Modification of the RINSC LEU Core to Increase Fluxes for BNCT Study

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1. Introduction

The Rhode Island Nuclear Science Center (RINSC) is owned by the State of Rhode Island and was constructed for education, research, and industrial applications. The RINSC research reactor achieved initial criticality in 1964 as a 1 MW reactor using fuel containing high enriched uranium (HEU). In 1968 the facility was upgraded to a power level of 2 MW, making it one of the highest flux university reactors in the United States. In 1993, RINSC converted the HEU core to a compact core using low enriched uranium (LEU) silicide fuel[1]. The combination of a compact core design and a higher LEU fuel density improved the neutron flux performance of the facility.

The Rhode Island Nuclear Science Center is presently working towards building an advanced Boron Neutron Capture Therapy (BNCT) facility for cancer treatment and research. As part of this project, consideration has been given to a core modification that involves the replacement of three graphite reflectors on the thermal column side of the core with new LEU fuel elements. The modified core has the same operating power of 2 MW. Neutronic and thermal hydraulic calculations were performed to determine how the neutron flux, control rod reactivity worths, and fuel and cladding temperatures would change as a consequence of this modification. This paper presents results of this study.

2. Computational Methods

The core configuration currently in use at RINSC is LEU Core 2, shown in Figure 1. Neutronic analyses were performed using diffusion code DIF3D[2] and continuous energy Monte Carlo code MCNP[3]. Seven group neutron cross sections for the diffusion calculations were generated by WIMS-ANL[4] using ENDF/B-VI data.

The present core is approaching the end of its core life (EOC) of 280 full power days. The core composition at EOC was calculated using the REBUS[5] code with a full core model and four axial zones in each fuel assembly. Diffusion calculations were performed to compute the excess reactivities and power distributions in the current and expanded cores. A detailed reactor model was used in the Monte Carlo calculations to verify results of the diffusion calculations and to calculate the worths of the control rods.

For the thermal analyses, the RINSC Startup Core was modeled along with a modified startup core that incorporates the proposed configuration changes. Fresh fuel was assumed for both cases, in order to provide the most conservative data. Coolant flow rate through the core was determined using the PLTEMP code. The power peaking factors were obtained from diffusion theory calculations with the control rods at 50% withdrawn and full out. A single plate model for the hot channel was run to determine the peak fuel, cladding and coolant temperatures.

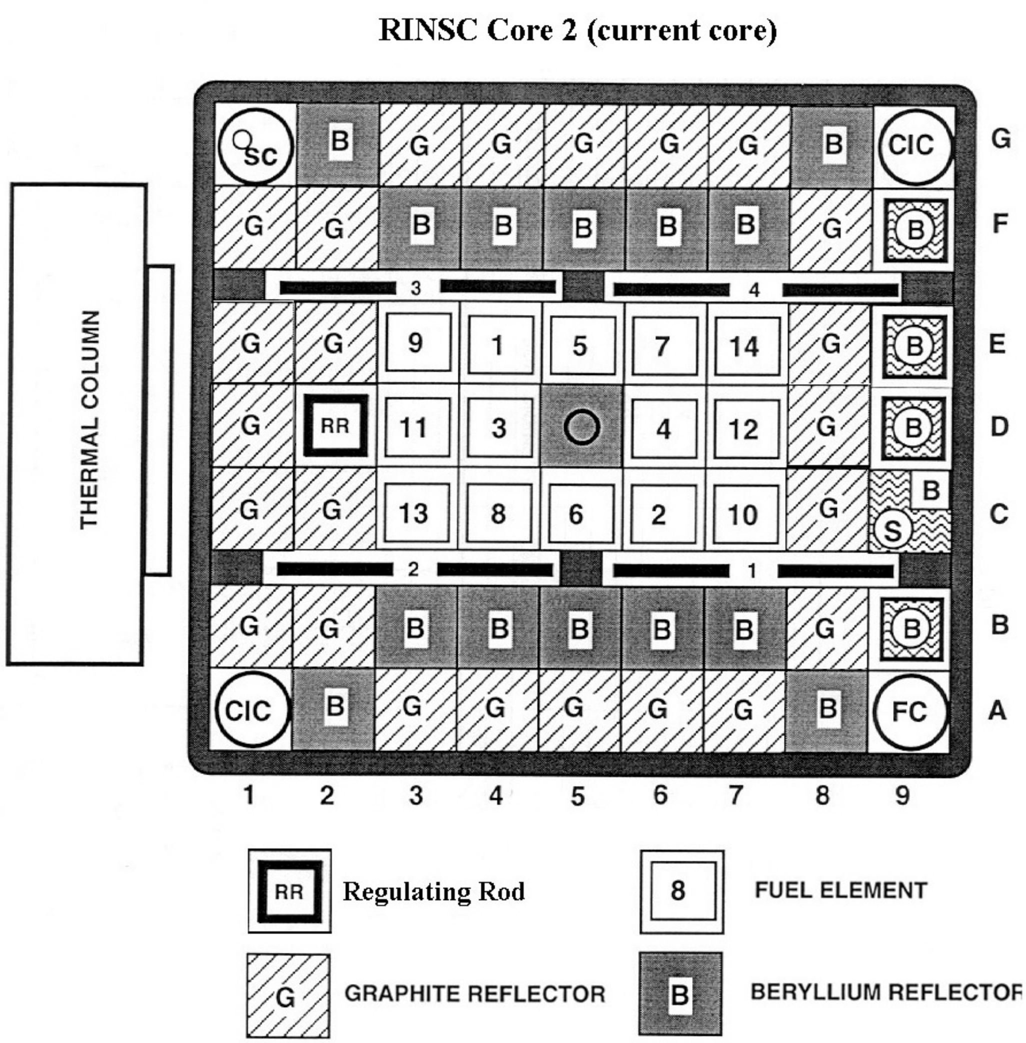


Figure 1. RINSC Core 2 (current core)

3. Modification of RINSC Reactor Core to Increase Epithermal Neutrons for BNCT Study

The main objective in the modification of the RINSC core is to find a technically and economically feasible model that, without changing the operating power, would increase the leakage of fast neutrons from the core to the thermal column for BNCT research and treatment. Several options were considered, including removing graphite reflectors near the thermal column and moving the fuel matrix toward the thermal column. However, many of these options were found to be technically infeasible and overly expensive. A suggestion by Dr. Nelson Hanan from the ANL RERTR group eventually lead to an expanded core design, in which three graphite reflector elements near the thermal column would be replaced by three fresh LEU fuel elements (see Figure 2). Such an arrangement leads to a significant increase in neutron leakage into the BNCT facility located behind the thermal column (see figure 3).

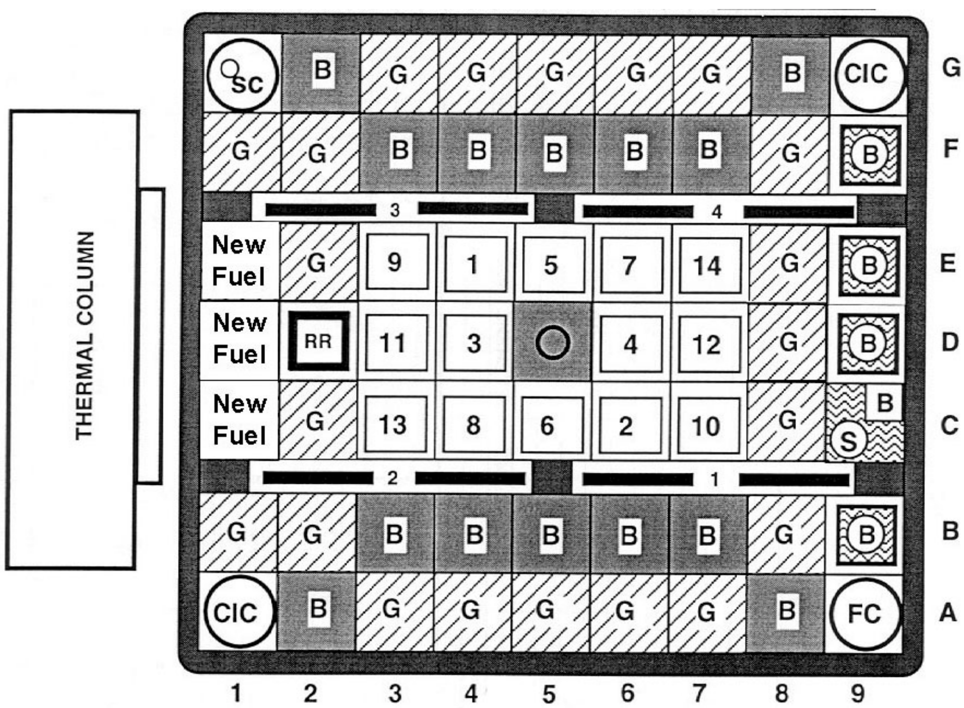


Figure 2. Expanded Core with Additional Fuel Elements to Increase Fast Leakage.

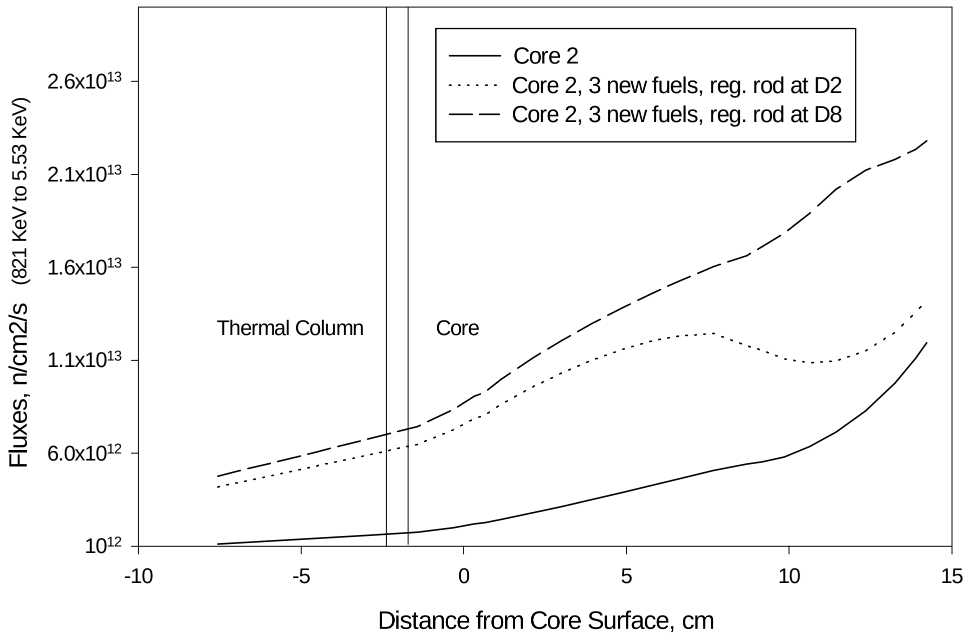


Figure 3. Neutron Leakage (821 KeV to 5.53 KeV) near the Core-Thermal Column Interface

Coupled with this expanded core design, consideration was also given to moving the stainless steel regulating rod from its present grid position at D2 to the opposite side of the core at grid position D8 to further skew the flux toward the thermal column. Figure 3 shows the intermediate-energy neutron fluxes (821 KeV to 5.53 KeV) predicted for each of these configurations. Unfortunately, in order to move the regulating rod to position D8, a new rod would have to be constructed using a higher neutron absorbing material such as borated stainless steel. Also, the support structures that would be required for the new regulating rod location, would interfere with fuel handling operations. Consequently, it was decided that the additional neutron flux that would be obtained by moving the regulating rod would not be worth the expense of moving it.

4. Regulating Rod Worth in the Expanded Core

The addition of three fuel elements near the stainless steel regulating rod in position D2 would cause an increase in the rod worth. The safety specifications require the worth of the regulating rod to be less than β (the prompt neutron fraction). Calculations were performed to predict the worth of the regulating rod in the expanded core. Results of these calculations are summarized in Table 2. The calculated worth of the regulating rod in the expanded core is estimated to be about 0.5% $\delta k/k$. Determination of the actual effect that the expanded core configuration would have on the regulating rod worth can only be made by making the core change, and performing a control rod calibration. If the calibration were to show that the rod had a measured worth close to the safety limit, the reflector elements could be rearranged to reduce the worth of the rod.

Table 1 Regulating Rod Worth in the Expanded Core

	DIF3D $\delta k/k$	MCNP $\delta k/k$
Core 2 EOC		
Equilibrium Xe	0.287%	0.198 ± 0.071%
No Xe	0.281%	
Expanded Core with 3 new fuel elements		
No Xe	0.586%	0.508± 0.056%

5. Reactor Excess Reactivities and Shutdown Margin

Table 2 shows total excess reactivity in terms of K_{eff}. The initial estimates were made with diffusion theory calculations, which are in good agreement with results of more detailed Monte Carlo calculations. Both computational methods suggest that the expanded core would have approximately 1.6% δk/k greater excess reactivity at equilibrium Xe, than the present core under the same conditions, and that the expanded core would have a total excess reactivity of 4% δk/k with no Xe poisoning in the core.

Table 2 K_{eff} obtained from DIF3D and MCNP Calculations

	DIF3D	MCNP
Core 2, EOC, equilibrium Xe	1.00028	0.99312 ± 0.00049
Expanded Core, 3 New Fuel, equilibrium Xe	1.01689	1.00915 ± 0.00052
Expanded Core, 3 New Fuel, no Xe	1.04402	1.04250 ± 0.00054

Table 3 shows the shutdown margin calculation for the expanded core. The calculations took into account the depletion of the fuel in core 2 and assumed that the core was at room temperature, with no Xe poisoning, and that an experiment worth +0.6% δk/k was present at the core. The most reactive shim safety rod and regulating rod were assumed to be fully out of the core. The estimated margin of -5% δk/k is far more conservative than the -1% δk/k technical specification limit.

Table 3. Reactor Shutdown Margin in the New Expanded Core

	Reactivity Worth
	δk/k
Core excess reactivity (cold/no Xe)	4.1%
Worth of control rods with the most reactive rod (CR2) and regulating rod fully out	-9.8%
Worth of experiments	0.6%
Shutdown Margin =	-5.1%

6. Thermal Analyses

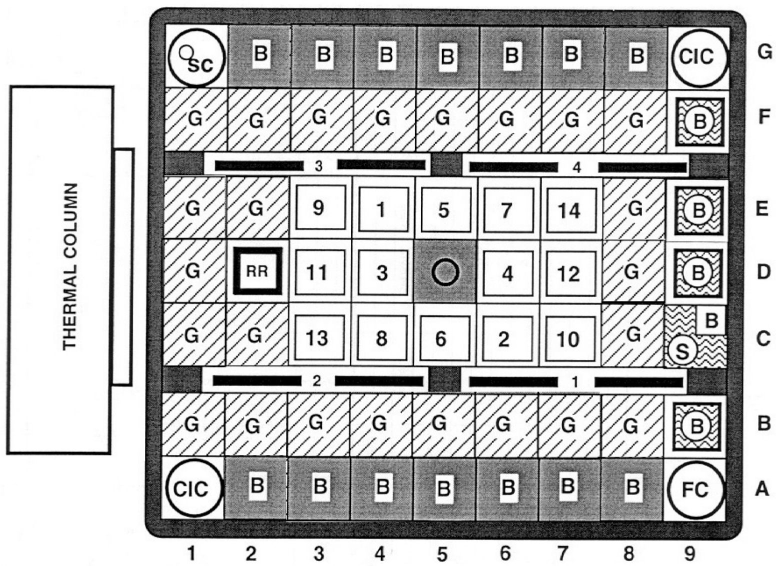
Two fresh core models were chosen for the thermal hydraulic calculations that would produce highest possible fuel and cladding temperatures under normal reactor operation at 2 MW reactor power. The upper diagram in Figure 4 represents the original RINSC startup core, while the lower diagram represents an expanded startup core with the three additional fuel elements on the thermal column side.

A full core model of the RINSC reactor was run using the PLTEMP[6] code in an iterative manner to determine the pressure drop that would yield the flow rate corresponding to one pump operation. A single plate model for the hot channel was then run to determine the peak fuel, clad and coolant temperatures. The coolant inlet temperature was assumed to be 46.1 °C (115 °F) because it is the design inlet temperature. The hot channel factors, bypass flow geometry, and axial power distribution remain unchanged from current operating conditions in the 14 element core.

Figure 5 shows the power peaking factors obtained from the DIF3D diffusion theory calculations with the control rods 50% withdrawn. From the point of view of thermal-hydraulic safety margins, the most important neutronic parameter is the total 3D power peaking factor (the absolute peak power density in a fuel assembly divided by the average power density in the core). The total power peaking factor is defined here as the product of three components: (1) a radial factor defined as the average power density in each assembly divided by the average power density in the core, (2) a planar factor defined as the peak power density in the limiting plane in each fuel assembly divided by the average power density in that plane, and (3) the axial peak to average power density in the assembly defined as the maximum ratio of the peak to average power density in the plate. The planar and axial factors are point wise factors computed at the mesh interval edge and includes both planar and axial power peaking.

Tables 4 and 5 show the results of the thermal analysis for both core configurations, with control rods withdrawn 50% and fully out. There is a substantial margin to boiling and flow instability in all cases.

RINSC startup core



Startup core with 3 new fuels

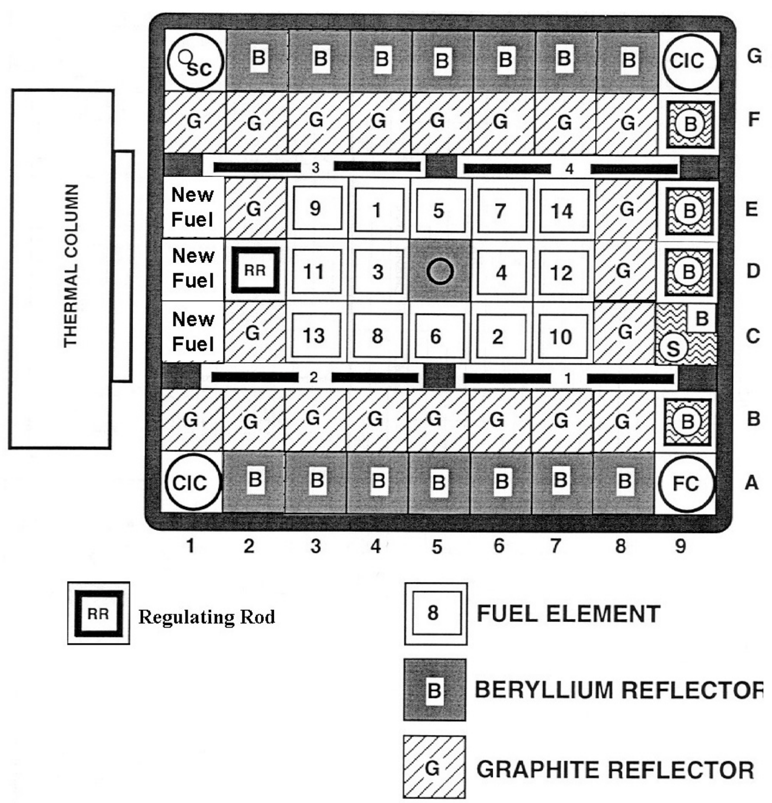


Figure 4. Fresh Core Models in the RINSC Thermal Hydraulic Calculations

E	D	C	
G	G	G	8
0.807 1,768 1.536 2.192 5.8%	0.930 1,169 1.536 1.670 6.6%	0.807 1.768 1.536 2.192 5.8%	7
1.011 1.630 1.536 2.531 7.2%	1.261 1.563 1.536 3.026 9.0%	1.011 1.630 1.536 2.531 7.2%	6
1.151 1.654 1.536 2.925 8.2%	Be	1.151 1.654 1.536 2.925 8.2%	5
1.009 1.632 1.536 2.530 7.2%	1.258 1.563 1.536 3.019 9.0%	1.009 1.632 1,536 2.530 7.2%	4
0.818 1.749 1.536 2.197 5.8%	0.961 1.312 1.536 1.936 6.9%	0.818 1.749 1.536 2.197 5.8%	3
G	Reg. Rod	G	2
G	G	G	1

Startup Core

E	D	C	
G	G	G	8
0.823 1.766 1.536 2.231 4.7%	0.944 1.191 1.536 1.727 5.4%	0.823 1.766 1.536 2.231 4.7%	7
1.052 1.631 1.536 2.635 6.1%	1.317 1.572 1.536 3.179 7.6%	1.052 1.631 1.536 2.635 6.1%	6
1.239 1.640 1.536 3.122 7.1%	Be	1.239 1.640 1.536 3.122 7.1%	5
1.134 1.591 1.536 2.770 6.5%	1.413 1.500 1.536 3.256 8.1%	1.134 1.591 1.536 2.770 6.5%	4
0.999 1.654 1.536 2.538 5.8%	1.180 1.410 1.536 2.556 6.8%	0.999 1.654 1.536 2.538 5.8%	3
G	Reg. Rod	G	2
0.526 2.228 1.536 1.800 3.0%	0.602 2.073 1.536 1.917 3.5%	0.526 2.228 1.536 1.800 3.0%	1

Startup Core with 3 more Fuel Elements

Radial peaking factor
Planar peaking factor
Axial peaking factor
Total peaking factor
% of total power in element

Figure 5 Power Peaking Factors in the Thermal Hydraulic Calculations

Table 4 Power Peaking Factors and Thermal Hydraulic Parameters in Hot Channel Calculations

No. of Elements in Core	Control Rod Position	Peaking Factor				Pressure Drop		Coolant Flow Rate
		Radial	Planar	Axial	Total	MPa	psi	GPM
14	50% Out	1.258	1.562	1.536	3.02	0.00677	0.98	1870
14	Out	1.178	1.695	1.32	2.64	0.00677	0.98	1870
17	50% Out	1.424	1.493	1.536	3.27	0.00677	0.98	2195[1]
17	Out	1.330	1.621	1.32	2.85	0.00677	0.98	2195[1]

[1]Estimated for 17 element core.
F_b (hot channel factor for bulk water temperature rise) = 1.25 .
F_q (hot channel factor for heat flux) = 1.26 .
F_h (hot channel factor for heat transfer coefficient) = 1.33 .

Table 5 Safety Margins and Peak Temperatures in Hot Channels

No. of Elements in Core	Control Rod Position	Margin to Onset of Nucleate Boiling	Margin to Flow Instability	Margin to Departure from Nucleate Boiling	Maximum Temperatures - °C / °F		
					Fuel	Clad	Coolant
14	50% Out	1.89	4.50	6.96	89.5 / 193	88.4 / 191	52.8 / 127
14	Out	2.14	5.14	7.96	84.8 / 185	83.8 / 183	53.1 / 128
17	50% Out	2.13	5.03	7.80	84.9 / 185	84.0 / 183	52.1 / 126
17	Out	2. 39	5.78	8.95	80.6 / 177	79.8 / 176	52.3 / 126

7. Conclusions

Expanding the RINSC core configuration to include three fuel elements in place of graphite reflector elements along the thermal column side of the core is expected to increase fast neutron leakage into the thermal column by over 300%. The worth of the current regulating rod would increase from 0.2% δk/k to about 0.5% δk/k. The estimated core excess reactivity for this configuration is about 4 % without Xe poisoning. The predicted shutdown margin is about 5 %δk/k subcritical with the assumption that the core is loaded with a maximum positive worth experiment. All of these reactivities are within the limits imposed by the facility license.

Replacing three graphite reflectors with fuel elements would lead to lower average fuel, cladding, and coolant temperatures, because the reactor power would be spread over more fuel elements, and because there would be an increase in the total coolant flow through the core due to the increase in the number of flow channels available. Results of the thermal analysis show there is substantial margin to boiling and flow instability in all cases.

References

1. A. F. DiMeglio, J. E. Matos, K. E. Freese and E. F. Spring, “The Conversion of the 2MW Reactor at the Rhode Island Nuclear Science Center”, 12th International RERTR Meeting, September 1989, Berlin, Germany.

2. K. L. Drestine, “DIF3D: A Code to solve One-, Two- and Three Dimensional Finite-Difference Diffusion Theory Problems,” ANL-82-64, April 1984.

3. J. F. Briesmeister, "MCNP-A General Monte Carlo N-Particle Transport Code," LA-12625-M, Los Alamos National Laboratory, Nov. 1993.

4. J. R. Deen, W. L. Woodruff and C. I. Costescu, "WIMS-D4M User Manual Rev. 0", ANL/RERTR/TM-23, Argonne National Laboratory, July 1995.

5. B. J. Toppel, "A User's Guide for the REBUS-3 Fuel Cycle Analysis Capability," ANL-83-2, March 1983.

6. K. Mishima, K. Kanda and T. Shibata, “Thermal-hydraulic Analysis for Core Conversion to Use of Low Enrichment-uranium Fuels in KUR,” KURRI-TR-258 (1984).