UNIVERSITY OF FLORIDA
TRAINING REACTOR
LICENSE NO. R-56
DOCKET NO. 50-83

SUBMITTAL OF RESPONSES DATED AUGUST 4, 2006, TO NRC REQUEST FOR ADDITIONAL INFORMATION FOR THE CONVERSION FROM HIGH-ENRICHED URANIUM TO LOW-ENRICHED URANIUM FUEL

REACTED VERSION

SECURITY RELATED INFORMATION REMOVED IN ACCORDANCE WITH 10 CFR 2.390(d)(1)

Redacted text and figures blacked out or denoted by brackets

UF | UNIVERSITY of FLORIDA

College of Engineering
Department of Nuclear & Radiological Engineering
UFTR Nuclear Facilities

202 Nuclear Sciences Bldg.
PO Box 118300
Gainesville, FL 32611-8300
352-392-1401
352-392-3380 Fax
www.nre.ufl.edu

August 4, 2006

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Amendment 26
(previously Amendment 25)
UFTR Technical Specifications
Addendum 2

University of Florida Training Reactor, Facility License: R-56, Docket No. 50-83
Request for Change in Technical Specifications Approving HEU to LEU Conversion
With Responses to Requests for Additional Information

A proposed amendment to the UFTR Technical Specifications (R-56 License) for conversion from high enriched uranium (HEU) fuel to low enriched uranium (LEU) fuel affecting pages 4, 5, 6, 7, 8, 9, 13, 15, 16, 21, 23, 24, 26 and 38 of the approved Tech Specs was submitted by letter dated June 19, 2006.

A second submittal with a letter dated July 20, 2006 corrected minor typographical errors in the Technical Specification pages of the June 19, 2006 submittal that were not intended to be changed per discussions with NRC Senior Project Manager Al Adams on July 19, 2006 plus other changes. Tech Spec pages in the July 20 submittal to replace those in the June 19 package were pages 4, 6, 8, 9, 15, 16 and 23.

The proposed changes, updated with this submittal, will continue to constitute Amendment 26 to the UFTR R-56 License as noted on the text pages. The corrected pages as submitted are considered to have minor safety significance. These two changed Tech Spec pages simply replace the corresponding pages in the previous two submittals to account for Limiting Safety System Settings necessary based on the analyses of the allowed tolerances in the fuel bundle fuel coolant channel spacing in *Attachment II* to this letter. Tech Spec pages affected which replace those in the earlier submittals are pages 5 and 8 in *Attachment I* to this letter.

First, on page 5, in Section 2.2, Limiting Safety System Settings, specification (2) is now split into two alternative specifications where the LSSS is set at 36 gpm for fuel channel spacing tolerances up to 15 mils and 41 gpm for fuel channel spacing tolerances up to 20 mils. Similarly, in specification (3), the average primary coolant inlet temperature specification in (3)(a) is now split into two alternative specifications at 109° F for fuel channel spacing tolerances up to 10 mils and 99° F for fuel channel spacing tolerances up to 20 mils.

Second, on page 8, in Table 3.1, Specifications for Reactor Safety System Trips, under Automatic Trips, the specification for Primary Cooling System, Low Inlet Water Flow is split into two alternative specifications at < 36 gpm for fuel coolant channel spacing tolerances at ≤ 15 mils and < 41 gpm for fuel coolant channel spacing tolerances ≤ 20 mils. Similarly, the specification for High Primary Coolant

A020

Average Inlet Temperature is split into two alternative specifications at ≥ 109° F for fuel coolant channel spacing tolerances at ≤ 10 mils and ≥ 99° F for fuel coolant channel spacing tolerances at ≤ 20 mils. These two corrected pages as submitted are considered to have minor safety significance. These two changed pages simply replace the corresponding pages in the previous submittals. These proposed changes have been reviewed in progress by UFTR management and some members of the Reactor Safety Review Subcommittee (RSRS), as well as formally by the RSRS Executive Committee prior to submittal, with all concurring on this evaluation.

Based on a telephone conversation on August 2, 2006 with project manager Al Adams, included as *Attachment III* is additional discussion and analyses showing that existing fuel storage facilities continue to meet the criticality requirements of Tech Spec Section 3.7(6), Fuel and Fuel Handling, requiring k_{eff} to be less than 0.8, as well as the criticality safety requirements of Tech Spec Section 5.8, Fuel Storage. This supporting analysis is supplied following the format of NUREG-1537.

Finally, the question about consistency of control blade reactivity worths is addressed in *Attachment IV* including replacement pages for the earlier submittal.

This entire submittal consists of one signed original letter of transmittal plus *Attachment I* containing the two replacement Tech Spec changed pages to the requested Amendment 26 plus *Attachment II* and *Attachment III* containing the reference analyses as well as *Attachment IV*.

We appreciate your consideration of this submittal. Please advise if further information is needed.

Sincerely,

William G. Ve

William G. Vernetson
Director of Nuclear Facilities

Email: vernet@ufl.edu
Phone: 352-392-1408 ext. 317

WGV/dms
Attachments I, II, III & IV

cc: Al Adams, NRC Project Manager
Craig Bassett, NRC Inspector
Reactor Safety Review Subcommittee

Sworn and subscribed this 4th day of August 2006.

Terri L. Sparks
Notary Public

Terri L. Sparks
Commission # DD346496
Expires August 12, 2008
Bonded Troy Fain - Insurance, Inc. 800-385-7019

ATTACHMENT I

REPLACEMENT TECH SPEC CHANGED PAGES
FOR ADDENDUM 2 TO THE
TECH SPEC AMENDMENT 26
JUNE 19, 2006 SUBMITTAL

Specifications: The limiting safety system settings shall be

(1) Power level at any flow rate shall not exceed 119 kW.

(2) The primary coolant flow rate shall be
  (a) greater than 36 gpm at all power levels greater than 1 watt if the fuel coolant channel spacing tolerance is ≤ 15 mils
  (b) greater than 41 gpm at all power levels greater than 1 watt if the fuel coolant channel spacing tolerance is ≤ 20 mils.

(3) The average primary coolant
  (a) inlet temperature shall not exceed 109° F when the fuel coolant channel spacing tolerance is ≤ 10 mils
  (b) inlet temperature shall not exceed 99° F when the fuel coolant channel spacing tolerance is ≤ 20 mils
  (c) outlet temperature shall not exceed 155° F when measured at any fuel box outlet.

(4) The reactor period shall not be faster than 3 sec.

(5) The high voltage applied to Safety Channels 1 and 2 neutron chambers shall be 90% or more of the established normal value.

(6) The primary coolant pump shall be energized during reactor operations.

(7) The primary coolant flow rate shall be monitored at the return line.

(8) The primary coolant core level shall be at least 2 in. above the fuel.

(9) The secondary coolant flow shall satisfy the following conditions when the reactor is being operated at power levels equal to or larger than 1 kW:

  (a) Power shall be provided to the well pump and the well water flow rate shall be larger than 60 gpm when using the well system for secondary cooling.

  or

  (b) The water flow rate shall be larger than 8 gpm when using the city water system for secondary cooling.

(10) The reactor shall be shut down when the main alternating current (ac) power is not operating.

(11) The reactor vent system shall be operating during reactor operations.

(12) The water level in the shield tank shall not be reduced 6 in. below the established normal level.

Bases: The University of Florida Training Reactor (UFTR) limiting safety system settings (LSSS) are established from operating experience and safety considerations. The LSSS 2.2.3 (1) through (10) are established for the protection of the fuel, the fuel cladding, and the reactor core integrity. The primary and secondary bulk coolant temperatures, as well as the outlet temperatures for the six fuel boxes, are monitored and recorded in the control room. LSSS 2.2.3 (11) are established for the protection of reactor personnel in relation to accumulation of argon-41 in the reactor cell and for the control of radioactive gaseous effluents from the cell. LSSS 2.2.3 (12) are established to protect reactor personnel from potential external radiation hazards caused by loss of biological shielding.

Table 3.1 Specifications for reactor safety system trips

Specification	Type of safety system trip
Automatic Trips	
Period less than 3 sec	Full
Power at 119% of full power	Full
Loss of chamber high voltage (≥10%)	Full
Loss of electrical power to control console	Full
Primary cooling system Loss of pump power Low-water level in core (< 42.5") No outlet flow Low inlet water flow (< 36 gpm for fuel coolant channel spacing tolerance at ≤ 15 mils; < 41 gpm for fuel coolant channel spacing tolerance at ≤ 20 mils)	Rod-drop
Secondary cooling system (at power levels above 1 kW) Loss of flow (well water < 60 gpm, city water < 8 gpm) Loss of pump power	Rod-drop
High primary coolant average inlet temperature (≥ 109° F for fuel coolant channel spacing tolerance at ≤ 10 mils; ≥ 99° F for fuel coolant channel spacing tolerance at ≤ 20 mils)	Rod-drop
High primary coolant average outlet temperature (≥155° F)	Rod-drop
Shield tank Low water level (6" below established normal level)	Rod-drop
Ventilation system Loss of power to dilution fan Loss of power to core vent system	Rod-drop
Manual Trips	
Manual scram bar	Rod-drop
Console key-switch OFF (two blades off bottom)	Full

ATTACHMENT II

ANALYSES OF ALLOWED TOLERANCES IN FUEL BUNDLE FUEL COOLANT CHANNEL SPACING FOR TECH SPEC AMENDMENT 26 CONSIDERATION

UNIVERSITY OF FLORIDA TRAINING REACTOR
SUPPLEMENTAL INFORMATION
August 3, 2006

References:
1. UFTR Conversion Proposal, December 2, 2005
2. UFTR Responses to NRC Request for Additional Information, June 9, 2006

INTRODUCTION

Initial inspection and measurement of LEU fuel assemblies manufactured by BWXT in July 2006 showed a variation in the water channel thickness that was larger than expected. In this design, the ends of the plates are separated by aluminum spacers and are bolted together. Aluminum spacers are welded onto the edges of the plates at about half their height. This fuel assembly design has been modified by BWXT to include the "comb" design shown in Figure 1 that will physically separate the fuel plates at the nominal quarter-points along the fuel plate length. In this modified design, the tolerance on the minimum water channel spacing is expected to be a maximum of ± 20 mils. The nominal water channel spacing at the bolted ends of the fuel assembly on the manufacturing drawings is 110 – 112 mils, giving a minimum water channel spacing of 90 mils.

The actual minimum water channel spacing of each water channel in each finished fuel assembly will be measured by BWXT and made available by August 10, 2006. In case any one of the channel spacings is less than 90 mils, another comb will be added to maintain a minimum channel spacing of 90 mils.

Thermal-hydraulic analyses are provided in this supplement assuming tolerances on the water channel spacing of 10, 15, and 20 mils, along with repositioning of the fuel assemblies in the fuel boxes due to the 65 mil protrusion of the combs beyond the ends of the fuel plates. The same hot-channel-factor methodology described in Ref. 2, Appendix Q8 and related appendices, was used to compute curves of true reactor power versus the true coolant flow rate at which onset of nucleate boiling (ONB) would occur for several values of the average coolant inlet temperature.

The three key parameters to be examined are the true minimum coolant flow rate, the Limiting Safety System Setting (LSSS) for the coolant flow rate, and the normal operating coolant flow rate. The corresponding three parameters for reactor power level are not affected and will remain the same as in Reference 2.

CALCULATIONAL MODEL AND RESULTS

New Hot Channel Factors and Revised Model

The steady-state thermal hydraulic analysis described in Ref. 2 assumed hot channel factors based on a 1 mil tolerance for the water channel spacing, as shown on the manufacturing drawings at the bolted ends of the fuel assemblies. In the model, the ends of the fuel plates were assumed to be in contact with the fuel box wall. These new analyses account for two effects – the larger tolerance on the water channel spacing described above and the repositioning of the fuel assemblies in the fuel boxes due to the thickness of the combs extending beyond the edges

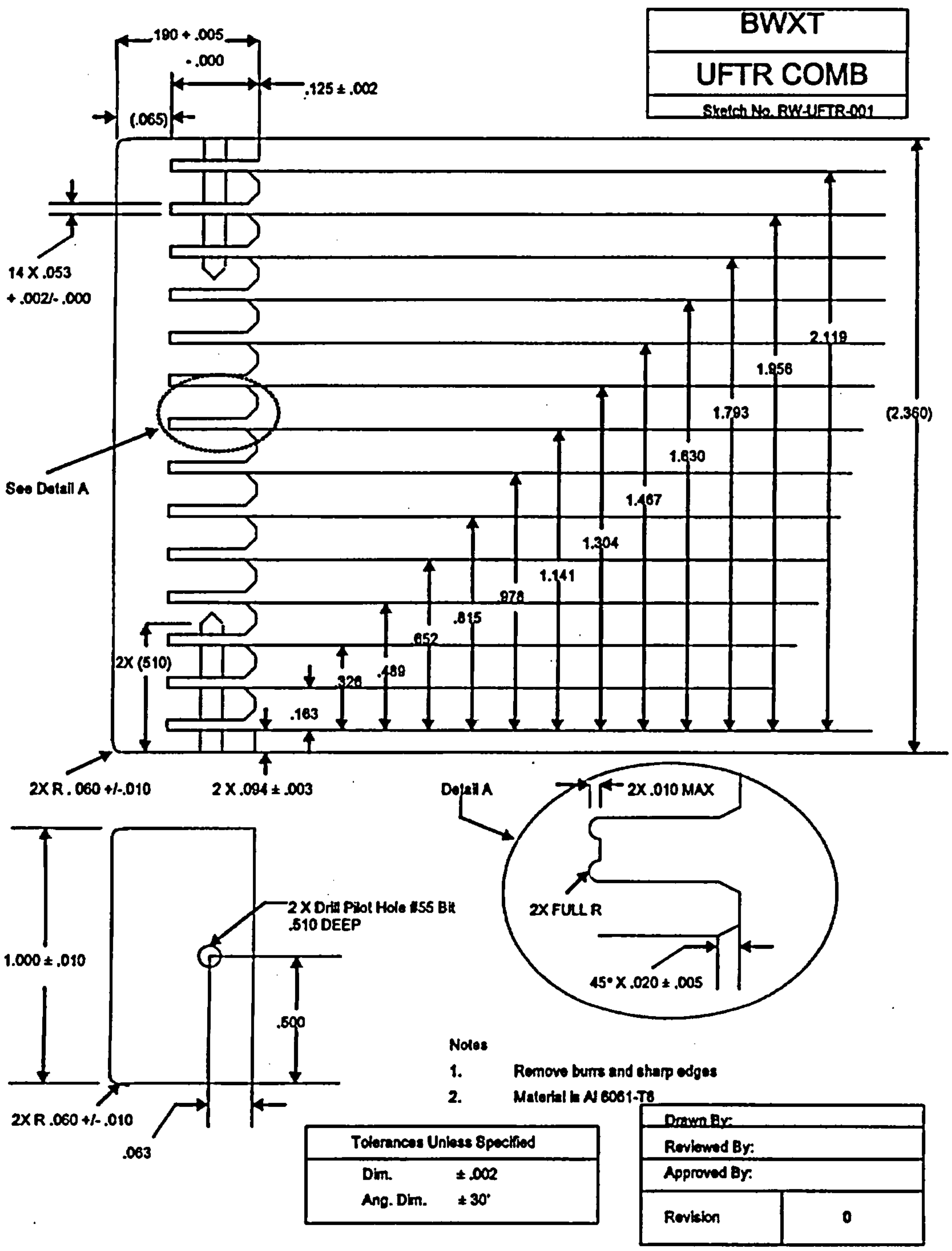

Figure 1 – Modified fuel assembly design by BWXT to include the "comb" design

of the fuel plates. The combs introduce into the computer model another bypass channel pathway that is located between the fuel plate ends and the box wall. In these new analyses, the ends of the fuel plates are moved away from the wall of the fuel box by 0.065 inches - the height of the comb above the "teeth", as shown in Figure 1.

However, the newly created slot is actually blocked by the width of the comb (0.190 inches), at the quarter-points along the length of the assembly. There will be a small bypass flow through this pathway that is conservatively accounted for by including it as a clear channel, ignoring the blockage created by the presence of the combs. The pressure drop introduced by the combs is not accounted for because it is very small. As a result, the computed coolant flow through the heated channels will be reduced, and the ONB margin and other parameters will be conservative. The presence of the comb also reduces the width of the central channel slot between fuel assemblies. This is a significant beneficial change because it reduces the bypass flow through the central slot.

The hot channel factors shown in Table 1 were computed for coolant channel spacing tolerances of 10, 15, and 20 mils using the methodology described in Ref. 2, Appendix Q8 and related appendices. All other hot channel factors shown in Ref. 2, Appendix Q8 are unchanged.

Table 1. Hot Channel Factors as a Function of Tolerance on Coolant Channel Width

Hot Channel Factor	1 mil	10 mil	15 mil	20 mil
F_b	1.21	1.39	1.58	1.84
F_{film}	1.23	1.24	1.26	1.29

Calculated Results

For convenience, Figure 2 shows the curves from Ref. 2, Question 8 and Appendix Q8 of true reactor power versus true coolant flow rate at which ONB occurs for average coolant inlet temperatures of 86, 100, and 110 °F. The key parameters are shown for reference purposes in Table 2.

Table 2. Key Parameters for Reactor Power and Coolant Flow Rate in Ref. 2.

Parameter	Value	Parameter	Value
True Maximum Power	125 kW	True Minimum Flow Rate	34 gpm
LSSS Power	119 kW	LSSS Flow Rate	36 gpm
Maximum Operating Power	100 kW	Operating Flow Rate	43 gpm

The new power versus coolant flow rate curves for water channel spacing tolerances of 10 mils, 15 mils, and 20 mils, along with repositioning of the fuel assemblies in the fuel box to account for the thickness of the combs are shown in Figures 3, 4, and 5, respectively.

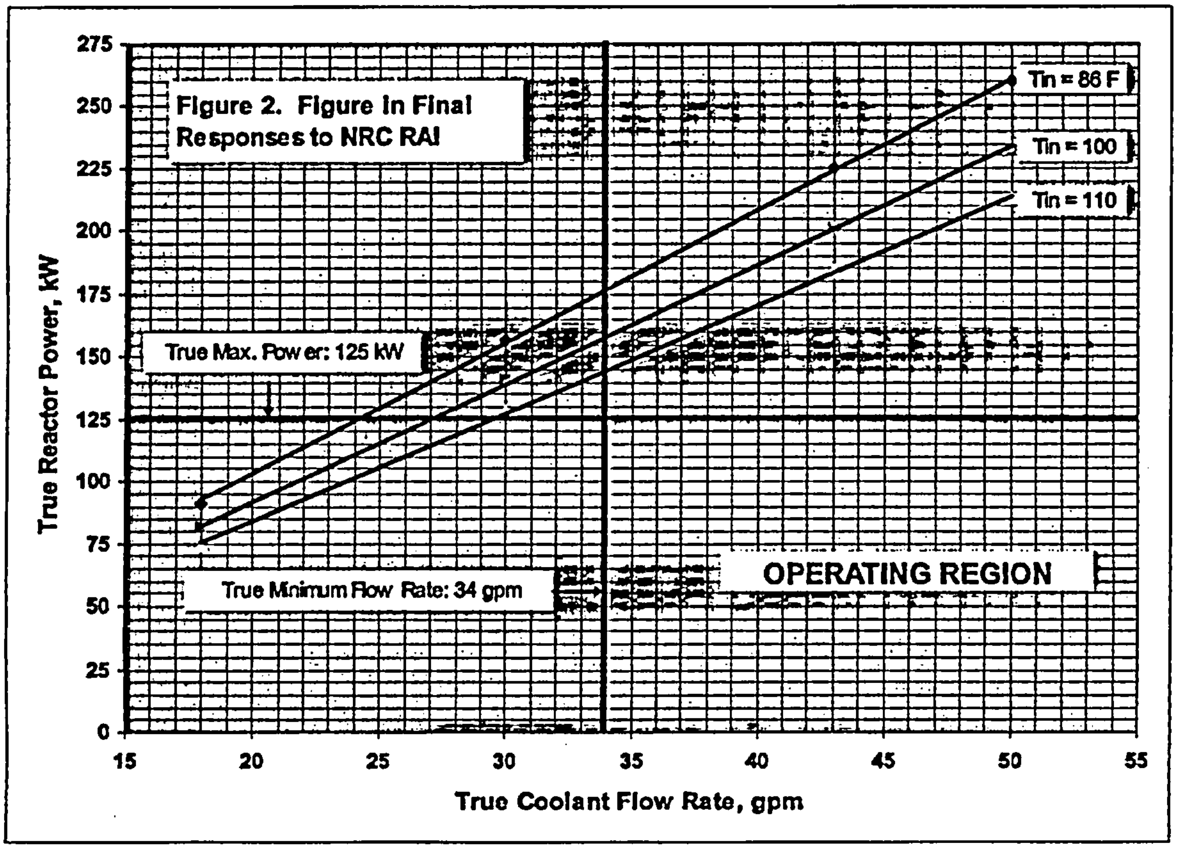

Figure 2. True Power versus True Coolant Flow Rate at Which ONB Occurs for Three Inlet Temperatures. (Figure from Ref. 2, Question 8 and Appendix Q8.)

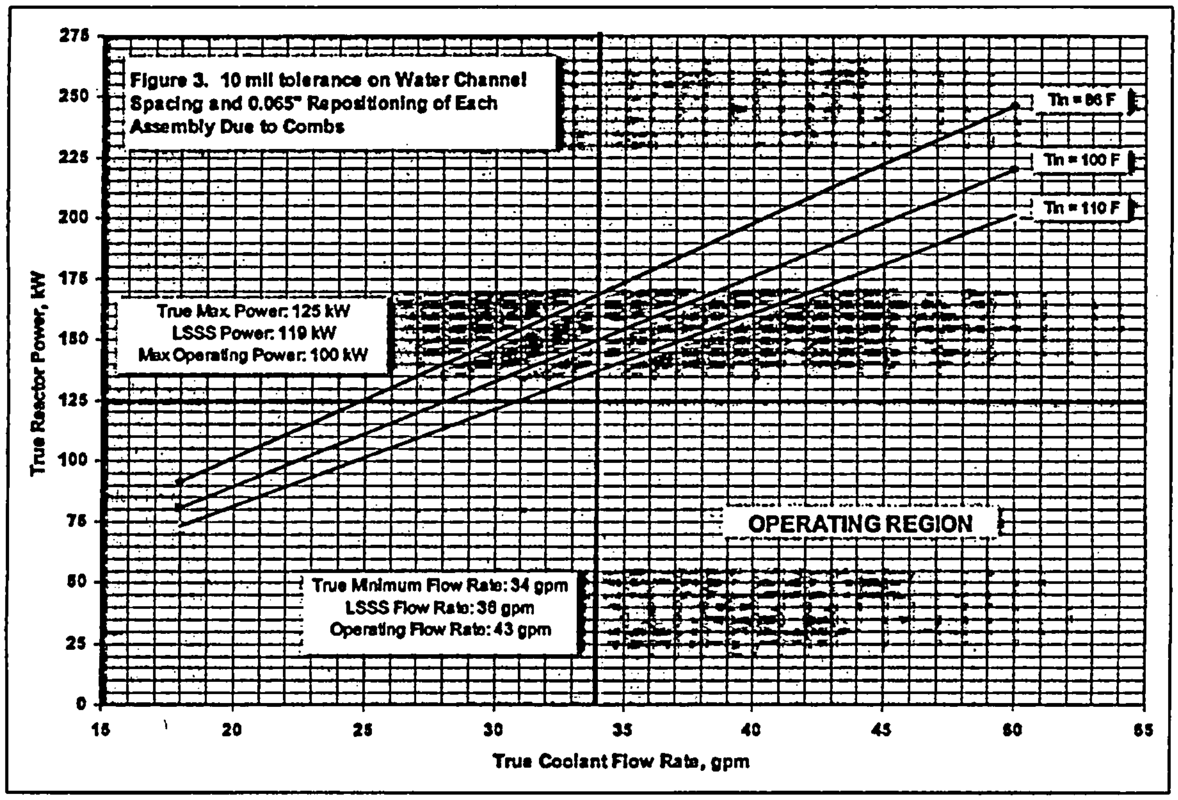

Figure 3. True Power versus True Flow Rate at Which ONB Occurs with 10 mil Tolerance on Water Channel Spacing and Repositioning of Assemblies in Fuel Box to Account for Thickness of Combs.

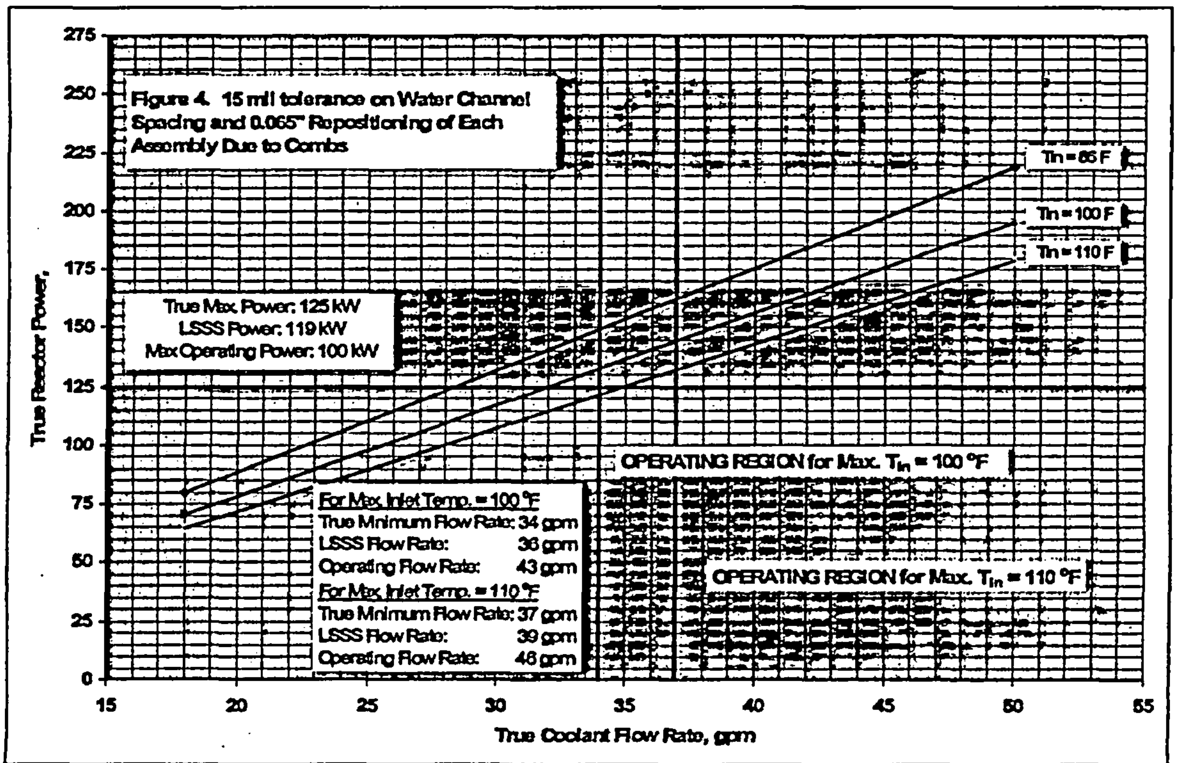

Figure 4. True Power versus True Flow Rate at Which ONB Occurs with 15 mil Tolerance on Water Channel Spacing and Repositioning of Assemblies in Fuel Box to Account for Thickness of Combs.

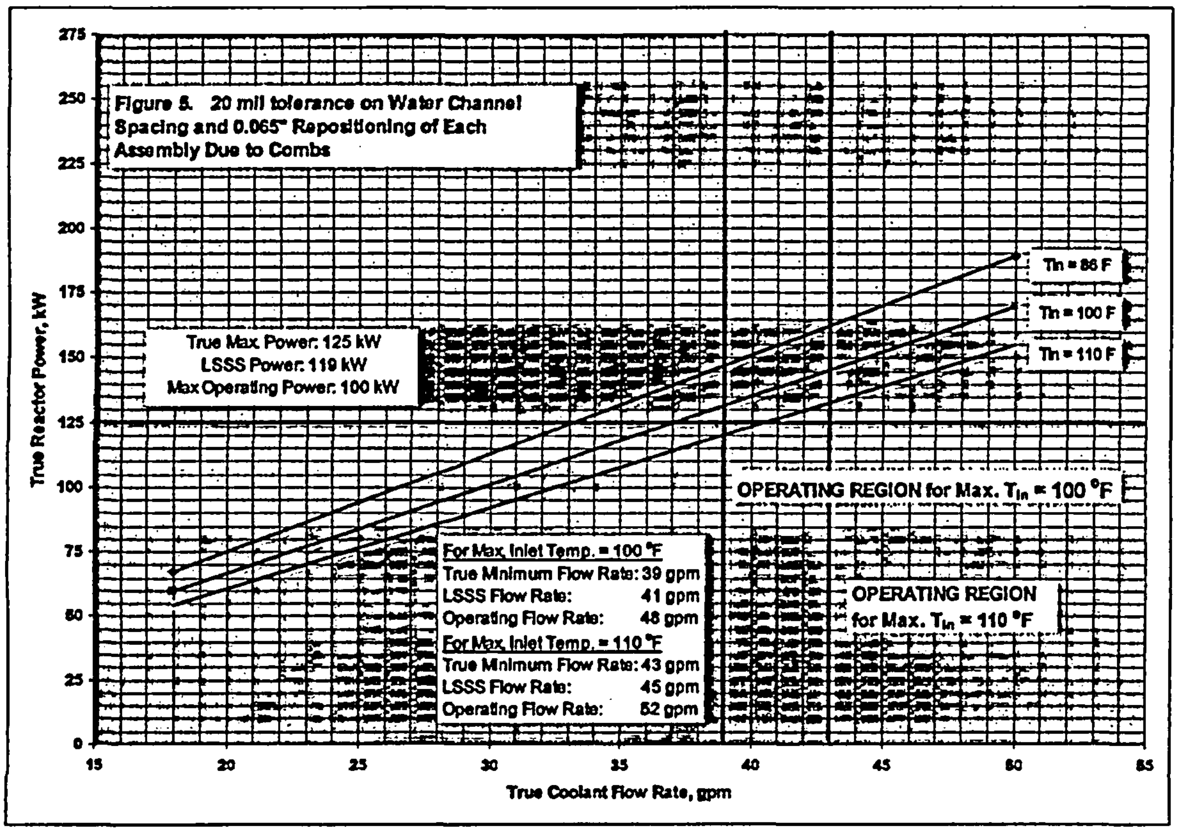

Figure 5. True Power versus True Flow Rate at Which ONB Occurs with 20 mil Tolerance on Water Channel Spacing and Repositioning of Assemblies in Fuel Box to Account for Thickness of Combs.

Based on these results, two sets of key flow rate settings are proposed, depending on the maximum value of the average inlet temperature:

True maximum inlet temperature = 100 °F (LSSS on inlet temperature of 99 °F)

Table 3 gives the proposed settings for different flow rates depending on the different water channel spacing tolerances.

Table 3 – Flow Rate Settings for Different Water-Channel-Spacing Tolerances

Type of flow rate	Water Channel Spacing tolerance (mil)			
	1	10	15	20
True Min. (gpm)	34	34	34	39
LSSS (gpm)	36	36	36	41
Operating (gpm)	43	43	43	48

Note that only the case with 20 mil tolerance requires changes to the values in Ref. 2.

True maximum inlet temperature =110 °F (LSSS on inlet temperature of 109 °F)

Table 4 gives the proposed settings for different flow rates depending on the different water channel spacing tolerances.

Table 4 – Flow Rate Settings for Different Water-Channel-Spacing Tolerances

Type of flow rate	Water Channel Spacing tolerance (mil)			
	1	10	15	20
True Min. (gpm)	34	34	37	43
LSSS (gpm)	36	36	39	45
Operating (gpm)	43	43	46	52

Cases with 15 mil tolerance and with 20 mil tolerance require changes to the values in Ref. 2.

CONCLUSIONS

Since a measured minimum water channel spacing of 90 mils (20 mil tolerance) is anticipated on a number of the LEU fuel assemblies being re-worked by BWXT, the following values for key coolant flow rate parameters are proposed in Table 5. The UFTR Technical Specifications will need to be changed.

Table 5 – Flow Rate Settings for 20 mil Water-Channel-Spacing Tolerance (measured minimum water channel spacing of 90 mils)

Type of flow rate	True Max. Inlet Temp.	
	100 °F	110 °F
True Min. (gpm)	39	43
LSSS (gpm)	41	45
Operating (gpm)	48	52

The accident analyses in Ref. 2 were performed using a true minimum coolant flow rate of 34 gpm, which is consistent with the proposed Technical Specifications for maximum water channel spacing tolerances of 10 mils and 15 mils. For a maximum tolerance of 20 mils, the accident analyses in Ref. 2 provide results that are more conservative than with the true minimum flow rate of 39 gpm proposed in the Technical Specifications. Therefore, none of the accident analyses in Ref. 2 need to be changed.

It is recognized that the thermal-hydraulic analyses with the hot-channel factor method used in the analyses in this supplement give conservative results. ANL is currently evaluating alternative methods for reducing some of the conservatism due the somewhat-unique variations in the water channel spacing anticipated in the UFTR LEU fuel assemblies. Measurements of the minimum channel spacing in each LEU fuel assembly are anticipated before August 10, 2006. Using the measured minimum channel spacing and less conservative, but more accurate, factors to represent more realistic coolant flow rate through these channels, it may be possible to reduce the coolant flow rate settings shown in Table 5.

Possible changes to the UFTR's Technical Specifications

Based on the current analyses, the UFTR technical specifications could be modified as follows:

a) If the maximum tolerance for the water channel spacing is 10 mil, we propose the following specifications:

Maximum T (inlet) = 110 F, and

Type of flow rate	Flow rate (gpm)
True Min. (gpm)	34
LSSS (gpm)	36
Operating (gpm)	43

b) If the maximum tolerance for the water channel spacing is 15 mil, we propose the following specifications:

Maximum T (inlet) = 100 F, and

Type of flow rate	Flow rate (gpm)
True Min. (gpm)	34
LSSS (gpm)	36
Operating (gpm)	43

c) If the maximum tolerance for the water channel spacing is 20 mil, we propose the following specifications:

Maximum T (inlet) = 100 F, and

Type of flow rate	Flow rate (gpm)
True Min. (gpm)	39
LSSS (gpm)	41
Operating (gpm)	48

ATTACHMENT III

**DISCUSSION AND ANALYSES
OF FUEL STORAGE FACILITIES
CRITICALITY AND SAFETY REQUIREMENTS
FOR TECH SPEC AMENDMENT 26
CONSIDERATION**

9. Auxiliary system

9.2 Handling and Storage of Reactor Fuel

This section presents the systems for secure storage of unirradiated and irradiated LEU fuel. These systems prevent criticality (k_{eff} not exceeding 0.80) under all conditions of moderation during storage.

9.2.1 Irradiated and Unirradiated Fuel Storage Descriptions

Irradiated Fuel Storage Area

Irradiated and unirradiated reactor LEU fuel can be stored in an irradiated fuel storage area [redacted] as illustrated in Figures 9-1 and 9-2.

Figure 9-1 Horizontal Projection of the Irradiated Storage Area

Figure 9-2 Vertical Projection of Two Irradiated Storage Locations

Unirradiated Fuel Storage Area

Figure 9-3 XZ and YZ Projections of the Unirradiated Storage Area

9.2.3 Criticality Analyses

In order to verify that the irradiated storage prevent criticality under normal and flooded conditions, it was analyzed for a single case where [redacted] bundle. For the unirradiated storage area, the following fuel arrangements are considered:

The k_{eff}'s for both systems were determined under two conditions of moderation using MCNP5. Table 9-1 summarizes the results of our Monte Carlo calculations.

Table 9-1 k_{eff} for both Systems Under Normal and Flooded Conditions

System	k_{eff}	
	Normal	Flooded
Irradiated Storage Area		
i) [redacted] locations containing fuel	0.13237 (0.00011)[1]	0.36486 (0.00025)
Unirradiated Storage Area		
i) 70 plates Pitch (cm) x=10.0, z=0.0	0.00923 (0.00001)	0.38733 (0.00039)
ii) 70 plates Pitch (cm) x=7.2263, z=0.0	0.01095 (0.00001)	0.43479 (0.00041)
iii) 70 plates/[redacted] Pitch (cm) x=10cm	0.01569 (0.00001)	0.45756 (0.00041)
iv) 98 plates Pitch (cm) x=7.2263, z=0.0	0.01110 (0.000001)	0.45590 (0.00044)
v) 196 plates Pitch (cm) x=7.2263, z=5.72516	0.01800 (0.00001)	0.73454 (0.00049)

[1] 1-σ relative statistical uncertainty

Based on the results presented in Table 9-1, the storage systems of the UFTR facility meet the criticality requirement of $k_{eff} < 0.8$ for all conditions of moderation.

ATTACHMENT IV

CONSISTENCY CHECKS OF CONTROL BLADE REACTIVITY WORTHS

UNIVERSITY OF FLORIDA TRAINING REACTOR
SUPPLEMENTAL INFORMATION
August 4, 2006

In the June 15th responses to the RAI (docket No. 50- 83), an inconsistency between the control blade worths given in Tables 4-1 and 4-13 was noticed. The current supplement addresses this issue by providing the correct set of values for both tables and the resulting modifications to the relevant pages of the submittal. These modifications were made to ensure that this inconsistency does not appear elsewhere in the submittal.

For convenience, the corrected values of Table 4-13 for the control blade integral reactivity worths are given below.

Table 4-13 Comparison of Control Blades Worth for the HEU and LEU Cores

Control Blade	HEU (calculated)	HEU (measured)	LEU-fresh (calculated)	LEU-depleted (calculated)
Regulating	0.87%	0.82%	0.63%	0.66%
Safety 1	1.35%	1.21%	1.62%	1.65%
Safety 2	1.63%	1.36%	1.81% [1]	1.76%
Safety 3	2.03% [1]	1.88%	1.42%	1.46%

[1] The total integral reactivity worth of this blade was evaluated by positioning all the blades at their critical position and then rotating the blade of interest (see following section).

The corrected Table 4-1 is included in Appendix A of this supplement while the modified pages of the submittal are included in Appendix B.

APPENDIX A

Table 4-1 Summary of Key Nominal Design Parameters of HEU (current) and LEU (expected) Cores

	HEU	LEU
DESIGN DATA		
Fuel Type	U-Al alloy	U_3Si_2-Al
Fuel Meat Size		
Width (cm)	5.96	5.96
Thickness (cm)	0.102	0.051
Height (cm)	60.0	60.0
Fuel Plate Size		
Width (cm)	7.23	7.23
Thickness (cm)	0.178	0.127
Height (cm)	65.1	65.1
Cladding	1100 Al	6061 Al
Cladding Thickness (cm)	0.038	0.038
Fuel Enrichment (nominal)	93.0 %	19.75%
"Meat" Composition (wt% U)	14.05	62.98
Mass of ^{235}U per Plate (nominal)	14.5 g	12.5 g
Number of Plates per Fuel Bundle	11	14
Number of Full Fuel Bundles (current/expected)	21	22
Number of Partial Fuel Bundles	1 (5 fuel plates + 5 dummy plates)	1 (10 fuel plates + 4 dummy plates)
Number of Dummy Bundles	2	1
REACTOR PARAMETERS		
Fresh Core Excess Reactivity (% Δk/k)	1.09	0.925
Shutdown Margin (Δk/k)	3.11	3.17
Control blade worth,		
Regulating (% Δk/k)	0.87	0.63
Safety 1 (% Δk/k)	1.35	1.62
Safety 2 (% Δk/k)	1.63	1.81
Safety 3 (% Δk/k)	2.03	1.42
Maximum Reactivity Insertion Rate (% Δk/k/s)	0.042	0.045
Ave. Coolant Void Coefficient, (% Δk/k/%void)		
Fresh Core	-0.148	-0.153
Depleted Core		-0.146
Coolant Temp. Coefficient, (% Δk/k/°C)		
Fresh Core	-5.91E-03	-5.68E-03
Depleted Core		-5.26E-03
Fuel Temp. Coefficient, (% Δk/k/°C)		
Fresh Core	-2.91E-04	-1.65E-03
Depleted Core		-1.49E-03
Effective Delayed Neutron Fraction		
Fresh Core	0.0079	0.0077
Depleted Core		0.00756

Neutron Lifetime (μs)		
Fresh Core	187.4	177.5
Depleted Core		195.1
THERMAL-HYDRAULIC PARAMETERS (100kW, 43 gpm, Tin=30 C)		
Max. Fuel Temperature [1] (°C)	66.5	64.5
Max. Clad Temperature [1] (°C)	66.5	64.4
Mixed Mean Coolant Outlet Temperature (°C)	40.8	40.5
Max. Coolant Channel Outlet Temp., (°C)	58.3	59.1
Minimum ONBR	1.98	2.09
Minimum DNBR	354	376

[1] At nominal operating conditions

APPENDIX B

B.1 Changes to page 28 of the December's Submittal

Table 4-13 Comparison of Control Blades Worth for the HEU and LEU Cores

Control Blade	HEU (calculated)	HEU (measured)	LEU-fresh (calculated)	LEU-depleted (calculated)
Regulating	0.87%	0.82%	0.63%	0.66%
Safety 1	1.35%	1.21%	1.62%	1.65%
Safety 2	1.63%	1.36%	1.81% [1]	1.76%
Safety 3	2.03% [1]	1.88%	1.42%	1.46%

[1] The total integral reactivity worth of this blade was evaluated by positioning all the blades at their critical position and then rotating the blade of interest (see following section).

For the HEU core, the calculated and experimental data differ in a range of 6.1% to 19.9%. These differences can be attributed to experimental uncertainty and inconsistency between the experimental procedure to measure the blade worth and the modeling procedure.

Further, the two control blades on the south part of the reference LEU core (safety 1 and 2) have higher worths as compared to the HEU core, while the two control blades on the north part of the LEU core (safety 3 and regulating) have lower worth than in the HEU core. This finding is expected because of the observed power shift presented in Table 4-10. This power shift is expected since more fuel is added to the south part of the core.

Maximum Reactivity Insertion Rate for HEU Core

In addition to calculations of the total reactivity worth for the UFTR control blades, an analysis of the integral worth as a function of position was performed for the most reactive blade. In the prior calculations, Safety Blade 3 was determined to be the most reactive blade. An MCNP model of the UFTR fueled with 21.5 HEU fuel bundles was utilized. The calculations were performed by positioning the Safety 1, Safety 2, and Regulating Blades at a critical position for the core, and then moving Safety Blade 3 through its full range of motion (2.5° to 47.5°). Results are provided in Table 4-14 and Figure 4-16. The total blade worth calculated here is 2.03% Δk/k, which is almost the same as the prior calculation for the total blade worth (2.06 % Δk/k). In the prior calculations, the other blades were fully-withdrawn, while in the calculations presented in Table 4-14, the safety blades were inserted at 38.5° and the regulating blade was at 18.7°.

Table 4-14 Integral Reactivity Worth versus Position for Safety 3 in the HEU Core

Time (s)[1]	Blade Position		k_{eff}	Reactivity (%Δk/k)	Reactivity Insertion Rate (%Δk/k/s)
	Degrees	Units			
0.0	2.5	0	0.98747 ± 0.04%	0.00%	n/a
5.6	5	56	0.98936 ± 0.03%	0.19%	0.034%
16.7	10	167	0.99330 ± 0.04%	0.59%	0.036%
27.8	15	278	0.99789 ± 0.04%	1.06%	0.042%

38.9	20	389	1.00158 ± 0.02%	1.43%	0.034%
50.0	25	500	1.00423 ± 0.02%	1.70%	0.024%
61.1	30	611	1.00576 ± 0.02%	1.85%	0.014%
72.2	35	722	1.00664 ± 0.02%	1.94%	0.008%
83.3	40	833	1.00704 ± 0.02%	1.98%	0.004%
100.0	47.5	1000	1.00747 ± 0.02%	2.03%	0.003%

[1]Assumes 100 seconds withdrawal time

B.2 Changes to page 29 of the December's Submittal

Figure 4-16 shows calculated the Safety Blade 3 (most reactive blade for HEU core) worth as a function of position.

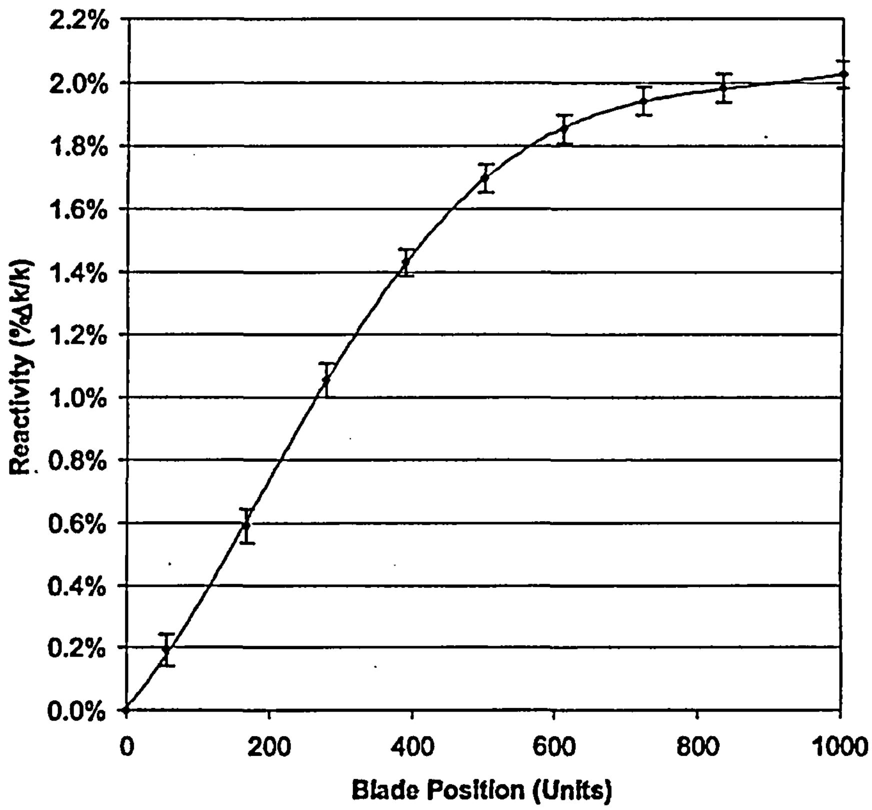

Figure 4-16 Integral Blade Worth versus Position for Safety 3 in the UFTR HEU Core.

The UFTR Technical Specifications require that the reactivity insertion rate from control blade withdrawal must be less than 0.06% Δk/k/s when averaged over a 10 second interval. The rate of reactivity insertion resulting from withdrawal of the highest worth blade was approximated by assuming a 100 second (minimum allowed) blade withdrawal time. As shown in Table 4-14, the highest rate of reactivity insertion from withdrawal of Safety Blade 3 is 0.042% Δk/k/s, which meets the requisite UFTR Technical Specification.

Maximum Reactivity Insertion Rate for LEU Core

The integral reactivity worth as a function of position was determined for Safety Blade 2 (most reactive blade for the LEU core) based on MCNP calculations in a manner similar to that employed for the HEU core calculations. The position of Safety Blade 1 and Safety Blade 3 was fixed at 26.3° and the Regulating Blade was positioned at 16.9°, while Safety Blade 2 was rotated from 2.5° to 47.5°. Results are provided in Table 4-15 and

Figure 4-17. The total worth for Safety Blade 2 calculated in this manner is similar to that obtained in the prior calculations with the other blades fully-withdrawn (1.77%Δk/k).