne National Laboratory.

The analysis in this work was intended to select the number of standard fuel plates per bundle for the new core loading and to assure such bundles can be reliably loaded into the same or equivalent fuel boxes with sufficient criticality margin to allow for uncertainties in the

calculations as well as for sufficient excess reactivity to meet operational expectations for maneuverability and some meaningful burnup over the 20 or more years the core could be expected to be licensed. When tolerances and variations in core and reflector/support geometry are incorporated, this task is non-trivial. Since the current HEU core has 2.5 dummy fuel bundles, and is nearing the need for an additional 5-plate half bundle, it was decided that the LEU core should have a similar number of dummy bundle locations initially to allow adding fuel bundles every 5-10 years to maintain excess reactivity without the need for full core replacement. The existence of only six fuel box locations represents a unique limitation to maintaining excess reactivity in this core design since, unlike a pool reactor, it does not allow adding fuel indefinitely to make up for burnup, though relatively low power operation does mitigate this consideration.

The first effort was to establish geometry and core constituents. The LEU plate-type fuel is of standardized design^[4] as a government cost savings measure in refueling university reactors with LEU fuel. The fuel meat is composed of U_3Si_2 dispersed in an aluminum matrix. The fuel material is rolled, and the plates custom cut to specific dimensions. The LEU fuel plate will be the same size as the HEU plates except thinner. Composition and dimensions of the LEU fuel in comparison to the HEU fuel are given in Table 3.1. The existing HEU fuel plate and fuel bundle geometric arrangement is shown in Figure 3.6; similarly, the drawing for the low-enriched uranium fuel plates assembly and detail is shown in Figure 3.7.

Table 3.1 HEU and LEU Fuel Plate Characteristics

	HEU Fuel	LEU Fuel*
Fuel Composition	93% Enriched, U-Al	19.75% enriched, U ₃ Si ₂ -Al
U Density	0.43 g/cc	3.47 g/cc
Plate Thickness	1.78 mm	1.27 mm
Width	72.1 mm	72.1 mm
Length	25.625 inches	25.625 inches
Al Clad Thickness	0.38 mm	0.38 mm
Meat Thickness	1.016 mm	0.51 mm
g U ²³⁵ per Plate	14.5 g	12.5 g
Meat Length	60.0075 cm	59.1 cm
Meat Width	58.4 mm	60.85 mm
*Nominal dimensions ²		

The LEU fuel bundle will be slightly larger in the north-south plane due to smaller coolant channels. The physical increase in the size of the fuel bundle is the difference between the coolant channel widths. The bundles including the end coolant channels are the same size. It is assumed that spacers like the current ones in the HEU fuel will be used with the LEU fuel. The LEU spacers will have dimensions of 2.54 cm x 0.48 cm x coolant channel thickness.

The coolant channel thicknesses are given in Table 3.2.

Table 3.2 Coolant Channel Dimensions

Core	Plates per Bundle	Coolant Channel Width	Cell Metal*/H ₂ O	Cell H/U ²³⁵
LEU	14	2.860 mm	0.44	211
LEU	15	2.586 mm	0.49	191
HEU	11	3.480 mm	0.51	222**
*Metal includes fuel and aluminum. **NE fuel box, other boxes varied from 220 to 223.				

Additional effort was expended to determine core geometry including the measurements and loading for the semaphore-type aluminum with cadmium-loaded absorber control blades. This effort was extensive with a considerable effort devoted to searching the existing design diagram and drawings with relatively large uncertainties on some dimensions such as fuel box width. The uncertainties in dimensions are attributed to the design drawing requirements which are very liberal apparently due to the age of construction. The reactor was first licensed in 1959 with facility construction occurring primarily in 1958. The basic UFTR geometry utilized for the criticality and flux calculations is illustrated in Figure 3.8. The details of the homogenization are available at the facility for review as desired.^[5]

3.2 Computational Methodology

The next effort in this work was to identify and implement the necessary computer codes and to benchmark the analysis methodology for the existing core and documented critical position. This benchmark work was performed for the existing core consisting of 21 full fuel bundles, a 5-plate half bundle, and 2.5 dummy bundles. The preliminary benchmark analysis was devoted to obtaining an accurate representation of UFTR core geometry including heterogeneities for the existing HEU fuel loading. The LEOPARD^[6] code was used for multi-group constants with the Battelle-Revised-THERMOS(BRT)^[7] code used for void correction factors to account for leakage. Subsequently, the UM3DB^[8] and then UM2DB^[9] codes were used to perform flux and criticality calculations. The calculational methodology is illustrated in the flow chart in Figure 3.9. This scheme resulted in a 1.6% $\Delta k/k$ bias in the calculated versus measured multiplication factor for the HEU core with several variations in the schematic methodology in Figure 3.9 serving to confirm the results. Subsequently, the same calculational

methodology was utilized to perform a parametric criticality analysis on the proposed LEU core to select a bundle design and proposed core loading.

The methodology used involved obtaining the most accurate description of the UFTR HEU reactor possible. From this point, atom densities were calculated for all reactor regions. Then, multi-group constants were generated using LEOPARD, and void correction factors were calculated with BRT. Calculations using the codes UM3DB and UM2DB built confidence in the reactor model. Then, calculations for the LEU core could proceed. Having the reactor geometry already defined, a simple fuel replacement as a scoping calculation was completed for fuel bundles containing from 14 to 16 plates per bundle. The 16 plates per bundle option was immediately eliminated due to excess reactivity beyond the licensed limit for the options of zero to 2.5 dummy bundles. This general methodology was developed as part of the RERTR program and has been used to license other facilities (Iowa State and University of Michigan) which is why the graduate student involved used training at Argonne National Laboratory to facilitate this work.

3.3 Neutronics Results

Key results of the LEU parametric criticality analyses include the calculated multiplication factors for 14-, 15- and 16-plate fuel bundles and for various numbers of dummy bundles.

Three fuel bundle options were given serious consideration as follows:

- (1) 15 plates with 2.5 dummy bundles
- (2) 14 plates with 2.5 dummy bundles
- (3) 14 plates with 2 dummy bundles

All were found to be viable from a neutronics perspective.

The reactivity coefficients for the LEU cores are essentially the same for the three configurations studied as shown in Table 3.4. As expected, the LEU values of the temperature coefficients are more negative than the HEU values. This characteristic is primarily due to the increased parasitic absorption in U-238 and to a lesser extent due to the thinner fuel meat (see Table 3.1) for the LEU fuel. The constant values for the reactivity coefficients are easily explained by the fact that there is actually little difference in the geometry for the three cores. The difference in coolant flow channel size (see Table 3.2) for 15 versus 14 plates per bundle is small. Also, going from 2.5 to 2 dummy bundles does not perturb the flux greatly.

Table 3.4 Reactivity Coefficients

	14/2	14/2.5	15/2.5	HEU Calculated	HEU Experimental
Uniform Temperature % $\Delta k/k$ per °F	-0.52×10^{-4}	-0.53	-0.52	-0.35	-0.3×10^{-4}
Uniform Water Void % $\Delta k/k$ /%Voids	-0.28	-0.28	-0.28	-0.19	-0.2

Table 3.5 gives the peak-to-average power densities by fuel box (fueled regions only) and averaged over all fuel bundles. The SW box contains the half dummy bundle where the highest fuel box peak-to-average ratio occurs for the LEU fuel.

Table 3.5 Peak-to-Average Power Densities

Region	HEU	14/2	14/2.5	15/2.5
NE	1.51	1.45	1.43	1.49
NC	1.44	1.43	1.42	1.50
NW	1.47	1.44	1.42	1.50
SE	1.53	1.45	1.45	1.51
SC	1.48	1.44	1.45	1.53
SW	1.50	1.45	1.43	1.51
All Boxes	1.39	1.35	1.38	1.43

Various flux profiles were calculated and compared to the calculated and experimental flux profiles for the existing HEU core. The peak thermal flux in the north-south direction is marginally greater for the 15/2.5 case compared to the other LEU configurations. In the fuel region, the LEU peak thermal flux is approximately six percent lower than the HEU reactor which is acceptable for all current UFTR applications. The 14/2 and 14/2.5 configurations show significant thermal flux peaking in the fuel region. This peaking is due to the lower metal-to-water ratio as compared to the 15/2.5 core. The same is true along the east-west direction where the throughport (or the rabbit system when inserted) is located. Three typical calculated flux profiles in the north-south direction are shown in Figure 3.11 for the existing HEU core, in Figure 3.12 for the 14/2 design and in Figure 3.13 for the 14/2.5 design, all at full 100 kW power level. Though not shown here, dozens of additional flux profiles are available for reference purposes at the UFTR facility.^[5] All are generally consistent with relatively small differences between HEU and LEU cases; generally the HEU cases agree reasonably well with experimental data providing additional confidence in the calculations.

The large peaks in the east-west flux profiles occur in the center of each of the fuel boxes. This is the region of greatest moderation. (Actually, the peak thermal neutron fluxes, Table 3.6, in the fuel boxes occur near the graphite due to thermal leakage into the fuel regions.) The two smaller peaks between the larger peaks (easily noted in the 14/2 core case) are caused by the non-lattice region between the fuel and blade shrouds. In the 15/2.5 case, these two peaks have merged.

The 14 plates per bundle cases have a higher peak thermal flux in the fuel boxes than the 15/2.5 configuration. This is due to its lower metal-to-water ratio. In the north slab, the 14/2 flux is close to that of the 14/2.5 configuration. In the south slab, the 14/2.5 flux profile deviates

significantly from the 14/2 case only in the dummy bundle region. The absolute peak thermal flux occurs in the graphite region between the two fuel slabs as listed in Table 3.6.

Table 3.6 Peak Fluxes

Core	Peak Total Flux	Peak Thermal Flux
15/2.5	3.75E+12 n/cm ² -s	1.67E+12 n/cm ² -s
14/2.5	3.79E+12 n/cm ² -s	1.65E+12 n/cm ² -s
14/2	3.57E+12 n/cm ² -s	1.62E+12 n/cm ² -s
HEU	3.98E+12 n/cm ² -s	1.86E+12 n/cm ² -s

The calculated values for the delayed neutron fraction and prompt neutron lifetime also agreed with the experimental values. All the LEU cores have the same values for the delayed neutron fraction while the prompt neutron lifetime varies by an inconsequential microsecond as summarized in Table 3.7.

Table 3.7 Reactor Kinetic Constants

Core	Delayed Neutron Fraction	Prompt Neutron Lifetime (10 ⁻⁴ sec)
15/2.5	7.3E-03	2.69
14/2.5	7.3E-03	2.68
14/2	7.3E-03	2.67
HEU Calculated	7.3E-03	2.76
HEU Experimental	7.3E-03	2.8

In summary, all three LEU options have acceptable flux profiles and similar, though somewhat reduced, peak thermal neutron flux values and relatively unchanged kinetics constants as compared to the current HEU system. The reactor kinetics constants, delayed neutron fraction and prompt neutron lifetime (0.0073 and 2.68 x 10⁻⁴ seconds, respectively) for the 14/2 LEU option are similar to those for the existing HEU core and are acceptable. The peak-to-average

power densities varied from 1.35 for the 14/2 option to 1.43 for the 15/2.5 case. On the basis of excess reactivity ($1.8\% \Delta k/k$ %), acceptable negative temperature reactivity coefficient ($-0.52 \times 10^{-4} \% \Delta k/k/^{\circ}F$) and void coefficient ($-0.28\% \Delta k/k$ % voids), the flattest flux profile and relative ease of fuel handling and loading, the 14-plate fuel bundle was selected with two dummy bundles as the most likely loading, though this will be verified during the actual fuel change out since the conversion itself is essentially an experimental procedure.

The peak thermal flux at full power is within two percent of 1.65×10^{12} n/cm²-s for all three LEU options. This is 11% less than the corresponding value for the HEU core (see Table 3.6). This peak occurs in the graphite between the two slabs. The peak total flux occurs within the fuel box. The 14/2 option has the lowest peak at 3.57×10^{12} n/cm²-s, while 14/2.5 has the highest at 3.79×10^{12} n/cm²-s. This variation is only about five percent, again relatively inconsequential for most UFTR applications and usages. The peak-to-average power ratios (see Table 3.5) for the LEU fuel are essentially the same for each fuel box except for the SW box where the 14/2 option has full fuel bundle instead of a half dummy bundle. Overall, the 14/2 option has the lowest peak-to-average flux ratio.

The 14/2 option has the most excess reactivity at 1.8%, while 14/2.5 has the least at 1%. The 15/2.5 option falls between at 1.4%. The temperature coefficients of reactivity remain essentially unchanged for the three cases (see Table 3.4) at values of $-0.28\% \Delta k/k$ % voids for the water coefficient and $-0.52 \times 10^{-4} \% \Delta k/k$ for the uniform temperature coefficient. Also, the reactor kinetics constants also remained unchanged (see Table 3.7).

The 14/2 option is marginally better than the other two. The 14/2 option has the flattest flux profile as inferred from the peak-to-average flux ratios. Its excess reactivity is high enough that burn up of the fuel over a long period of time will not be a problem which is important since

the UFTR expects to renew its license for 20 years in 2002. Also, the ease of fuel handling must be considered. Fuel loading for maintenance and inspection can be facilitated by not having half dummy bundles. These have been known to be difficult to load, and may be harder to load for the larger LEU fuel bundles. Therefore, the 14 plates per bundle, probably with two dummy bundles in the initial loading, is the selected option if allowed experimentally during the actual loading.

3.4 Supporting Analysis

The neutronics results reported in Section 3.3 are further supported by analysis using a three dimensional neutronics calculation.^[10] The cell calculation code used was the one dimensional transport code: WIMS-D4M.^[11] This code is the ANL version of the WIMS-D4 code developed at Winfrith, England.^{[12] [13]} The core calculation code used was DIF3D 7.0, a code system for solving finite difference diffusion theory problems.^[14] In this analysis, the HEU core criticality benchmarking results show a 1 mk difference between the k_{eff} value calculated using the fuel number densities from the Section 3.3 work and the value calculated using the total U-235 loading for the DIF3D 7.0 work. Both values agree well with the value based on a rod worth experiment. The U-235 loading adjustment was done in DIF3D by assigning to all fuel zones a fictitious volume fraction slightly greater than 1.^[10]

When going from HEU to LEU fuel, the peak thermal flux in the center vertical port (CVP) decreases by 7% while the peak fast flux decreases by 2%. The result is a small hardening of the energy spectrum: the total-to-fast flux ratio (essentially the cadmium ratio) decreases from 2.15 to 2.10. This effect is due to the increased U-238 resonance absorption but is essentially negligible and agrees with the earlier work.^[5]

The calculated peak-to-average power densities are slightly higher in the LEU case. The three dimensional results are consistently higher than the two dimensional values of DeMartino.^[5]

The HEU U-235 mass coefficient was calculated using DIF3D values which represent a uniform mass increase of 12 g. This value agrees with the existing value in the UFTR Safety Analysis Report.^[15] A similar calculation was done for LEU. The fuel temperature coefficient is also calculated since it is the most significant one—for being prompt. In going from HEU to LEU, changes are caused by the large increase in U-238. There is a decrease in the U-235 mass coefficient; the other reactivity coefficients become more negative.

In general, there are discrepancies between the experimental and the calculated values. In part, this may be due to the fact that the experiments are not physically “clean,” mainly due to safety limitations. The isothermal temperature coefficient measurement is a time consuming procedure, which is affected by heat loss to the graphite reflector. The uniform water void coefficient is estimated by bubbling nitrogen into the reactor. Fortunately, the UFTR Technical Specifications only require that the reactivity coefficients be negative, not that the measurement be quantified.

These additional analyses yield results that are in essential agreement with those presented previously with some new information on mass and temperature coefficients of reactivity. This work provides independent supporting documentation for the basic choice of a 14-plate fuel bundle for the UFTR HEU to LEU fuel conversion to assure that the LEU core can be licensed by the NRC and operated by the licensee.

The basic static neutronics analysis for fuel conversion has been successfully completed. The critical condition, required excess reactivity, kinetics constants, reactivity coefficients and

flux profiles have been determined to be adequate for the LEU conversion core with important experimental facilities relatively unaffected. The stated condition that the outage for the conversion be minimized has also been met with the 14-plate bundles and no half bundles. In addition, the use of the silicide fuel avoids core and structural modifications as well as changes in auxiliary equipment and is probably the better economical choice to say nothing of being in agreement with minimizing personnel radiation doses per the as-low-as-reasonably-achievable (ALARA) principle. When coupled with thermal hydraulics analysis, the results of this neutronics work will be summarized to support the required NRC licensing submittal for the mandated conversion.

Figure 3.1 Isometric of UFTR Fuel Boxes

Figure 3.2 Horizontal Section at Beam Tube Level

Figure 3.3 Isometric Diagram of UFTR Components

Figure 3.6 UFTR Plate and Fuel Bundle Geometric Arrangement

Figure 3.7 UFTTR LEU Fuel Plates Assembly and Detail

Figure 3.8 Geometry of the UFTR as Used in UM2DB

Figure 3.9 Neutronics Calculation Methodology for Fuel Conversion Analyses

Multiplication Factor vs Dummy Bundles
calculated values w/o bias factor

+ 14 plates/bundle * 15 plates/bundle x 16 plates/bundle

Figure 3.10 Multiplication Factor vs. Number of Dummy Bundles

Figure 3.11 HEU Core Calculated Flux in North-South Direction at Full Power

Flux in North-South Direction LEU Core - 14/2

Figure 3.12 LEU 14/2 Flux in North-South Direction at Full Power

Flux in North-South Direction LEU Core - 14/2.5

Figure 3.13 LEU 14/2.5 Flux in North-South Direction at Full Power

4.0 Thermal Hydraulics Analysis

4.1 Introduction

The results of steady-state thermal hydraulics characteristics of the University of Florida Training Reactor (UFTR), fueled by the proposed low (19.75%) enriched uranium (LEU) plate fuel, are analyzed in this section.^[15] The suitability of the LEU (U_3Si_2 -Al) plate fuel is demonstrated (a) by comparing steady-state calculation results for the UFTR using proposed LEU fuel plate core configurations with calculation results for the UFTR using the existing high (93%) enriched uranium (HEU) plate fueled core configuration, and (b) through conservative estimates of the maximum fuel and clad temperatures encountered in an overpower transient.

As noted earlier, the HEU fuel bundles of the current UFTR configuration consist of 11 fuel plates; each fuel plate is composed of a U-Al fuel meat enclosed within aluminum clad. One half bundle contains five (5) active fuel plates. The HEU fuel characteristics and dimensions are provided in Table 3.1, repeated here as Table 4.1.

Table 4.1 HEU and LEU Fuel Plate Characteristics

	HEU Fuel	LEU Fuel*
Fuel Composition	93% Enriched, U-Al	19.75% enriched, U_3Si_2 -Al
U Density	0.43 g/cc	3.47 g/cc
Plate Thickness	1.78 mm	1.27 mm
Width	72.1 mm	72.1 mm
Length	25.625 inches	25.625 inches
Al Clad Thickness	0.38 mm	0.38 mm
Meat Thickness	1.016 mm	0.51 mm
g U^{235} per Plate	14.5 g	12.5 g
Meat Length	60.0075 cm	59.1 cm
Meat Width	58.4 mm	60.85 mm
*Nominal dimensions ²		

As shown in Table 4.1, the LEU plate has the same dimensions as the HEU fuel plate with the exception that LEU plates are thinner to accommodate more plates per bundle and a higher total required fuel loading due to lower U-235 content per plate.

The steady-state analysis of the LEU fuel plate reactor configuration is performed in this section. The steady-state theory is first provided in Section 4.1. The calculational results from the developed thermal hydraulics solver are then used to demonstrate the suitability of the LEU plate fueled reactor configurations.

4.1 Thermal-Hydraulics Theory

This section provides an overview of the theoretical modeling of the “hot channel” thermal hydraulics. A steady-state two-dimensional heat conduction solver is developed following the same assumptions used in the steady-state thermal analysis of the current UFTR Final Safety Analysis Report (FSAR).^{[16] [17] [18]} This methodology was used to assure in-house capabilities for relicensing.

The purpose of the developed thermal hydraulics solver is to obtain the two-dimensional (x,z) temperature distribution in the “hot plate” and the axial (z-direction) coolant temperature distribution during normal (e.g., 100 kW_{th}) operation. The hot plate or hot channel conditions are set using conservative hot channel factors.

The hot plate geometry of interest in this problem is provided in Figure 4.1. As indicated in the lower view, the heat conduction problem is restricted to two dimensions, x and z; the temperatures are considered uniform in the y-directions, as if the plate were infinite in the y-direction.

The principal assumptions under which the modeling is performed are listed below.

- Material properties are uniform and constant.
- The fuel volumetric heat generation rate is independent of y-direction position (see Figure 4.1 for geometry) and is symmetric about the mid-plane of the fuel plate (i.e., in the x-direction); though not a necessary assumption, in this work the volumetric heat generation rate is uniform in the x-direction as well.
- The volumetric heat generation rate is sinusoidal function of axial position.
- The hot fuel plate average volumetric heat generation rate is related to that average fuel volumetric heat generation rate of the reactor through a hot channel factor.
- The upper and lower ends of the fuel plates are fully insulated (adiabatic) walls.
- The fluid flow in the hot channel is fully developed laminar flow—this assumption is justified later in this text—and the fuel plate surface is (conservatively) taken to be uniform for purposes of obtaining a Nusselt number for the clad surface-to-coolant heat transfer coefficient.

With the above listed assumptions, the fuel volumetric heat generation rate in the hot plate may be expressed as a sinusoidal function with f_z designated as the peaking factor due to the axial sinusoidal shape of the fuel power density; note that the average power density in the hot plate is related to the average power density in the reactor (fuel) through the hot channel factor f_H .

The conduction heat transfer in the fuel plate is governed by the general heat conduction equation which is utilized in two dimensions. The appropriate coordinate transformation and boundary conditions for the heat transfer problem as under the assumptions listed earlier are

applied along with interface conditions to account for the surface between the fuel and the clad where the temperature distribution is assumed continuous between the two regions.

The "hot channel" or fluid passages corresponding to the hot plate is modeled as a passage containing a single phase water coolant flowing axially upward between the fuel plates. For the LEU fuel plates, the Reynolds number, Re (based on the hydraulic diameter, D_H) is estimated to be ≈ 240 (at 15 plates per bundle) to ≈ 350 (at 14 plates per bundle); at such low Reynolds numbers the flow may be assumed to be laminar. It may be estimated that the laminar flow at the conditions in UFTR fuel plates becomes fully developed between 4 to 6 mm into the plate; i.e., for the sake of the thermal hydraulics analysis in this work, the low Reynolds number flow is laminar and if fully developed.

In order to determine a heat transfer coefficient for the fully developed laminar flow between two flat plates, a convection correlation for internal flow between noncircular tubes is used.^[19] The heat transfer coefficient, h_∞ , is determined using the formulation

$$h_\infty = \frac{Nu D_H}{k_1}$$

where Nu is the local Nusselt number, D_H is the hydraulic diameter, and k_1 is the thermal conductivity of the water coolant. The Nusselt number is obtained depending on the width of the fuel, b , and the channel thickness (distance from plate to plate), a , per Figure 4.1. At a ratio of $b/a = \infty$, the Nusselt number is 7.54. For the dimensions of the UFTR (LEU and HEU) fuel plates, the Nusselt number is $7.35 \ll Nu \ll 7.5$. A conservative assumption is used that the clad surface temperature is uniform rather than assuming that the clad surface heat flux is uniform. It is also conservative, in terms of Nusselt number, to include the end effects of finite width (b)

plates. Typical heat transfer coefficients in this analysis are $\approx 870 \text{ W/m}^2\text{-K}$ for the 14 plate fuel bundle and $\approx 960 \text{ W/m}^2\text{-K}$ for the 15 plate fuel bundles.

In order to compute the axial coolant temperature distribution, a macroscopic energy balance was used. The mass flow rate corresponds to one-half the channel mass flow rate (as half plate space is considered in the solution of the conduction problem); the fluid is assumed to be evenly distributed among all the fuel boxes, and among all the fuel plates (active or dummy).

The heat conduction equation is then solved by integrating finite-difference approximations to the spatial derivatives. The equation is solved in the computational grid space using the Alternating-Direction-Implicit (ADI) method^[20] with appropriate boundary condition and interface condition representations and assuring convergence in the numerical integration.

After the fuel and clad temperature field is converged upon in the inner ADI solution, a new coolant temperature axial distribution is determined using an Euler integration of the energy balance equation. The overall problem is considered solved—the solution converged upon—when the norm of the axial coolant temperature distributions between two successive outer iterations is less than some specified value (for this work, 10^{-6} was used), given that the fuel and clad temperatures have already been converged upon during the inner iterations. The developed thermal hydraulics solver is available at the UFTR facility as is the free format input data description.^[15]

4.2 Steady-State Results and Analysis

Steady-state calculations were performed for the 14/2 (14 plates per bundle/2 dummy bundles per reactor core), 14/2.5, and 15/2.5 LEU fuel assembly configurations. A “base case” is treated in this analysis as that case in which the UFTR is operating at full power, $100 \text{ kW}_{\text{th}}$, the reactor volumetric flow rate is 40 gallons per minute (gpm), the hot channel factor (f_H) is 1.5,

and the reactor coolant core inlet temperature is 319.11 K. A parametric analysis is performed for each of the LEU core configurations by setting the volumetric flow rate at either 30 or 40 gpm and the hot channel factor at 1.5 or 3.0. In addition, for each LEU core configuration, the steady-state analysis is performed using what are the best estimate of current UFTR operating parameters—100 kW_{th} full power, 313.56 K coolant inlet temperature, 43.5 gpm volumetric flow rate, and a hot channel factor of 1.5. At these best estimate values, comparisons are made between the LEU configurations and the current HEU 11/2.5 configured core. Table 4.2 summarizes the “base case” and best estimate operating conditions. The detailed results of these steady-state calculations are available to support relicensing.^[15]

Table 4.2 Parametric Study “Base Case” and Current Best Estimate UFTR Steady-State Operating Conditions at 100 kW_{th}

UFTR Base Case Conditions	
Reactor Power Level (kW _{th})	100
Reactor Coolant Flow Rate (gpm)	40.0
Reactor Coolant Inlet Temperature (°K, °F)	319.11, 115
Reactor Hot Channel Factor	1.5
UFTR Current Best Estimate Conditions	
Reactor Power Level (kW _{th})	100
Reactor Coolant Flow Rate (gpm)	43.5
Reactor Coolant Inlet Temperature (°K, °F)	313.56, 105
Reactor Hot Channel Factor	1.5

Table 4.3 provides a summary of the UFTR hot plate fuel temperatures for the different LEU core configurations for the case in which the hot channel factor is varied between 1.5 (base case) and 3.0 (highly skewed) and in which the reactor coolant flow rate is 40 gpm. The average fuel power density is seen to be highest in the 14/2.5 configuration. This is because of

the 2.5 dummy fuel bundles, requiring that fewer active bundles contain the fuel; further, the average power density in the fuel is higher in the 14/2 and 14/2.5 configurations than in the 15/2.5 configuration which also has 2.5 dummy bundles but has 15 plates per bundle to accommodate the active fuel. The maximum (centerline) fuel temperatures in the hot plate of the hot bundle lie in the range of 341 to 362 K. The hot spot (hottest centerline temperature in the hot plate of the hot bundle) is located nearly 41 cm above the bottom of the active core (beginning of the active fuel region); i.e., the hot spot is located at a distance nearly 70% of the way up through the core. The maximum clad temperature locations typically correspond to the position of the maximum fuel temperature. For each configuration, the maximum clad outer-to-coolant ΔT occurs halfway up through the bundle; i.e., at the location of the maximum heat flux (29.55 cm up through the core). It is noted that the temperatures (and the ΔT s) are lower in the 15/2.5 configuration than in the 14/2 or 14/2.5 configurations, again because of the lower average (and therefore peak) power densities in the 15-plate fuel configuration relative to the 14-plate configurations.

It is seen in Table 4.3 that although the hot bundle ΔT —i.e., core inlet to outlet coolant temperature difference—is doubled when the hot channel factor is doubled from 1.5 to 3.0, the fuel centerline temperature and the clad outer surface temperatures do not vary dramatically from those found for the $f_H = 1.5$ case as expected due to large surface-to-volume/thickness ratio and good fuel and clad thermal conductivity.

Table 4.4 provides steady-state calculational results for the same LEU core configurations; however, in Table 4.4 the f_H is fixed at 1.5 and the reactor coolant volumetric flow rate is varied and is either 40 gpm (base center) or 30 gpm (low flow). The maximum fuel and clad temperatures and clad outer-to-coolant ΔT are again lowest in the configuration due to the

**Table 4.3 Steady-State Thermal Hydraulics Analysis Results: Comparison of UFTR
Temperatures at Full Power (100 kW_{th}), 40 gpm Flow Rate, Reactor Inlet
Temperature of 319.11 K, and Variable Hot Channel Factor (f_H)**

Plates/Dummies:	14/2		14/2.5		15/2.5	
Case:	A	E	B	F	K	M
Hot Channel Factor (f_H):	1.5	3.0	1.5	3.0	1.5	3.0
Fuel Average Power Density (MW/m ³):	17.70	17.70	18.11	18.11	16.91	16.91
Maximum Fuel Temperature (K) (at height, cm):	341.41 (40.19)	363.80 (40.19)	341.97 (40.19)	364.83 (40.19)	340.41 (41.37)	361.71 (41.37)
Maximum Clad Temperature (K) (at height, cm):	341.41 (40.19)	363.70 (40.19)	341.93 (40.19)	364.74 (40.19)	340.37 (40.19)	361.62 (41.37)
Maximum Clad-to-Coolant ΔT (K) (at height, cm):	12.16 (29.55)	24.32 (29.55)	12.44 (29.55)	24.89 (29.55)	10.54 (29.55)	21.08 (29.55)
Hot Bundle ΔT (K):	15.69	31.37	16.05	32.1	16.05	32.1
Reactor Average ΔT (K):	9.59	9.59	9.59	9.58	9.59	9.58
Hot Bundle Outlet Temperature (K):	334.80	350.48	335.16	351.21	335.16	351.21
Avg Reactor Outlet Temperature (K):	328.70	328.70	328.70	328.70	328.69	328.70

**Table 4.4 Steady-State Thermal Hydraulics Analysis Results: Comparison of UFTR
Temperatures at Full Power (100 kW_{th}), Hot Channel Factor (f_H) of 1.5,
Reactor Inlet Temperature of 319.11 K, and Variable Volumetric Flow Rate**

Plates/Dummies:	14/2		14/2.5		15/2.5	
Case:	A	C	B	D	K	L
Volumetric Flow Rate (gpm):	40	30	40	30	40	30
Fuel Average Power Density (MW/m ³):	17.70	17.70	18.11	18.11	16.91	16.91
Maximum Fuel Temperature (K) (at height, cm):	341.45 (40.19)	345.62 (42.55)	341.97 (40.19)	346.24 (42.55)	340.41 (41.37)	344.84 (44.92)
Maximum Clad Temperature (K) (at height, cm):	341.41 (40.19)	345.58 (42.55)	341.93 (40.19)	346.20 (42.55)	340.37 (41.37)	344.80 (44.92)
Maximum Clad-to-Coolant ΔT (K) (at height, cm):	12.16 (29.55)	12.16 (29.55)	12.44 (29.55)	12.44 (29.55)	10.54 (29.55)	10.54 (29.55)
Hot Bundle ΔT (K):	15.69	20.91	16.05	21.40	16.05	21.40
Reactor Average ΔT (K):	9.59	12.78	9.59	12.78	9.59	12.78
Hot Bundle Outlet Temperature (K):	334.80	340.02	335.16	340.51	335.16	340.51
Avg Reactor Outlet Temperature (K):	328.70	331.89	328.70	331.89	328.69	331.89

larger number of active plates in which the LEU fuel is distributed. The fuel centerline temperature (and clad outer surface temperature) remain in a tight band between 340 to 346 K as the coolant flow rate is reduced from 40 to 30 gpm for the considered configurations. The hot bundle ΔT (inlet-to-outlet) increases by 33% for the same reduction in coolant flow rate, from 40 to 30 gpm. Even at the low flow (30 gpm) conditions considered, the peak hot bundle (hot plate) outlet temperature—340.5 K in the case of 14/2.5 and 15/2.5 configurations—there is a 32 or 33 K temperature margin between the hot bundle outlet temperature and the 373 K saturation temperature.

Note from both Table 4.3 and Table 4.4, core configurations with the same number of dummy bundles per core (e.g., 14/2.5 and 15/2.5) will have the same steady-state hot bundle coolant inlet-to-outlet ΔT , and therefore the same hot bundle outlet temperature. These ΔT s are only slightly higher than the hot bundle inlet-to-outlet ΔT of the 14/2 configuration. The higher the number of plates per bundle, however, for the same number of dummy bundles per core then the lower the resultant system temperatures. Again, this result is expected and serves again to confirm the methodology.

Table 4.5 provides a comparison of the same steady-state thermal hydraulics characteristics considered in Tables 4.3 and 4.4; however, in Table 4.5 the “best estimate” parameters, or current UFTR operating conditions— $f_H = 1.5$, 43.5 gpm flow rate, $T_{in} = 313.56$ K and 100 kW_{th} full power operation—are considered. The current UFTR 11/2.5 HEU fuel configuration is compared with the 14/2, 14/2.5 and 15/2.5 LEU configurations using the steady-state analysis.

The average (full) power density of the high-enriched (HEU) 11/2.5 configuration is significantly lower than that in the LEU configurations as seen in Table 4.5; however, note that the maximum fuel temperature and maximum clad outer surface temperature (342 K) are higher

than the same corresponding temperatures in the LEU fueled configurations (334 to 335 K). Although the power density is lower in the HEU plates, the power produced in each of the HEU plates are generally higher than the power produced in corresponding plates in the LEU configured cores because of the large active fuel volume per plate in the HEU fuel. For the example operating conditions considered in Table 4.5, the average HEU fuel plate produces 634 $W_{th}/plate$ while, for discussion, the 15/2.5 plate produces 465 $W_{th}/plate$. Note that this means that although the HEU fuel has lower power density, the average (and therefore peak) clad outer surface heat flux is higher in the HEU fuel configured core than in the LEU core configurations. As far as reactor temperatures, clearly the move to LEU fuel results in lower fuel and clad temperatures as well as lower clad outer surface-to-coolant ΔT s for the same reactor operating conditions. The 15/2.5 configuration is again, as expected, the best of the LEU configurations in terms of lower fuel and clad temperatures.

Table 4.5 HEU and LEU Configuration Comparison of UFTR Temperatures at Full Power (100 kW_{th}) with Current Best Estimate of UFTR Volumetric Flow Rate (43.5 gpm), Hot Channel Factor ($f_H = 1.5$), and Coolant Inlet Temperature (313.56 K)

Fuel Type:	HEU	LEU		
Plates/Dummies:	11/2.5	14/2	14/2.5	15/2.5
Case:	HEU-B	I	J	O
Fuel Average Power Density (MW/m ³):	11.88	17.70	18.11	16.91
Maximum Fuel Temperature (K) (at height, cm):	342.36	334.94 (39.01)	335.43 (39.01)	333.83 (41.37)
Maximum Clad Temperature (K) (at height, cm):	342.22	334.89 (39.01)	335.39 (39.01)	333.79 (41.37)
Maximum Clad-to-Coolant ΔT (K) (at height, cm):	19.98	12.16 (29.55)	12.44 (29.55)	10.54 (29.55)
Hot Bundle ΔT (K):	14.76	14.42	14.76	14.76
Reactor Average ΔT (K):	8.81	8.81	8.81	8.81
Hot Bundle Outlet Temperature (K):	328.32	327.98	328.32	328.32
Avg Reactor Outlet Temperature (K):	322.37	322.37	322.37	322.37

4.3 Overpower Transient Considerations

A method that provides a conservative estimate of the maximum centerline fuel temperature that will occur during an idealized prompt overpower transient—e.g., due to a neutronic excursion—was also developed. The overpower transient analysis is consistently applied to the LEU and HEU fuel configurations so that, even though the modeling is based on many simplifying assumptions, the relative merits of the LEU and HEU configurations can be discussed. The simplicity allowed here is one of the reasons for not selecting the SPERT pin-type fuel option. The method is based on simple steady-state relationships. The calculations can accurately be estimated using hand calculations; however, exact calculations are performed in the same thermal hydraulics code used for the steady-state calculations of Section 4.2.^[15] A simple theory is presented first below, followed by example results that demonstrate the viability of the LEU fuel. For example, the fuel is not compromised in terms of centerline fuel melt even during the severe overpower transients considered.

4.3.1 Theory

A simple theory for providing a conservative estimate of the maximum fuel centerline temperature and maximum clad surface temperature during an overpower transient is developed in this section. It is imagined that the reactor is operating at a nominal steady-state power level (e.g., at 100 kW_{th}) when suddenly a neutronic excursion occurs resulting in an overpower transient. It is assumed that a prompt (step-function) rise in neutron flux level occurs, the reactor scrams on high flux level, and the core thermal power—having peaked with the peak flux—quickly diminishes. The overpower transient is considered to be short with respect to the heat transfer time constants (i.e., occurs on the order of milliseconds).

It is assumed that the overpower transient of concern in this section may be modeled as a step function increase in power to a good approximation to the actual situation; i.e., the transient reactor power level increases instantaneously or apparently so in terms of the thermal inertia of the coolant within the reactor core. In a time on the order of milliseconds the excursion is initiated, the reactor scrams, and the power excursion is ended. For the sake of estimating the peak fuel temperatures, it is assumed that during this short excursion in reactor power—i.e., during the time required for the fuel to heat up to a maximum in temperature—the heat transfer characteristics of the water coolant are assumed to be those extant before the excursion. It is noted that in the event that the reactor rupture disk breaks (all coolant is lost) or that the primary coolant boils momentarily prior to the rupture disk breaking, the heat transfer coefficient will change.

The fuel centerline temperature is estimated by determining what would be the steady-state centerline fuel temperature for a reactor operating at the maximum power level (the overpower) experience in the course of the transient; in this work, 500% and 625% overpower transients are considered. In an actual transient, the fuel would experience the high heating rates immediately as the neutron flux level increased (i.e., nearly in a step function) but the fuel temperature would respond with a time constant based on its heat capacitance (ρc_p). In general the time corresponding to maximum power density in the transient would not coincide—would be earlier—than the time at which the fuel (centerline) temperature would reach a maximum. It is expected then that it is quite conservative to neglect fuel heat capacity altogether and to estimate the centerline fuel temperature assuming a steady-state calculation at the maximum reactor power of the transient.

In addition to the scenarios specified above, for the purposes of the overpower transient calculations, the heat transfer is assumed one-dimensional (in the x-direction, normal to the face of the plate). Further, although in steady-state, the location of the maximum fuel temperature occurs usually 40 cm from the coolant inlet, or nearly two-thirds of the way through the channel; for the purpose of discussion below, it is assumed that the maximum fuel temperature will occur at the fuel hot spot halfway up the hot plate (≈ 29.5 cm) from the coolant inlet. This should be expected because the difference between the pre-transient steady-state coolant temperature at the pre-transient (nominal) hot spot (40 cm) and the pre-transient steady-state coolant temperature at the post-transient hot spot (29.6 cm) is small. On the other hand, the power density at the halfway position (29.6 cm) is significantly (15%) higher than that at the pre-transient hot spot (40 cm). The shift in hot spots discussed here reflects the assumption that the reactor coolant characteristics do not change in the course of the transient. The fuel and clad are therefore in local steady-state with the coolant at the same axial position, but the coolant axial temperature distribution is not a steady-state distribution for the fuel and clad temperatures calculated at the maximum power level of the transient.

It is noted that the actual calculations for the overpower transient maximum fuel temperature are performed in the same thermal hydraulics solver described previously.^[15] The calculations are performed at both halfway through the duct (near 29 cm) and at the pre-transient hot spot (near 40 cm). As discussed here, it is found that the post-transient hot spot is located midway through the reactor core.

4.3.2 Overpower Analysis Results

Because existing UFTR safety analysis considers overpower transients at 500 kW and 625 kW, the same overpower transients are considered for comparison purposes with the LEU configuration. Tables 4.6 and 4.7 provide the calculated results for the fuel temperatures for 500% and 650% overpower transients; Table 4.6 provides results for a reactor with pre-transient steady-state conditions corresponding to the "base case" in Tables 4.3 and 4.4. Table 4.7 provides results for a reactor with pre-transient conditions corresponding to the "best estimate" or current UFTR full power operating conditions (see Table 4.5). More detailed results from the simple transient analysis considered here are available for reference and to support relicensing documentation to be submitted to the NRC.^[15]

An immediately evident significant feature of the overpower transient estimate results provided is that the peak LEU fuel and clad temperatures predicted for these transients are significantly below those predicted for the current HEU core configuration. Peak fuel temperatures in the LEU configurations may be expected to be between 370 and 390 K at 500% overpower (390 to 405 K at 625% overpower), using the simple analysis, while the HEU configuration leads to maximum fuel temperatures as high as 430 K at 500% overpower (455 K at 625% overpower). This relaxation of maximum fuel temperatures by a considerable percentage is again a consequence of the lower power per plate produced—both during steady-state and during an excursion—with the LEU core configurations versus the power per plate produced with the current HEU configuration. The overpower results are generally in agreement with calculations performed previously on HEU fuel for certain overpower conditions.^[21]

Note that the clad outer surface-to-coolant ΔT obtained during the overpower transients—for both 500% and 625% overpower and also both HEU to LEU configurations—

suggest that the heat transfer mechanisms during an actual transient will not remain subcooled convection but will quickly, on the order of the thermal time constants in the fuel and clad, transition to boiling or radiation heat transfer.

It is also noteworthy that the 15/2.5 maximum fuel temperature is 10 K lower than either the 14/2.5 or 14/2 configurations for the 500% overpower transient and 10 to 12 K lower during the 625% overpower transient. These lower temperatures indicate, once again, that for reactor temperature considerations, both during steady-state and during a transient, the 15/2.5 LEU core configuration has advantages over the 14/2 and 14/2.5 configurations, all of which have advantages—again in the sense of the lower reactor temperatures—over the HEU fuel configured cores. This advantage is expected again because of more surface area through which to remove heat.

Finally, note that the solidus temperature of Al (6061) is near 855 K (NUREG 1313); the melting point of pure Al is 775 K^[19] and the onset of metal-salt reaction in U_3Si_2 -Al is 863 K. It is evident from Tables 4.6 and 4.7 that the simple overpower transient estimates indicate that there is much fuel and clad temperature margin (ΔT) allowed for in the LEU fuel configurations versus onset of metal/salt reaction or clad or fuel melting. Further, previous overpower transient safety analysis work for the HEU core configurations will prove even more conservative for the LEU configurations due to the lower power per plate in the LEU configurations compared with the HEU configured reactors. Indeed, in the final analysis, the increased number of plates in all LEU configurations required by the neutronics automatically assures increased safety margins for the temperatures versus the existing HEU configuration.

Table 4.6 Comparison 1 of Estimated UFTR Peak Temperatures Midway Through (or at the Axial Centerline of) the Hot Plate During 500% and 625% Overpower Transients¹

Plates/Dummies:	14/2	14/2.5	15/2.5	11/2.5
Case:	A	B	K	HEU-A
500% Overpower				
Fuel Centerline-to-Coolant ΔT (K):	61.1	62.5	53.0	100.9
Fuel Centerline Temperature (K):	387.6	389.2	378.6	427.5
Clad Outer Temperature (K):	387.3	388.9	379.4	426.8
Clad Outer-to-Coolant ΔT (K):	60.8	62.2	52.7	100.1
625% Overpower				
Fuel Centerline-to-Coolant ΔT (K):	76.4	78.2	66.2	126.1
Fuel Centerline Temperature (K):	402.9	404.8	392.9	452.7
Clad Outer Temperature (K):	402.5	404.6	392.6	451.8
Clad Outer-to-Coolant ΔT (K):	76.0	77.8	65.9	125.1

¹Pre-transient Conditions: Volumetric flow rate is 40 gpm; reactor coolant inlet temperature is 319.11 K, hot channel factor (f_H) is 1.5, and reactor is at full power (100 kW_{th}).

Table 4.7 Comparison 2 of Estimated UFTR Peak Temperatures Midway Through (or at the Axial Centerline of) the Hot Plate During 500% and 625% Overpower Transients^{1, 2}

Plates/Dummies:	14/2	14/2.5	15/2.5	11/2.5
Case:	I	J	O	HEU-B
500% Overpower				
Fuel Centerline-to-Coolant ΔT (K):	61.1	62.5	53.0	100.9
Fuel Centerline Temperature (K):	381.4	383.0	373.5	421.4
Clad Outer Temperature (K):	381.2	382.7	373.2	420.6
Clad Outer-to-Coolant ΔT (K):	60.8	62.2	52.7	100.1
625% Overpower				
Fuel Centerline-to-Coolant ΔT (K):	76.4	78.2	66.2	126.1
Fuel Centerline Temperature (K):	396.7	398.7	386.7	446.6
Clad Outer Temperature (K):	396.4	398.3	386.4	445.6
Clad Outer-to-Coolant ΔT (K):	76.0	77.8	65.9	125.1

¹Pre-transient conditions are for UFTR current best estimate operating conditions per Table 4.5.

²Volumetric flow rate is 43.5 gpm; reactor coolant inlet temperature is 313.56 K, hot channel factor (f_H) is 1.5, and reactor is at full power (100 kW_{th}).

4.4 Summary Discussion and Conclusions

This section provides summary discussion of the steady-state and overpower transient thermal hydraulic analysis. Concluding statements on the relative merits of the LEU fuel configurations and HEU fuel configurations are provided.

4.4.1 Summary of Steady-State Analysis

The steady-state analysis provides data for discussion on the relative merits of the thermal hydraulic characteristics of the proposed low-enriched uranium (LEU) fueled UFTR and the existing high-enriched uranium (HEU) fueled UFTR. The steady-state thermal hydraulics characteristics of three LEU core configurations were compared. Parametric calculations for the three LEU core configurations around 100 kW_{th} "Base case" operating conditions (see Table 4.2) were performed. For all three LEU configurations, hot channel factors of 1.5 (base case) and 3.0 (highly skewed) were considered at coolant flows of 40 gpm (base case) and 30 gpm (low flow). Comparisons among the LEU core configurations demonstrate the following:

- The 14/2.5 configuration has the highest average fuel power density among the LEU fueled configurations. The 14/2 configuration has a lower average fuel power density because the active fuel is spread over 308 plates rather than 301 plates in the case of the 14/2.5 configuration. The 15/2.5 configuration has the lowest average fuel power density because, although this configuration has 2.5 dummy bundles, the active fuel is distributed among 322 plates. Therefore, in the steady-state calculations, the 15/2.5 configuration consistently demonstrates lower peak fuel-centerline and peak clad surface temperatures.
- The maximum core fuel centerline temperature and maximum clad-outer surface temperatures (in the hot plate of the hot bundle) occur approximately 70% of the

way up through the active core length; i.e., at a distance of 40 to 41 cm along the active core length of 59.1 cm. The peak heat flux and therefore clad surface-to-coolant temperature difference occurs halfway up the active core length, at 29.6 cm.

- The effect of increasing the hot channel factor (ratio of hot bundle power density to average bundle power density) correspondingly doubles the core inlet-to-outlet ΔT for a fixed coolant flow rate. The increase in the hot bundle/hot plate maximum centerline fuel temperature and the corresponding clad-outer temperature is not dramatic; e.g., in the 15/2.5 configuration, the maximum fuel temperature increases from 340 K (at $f_H = 1.5$) to 362 K (at $f_H = 3.0$) (see Table 4.3) as expected.
- At a fixed hot bundle factor, the change from nominal flow conditions (40 gpm) to low flow conditions (30 gpm) results in a 33% increase in core inlet to core outlet ΔT ; however, only a small change in reactor fuel and clad temperatures occur. For example, during low flow conditions, the maximum fuel centerline temperature and corresponding clad surface temperature in the hot plate of the hot bundle are both nearly 345 K as compared to 340 K at nominal coolant flow conditions (see Table 4.4) as expected.
- The highest hot bundle outlet temperature encountered in the steady-state parametric study, 351.2 K, occurs in the 15/2.5 configuration with a hot channel factor of 3.0 (at nominal 40 gpm flow). There exists therefore a 21 K margin between the highest encountered hot bundle exit temperature and the saturation temperature of 373 K. Note that in the case that the hot channel factor was

indeed 3.0 and a low flow (30 gpm) situation was in effect, the expected hot bundle exit temperature would be 362 K (the overall core inlet-to-outlet ΔT being 33% higher than the ΔT at nominal flow in this case), so that even in this unlikely and extreme instance, an 11 K margin exists between the core outlet temperature and the saturation temperature.

In Section 4.3, the three proposed LEU configurations discussed above are compared with the existing HEU UFTR fuel configuration at the steady-state 100 kW_{th} best estimate conditions (see Table 4.2). The LEU and HEU comparisons indicate the following:

- The proposed LEU fuel bundle configurations all have higher average fuel power densities (e.g., 16.91 MW_{th}/m³ in the 15/2.5 configuration) than the power density (11.88 MW_{th}/m³) of the existing HEU fueled UFTR; however, because the existing HEU fuel has only 11 plates per bundle, the thicker HEU plate produces significantly more power (634 W_{th}/plate) than do the plates of the proposed LEU fuel bundle configurations (e.g., 465 W_{th}/plate in the case of the 15/2.5 configuration). Therefore, the HEU fuel plates experience higher peak heat fluxes and system temperatures for given reactor conditions (see Table 4.5). For example, the maximum HEU fuel centerline temperature of the hot plate of the hot bundle is calculated to be 342 K at the current best estimate UFTR conditions as compared to a maximum fuel temperature of 334 K for the proposed 15/2.5 fueled UFTR at the same conditions.

The steady-state thermal hydraulics analyses therefore show that the proposed LEU fuel bundle configurations have higher average fuel power densities, but generate lower power per plate than do the existing HEU plates. Therefore, the LEU fueled UFTR reactor will experience

lower fuel and clad-surface temperatures. Among the proposed LEU configurations, the 14/2.5 configuration plates have the highest fuel power density and experience the highest maximum fuel and clad-surface temperatures. The 15/2.5 configuration has the lowest average fuel power density of the three proposed LEU configurations and experiences the fuel and clad temperatures of all the proposed LEU and existing HEU configurations.

4.4.2 Summary of Overpower Transient Analysis

Using the simple overpower transient analysis of this chapter, the following results are found for 500% and 625% overpower transients relative to pre-transient steady-state operation at the "base case" and current best estimate conditions of Table 4.2.

- The maximum centerline fuel and corresponding clad-surface temperatures during the transient are expected to occur halfway up the active core height (at 29.55 cm of 59.1 cm) rather than at the nominal steady-state operation hot spot (of nearly 40 to 41 cm of the 59.1 cm active core height).
- The peak fuel centerline temperatures of the proposed LEU fuel configurations are significantly lower than those predicted for the existing HEU fuel UFTR, again due to the much larger power produced per plate in the HEU fuel configuration. For example, the peak HEU configuration temperatures are 40 to 45 K higher in the 500% overpower transient than those of the proposed LEU fuel configurations; the HEU peak temperatures are 50 to 60 K higher than the LEU temperatures for the 625% overpower transient.
- The 15/2.5 LEU configuration has a 10 K lower maximum centerline temperature as compared to the maximum fuel temperatures of the 14/2 and 14/2.5 LEU configurations during the 500% overpower transient and is 10 to 12 K lower in

the 625% overpower case. This is consistent with the steady-state results, due to the distribution of fuel over many more plates in the 15/2.5 fuel configuration.

- The clad out surface-to-coolant ΔT (or heat fluxes) experienced by all LEU and HEU configurations during the 500% and 625% overpower transients suggest that the heat transfer coolant heat transfer mode in an actual transient will quickly change away from subcooled convection.
- The simple, conservative overpower transient analysis provided in this chapter predicts fuel and clad temperatures much below the melting points of aluminum and below the onset of metal-salt reactor in U_3Si_2 -Al (see Tables 4.6 and 4.7).

4.4.3 Conclusions for Thermal Hydraulics

The summary comments provided above show that the proposed LEU fuel configured UFTR has higher average fuel power density than does the existing HEU fueled UFTR; however, the proposed LEU fueled UFTR will experience lower hot plate maximum fuel centerline and clad out-surface temperatures. Further, a simple overpower transient analysis, consistently performed for the proposed LEU fueled UFTR and the existing HEU fueled UFTR indicates that lower peak fuel and clad temperatures will be encountered in the LEU fueled reactor than in the HEU reactor. Therefore, from a purely thermal hydraulics perspective, and in light of the HEU UFTR experience, the LEU fuel configurations are acceptable in terms of anticipated reactor fuel, clad, and coolant temperatures. Further, the analysis presented here suggests increased safety margin, in terms of fuel and clad temperatures and peak heat fluxes, in the switch from HEU to LEU fueled reactor.

Among the LEU fuel configurations, the 15/2.5 configuration has the lowest average fuel power density due to the large number of active fuel plates over which the active fuel is distributed. The maximum fuel and clad temperatures experienced by the 15/2.5 LEU configuration are lower than those in the 14/2 or 14/2.5 LEU configurations. Therefore, based on thermal hydraulics analysis, the 15/2.5 LEU configuration is recommended over the 14/2 or 14/2.5 LEU configurations.

Figure 4.1 Fuel Plate/Coolant Channel Enlargement Showing UFTR Fuel Arrangement Detail

5.0 Other Licensing Considerations

Because of the selection of the uranium silicide (U_3Si_2 -Al) fuel design, other considerations following the neutronics and thermal hydraulics analysis are greatly simplified. Few calculations are required to provide the licensing basis in these areas once the fuel selection is made. These areas include effluent analysis, shielding analysis, ex-core criticality analysis, emergency plan analysis, and technical specification changes. These areas are each considered in turn and, because of the fuel selection, each is found to require little additional analysis to meet the NRC licensing requirements for HEU to LEU fuel conversion for the UFTR.

5.1 Effluent Analysis

The only effluent for the UFTR that is affected by the HEU to LEU conversion is the gaseous Argon-41 resulting from neutron activation of the approximately 1% Argon-40 constituent in the air flowing past and through the reactor core and, to a lesser extent, dissolved in the reactor coolant. Since all neutron fluxes are about the same for the LEU core versus the HEU core, and smaller in most cases as shown in Chapter 3.0, the Argon-41 effluent for the LEU core is expected to fall within the corresponding effluent values for the HEU core. Therefore, the LEU conversion will not affect operating restrictions relative to gaseous Argon-41 effluent values. Nevertheless, this effluent value will be measured after the new LEU core is loaded upon initial full power operation to assure documentation that the existing restrictions on monthly energy generation will assure operating within Argon-41 effluent limits.

5.2 Shielding Analysis

For shielding considerations, it is again noted that the UFTR LEU core will present shielding requirements that are within the limits required for the existing HEU core. The total fission product inventory will be similar based upon the similar spectrum and same operating

power. Minor variations in the fission product inventory makeup will not require changes in shielding. At full power ($100 \text{ kW}_{\text{th}}$) operation, the existing shielding will be adequate, meeting the same limits as for the HEU fueled core, especially since the core is essentially the same size as for the HEU fuel continuing to fill the six existing fuel boxes.

During fuel transfer operations, the fuel bundles are essentially the same physical size and will contain the same maximum fraction of the core radioactivity (fission product) inventory. As a result, the existing fuel transfer shield cask will continue to provide more than adequate shielding during fuel handling. Fuel handling will continue to be restricted to one fuel bundle at a time so the source terms and shielding requirements will be essentially unchanged.

5.3 Criticality Analysis (Ex-Core)

First, the irradiated fuel pits accept one bundle only per pit; since the LEU neutronics analysis indicates the core will have the same fuel bundle loading as the existing HEU core with the same approximate critical conditions and neutron multiplication factor, the pits will also serve for storage of LEU irradiated fuel to fit within the HEU irradiated fuel design envelope. Second, the unirradiated fuel storage facility will also continue to be usable for the same reasons, again fitting within the design envelope.

5.4 Emergency Plan Analysis

Again, the selection of the silicide fuel in the same core configuration as for the HEU fuel but with 14 plates per bundle assures that all emergency response limitations fall within the envelope existing for the HEU fueled core per the existing UFTR Emergency Plan.^[22] The total fission product source term will be essentially unchanged since operation will continue to be at $100 \text{ kW}_{\text{th}}$ for full power. Similarly, hours of operation will be limited to assure meeting the emergency plan assumptions for radioactivity inventory.

It is also noteworthy that the limitation on moving and handling only one fuel bundle at a time will continue. However, each LEU bundle will have 14 plates versus 11 plate per bundle in the HEU fueled core. Therefore, when the assumption is made that cladding can be stripped from one plate for the maximum credible accident, as is currently done for the maximum credible accident, the source term involved will be less than 80% ($^{11/14}$) of that for the HEU fueled core case. Even allowing for minor variations in core radioactivity constituents, it is clear here that the LEU accident case falls well within the envelope of the case for the existing HEU core. So again, the emergency plan analysis is simplified by the choice of LEU silicide fuel in the same core configuration as for the existing HEU core.

5.5 Technical Specification Changes

The only changes in the UFTR Technical Specifications are in the sections describing the fuel—Section 5.3 (Reactor Fuel) and Section 5.4 (Reactor Core). These changes are minor as the low-enriched silicide fuel replaces the existing high-enriched U-Al fuel and can be easily incorporated in the licensing package for NRC consideration based on the results of this report.

6.0 Discussion and Conclusions

This report contains the results of basic analyses required to support conversion of the University of Florida Training Reactor from high-enriched uranium (U-Al) fuel plates to standard low-enriched silicide (U_3Si_2 -Al) fuel plates.

6.1 Conclusions

The work reported in Chapter 2.0 allowed the choice of the silicide (U_3Si_2 -Al) standard plate-type fuel over the SPERT pin-type fuel stored at the University of Florida under the SNM-1050 license. As discussed in Chapter 2.0 and amplified in Chapter 5.0, this fuel selection allows the minimum number of changes in the UFTR core as the fuel bundles are essentially the same physical size and will have similar fission products/radioactive material inventory when compared to the existing HEU fuel bundles. The remainder of the UFTR core, shielding and even auxiliary facilities will require no modification.

The neutronics results reported in Chapter 3.0 and the thermal hydraulics results reported in Chapter 4.0 show that the silicide fuel in 14-plate bundles is the optimal fuel bundle choice. This 14-plate design will assure a critical configuration can be assembled with two dummy fuel bundle locations expected initially to allow working margin on the loading which is an experimental procedure. Since the loading is experimental, some additional fuel loading may be deemed necessary or desirable during the loading as any additional loading makes all safety-related limits less restrictive. The criticality and flux variations have been shown to be acceptable by the neutronics analyses summarized in Chapter 3.0. Again, the facilities used to store unirradiated or irradiated LEU fuel will continue to be usable with no modification.

First, the irradiated fuel pits accept one bundle only per pit; since the LEU neutronics analysis indicates the core will have the same loading as the existing HEU core with the same

approximate critical conditions and neutron multiplication factor, the pits will also serve for storage of LEU irradiated fuel with the HEU irradiated fuel design envelope. Second, the unirradiated fuel storage facility will also continue to be usable for the same reasons. Further, not only will the core internal structure remain unchanged but shielding needs will also remain the same. In addition, the lead shield fuel transfer cask will continue to be usable for removing fuel bundles for inspection or for core access which means the overhead crane system will also not need to be considered for change. The same is true for the storage container system used for fresh unirradiated fuel. Finally, the storage pits used for off-loading fuel assemblies whether for inspection, core maintenance, or perhaps because of damage to fuel bundles or plates, will be able to continue in use without modification. All these systems that can remain in use without modification represent significant savings in both financial resources and human resources via the ALARA principle for minimizing radiation dose commitment.

In Chapter 4.0, the steady-state thermal hydraulics analysis shows the UFTR operating with LEU fuel will be adequately cooled with the existing cooling system, perhaps with an increase in pumping power which is available with the existing primary coolant pump which is currently valved down by nearly 30% from the maximum flow condition. Not only are steady-state fuel temperatures lower for the 14-plate LEU bundles versus the 11-plate HEU bundles, but the same condition of increased safety margin applies for the overpower transient analyses performed for the 500 kW_{th} and 625 kW_{th} power levels considered in the current HEU safety analysis.^[22] Again, the silicide fuel choices drive these conclusions.

The discussion in Chapter 5.0 shows that the effluent and shielding considerations will not change with the HEU to LEU conversion. Similarly, emergency planning considerations for the LEU fuel are shown to fall within the design envelope for emergency response for the

existing HEU fueled core. Similarly, the only changes in technical specifications required involve minor changes to the fuel physical descriptions in Section 5.3 (Reactor Fuel) and Section 5.4 (Reactor Core). Again, these changes will be easily incorporated into the licensing package for submittal to the NRC.

The results of this work are now ready for incorporation into a format for submittal to NRC for licensing conversion of the UFTR for HEU to LEU fuel. With the results of this study and some additional document preparation, it is expected that the NRC will approve HEU to LEU conversion of the UFTR with no significant problems.

6.2 Suggestions for Additional Work

Although the basic neutronics and thermal hydraulics analyses have been accomplished as reported earlier, two additional pieces of analysis will be required prior to submitting the HEU to LEU conversion document to NRC based on NUREG-1537 which is the document providing NRC recommendations on content and format for the submission.^[23] Some basic control blade reactivity worths must be calculated so the actual control blade reactivity worths can be compared with them following the initial experimental LEU fuel loading. Similarly, a few reactivity transients will have to be run using the data, especially reactivity coefficients presented and/or referenced in Chapter 3.0. When these calculations are complete, then the entire licensing document for conversion will be assembled and submitted to NRC for approval of the HEU to LEU conversion.

The NRC will require up to a year to review this document submittal. During this time, all procedures, the final safety analysis report and any other internal documents, such as training materials, will be reviewed and changed as appropriate. With the results of this study providing the basis, additional support will then be needed to implement the actual conversion of the

UFTR including the items noted above as well as considerations for accepting and loading LEU fuel, shipping the used HEU fuel and allowing for the required reactor down time for conversion and the testing and documentation required for the converted core.

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