ANALYSES FOR CONVERSION OF THE GEORGIA TECH RESEARCH REACTOR FROM HEU TO LEU FUEL

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SUMMARY

This report contains the results of design and safety analyses performed by the RERTR Program at the Argonne National Laboratory (ANL) for conversion of the Georgia Tech Research Reactor (GTRR) from the use of HEU fuel to the use of LEU fuel. The objectives of this study were to: (1) maintain or improve upon the present reactor performance and margins of safety, (2) maintain as closely as possible the technical specifications and operating procedures of the present HEU core, and (3) utilize a proven fuel assembly design that is economical to manufacture. Extensive collaboration with Dr. R. Karam, Director of the Neely Nuclear Research Center at Georgia Tech, took place on all aspects of this work.

The LEU fuel assembly has the same overall design as the present HEU fuel assembly, except that it contains 18 fueled plates with LEU U_3Si_2-Al fuel instead of 16 fueled plates with HEU U-Al alloy fuel. This LEU silicide fuel has been approved by the Nuclear Regulatory Commission for use in non-power reactors.

Documents that were reviewed by ANL as bases for the design and safety evaluations were the GTRR Safety Analysis Reports, the GTRR Technical Specifications, and responses by the reactor organization to AEC questions in licensing the reactor for 5 MW operation.

The methods and codes that were utilized have been qualified using comparisons of calculations and measurements of LEU demonstration cores in the Ford Nuclear Reactor at the University of Michigan and in the Oak Ridge Research Reactor at the Oak Ridge National Laboratory. Additional qualification has been obtained via international benchmark comparisons sponsored by the IAEA for heavy water research reactors.

Only those reactor parameters and safety analyses which could change as a result of replacing the HEU fuel in the core with LEU fuel are addressed. The attached summary table provides a comparison of the key design features of the HEU and LEU fuel assemblies and a comparison of the key reactor and safety parameters that were calculated for each core. The results show that all of the objectives of this study were fully realized and that the GTRR reactor facility can be operated as safely with the new LEU fuel assemblies as with the present HEU fuel assemblies.

SUMMARY TABLE

HEU and LEU Design Data, Core Physics, and Safety Parameters for Conversion of the Georgia Tech Research Reactor

DESIGN DATA	HEU Core	LEU Core
Minimum Number of Fuel Assemblies	14	14
Maximum Number of Fuel Assemblies	19	19
Fuel Type	U-Al Alloy	U_3Si_2-Al
Enrichment, %	93	19.75
Uranium Density, g/cm^3	0.65	3.5
Number of Fueled Plates per Assembly	16	18
Number of Non-Fueled Plates per Assembly	2	2
^{235}U per Fuel Plate, g	11.75	12.5
^{235}U per Fuel Assembly, g	188	225
Fuel Meat Thickness, mm	0.51	0.51
Cladding Thickness, mm	0.38	0.38
Cladding Material	1100 Al	6061 Al

REACTOR PARAMETERS	HEU Core	LEU Core	Number of Assemblies
Cold Clean Excess Reactivity, % Δk/k	11.7 ± 0.4	9.4 ± 0.4	17
Coolant Temperature Coefficient, % Δk/k/°C	- 0.0076	- 0.0067	14
Doppler Coefficient, % Δk/k/°C	~ 0.0	- 0.0017	14
Whole Reactor Isothermal Temp. Coeff., % Δk/k/°C	- 0.0224	- 0.0232	14
Coolant Void Coefficient, % Δk/k/% Void	- 0.0383	- 0.0333	14
Limiting Power Peaking Factor	1.54	1.58	14
Prompt Neutron Lifetime, μs	780	745	14
Effective Delayed Neutron Fraction	0.00755	0.0075-0.0076	14
Shutdown Margin, % Δk/k (Max. Worth Shim Blade and Reg. Rod Stuck Out)	- 7.1 ± 0.2	- 8.8 ± 0.2	17
Top D_2O Reflector Worth, % Δk/k (For D_2O 2” Above Fuel Meat)	- 2.1 ± 0.3	- 2.4 ±0.3	17
Reactor Power Limits -1625 gpm Flow Rate			
Based on Departure from Nucleate Boiling, MW	11.5	10.8	14
Based on Flow Instability Criterion, MW	10.6	10.6	14
Limiting Reactor Inlet Temperature, °F	172	170	14
Limiting Reactor Outlet Temperature, °F	188	187	14
Limiting Safety System Settings - Forced Convection			
Reactor Power, MW	5.5	5.6	14
Coolant Flow Rate, gpm	1625	< 1625	14
Reactor Outlet Temperature, °F	139	145	14
Margin to D_2O Saturation Temperature, °F	8	11	14
Max. Fuel Plate Temp. for LOCA after 8 Hours Cooling, °C	425	400	14
Maximum Positive Reactivity Insertion, % Δk/k	> 2.2	> 2.2	14

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1. INTRODUCTION

This report contains the results of design and safety analyses performed by the RERTR Program at the Argonne National Laboratory (ANL) for conversion of the Georgia Tech Research Reactor (GTRR) from the use of HEU fuel to the use of LEU fuel. The objectives of this study were to: (1) maintain or improve upon the present reactor performance and margins of safety, (2) maintain as closely as possible the technical specifications and operating procedures of the present HEU core, and (3) utilize a proven fuel assembly design that is economical to manufacture.

The design and safety analyses in this report provide comparisons of reactor parameters and safety margins for the GTRR HEU and LEU cores. Only those parameters which could change as a result of replacing the HEU fuel in the core with LEU fuel are addressed. Documents that were reviewed by ANL as bases for the design and safety evaluations were the GTRR Safety Analysis Reports,[1] the GTRR Technical Specifications[2], and responses[3,4] by the reactor organization to AEC questions in licensing the reactor for 5 MW operation.

The LEU fuel assembly has the same overall design as the present HEU fuel assembly, except that it contains 18 fueled plates with LEU U_3Si_2-Al fuel and two non-fueled plates instead of 16 fueled plates with HEU U-Al alloy fuel and 2 non-fueled plates. A detailed safety evaluation of LEU U_3Si_2-Al fuel can be found in Reference 5.

The methods and codes that were utilized by ANL have been qualified using comparisons of calculations and measurements of LEU demonstration cores[6-10] in the Ford Nuclear Reactor at the University of Michigan and in the Oak Ridge Research Reactor (ORR) at the Oak Ridge National Laboratory. Additional qualification has been obtained via international benchmark comparisons[11,12] sponsored by the IAEA.

2. REACTOR DESCRIPTION

The GTRR is a heterogeneous, heavy-water moderated and cooled, tank-type reactor fueled with 93% enriched MTR-type U-Al alloy fuel. Horizontal and vertical sections through the reactor are shown in Figs. 1 and 2, respectively. Provision is made for up to 19 fuel assemblies spaced 6 inches apart in a triangular array. The current core consists of 17 fuel assemblies. Each assembly consists of 16 fueled and two non-fueled plates with a fissile loading of about 188 g ^{235}U. The total fissile loading of a fresh 17 assembly core would be about 3.2 kg ^{235}U.

The fuel is centrally located in a six foot diameter aluminum reactor vessel which provides a two foot thick D_2O reflector completely surrounding the core. The reactor vessel is mounted on a steel support structure and is suspended within a thick-walled graphite cup. The graphite provides an additional two feet of reflector both radially and beneath the vessel. The core and reflector system is completely enclosed by the lead and concrete biological shield.

The reactor is controlled by means of four cadmium shim-safety blades and one cadmium regulating rod. The four shim-safety blades are mounted at the top of the reactor vessel and swing downward through the core between adjacent rows of fuel assemblies. The regulating rod is supported on the reactor top shield and extends downward into the radial D_2O reflector region. This rod moves vertically between the horizontal midplane and the top of the core.

The heat removal system is composed of a primary heavy-water system and a secondary light-water system. The heavy-water system includes the reactor vessel, the primary D_2O coolant pumps, the D_2O makeup pump, the heat exchangers, and the associated valves and piping. The light-water secondary system is composed of the circulating water pumps, the cooling tower, and associated valves and piping.

The LEU reference core used in this analysis consists of 17 fuel assemblies with the same arrangement as the present HEU core. Each fuel assembly contains 18 fueled plates with 225 g ^{235}U when fresh. The LEU core will use the same control system, heat removal system, and auxiliary systems as the current HEU core.

Fig. 1. Horizontal Section of GTRR at the Core Midplane.

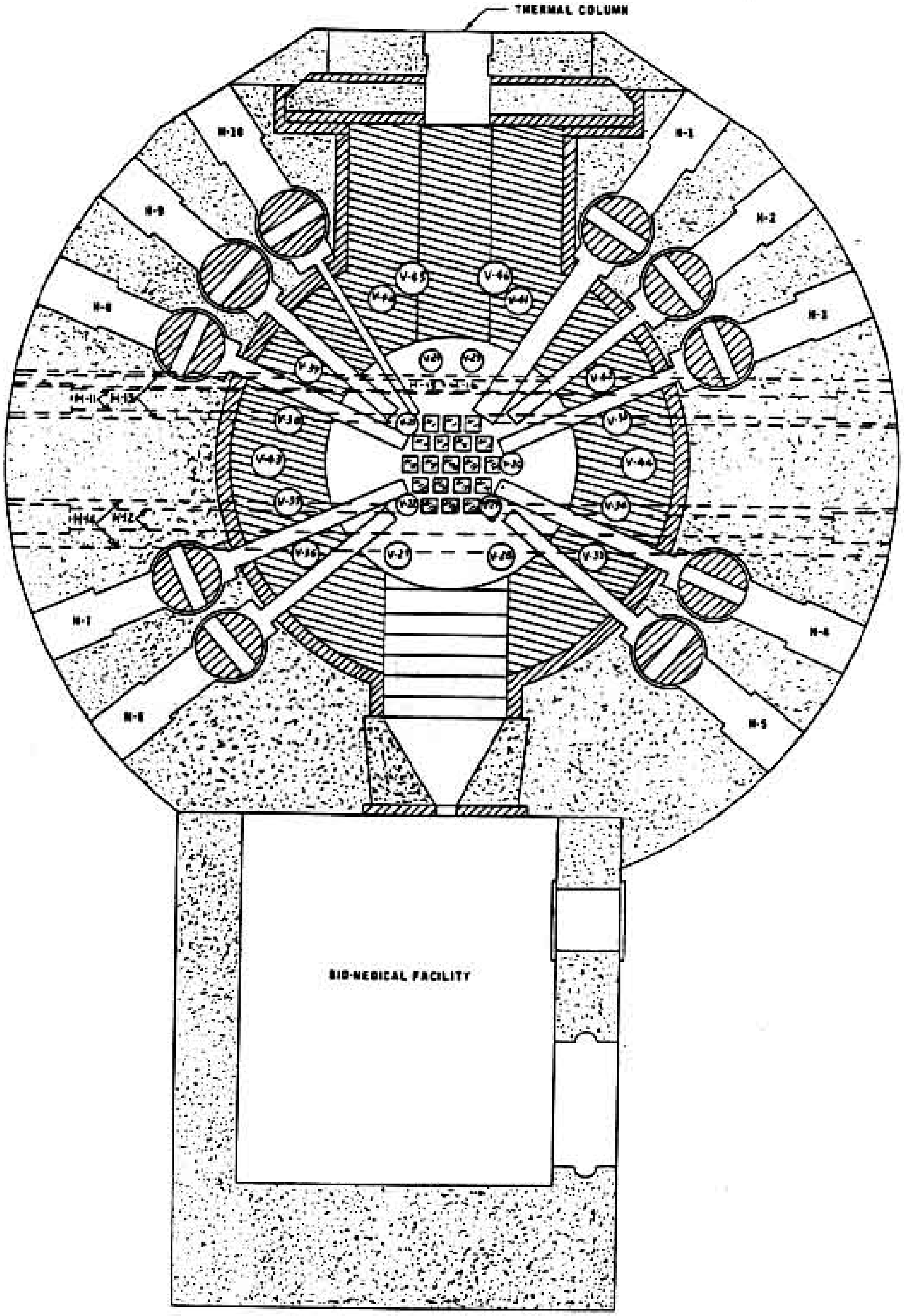
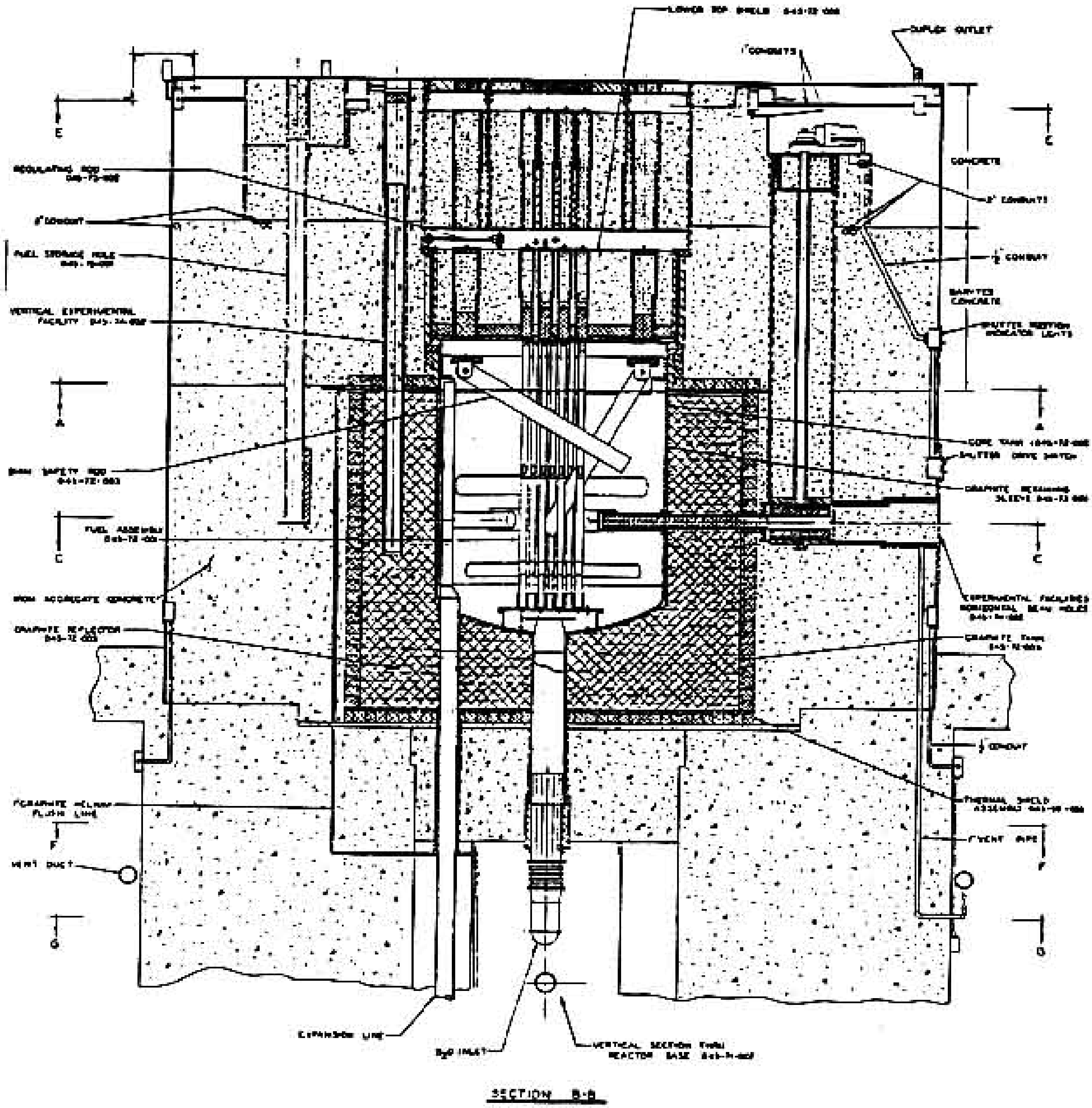


Fig. 2. Vertical Section Through Reactor.



3. FUEL ASSEMBLY DESCRIPTIONS

The geometries, materials and fissile loadings of the current HEU fuel assemblies and the replacement LEU fuel assemblies are described in Table 1. A schematic diagram of the HEU fuel assembly is shown in Fig. 3. The LEU fuel plate is the standard DOE plate containing U_3Si_2-Al fuel with ~3.5 g U/cm^3 and 12.5 g ^{235}U. The external dimensions and structural materials of both assemblies are identical, except that the LEU assemblies utilize 6061 Al instead of 1100 Al.

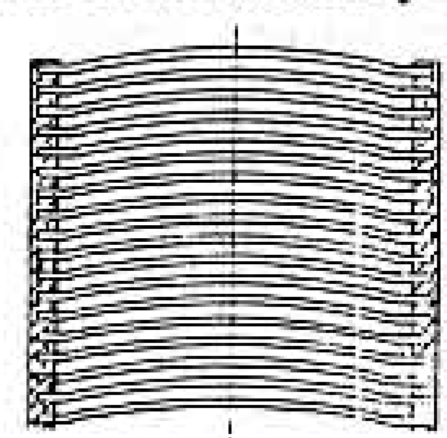
Table 1. Descriptions of the HEU and LEU Fuel Assemblies

		HEU	LEU
Number of Fueled Plates/Assembly		16	18
Number of Non-Fueled Plates/Assembly		2	2
Fissile Loading/Plate, g ^{235}U		11.75	12.5
Fissile Loading/Assembly, g ^{235}U		188	225
Fuel Meat Composition		U-Al Alloy	U_3Si_2-Al
Cladding Material		1100 Al[1]	6061 Al[2]
Fuel Meat Dimensions			
Thickness,	mm	0.510.51	
Width, mm	63.5	58.9 - 62.8	
Length, mm	584 - 610	572 - 610	
Cladding Thickness, mm		0.380.38	

[1] 10 ppm natural boron was added to the composition of the cladding and all fuel assembly structural materials to represent the alloying materials, boron impurity, and other impurities in the 1100 Al of the HEU assemblies.

[2] 20 ppm natural boron was added to the composition of the cladding and structural materials of the LEU assemblies to represent the alloying materials, boron impurity, and other impurities in 6061 Al. Aluminum with no boron or other impurities was used in the fuel meat of both the HEU and LEU assemblies.

Fig. 3. HEU Fuel Assembly Schematic



4. CALCULATIONAL MODELS

4.1 Nuclear Cross Sections for Diffusion Theory Models

Microscopic cross sections in seven energy groups (Table 2) were prepared at 23°C using the EPRI-CELL code[13] for the HEU and LEU fuel assembly geometries and fissile loadings. The integral transport calculations in EPRI-CELL were performed for 69 fast groups and 35 thermal groups (<1.855 eV), which were then collapsed to seven broad energy groups for use in diffusion theory calculations.

Table 2. Seven Group Energy Group Boundaries

Group No.	Upper Energy	Lower Energy	Group No.	Upper Energy	Lower Energy
1	10.0 MeV	0.821 MeV	5	0.625 eV	0.251 eV
2	0.821 MeV	5.531 keV	6	0.251 eV	0.057 eV
3	5.531 keV	1.855 eV	7	0.057 eV	2.53×10^{-4} eV
4	1.855 eV	0.625 eV			

Figure 4 shows the dimensions of the HEU and LEU fuel assemblies and the fuel assembly models that were used in the diffusion theory calculations for the reactor. The fueled and non-fueled regions were modeled separately. A non-fueled region consists of a sideplate and the fuel plate aluminum (plus associated water) between the fuel meat and the sideplate.

Figure 5 shows the unit cell geometry and dimensions that were used in EPRI-CELL to generate microscopic cross sections for the fueled and non-fueled regions of the HEU and LEU assemblies. The non-fueled region inside the assembly is represented by the “extra region 1” containing calculated volume fractions of aluminum and heavy water associated with each fuel plate. “Extra region 2” was modeled to represent the heavy water outside the assembly that is associated with each fuel plate. Its thickness was chosen to preserve the water volume fraction in the physical unit cell of each fuel assembly. All cell calculations were done using a fixed buckling of 0.00373 cm^{-2}, which corresponds with the anticipated axial extrapolation length of about 21 cm in each fuel assembly in the reactor diffusion theory calculations.

Each EPRI-CELL case was run three times using the local fine-group spectra over the fueled region and the two extra regions to collapse the fine group cross sections into 7 broad groups. This procedure was performed because the fueled region, the non-fueled region inside the fuel assembly and the water outside each fuel assembly were modeled as separate regions in the diffusion theory model of the reactor. Cross sections for the heavy water and graphite reflectors and for the fuel assembly end fittings were calculated using a unit cell model consisting of a pure ^{235}U fission spectrum on a 10 cm thick slab of water.

Fig. 4 Models for HEU and LEU Fuel Elements
(Dimensions in mm)

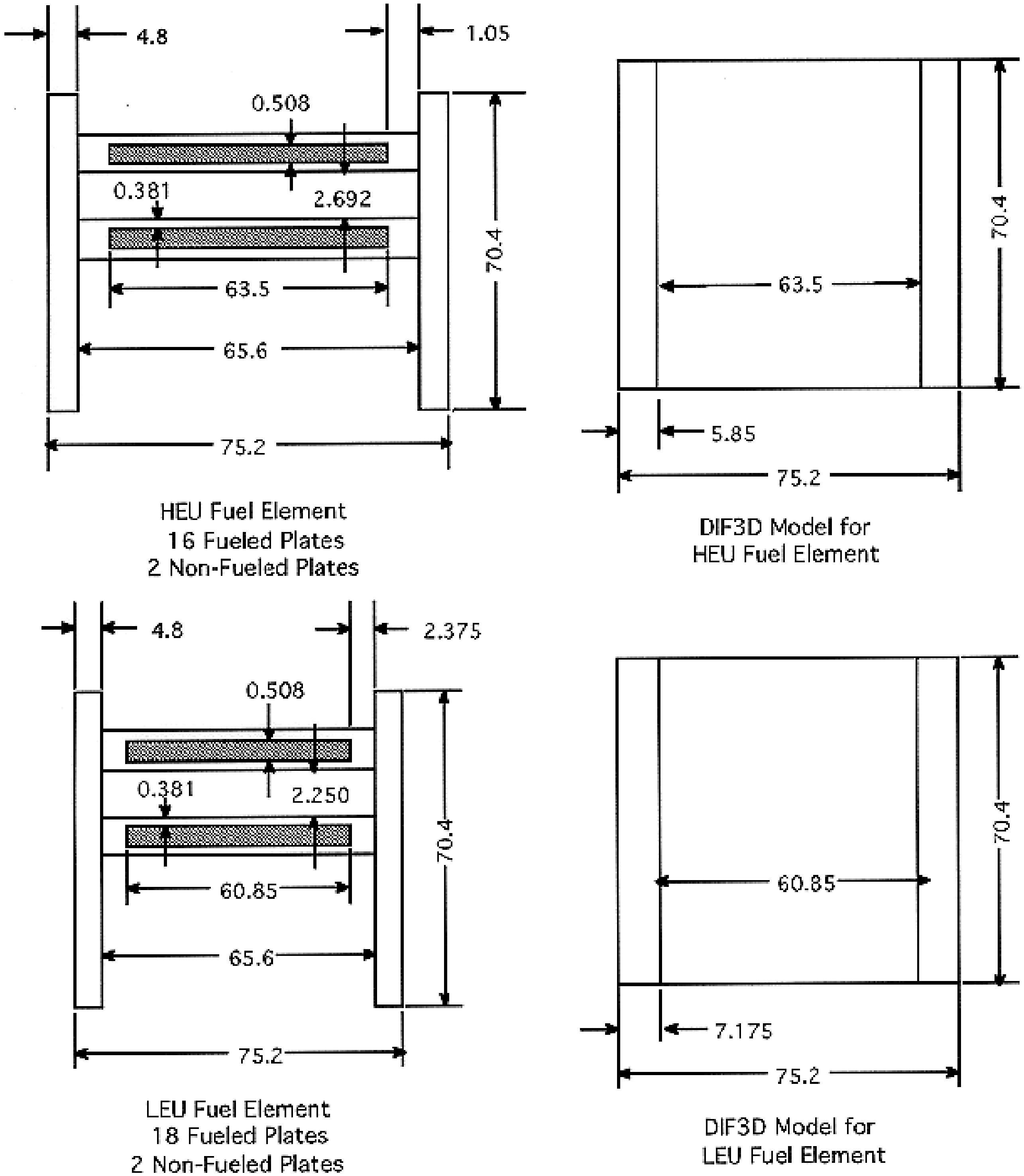
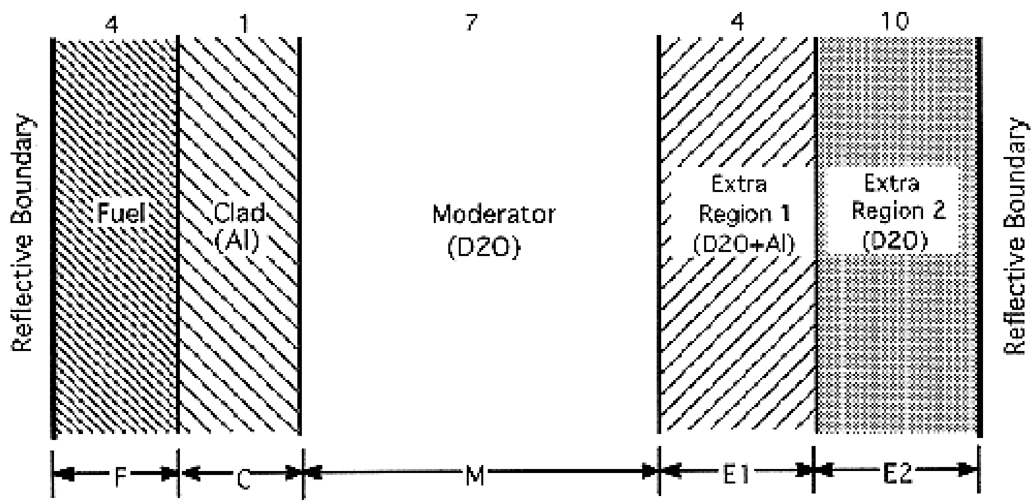


Fig. 5. EPRI-CELL Model for Generating Fuel Element Cross Sections (Dimensions in mm)

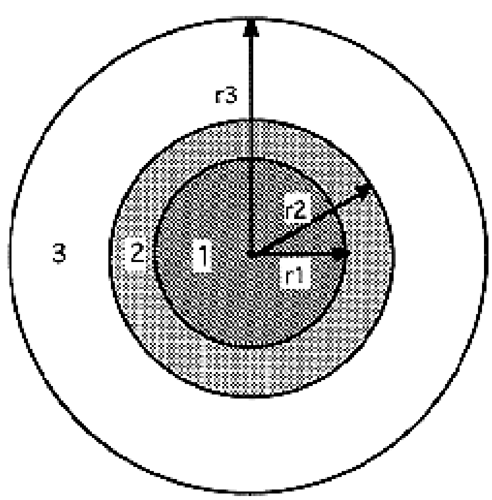
Core	Fuel Plates /Elem.	Half Fuel Region (F)	Clad Region (C)	Half Moderator Region (M)	Half Extra Region1 (E1)	Extra Region2 (E2)	Extra Region1 Al/D2O Volume Fractions
HEU	16	0.254	0.381	1.346	0.6244	7.2934	0.6499/0.3501
LEU	18	0.254	0.381	1.125	0.6567	6.7654	0.6611/0.3389

Number of Mesh Points in Each Region



Unit-Cell Specifications for Fueled and Non-Fueled Portions of Fuel Element

(Fuel Region Cross Sections: Collapse using Fluxes over F, C, and M;
Non-Fuel Cross Sections: Collapse using Fluxes over E1 Only)



Region 1: Homogenized Fuel
r1 = 35.8 mm
Region 2: Mixture of Al + D2O
r2 = 41.1 mm
Region 3: D2O
r3 = 80.0 mm

Unit-Cell Specifications for D2O Between Fuel Elements
(Collapse Cross Sections using Fluxes over Region 3 Only.)

4.2 Reactor Models

Reactor calculations were performed in three dimensions using the VIM continuous energy Monte Carlo code[14,15] and the DIF3D diffusion theory code[16].

A detailed Monte Carlo model of the reactor was constructed including all fuel assemblies, the shim-safety rods, the regulating rod, beam tubes and experiment penetrations, the bio-medical facility, and the top and bottom reflector regions in order to obtain absolute excess reactivities and shutdown margins for comparison with limits specified in the Technical Specifications. Nuclear cross sections were based on ENDF/B V data. The experiment facilities that were modeled are shown in Table 3.

In diffusion theory, the reactor was modeled in rectangular geometry with a heterogeneous representation of the fueled and non-fueled portions of the fuel assemblies and the water between fuel assemblies (see Fig. 6). The four shim-safety rods (control arms) that swing between the fuel assemblies, the regulating rod, and the various reactor penetrations (reactivity worth ~4.5% Æk/k) were not included in diffusion theory model. The bottom axial reflector and the radial reflector were also simplified. The LEU model is identical with the HEU model except for the fuel assembly materials.

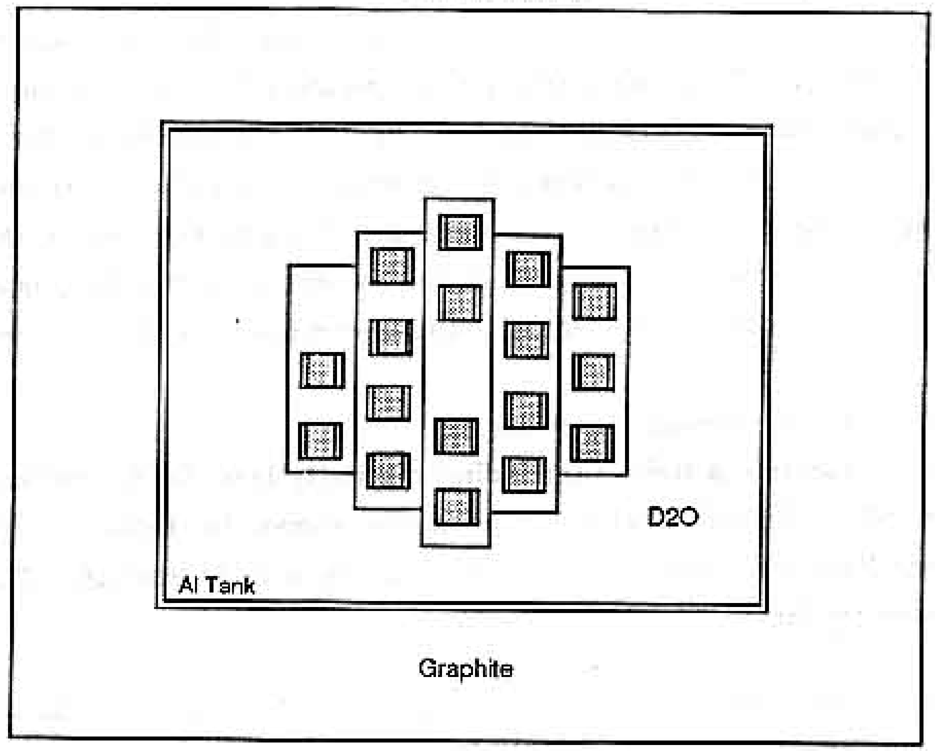
A simplified Monte Carlo model corresponding with the diffusion model was also constructed in order to verify that the diffusion theory model was correct.

Table 3. Experimental Facilities Included in the Detailed Monte Carlo Model.

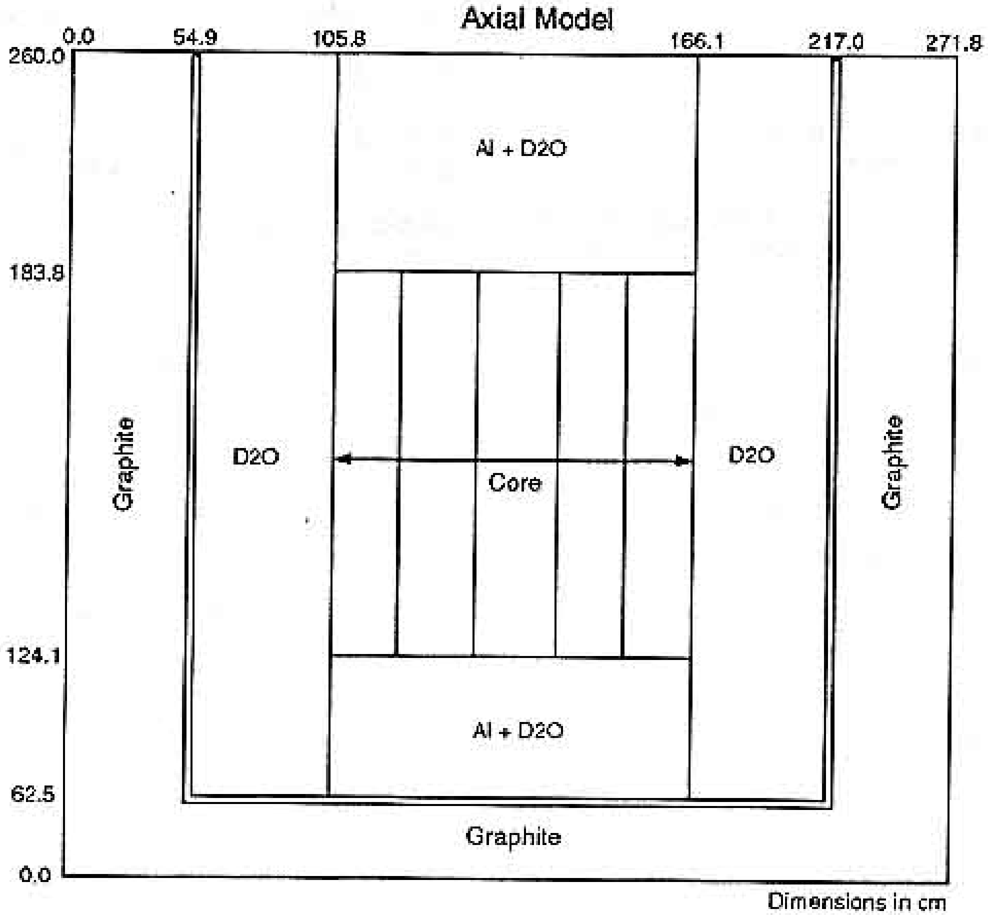
8	vertical experiment tubes filled with air in the D_2O reflector
2	vertical experiment tubes filled with air in the graphite reflector
12	vertical experiment tubes filled with graphite in the graphite reflector
14	horizontal beam tubes filled with air and penetrating the D_2O and graphite reflectors
8	horizontal beam tubes filled with graphite and penetrating both reflectors
2	horizontal beam tubes filled with 12” graphite, remainder air and penetrating both reflectors
Biomedical Facility:	A portion of the graphite reflector between the vessel and the biomedical facility consists of a bismuth shield and air (see Fig. 1).
Thermal Column	

Fig. 6. Radial and Axial Models for Diffusion Theory Calculations

Radial Model



Axial Model



5. NEUTRONIC PARAMETERS

5.1 Critical Experiment for HEU Core

In 1974, a critical experiment was built using 9 fresh HEU fuel assemblies. The core was made critical at different shim-safety blade positions[17] with the regulating rod nearly fully-withdrawn and nearly fully-inserted. The k_{eff}'s calculated for these critical configurations using the detailed Monte Carlo model were 0.991 ± 0.002 and 0.988 ± 0.002. The corresponding reactivity values were -0.91 ± 0.20% Æk/k and -1.22 ± 0.22% Æk/k, respectively. The reactivity bias of about -1.0 ± 0.3% Æk/k in the calculations is attributed to uncertainties in the nuclear cross sections and uncertainties in the reactor materials.

5.2 Cold Clean Excess Reactivities

Calculated excess reactivities (including reactivity bias) for the reference HEU and LEU cores with 17 fresh fuel assemblies are shown in Table 4. The Technical Specifications limit the excess reactivity to a maximum of 11.9% Æk/k. The LEU core is expected to satisfy this requirement.

Table 4. Excess Reactivities of HEU and LEU Cores with 17 Fuel Assemblies

	Calculated Excess React.[1], % Æk/k ± 1σ	
	Fresh HEU Core	Fresh LEU Core
Detailed Monte Carlo Model	11.7 ± 0.4	9.4 ± 0.4
Simplified Monte Carlo Model[2]	16.8 ± 0.4	14.3 ± 0.4
Diffusion Theory Model[2]	16.6	14.6

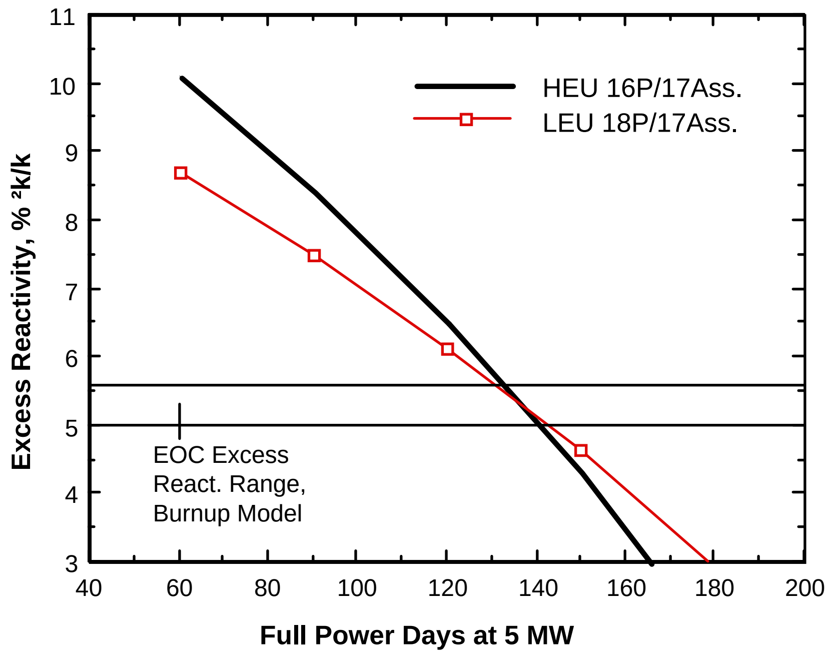
[1] The reactivity bias of -1.0 ± 0.3% Æk/k was added to calculated values.
[2] Without experiment penetrations, shim-safety blades, and regulating rod.

Differences between the detailed and simplified Monte Carlo models were described in Section 4.2. The reactivity effect of (1) replacing all vertical and horizontal experiment facilities inside the heavy water vessel with D_2O, (2) replacing all air-filled experiment facilities in the graphite reflector with graphite, and (3) replacing the bismuth shield and air in front of the biomedical facility with graphite was calculated[18] to be 4.5 ± 0.3% Æk/k. The worth of replacing the control absorbers in their fully-withdrawn position with D_2O was calculated to be 0.1 ± 0.3% Æk/k, a value consistent with zero worth. Thus, the simplified Monte Carlo model and the diffusion theory model are reasonable representations of the reactor if the reactivity worth of the experiment facilities is taken into account.

5.3 Burnup Calculations

Burnup calculations were run using the REBUS code[19] for HEU and LEU cores with 17 fuel assemblies to estimate fuel lifetimes. Reactivity profiles (including the 1% Æk/k reactivity bias) are shown in Fig. 7 over a limited burnup range. Excess reactivity values for fresh cores computed using the diffusion theory model are shown in Table 4. The dashed lines show the end-of-cycle excess reactivity range that accounts for reactivity losses due to experiment facilities (4.5 ± 0.3% Æk/k), cold-to-hot swing (~0.3% Æk/k), and control provision (~0.5% Æk/k) that are not included in the diffusion theory burnup model. Reactivity losses due to equilibrium Xe and Sm are included in the curves. We conclude that the lifetime of the LEU core will be about the same as that of the HEU core when absolute errors in the calculations are taken into account.

Fig. 7. Burnup Reactivity Profiles



5.4 Power Distributions and Power Peaking Factors

Power distributions and nuclear power peaking factors were calculated using the diffusion theory model for HEU and LEU cores with 14 and 17 fuel assemblies. As stated previously, the shim-safety rods, regulating rod, and experiment penetrations were not represented. The results are shown in Fig. 8 for the 14 element cores and in Fig. 9 for the 17 element cores. The reason for calculating cores with 14 fuel assemblies is that this is the minimum GTRR core size and cores with 14 assemblies will be used to compute the thermal-hydraulic safety margins.

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 two components: (1) a radial factor defined as the average power density in each assembly divided by the average power density in the core and (2) an assembly factor defined as the peak power density in each assembly divided by the average power density in that assembly. The assembly factor is a pointwise factor computed at the mesh interval edge and includes both planar and axial power peaking.

The data in Figs. 8 and 9 show that the power distributions and power peaking factors are nearly the same in fresh HEU and LEU cores with 14 fuel assemblies and in fresh HEU and LEU cores with 17 fuel assemblies. The percentages of reactor power shown in Figs. 8 and 9 do not add to 100% because ~2.5% of the energy is deposited outside the fuel assembles.

Fig. 8. Power Distributions and Power Peaking Factors
HEU and LEU Cores with 14 Fuel Assemblies

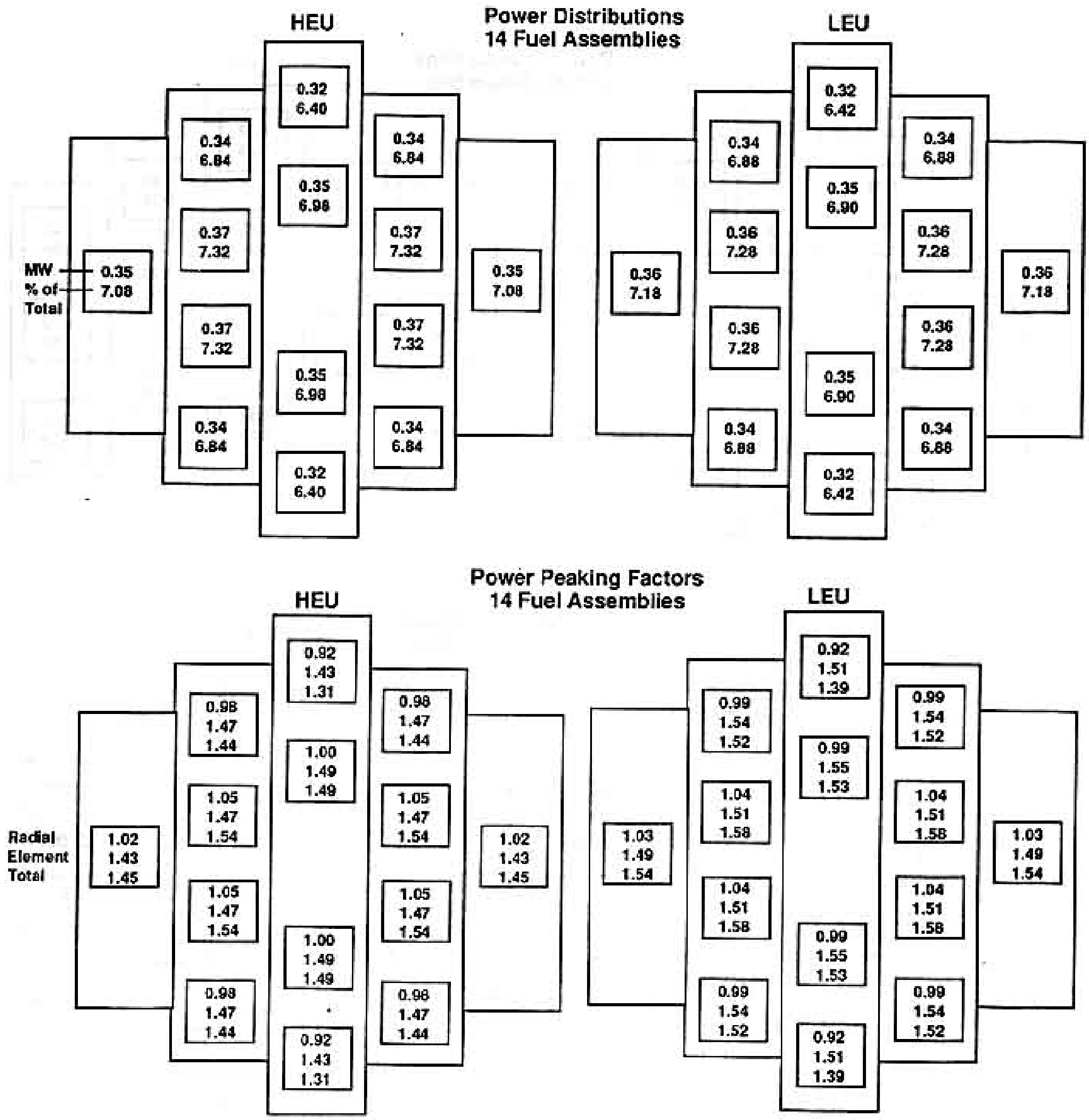
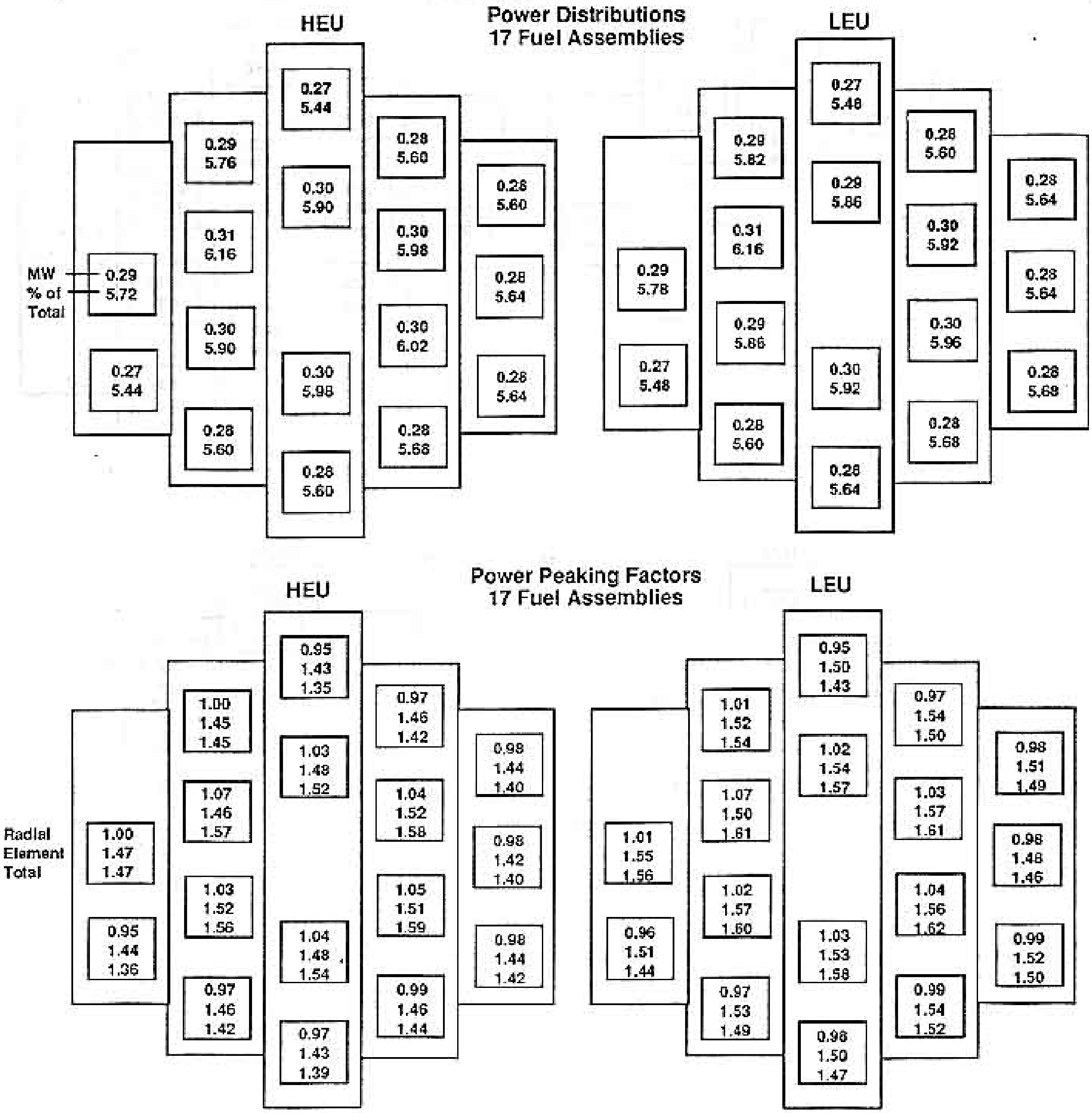


Fig. 9. Power Distributions and Power Peaking Factors
HEU and LEU Cores with 17 Fuel Assemblies



5.5 Reactivity Coefficients and Kinetics Parameters

Reactivity coefficients were computed for HEU and LEU cores with 14 and 17 fresh fuel assemblies as functions of temperature and void fraction using the 3D diffusion theory model. Also computed were the whole-core void coefficient, the reactor isothermal temperature coefficient, and the prompt neutron lifetime. Fresh cores were calculated because they are limiting cores. As fuel burnup increases, the neutron spectrum becomes softer and the reactivity coefficients become more negative.

Reactivity changes were calculated separately for changes in coolant temperature, coolant density, and fuel temperature while holding all heavy water outside the fuel assemblies at 23°C. Slopes of the reactivity feedback components at 45°C are shown in Table 5 along with the void coefficient for a uniform 1% change in the coolant density in all fuel assemblies. These reactivity feedback coefficients will be used in the transient analyses in Sections 9 and 10 because the transients considered involve heating of the fuel and coolant. Heating of the heavy water outside the fuel assemblies would have only a small effect because of the time constants involved in the transients.

Table 5. Reactivity Coefficients (% Æk/k/°C at 45°C) and Kinetics Parameters

	HEU		LEU	
	14 Ass.	17 Ass.	14 Ass.	17 Ass.
Coolant Temperature	-0.0062	-0.0055	-0.0055	-0.0053
Coolant Density	-0.0014	-0.0014	-0.0012	-0.0013
Fuel Doppler	~0.0	~0.0	-0.0017	-0.0020
Sum	-0.0076	-0.0069	-0.0084	-0.0086
Whole Reactor Isothermal[1]	-0.0224	-0.0201	-0.0232	-0.0215
Void Coefficient[2],	-0.0383	-0.0392	-0.0333	-0.0350
l_p[3], µs	780	704	745	695
β_{eff}	0.00755[4]	0.00755[4]	0.0075 - 0.0076[5]	

[1] Includes fuel, coolant, inter-assembly water, and reflector.
[2] % Æk/k/% Void. Uniform voiding of coolant in all fuel assemblies.
[3] Calculated prompt neutron lifetime.
[4] Measured effective delayed neutron fraction.
[5] Estimated value.

The sum of the coolant and fuel Doppler reactivity coefficients in Table 5 are slightly more negative in the LEU cores than in the HEU cores. The Doppler coefficient actually has a larger weight than shown in Table 5 because the fuel temperature normally increases more rapidly than the coolant temperature. The coolant void coefficient for all fuel assemblies in the core is slightly more negative in the HEU cores than in the LEU cores.

The reactor isothermal temperature coefficient for the 5 MW clean core with 16 HEU fuel assemblies was calculated in the GTRR Safety Analysis Report (Ref. 1, p. 98) to be -0.0232% Æk/k/°C at 45°C. The reactor isothermal temperature coefficients shown in Table 5 for clean cores with 14 and 17 HEU assemblies are in good agreement with this value. The corresponding reactor isothermal temperature coefficients for LEU cores with 14 and 17 assemblies are slightly more negative than those for the HEU cores. A breakdown of calculated isothermal reactivity feedback components for the coolant, inter-assembly water, and reflector of an HEU core with 17 fresh fuel assemblies is shown in Attachment 1.

In April 1992, the whole-reactor isothermal temperature coefficient was measured to be -0.0338 Æk/k/°C in a 17 assembly HEU core with about 10,000 MW-hr burnup over the period 1974-1992 (R. Karam, GTRR; private communication). Although these measured and calculated data cannot be compared directly (temperature coefficients normally become more negative with increasing burnup), it does indicate that measured temperature coefficients in the GTRR may be more negative than calculated values.

The calculated prompt neutron lifetimes shown in Table 5 for the LEU cores with 14 fuel assemblies and with 17 fuel assemblies are slightly smaller than those in the corresponding HEU cores because the LEU cores have a slightly harder neutron spectrum.

The fission component of the delayed neutron fraction in both the HEU and LEU cores was calculated to be 0.0071. The difference between this value and the β_{eff} of 0.00755 measured in the HEU core is attributed to delayed neutrons resulting from dissociation of heavy water by neutrons and gamma rays. The latter component of β_{eff} has not been computed here. Since the fission components of β_{eff} were computed to be the same in the HEU and LEU cores, we expect that the heavy water components of β_{eff} and thus the total effective delayed neutron fractions will be very similar as well.

6. SHUTDOWN MARGINS

The Technical Specifications require that the reactor have a shutdown margin of at least 1% Æk/k with the most reactive shim-safety blade and the regulating rod fully withdrawn. Measured reactivity worths[20] of the shim-safety blades in the present HEU core are shown in Table 6. The blade with the highest reactivity worth is blade #3.

Table 6. Measured Reactivity Worths of Shim-Safety Blades in HEU Core (9/26/90)

Shim-Safety Blades	Reactivity Worth, % Æk/k
Blade #1	5.55
Blade #2	4.66
Blade #3	6.21
Blade #4	4.41

Table 7 compares shutdown margins calculated using the detailed Monte Carlo model for HEU and LEU cores with 17 fresh fuel assemblies. The regulating rod and shim-safety blade #3 were fully-withdrawn and the other three shim-safety blades were fully-inserted. The results show that both cores satisfy the 1% Æk/k shutdown margin requirement of the Technical Specifications.

Table 7. Calculated Shutdown Margins for HEU and LEU Cores with 17 Fresh Fuel Assemblies.

Core	Shutdown Margin, % Æk/k
HEU	-7.14 ± 0.25
LEU	-8.84 ± 0.21

In addition to the automatic protective systems, manual scram and reflector drain provide backup methods to shut the reactor down by operator action. The top of the core is covered by 29.75 inches of D_2O, measured from the top of the fuel meat. The top 28 inches of D_2O can be drained through a 4 inch pipe which connects the reactor vessel to the storage tank of the primary D_2O system. The reactivity worth of the top 28 inches of reflector was measured[21] to be 2.75% Æk/k in an HEU core composed of 15 fuel assemblies with 142 g ^{235}U per assembly.

Monte Carlo calculations using the detailed Monte Carlo model were done to compare reactivity worths of the top D_2O reflector in HEU and LEU cores with 17 fresh fuel assemblies (188 g ^{235}U HEU, 225 g ^{235}U LEU). Several calculations were first done for each core to determine shim-safety blade positions that would bring the reactor near critical. Results in Table 8 for cases with 1” and 2” of D_2O reflector above the top of the fuel meat show that the top reflector worths of the HEU and LEU cores are very similar. Thus, the shutdown capability of reflector drain in the LEU core will be very similar to that in the present HEU core.

Table 8. Calculated Top Reflector Worths (% Æk/k) of HEU and LEU Cores with 17 Fuel Assemblies and Control Blades near Critical Positions

Top D_2O Reflector	HEU Core	LEU Core
D_2O 1" Above Fuel Meat	- 2.58 ± 0.29 (1σ)	- 2.73 ± 0.31 (1σ)
D_2O 2" Above Fuel Meat	- 2.05 ± 0.28	- 2.42 ± 0.30

7. THERMAL-HYDRAULIC SAFETY PARAMETERS

Thermal-hydraulic safety limits and safety margins calculated using the PLTEMP code[22] for the LEU core with 14 fuel assemblies (see Fig. 8) were compared with the thermal-hydraulic safety parameters used as bases for the current Technical Specifications. The analyses by ANL for the LEU core used a combined multiplicative and statistical treatment of a revised set of engineering uncertainty factors. Attachment 2 lists the engineering uncertainty factors used by Georgia Tech for analyses[23] of the HEU core and discusses the factors used by ANL, the rationale for their choice, and the method used to combine them. Results for the HEU core obtained using ANL's statistical treatment of the engineering uncertainty factors agree well with the analyses performed by Georgia Tech.

7.1 Safety Limits in the Forced Convection Mode

The current Technical Specifications utilize departure from nucleate boiling (DNB) as a basis for establishing safety limits on reactor power, coolant flow, and coolant inlet (or outlet) temperature. This report evaluates these limits based on flow instability as well as DNB criteria. The modified Wheatherhead correlation[23,24] was used for DNB and the Forgan-Whittle correlation[25,26] was used for flow instability.

Calculated reactor power limits based on DNB and flow instability are shown in Table 9 for 14-assembly HEU and LEU cores with the minimum coolant flow of 1625 gpm and with the coolant lowflow limit of 760 gpm. A maximum inlet temperature of 123°F was used in all cases. Power limits based on the flow instability criterion are smaller than those based on DNB, but are still adequate to ensure the safety of the facility. The main reason for the difference in reactor power limits in the HEU and LEU cores is that the manufacturing specifications for LEU silicide dispersion fuel plates contain a factor of 1.2 for homogeneity of the fuel distribution while the HEU alloy fuel has a corresponding factor of 1.03.

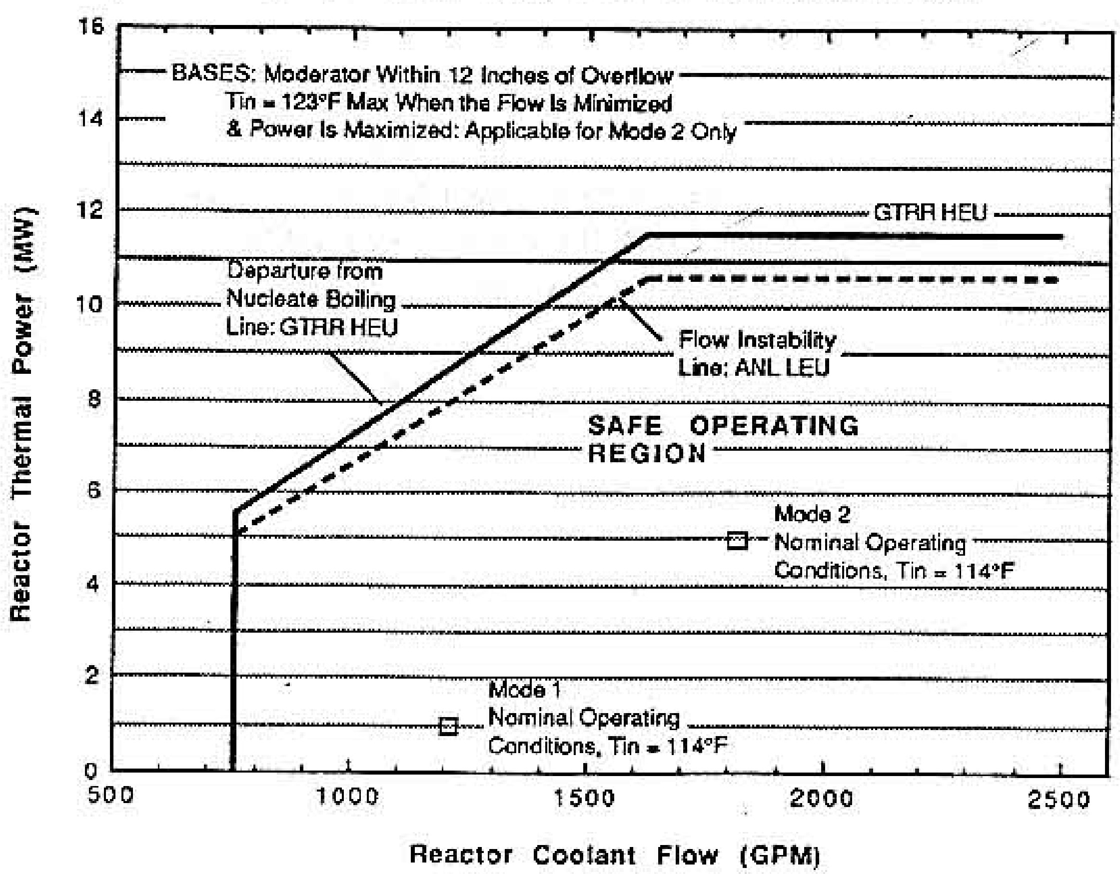
Table 9. Reactor Power Limits in 14-Assembly Cores for a Maximum Inlet Temperature of 123°F Based on Departure from Nucleate Boiling and Flow Instability.

Reactor Coolant Flow, gpm	GTRR-HEU[1]	ANL-LEU[2]
	Reactor Power Level (MW) for DNB[23,24]	
760	5.5	5.3
1625	11.5	10.8
	Reactor Power Level (MW) for Flow Instability[25,26]	
760	5.3	5.0
1625	10.6	10.6

[1] Calculated by ANL using GTRR engineering uncertainty factors in Ref. 23.
[2] Calculated by ANL using revised engineering uncertainty factors (see Attach. 2).

Figure 10 shows the calculated reactor power limits as functions of reactor coolant flow based on DNB for the HEU core and on flow instability for the LEU core. In the LEU core, we recommend a power limit of 10.6 MW based on the flow instability criterion for the minimum coolant flow of 1625 gpm and the maximum inlet temperature of 123°F..

Fig. 10. GTRR Safety Limit for Forced Convection



More detailed data for the minimum coolant flow rate of 1625 gpm and the maximum inlet temperature of 123°F are shown in Table 10. The LEU fuel assembly has reduced power per plate, a smaller flow area, a higher coolant velocity, and a larger pressure drop due to friction. The peak cladding surface temperature is larger by about 5°F and the margins to DNB and flow instability are adequate.

Table 10. Thermal-Hydraulic Data for 14-Assembly Cores with the Minimum Coolant Flow of 1625 GPM and the Maximum Inlet Temperature of 123°F.

	GTRR-HEU[1]	ANL-LEU[2]
Coolant Velocity, m/s	2.44	2.61
Friction Pressure Drop[3], kPa	10.9	15.0
Power/Plate[4], kW	21.2	18.8
Outlet Temperature of Hottest Channel, °F	157	156
Peak Clad Surface Temperature, °F	219	224
Minimum DNBR[5]	2.29	2.17
Limiting Power Based on Min. DNBR, MW	11.5	10.8
Flow Instability Ratio (FIR)[6]	2.12	2.11
Limiting Power Based on FIR, MW	10.6	10.6

1 Calculated by ANL using engineering uncertainty factors used in Ref. 23.
2 Calculated by ANL using revised engineering uncertainty factors (see Attachment 2).
3 Pressure drop across active fuel only.
4 Assuming 95% of power deposited in fuel.
5 Using modified Weatherhead Correlation[23,24] for DNB.
6 Using Forgan-Whittle Correlation[25,26] with $\eta = 25$.

Safety limits for the reactor inlet temperature were calculated at the maximum reactor power of 5.5 MW and the minimum coolant flow of 1625 gpm. The results are shown in Table 11. Data for the GTRR-HEU core are based on DNB. ANL results for the LEU core are based on both DNB and flow instability criteria. A safety limit for the reactor outlet temperature was then established by adding the average temperature rise across the core to the limiting inlet temperature. These results show that the HEU and LEU cores have nearly identical safety limits on the reactor inlet and oulet temperatures.

Table 11. Safety Limits on Reactor Inlet and Outlet Temperatures.

	GTRR-HEU[1]	ANL-LEU[2]	
Parameter	DNB	DNB	Flow Instability
Limiting Reactor Inlet Temp., °F	172	171	170
Ave. Coolant Temp. Rise across Core, °F	16	17	17
Limiting Reactor Outlet Temp., °F	188	188	187

1 Data from Ref. 23 based on DNB criterion.
2 Calculated using ANL engineering uncertainty factors in Attachment 2.

7.2 Safety Limits in the Natural Convection Mode

The current Technical Specifications state that the reactor thermal power shall not exceed two (2) kW in the natural convection mode. This specification is based on GTRR experience showing that no damage to the core and no boiling occurs without forced convection coolant flow at power levels up to 2 kW. We expect that this specification will also hold in the LEU core because the average power per fuel plate will be lower in the LEU core. Each LEU fuel assembly will contain 18 fuel plates while each HEU assembly contains 16 fuel plates.

7.3 Limiting Safety System Settings in the Forced Convection Mode

The safety system trip setting in the current GTRR Technical specifications for power levels >1 MW and for power levels ≤ 1 MW are shown in Table 12.

Table 12. Safety System Trip Settings

Parameter	Reactor Power Level >1 MW	Reactor Power Level ≤1 MW
Thermal Power	5.5 MW	1.25 MW
Reactor Coolant Flow	1625 GPM	1000 GPM
Reactor Outlet Temperature	139°F	125°F

These safety system trip settings are based on a criterion[3] that there shall be no incipient boiling during normal operation. The criterion is applied by ensuring that the surface temperature at any point on a fuel assembly does not exceed the coolant saturation temperature at that point. This criterion is conservative because there is an additional margin of ~26°F between the D_2O saturation temperature and the temperature at which onset of nucleate boiling occurs.

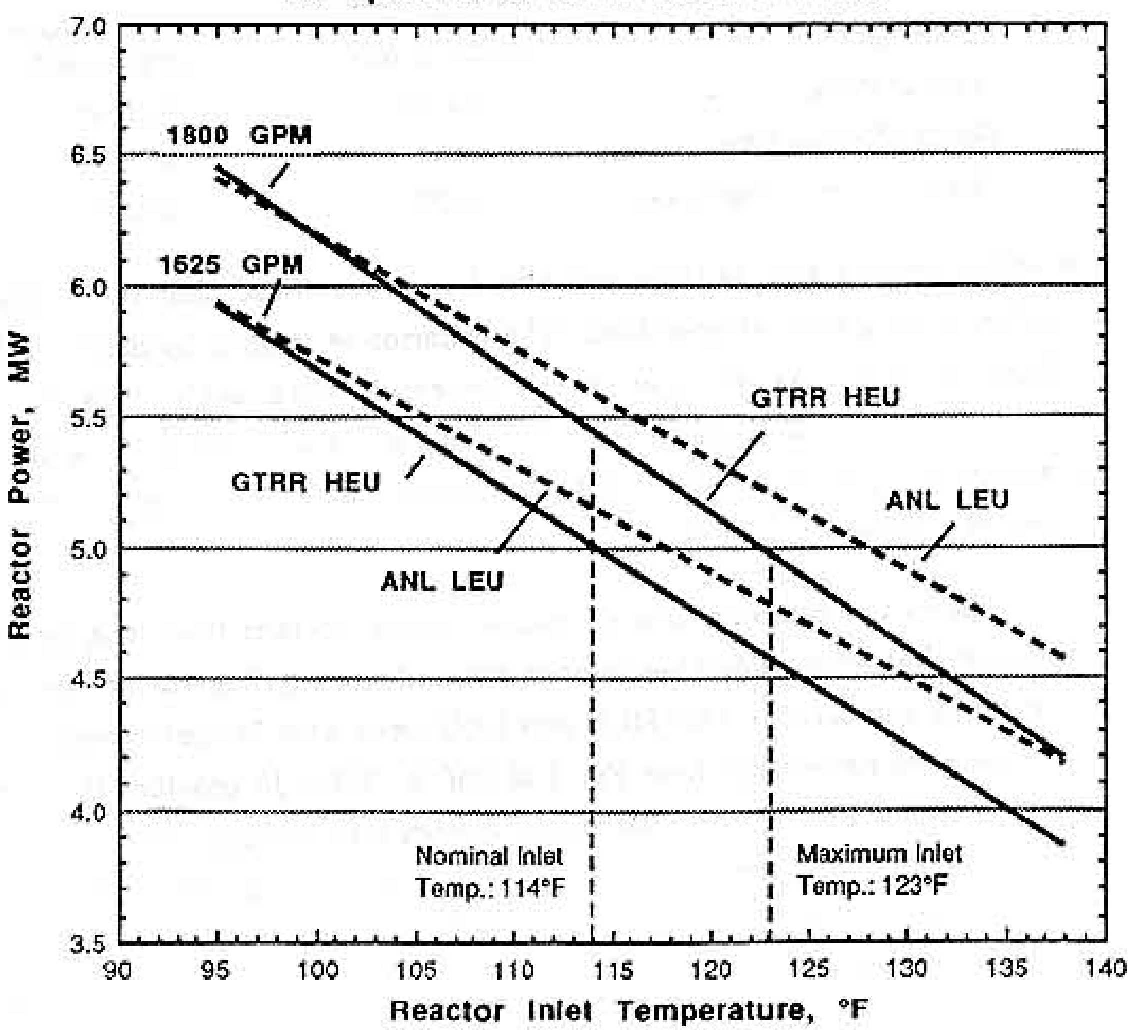
Figure 11 shows the combinations of reactor power, coolant flow rate, and reactor inlet temperature that were calculated to have zero subcooling (fuel surface temperature = coolant saturation temperature) for HEU and LEU cores with 14 fuel assemblies. Data for the HEU core were reproduced from Fig. 1 of Ref. 3. Table 13 provides the parameter combinations which correspond with the safety system trip settings shown in Table 12. The trip setting of 139°F on reactor outlet temperature was obtained by adding the 16°F temperature rise across the core to the maximum inlet temperature of 123°F. Similar considerations based on operation during the period 1964 to 1973 were applied to determine the safety system trip settings for power levels equal to or less than 1 MW.

Parameter combinations that have zero subcooling in the LEU core are shown in Table 13 and in Fig. 11. Since the values for the LEU core are more conservative than those for the HEU core, the current safety system trip settings for the HEU core can also be used for the LEU core.

Table 13. Parameter Combinations for Zero Subcooling with 14-Assembly HEU and LEU Cores

	GTRR HEU			ANL LEU		
Reactor Power, MW	5.5	5.0	5.0	5.6	5.0	5.0
Coolant Flow Rate, gpm	1800	1625	1800	1800	<1625	1800
Reactor Inlet Temp., °F	114	114	123	114	114	128
Temp. Rise Across Core, °F	16	16	16	17	17	17
Reactor outlet Temp., °F	130	130	139	131	131	145

Fig. 11. Thermal-Hydraulic Limits Based on Zero Subcooling For Operation at Power Levels ≤ 5 MW.



The results in Table 14 show that the degree of subcooling (ÆT$_{sub}$) at the hottest spot of the limiting fuel assembly under normal operating conditions is expected to be 11°F in the LEU core and 8°F in the HEU core. Another criterion that is often used in research reactors is that the margin to onset of nucleate boiling (ONB) should be equal to or greater than 1.2. ONB occurs at a temperature of about 246°F, which is ~26°F above the D_2O saturation temperature of 220°F. The margin to ONB in the LEU core was computed by increasing the reactor power until ONB occured and dividing by the nominal reactor power of 5 MW. These margins are adequate to ensure that the LEU core can be operated safely at a power level of 5 MW.

Table 14. Margins to D_2O Saturation Temperature and ONB for 14-Assembly Cores

Parameter	GTRR-HEU[1]	ANL-LEU[2]
Thermal Power, MW	5.0	5.0
Reactor Coolant Flow, gpm	1800	1800
Reactor Inlet Temp., °F	114	114
ÆT$_{sub}$, °F	8	11
Margin to ONB[3]	-	1.44
Limiting Power Based on ONB, MW	-	7.2

1 Data from Refs. 3 and 23.
2 Calculated using ANL engineering uncertainty factors in Attachment 2.
3 Using the Bergles and Rohsenow correlation[27].

Calculations were also done to examine the adequacy of the current safety system trip settings shown in Table 12 for operation at power levels equal to or less than 1 MW. Since data from analyses of the HEU core by Georgia Tech were not available, calculations were done using the GTRR-HEU and the ANL-LEU engineering uncertainty factors shown in Attachment 2, a thermal power of 1.25 MW, a reactor coolant flow of 1000 gpm, and an inlet temperature of 123°F. The results shown in Table 15 for the degree of local subcooling (ÆT$_{sub}$) and the flow instability ratio indicate that the current trip settings on reactor power and coolant flow are conservative and are adequate to ensure the safety of the facility for operation at power levels that are ² 1 MW.

Table 15. Selected Thermal-Hydraulic Safety Margins with 14-Assembly Cores and Power ² 1 MW.

Parameter	GTRR-HEU[1]	ANL-LEU[2]
Thermal Power, MW	1.25	1.25
Reactor Coolant Flow, gpm	1000	1000
Reactor Inlet Temp., °F	123	123
Peak Surface Clad Temp., °F	162	164
ÆT$_{sub}$, °F	58	56
Flow Instability Ratio	5.4	5.3

1 Calculated using GTRR HEU engineering uncertainty factors in Attachment 2.
2 Calculated using ANL LEU engineering uncertainty factors in Attachment 2.

7.4 Limiting Safety System Settings in the Natural Convection Mode

The Technical Specifications state that the reactor thermal power safety system setting shall not exceed 1.1 kW when operating in the natural convection mode. This specification is based on GTRR experience showing that the reactor can be operated at one kW indefinitely without exceeding a bulk reactor temperature of 123°F. We expect that this safety system trip setting will also be adequate for the LEU core.

8. COOLING TIME REQUIREMENTS

The Technical Specifications for the HEU core state that containment integrity shall be maintained when the reactor has been shutdown from a power level greater than 1 MW for less than eight hours. In addition, a minimum cooldown time of twelve hours is required before fuel assemblies are transferred out of the reactor.

Fuel melting and subsequent release of fission products could result from a loss-of-coolant accident following reactor shutdown if sufficient decay heat is present. Containment integrity is therefore required until the decay heat generation rate is less than that required to melt the fuel plates. A limit of 450°C was set in the Technical Specifications as the upper value for a fuel plate temperature to preclude melting of the plates. The decay time needed to ensure that this temperature would not be reached was calculated in Ref. 23.

The analysis method and input parameters described in Ref. 23 were used to reproduce the results for the HEU core. The same methodology was then used for the LEU core, with modification of the input parameters appropriate for the LEU fuel assembly design. A standard 3-week operating history consisting of 4.33 days at full power of 5 MW and 2.67 days shutdown was used for 14 assembly cores with HEU and LEU fuel. The analysis in both cases was applied to a fuel assembly which has been subjected to a power peaking factor of 1.5 (see Fig. 8). As in Ref. 23, the peak power was increased by 17% to account for the incremental heat contribution due to additional gamma heating from surrounding fuel assemblies in the core and was decreased by 15% to take credit for an improved convection condition in the reactor vessel.

Three input parameters that were used for the HEU fuel assembly in Ref. 23 were modified for the LEU fuel assembly design: (1) the parameter hA_{GTRR} was reduced from 3.03 x 10^{-4} kW/°C for an HEU plate to 2.88 x 10^{-4} kW/°C for an LEU plate based on the heat transfer areas of the HEU and LEU fuel meat shown in Table 1, (2) the mass of aluminum associated with one fuel plate was reduced from 0.418 lbm for an HEU plate to 0.377 lbm for an LEU plate, mainly because U_3Si_2 fuel particles occupy approximately 31% of the fuel meat volume in an LEU plate; no credit was taken for the specific heat of the U_3Si_2 particles, and (3) most importantly, the maximum power per fuel plate in the LEU assembly was reduced by a factor of 16/18 since an HEU assembly contains 16 fueled plates and an LEU assembly contains 18 fueled plates.

The results for loss-of-coolant from the reactor vessel after eight hours of cooling showed a maximum plate temperature of 425°C in the HEU core and 400°C in the LEU core. The maximum temperature occurred 45 minutes after loss-of-coolant in the HEU core and 50 minutes after loss-of-coolant in the LEU core. For the more confined heat transfer situation, without gamma rays from other fuel assemblies, but with a restricted heat transfer volume, the maximum fuel plate temperature after a twelve hour cooldown was calculated to be 361°C for an HEU plate and 340°C for an LEU plate. The maximum temperature occurred 60 minutes after removal from the HEU core and 50 minutes after removal from the LEU core.

We conclude that the current Technical Specification requirements on cooling times are more conservative for the LEU core than for the HEU core. The most important factor is the reduced power per plate in the LEU core. However, any reduction of technical specification cooling time requirements for the LEU core should be based on measurements in the GTRR.

9. LIMITATIONS OF EXPERIMENTS

The Technical Specifications contain three limitations of experiments that could be affected by changing the fuel in the core from HEU to LEU:

a) The magnitude of the potential reactivity worth of each unsecured experiment is limited to 0.004 Æk/k.

b) The potential reactivity worth of each secured removable experiment is limited to 0.015 Æk/k.

c) The sum of the magnitudes of the static reactivity worths of all unsecured experiments which coexist is limited to 0.015 Æk/k.

The objective of these specifications is to prevent damage to the reactor and to limit radiation dose to personnel and the public in event of experiment failure. Qualification of the PARET code that was used for the transient analysis is discussed first, followed by the calculated results.

9.1. Comparison of Calculations with SPERT-II Experiments

The PARET code[28] was originally developed at the Idaho National Engineering Laboratory for analysis of the SPERT-III experiments, which included both pin-type and plate-type cores and pressures and temperatures in the range typical of power reactors. The code was modified by the RERTR Program at ANL to include a selection of flow instability, departure from nucleate boiling, single- and two-phase heat transfer correlations, and properties libraries for light water and heavy water that are applicable to the low pressures, temperatures, and flow rates encountered in research reactors.

To validate the PARET code for use with heavy water reactors, calculated and measured data were compared[29] for the SPERT-II BD-22/24 HEU core[30] (24 MTR-type fuel elements with 22 plates per element). This core is similar to the GTRR in design. The tests performed in the BD-22/24 core included only nondestructive transients. Calculated transient parameters shown in Ref. 29 are in very good agreement with the measured data and validate the PARET code for use in calculating transients in heavy water research reactors.

9.2 Inadvertent Reactivity Insertions Due to Experiment Failure

The consequences of inadvertent step reactivity insertion of 0.4% Æk/k and 1.5% Æk/k in HEU and LEU cores with 14 fuel assemblies were evaluated. The model and methods that were used for analysis of the SPERT-II BD-22/24 HEU cores were also used to analyze the HEU and LEU cores of the GTRR.

Inputs to the code for analysis of the GTRR included the prompt neutron lifetime, effective delayed neutron fraction, temperature coefficients of reactivity, and power distributions discussed in Sections 5.4 and 5.5. Temperature coefficients included contributions from only the coolant and the fuel. Axial power distributions for the average channel of the HEU and LEU cores were represented by chopped cosine shapes having peak-to-average power densities of 1.19. In the hot channel, these axial shapes were scaled to produce peak power densities in the limiting fuel assemblies of the HEU and the LEU cores that are consitent with the power distributions shown in Fig. 8.

Calculations were performed for step reactivity insertions of 0.4% and 1.5% Æk/k with the reactor at nominal operating conditions of 5 MW thermal power, a coolant flow rate of 1800 gpm, and a reactor inlet temperature of 114°F. A scram signal was initiated when the reactor power reached the safety system overpower trip setting of 5.5 MW. A time delay of 100 ms was assumed between introduction of the scram signal and release of the shim-safety blades. The results of these calculations are shown in Table 16.

Table 16. Results of Assumed Step Reactivity Insertions Due to Experiment Failure

Parameter	HEU Core		LEU Core	
Step Reactivity Insertion, % Æk/k	0.4	1.5	0.4	1.5
Asymptotic Period, s	0.18	0.05	0.18	0.05
Peak Power, MW	7.4	27.5	7.4	27.2
Peak Surface Cladding Temp., °F	184	277	179	267
Peak Coolant Outlet Temp., °F	135	-	135	-

A positive step reactivity change less than 0.4% Æk/k caused by the ejection or insertion of experiments would result in transient behavior that would not exceed the safety limits for the HEU or LEU cores that were discussed in Section 7.1. The peak power of 7.4 MW in both cores is well below the safety limits of 11.5 MW in the HEU core and 10.6 MW in the LEU core. Similarly, the peak coolant outlet temperatures are well below the limiting reactor outlet temperature of 188°F.

Step reactivty insertions of 1.5% Æk/k would result in peak surface cladding temperatures that are far below the solidus temperature of 1220°F (660°C) in the 1100 Al cladding of the HEU core and far below the solidus temperature of 1080°F (582°C) in the 6061 Al cladding of the LEU core. Thus, no damage to the fuel and no release of fission products is expected.

10. ACCIDENT ANALYSES

A spectrum of accident scenarios was evaluated by Georgia Tech in its safety documentation[1,3,4] for 5 MW operation. These scenarios included (1) failure of electrical power, (2) failure of various reactor components, (3) a startup accident in which one shim blade and the regulating rod were withdrawn simultaneously, (4) reactivity effects resulting from the melting of fuel plates, (5) assumed maximum positive reactivity insertion, and (6) the Design Basis Accident. A review of these scenarios concluded that only scenarios (3) - (6) could be affected by changing the fuel assemblies from HEU to LEU, and only these scenarios are addressed here.

10.1 Startup Accident

The worst case for a possible startup accident in the current HEU core was determined[3] to result from the simultaneous withdrawal of one shim blade and the regulating rod. An experiment was done in the GTRR to simulate reactor behavior when reactivity was added at rate of approximately 0.005 Æk/k per second starting from a power level of 5 kW. Within 3 seconds, the reactor was automatically scrammed by a positive period trip. The power level at the scram point was 6.5 kW. On this basis, it was concluded[3] that if the reactor were operating at 5 MW, the reactor would be scrammed by the overpower trip at 5.5 MW or the log-N period systems would scram the reactor at a power level of no more than 7 MW. Since this is well below the 11.5 MW burnout power level of the GTRR, no fuel plate melting would be expected.

Calculations were done here using the PARET code for the HEU and LEU cores with 14 fuel assemblies in which reactivity was added at a rate of 0.005 Æk/k per second starting from a power level of 5 MW. Except for the reactivity addition rate, inputs to the code were the same as those described in paragraphs 2 and 3 of Section 9.2. Both the HEU and LEU cores were scrammed by the overpower trip at 5.5 MW. A time delay of 100 ms was assumed between introduction of the scram signal and release of the shim-safety blades. Both cores reached a peak power of 5.9 MW at a time of 0.335 s after the transient was initiated. Peak surface cladding temperatures of 177°F and 172°F were reached in the limiting fuel assembly of the HEU and LEU cores, respectively. The peak power is well below the safety limits of 11.5 MW in the HEU core and 10.6 MW in the LEU core. The peak surface cladding temperatures are far below the solidus temperature of 1220°F in the 1100 Al cladding of the HEU core and far below the solidus temperature of 1080°F in the 6061 Al cladding of the LEU core. Thus, no damage to the fuel and no release of fission products is expected.

10.2 Reactivity Effects of Fuel Plate Melting

The reactivity effect of melting individual fuel plates within an assembly due to the blockage of individual flow channels was analyzed[4] for the current GTRR HEU core by estimating the reactivity change caused by removing the two central fuel plates in a fuel assembly at the core center. It was concluded that the loss of one or more fuel plates would result in a negative reactivity effect.

Calculations were done for HEU and LEU cores with 14 and 17 fresh fuel assemblies using the reactor diffusion theory model described in Section 4.2 and in Figs. 8 and 9. The results in Table 17 show that the reactivity effect of removing one or two fuel plates from a fuel assembly near the center of the HEU and LEU cores and replacing the fuel plate volume with D_2O is expected to be negative.

Table 17. Calculated Reactivty Effect of Removing Fuel Plates from a Fuel Assembly Near the Center of the HEU and LEU Cores.

	Reactivity Change, % Æk/k			
	14 Assembly Cores		17 Assembly Cores	
	HEU	LEU	HEU	LEU
1 Fuel Plate Removed	- 0.060	- 0.037	- 0.043	- 0.028
2 Fuel Plates Removed	- 0.127	- 0.078	- 0.090	- 0.060

10.3 Fuel Loading Accident

During refueling operations, all control blades are required to be fully inserted and the top D_2O reflector drained to storage. Calculations in Section 6 indicated that the shutdown margin with the blade of maximum worth stuck out of the core is expected to be - 7.1 ± 0.3% Æk/k in the HEU core and - 8.8 ± 0.2% Æk/k in the LEU core. The shutdown margins will be more negative with all shim safety blades inserted. In addition, the reactivity worth of the top reflector is at least 2% Æk/k.

The current GTRR safety analysis report[1] analyzed a hypothetical fuel loading accident scenario assuming, in violation of established startup procedures, that the shim-safety blades are withdrawn so that the reactor is just sub-critical and that the D_2O is at the normal operating level. A fresh fuel assembly was then assumed to be dropped into the center core position, resulting in a sudden reactivity insertion of 2.5% Æk/k. We consider this postulated scenario to be incredible and no analysis of this scenario is presented in this report. The maximum positive reactivity insertion is addressed in Section 10.4.

10.4 Maximum Positive Reactivity Insertion

The Technical Specifications limit the potential reactivity worth of each secured removable experiment to 1.5% Æk/k and the sum of the magnitudes of the static reactivity worths of all unsecured experiments which coexist to 1.5% Æk/k. The purpose of this analysis is to show that there is a sufficient margin between the maximum allowable reactivity worth of a single experiment and the maximum step reactivity insertion that can be tolerated without fuel damage, assuming failure of reactor scram systems.

Analysis[1] for the current HEU core used SPERT-II experimental data[30] as a basis for estimating the step reactivity insertion that would result in the onset of steam blanketing in the GTRR. In the present analysis, the PARET code was used to compute the step reactivity insertion required to initiate steam blanketing (film boiling) in both the SPERT-II B22/24 core and 14-assembly GTRR cores with HEU and LEU fuel. Some of the kinetics parameters and key PARET results are provided in Table 18. Power peaking factors are similar in the SPERT-II and GTRR cores. The inverse period corresponding to the onset of steam blanketing as determined from the SPERT experimental data[1,30] is about 13 s^{-1}. The PARET code predicts the onset of film boiling for a step insertion of $2.0 (1.5% Æk/k) with an inverse period of 12 s^{-1}, in good agreement with experiment.

The same methodology was used to compute GTRR cores with 14 fuel assemblies. These cores have smaller coolant void coefficients than the SPERT-II B22/24 core, but the step insertions needed to initiate film boiling (~$2.0) and the peak surface cladding temperatures (250-260°C) at the onset of steam blanketing are nearly the same. At the time of peak power, the energy deposited per plate is about the same in the SPERT and GTRR cores. The peak surface cladding temperature at the time of peak power is about 220°C in the GTRR cores and about 204°C in the SPERT core.

The SPERT-II B22/24 tests[30] indicate that even more extensive film boiling (or steam blanketing) does not result in temperatures that exceed the solidus temperature of the cladding. The most extreme case in the test series with a reactivity insertion of $2.95 (2.2% Æk/k) resulted in a peak surface cladding temperature of 337°C, a temperature far below the solidus temperature of 582°C for 6061 Al cladding. The GTRR SAR[1] also notes that the maximum temperature for large insertions is primarily limited by the energy deposited in the plate with very little effect from the boiling heat transfer.

Since the behavior of the SPERT-II B22/24 and GTRR 14-assembly cores is very similar, a step reactivity insertion greater than 2.2% Æk/k would be required to initiate melting of the

GTRR LEU core. The margin of at least 0.7% Æk/k above the maximum allowed reactivity worth of 1.5% Æk/k for a single experiment is sufficient to ensure that the facility is safe in the unlikely event that the maximum allowed reactivity were inserted in a step and the reactor scram system failed to function.

Table 18. Comparison of Kinetics Parameters and Onset of Steam Blanketing Results

	SPERT-II	14 Assembly GTRR	
	B-22/24	HEU	LEU
Prompt Neutron Generation Time, µs	660	780	745
Beta Effective	0.0075	0.00755	0.00755
Coolant Temperture Coeff., $/°C	-0.00867	-0.00874	-0.00689
Void Coefficient, $/% Void	-0.0729	-0.0509	-0.0442
Doppler Coefficient, $/°C	~0.0	~0.0	-0.00096
Operating Pressure, kPa	122	127	127
Step Reactivity Insertion, $ (% Æk/k)	2.00	1.99	1.95
Inverse Period, s^{-1}	12	19	19
Energy/Plate at t_m, kWs	31.8	31.2	32.0
Peak CladdingTemperature at t_m, °C	204	218	225
Peak Cladding Temperature at Onset of Steam Blanketing, °C	252	257	257

where t_m is the time of peak power.

10.5 Design Basis Accident

The Design Basis Accident for the HEU core was determined[4] to be the melting and release of the fission products from one fuel assembly into the containment atmosphere. This accident was assumed to occur during a fuel transfer operation in which an irradiated fuel assembly was being moved from the core to the fuel storage area using a shielded transfer cask. Fuel assemblies are not normally discharged from the reactor until at least 12 hours after reactor shutdown. This ensures that sufficient fission product decay heat has been removed from the assembly and that the surface temperature of the fuel plates will not reach 450°C when the assembly is moved into the cask.

In spite of administrative controls, it is conceivable that a fuel assembly could be withdrawn from the reactor prior to a 12 hour cooldown period. Some or all of the fuel plates within the assembly could then melt and release some of their fission products into the containment atmosphere.

The source term for evaluating the radiological consequences of this accident was obtained[4] by assuming that an HEU fuel assembly with equilibrium burnup was removed from the core before the 12 hour cooldown period. All of the plates in the fuel assembly melt and the isotopes of iodine, krypton, and xenon were released to the containment. The methodology for the dose calculations and the results are shown in Ref. 4. The limiting dose is the thyroid dose from the iodine isotopes.

Since the HEU and LEU cores operate at 5 MW, neutron flux levels and equilibrium concentrations of iodine, xenon, and krypton will be about the same in the two cores. Burnup calculation results shown in Section 5.3 concluded that the lifetime of the LEU core will be comparable to but probably less than that of the HEU core. As a result, concentrations of the other fission products in LEU fuel assemblies will be the same or less than those in HEU fuel assemblies. The exception is that the LEU assembly will contain larger concentrations of plutonium isotopes. Reference 31 contains a detailed analysis comparing the radiological consequences of a hypothetical accident in a generic 10 MW reactor using HEU and LEU fuels. This analysis concluded that the buildup of plutonium in discharge fuel assemblies with ^{235}U burnup of over 50% does not significantly increase the radiological consequences over those of HEU fuel. Because fission product concentrations in the GTRR HEU and LEU cores are expected to be comparable, the thyroid dose shown in Ref. 4 will be the limiting dose for both cores.

11. FUEL HANDLING AND STORAGE

Three Technical Specifications apply to the handling and storage of fuel assemblies. The objective of these specifications is to prevent inadvertent criticality outside of the reactor vessel and to prevent overheating of irradiated fuel assemblies.

Irradiated fuel assemblies are stored in aluminum racks fastened to the side walls of a light water pool. There is one rack along each of the two walls and each rack can accomodate up to 20 assemblies in a linear array. The center-to-center spacing of the assemblies is six inches and the separation between assemblies is about three inches.

A systematic nuclear criticality assessment[32] been done for infinite-by-infinite arrays of fresh LEU fuel assemblies with ^{235}U contents between 225 and 621 grams using the ORR fuel storage rack spacing specifications[33] of 0.7 inch assembly separation and 6.8 inch row separation. An assembly similar to the GTRR LEU assembly with a ^{235}U content of 225 grams gave a k_{eff} of 0.72, well below the maximum k_{eff} of 0.85 needed to ensure an adequate margin below criticality for storage of irradiated fuel assemblies. The GTRR storage configuration discussed above will have k_{eff} less than 0.72.

Calculations[1] with HEU fuel assemblies have shown that four unirradiated fuel assemblies cannot achieve criticality. Calculations of HEU and LEU cores shown in Section 5.2 indicate that a grouping of four LEU assemblies will be less reactive than the same configuration of HEU assemblies. Thus, the current specification that no more than four unirradiated fuel assemblies shall be together in any one room outside the reactor, shipping container, or fuel storage racks will also hold for the LEU assemblies.

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REFERENCES

1. Safety Analysis Report for the 5 MW Georgia Tech Research Reactor, December 1967. Final Safeguards Report for the Georgia Tech Research Reactor, February 1963.

2. Appendix A to Facility License No. R-97: Technical Specifications for the Georgia Tech Research Reactor, Docket No. 50-160, June 6, 1974.

3. Letter, R. S. Kirkland, GTRR Reactor Supervisor, to USAEC, October 22, 1971.

4. Letter, R. S. Kirkland, GTRR Reactor Supervisor, to USAEC, June 23, 1972.

5. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Evaluation of Low-Enriched Uranium Silicide-Aluminum Dispersion Fuel for Use in Non-Power Reactors", NUREG-1313, July 1988.

6. M. M. Bretscher and J. L. Snelgrove, "Comparison of Calculated Quantities with Measured Quantities for the LEU-Fueled Ford Nuclear Reactor," Proc. International Meeting on Research and Test Reactor Core Conversions from HEU to LEU Fuel, Argonne National Laboratory, Argonne, IL, November 8-10, 1982, ANL/RERTR/TM-4, CONF-821155, pp. 397-425 (1983).

7. M. M. Bretscher, "Analytical Support for the Whole-Core Demonstration at the ORR," Proc. 1986 International Meeting on Reduced Enrichment for Research and Test Reactors, Gatlinburg, TN, November 3-6, 1986, ANL/RERTR/TM-9 ,CONF-861185, pp. 287-301 (1988).

8. R. J. Cornella and M. M. Bretscher, "Comparison of Calculated and Experimental Wire Activations," Proc. 1986 International Meeting on Reduced Enrichment for Research and Test Reactors, Gatlinburg, TN, November 3-6, 1986, ANL/RERTR/TM-9, CONF-861185, pp. 302-309 (1988).

9. M. M. Bretscher, J. L. Snelgrove, and R. W. Hobbs, "The ORR Whole-Core LEU Fuel Demonstration", Trans. Am. Nucl. Soc. 56, 579-581 (1988).

10. M. M. Bretscher and J. L. Snelgrove, “The Whole-Core U_3Si_2 Fuel Demonstration in the 30-MW Oak Ridge Research Reactor”, ANL/RERTR/TM-14, July 1991.

11. IAEA Guidebook on Research Reactor Core Conversion from the Use of Highly Enriched Uranium to the Use of Low Enriched Uranium Fuels, Addendum on Heavy Water Moderated Reactors, IAEA-TECDOC-324, Appendix F, pp. 182-250 (1985).

12. J. E. Matos, E. M. Pennington, K. E. Freese, and W. L. Woodruff, "Safety-Related Benchmark Calculations for MTR-Type Reactors with HEU, MEU, and LEU Fuels," paper included in IAEA Safety and Licensing Guidebook on Research Reactor Core Conversions from HEU to LEU Fuel, Volume 2, Analytical Verification, Draft #7, June 1985.

13. B.A. Zolotar et al., "EPRI-CELL Code Description," Advanced Recycle Methodology Program System Documentation, Part II, Chapter 5 (Oct. 1975).

14. E. M. Gelbard and R. E. Prael, "Monte Carlo Work at Argonne National Laboratory," in Proc. NEACRP Mtg. Monte Carlo Study Group, July 1-3, 1974, Argonne, Illinois, ANL-75-2 (NEA-CRP-L-118), Argonne National Laboratory , p. 201 (1975).

15. R. Blomquist, "VIM - A Continuous Energy Neutronics and Photon Transport Code", Proc. Topl. Mtg. Advances in Reactor Computations, Salt Lake City, Utah, March 28-31, 1983, p. 222, American Nuclear Society (1983).

16. K.L. Derstine, "DIF3D: A Code to Solve One-, Two-, and Three-Dimensional Finite-Difference Diffusion Theory Problems," ANL-82-64, April 1984.

17. GTRR Operation Log Book, June 10 and June 11, 1974.

18. ANL Internal Memorandum, K.E. Freese to J.E. Matos, “Monte Carlo Calculations for the GTRR Reference Core”, March 12, 1991.

19. R. P. Hosteny, “REBUS-2, Fuel Cycle Analysis Capability,” ANL-7721, October 1978.

20. “Measured Reactivity Worths of Shim-Safety Blades and Regulating Rod”, GTRR Work Order #90317, Record No. 188, 26 September 1990.

21. GTRR Operation Log Book, “Top Reflector Worth”, May 17, 1966.

22. K. Mishima and T. Shibata, “Thermal-hydraulic Calculations for KUHFR with Reduced Enrichment Uranium Fuel,” KURRI-TR-223 (1982), and K. Mishima, K. Kanda, and T. Shibata, “Thermal-hydraulic Analysis for Core Conversion to Use of Low-enriched Uranium Fuels in the KUR,” KURRI-TR-258 (1984).

23. Appendix A to Facility License No. R-97: Draft Technical Specifications for the Georgia Tech Research Reactor, Docket No. 50-160, May 26, 1972.

24. R. J. Weatherhead, “Nucleate Boiling Characteristics and the Critical Heat Flux Occurrence in Subcooled Axial-Flow Water Systems,” ANL-6674, 1963.

25. R. H. Whittle and R. Forgan, “A Correlation for the Minima in the Pressure Drop Versus Flow-Rate Curves for Subcooled Water Flowing in Narrow Heated Channels,” Nuclear Engineering and Design, Vol. 6, (1967) pp. 89-99.

26. “Guidebook on Research Reactor Core Conversion from the Use of Highly Enriched Uranium to the Use of Low Enriched Uranium Fuels”, IAEA-TECDOC-233 (1980) pp. 99-106.

27. A.E. Bergles and W.M. Rohsenow, “The Determination of Forced-Convection Surface-Boiling Heat Transfers,” Transactions of the ASME 86 (Series C - Journal of Heat Transfer), pp. 365-371 (August 1964).

28. C.F. Obenchain, "PARET - A Program for the Analysis of Reactor Transients," IDO-17282, Idaho National Engineering Laboratory (1969).

29. W. L. Woodruff, "Additional Capabilities and Benchmarking with the SPERT Transients for Heavy Water Applications of the PARET Code," Proc. XIIth International Meeting on

Reduced Enrichment for Research and Test Reactors, Berlin, 10.-14. September 1989., Konferenzen des Forschungs-zentrums Jülich(1991).

30. J. E. Grund, “Self-Limiting Excursion Tests of a Highly Enriched Plate-Type D2O-Moderated Reactor, Part I. Initial Test Series”, USAEC Report IDO-16891, Phillips Petroleum Co., July 12, 1963.

31. W.L. Woodruff, D.K. Warinner, and J.E. Matos, “A Radiological Consequence Analysis with HEU and LEU Fuels,” Proc. 1984 International Meeting on Reduced Enrichment for Research and Test Reactors, Argonne National Laboratory, Argonne, IL, October 15-18, 1984, ANL/RERTR/TM-6, CONF-8410173, pp. 472-490 (July 1985).

32. R.B. Pond and J.E. Matos, “Nuclear Criticality Assessment of LEU and HEU Fuel Element Storage,” Proc. 1983 International Meeting on Reduced Enrichment for Research and Test Reactors, Japan Atomic Energy Research Institute, Tokai, Japan, October 24-27, 1983, JAERI-M 84-073, pp. 416-425 (May 1984).

33. J.T. Thomas, “Nuclear Criticality Assessment of Oak Ridge Research Reactor Fuel Element Storage,” ORNL/CSD/TM-58, Oak Ridge National Laboratory (1978).

34. W.L. Woodruff, “Evaluation and Selection of Hot Channel (Peaking) Factors for Research Reactor Applications,” Proc. X International Meeting on Reduced Enrichment for Research and Test Reactors, Buenos Aires, Argentina, September 28 - October 1, 1987, pp. 443-452.

ATTACHMENT 1

ISOTHERMAL REACTIVITY CHANGE COMPONENTS FOR AN HEU CORE WITH 17 FRESH FUEL ASSEMBLIES

The purpose of this attachment is to analyze the components of the reactor isothermal temperature coefficient for the heavy water in various regions of the reactor tank. The calculations were done for an HEU core with 17 fresh fuel assemblies. Reactivity component values for heavy water outside of the fuel assemblies are expected to be very similar in LEU cores. Reactivity coefficients for the fuel and coolant shown in Table 5 of Section 5.5 are also very similar in HEU and LEU cores.

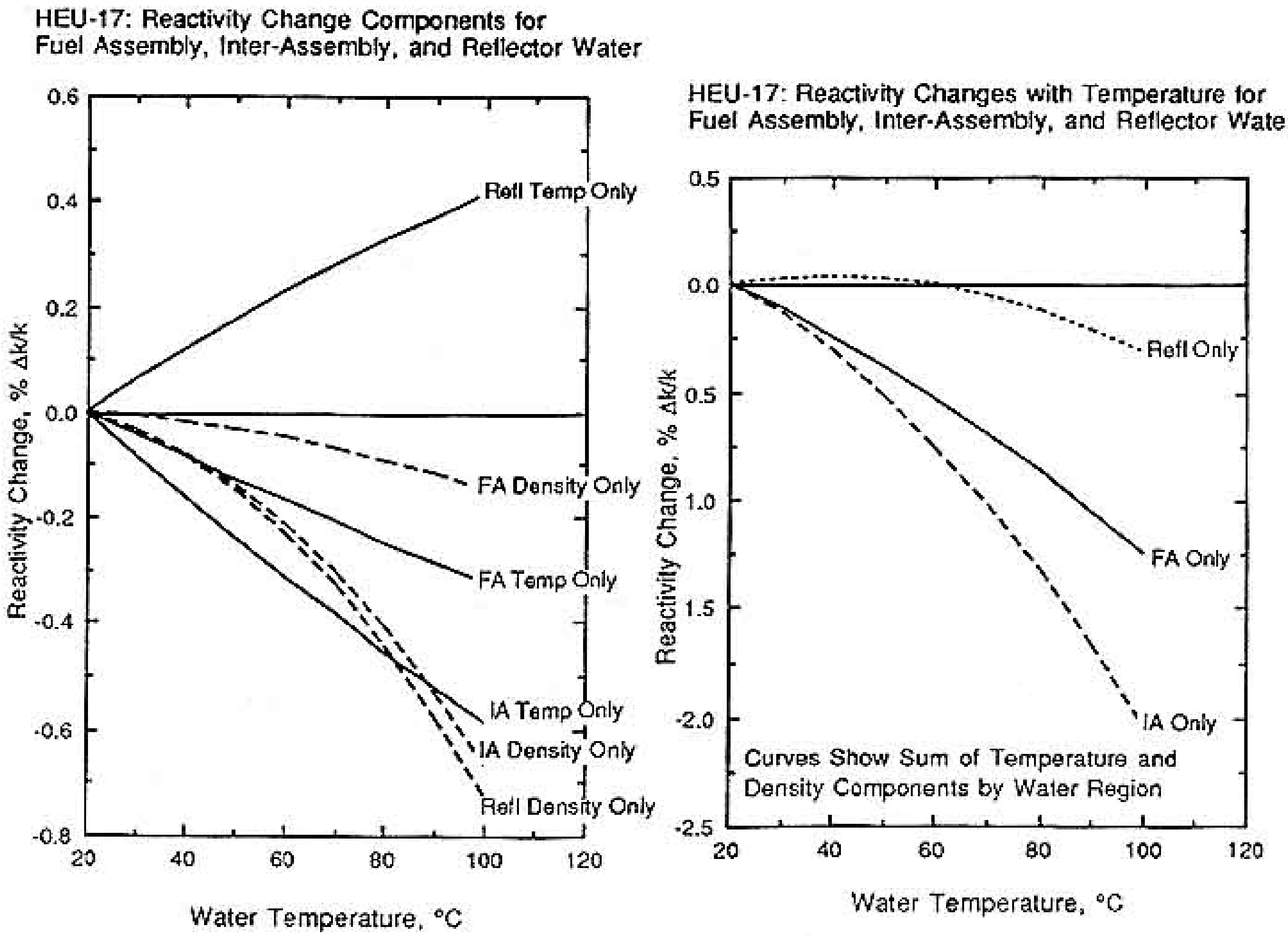
The reactor was divided into three regions: (1) the heavy water inside the fuel assemblies, (2) the heavy water between fuel assemblies, and (3) the heavy water reflector. On the outer edges of the core, a heavy water thickness equal to one-half the water thickness between fuel assemblies was included as part of the inter-assembly water. The remaining heavy water in the tank is referred to as the reflector. Calculations were performed by separately changing the water temperature and density in each region while holding the water in the other two regions at 23°C. Least-squares fits were then done to obtain reactivity values at intermediate temperatures.

Reactivity changes relative to 20°C for water temperature and density changes in each region are shown in the attached figure. Increasing the heavy water temperature and decreasing its density in the fuel assemblies and between fuel assemblies results in negative reactivity changes for both the temperature and density components. In the reflector, the water density component is negative, but the water temperature component is positive. Combined temperature and density effects for each heavy water region show that reactivity changes with increasing water temperature are negative for the fuel assembly and inter-assembly water. In the reflector, net reactivity changes are slightly positive for heavy water temperatures up to about 60°C and then become negative with further increases in temperature.

The sum of the temperature and density components over the three heavy water regions is negative for the entire temperature range between 20°C and 100°C. A direct calculation of the isothermal temperature coefficient in which all changes were made simultaneously gave results which are in good agreement with those obtained by summing the various components.

Figure 1-1. Calculated Reactivity Changes (in % Δk/k) with Temperature for a Fresh GTRR HEU Core with 17 Fresh Fuel Assemblies

FA = Fuel Assembly Water; IA = Inter-Assembly Water; Refl = Reflector Water



ATTACHMENT 2

ENGINEERING UNCERTAINTY FACTORS

This attachment addresses the engineering uncertainty factors (or hot channel factors) that were used to compute the thermal-hydraulic safety limits, safety margins, and safety system trip settings in HEU and LEU cores with 14 fuel assemblies. The rationale for choosing these factors and the method used to combine them are outlined along with a summary of results for the HEU and LEU cores.

The PLTEMP code[22] used in the ANL analyses allows for introduction of three separate engineering hot channel factors as they apply to the uncertainty in the various parameters (as opposed to a single lumped factor). The three hot channel factors are:

F_q for uncertainties that influence the heat flux q
F_b for uncertainties in the temperature rise or enthalpy change in the coolant
F_h for uncertainties in the heat transfer coefficient h.

The code also allows introduction of nuclear peaking factors for the radial, F_r, and axial, F_z, distributions of the heat flux.

While there is no generally accepted method for the selection of hot channel factors, these factors are normally a composite of sub-factors, and the sub-factors can be combined either multiplicatively, statistically [$F_b = 1 + \sqrt{\sum(1 - f_{bi})^2}$], or as a mixture of the two. A detailed description of methods for calculating hot channel factors is contained in Ref. 34. The multiplicative method of combining the sub-factors is very conservative and somewhat unrealistic. The statistical method recognizes that all of these conditions do not occur at the same time and location.

The engineering uncertainty factors that were combined multiplicatively and used by Georgia Tech in analyses[3,23] of the HEU core are shown in Table 2-1. The factors that were combined statistically and used by ANL for calculations of the HEU and LEU cores are shown in Table 2-2.

Key thermal-hydraulic safety limits and safety margins for the HEU and LEU cores computed using the Georgia Tech factors and the ANL factors are compared in Table 2-3. Results for the HEU core obtained using ANL's statistical treatment of the engineering uncertainty factors agree well with the analyses performed by Georgia Tech. Except for the reactor power limit, data for the LEU core are comparable to or more conservative than those for the HEU core. An LEU core power limit of 10.6 MW based on the flow instability criterion is considered to be adequate.

Table 2-1. GTRR-HEU Engineering Uncertainty Factors[23]

Uncertainty	F_q	F_b	F_h
Equivalent Diameter	-	1.09	-
Fuel Distribution	1.03	1.03	-
Axial Flux Peaking	1.19	-	-
Power Level Measurement	1.03	1.03	-
Flow Distribution - Plenum	-	1.07	-
Flow Distribution - Channel	-	1.10	-
Multiplicative Combination	1.26	1.36	1.0

Table 2-2. ANL-HEU and ANL-LEU Engineering Uncertainty Factors

	ANL-HEU Factors			ANL-LEU Factors		
Uncertainty	F_q	F_b	F_h	F_q	F_b	F_h
Fuel Meat Thickness [a]	1.04	-	-	1.04	-	-
^{235}U Loading	1.03 [b]	1.03 [b]	-	1.03 [c]	1.03 [c]	-
^{235}U Homogeneity	1.03 [d]	1.03 [d]	-	1.20 [e]	1.10 [e]	-
Coolant Channel Spacing	-	1.17 [f]	1.03 [f]	-	1.22 [g]	1.04 [g]
Power Level Measurement [d]	1.03	1.03	-	1.03	1.03	-
Calculated Power Density [h]	1.10	1.10	-	1.10	1.10	-
Coolant Flow Rate [h]	-	1.10	1.08	-	1.10	1.08
Heat Transfer Coefficient [h]	-	-	1.20	-	-	1.20
Statistical Combination	1.12	1.23	1.30	1.23	1.28	1.31
Multiplicative Combination	1.26	1.55	1.33	1.41	1.72	1.35

a Derived from fuel plate thickness specification of 50 ± 2 mils.
b Assumed to be the same as for the LEU plate.
c From LEU fuel plate loading specification of 12.5 ± 0.35 g ^{235}U.
d GTRR-HEU value from Table 2-1.
e From LEU plate fuel homogeneity specification.
f Computed based on coolant channel spacing of 106 ± 10 mils and fuel plate thickness specification of 50 ± 2 mils in HEU assembly (see Ref. 34 for calculation method).
g Computed based on coolant channel spacing of 89 ± 10 mils and fuel plate thickness specification of 50 ± 2 mils in LEU assembly (see Ref. 34 for calculation method).
h Assumed values.

The ANL factors for F_q and F_b were combined statistically using the relation $F = 1+ \sqrt{\sum(1 - f_i)^2}$.

The corresponding factor for F_h was obtained by statistically combining the factors for the coolant channel spacing and the coolant flow rate and multiplying the result by the factor for the heat transfer coefficient.

Table 2-3. Comparison of Key Thermal-Hydraulic Safety Parameters for HEU and LEU Cores wth 14 Fuel Assemblies.

Reactor Power Limits for a Maximum Inlet Temperature of 123°F

Reactor Coolant Flow, gpm	GTRR-HEU	ANL-HEU	ANL-LEU
	Reactor Power Level (MW) for DNB[23,24]		
760	5.5	5.7	5.3
1625	11.5	11.9	10.8
	Reactor Power Level (MW) for Flow Instability[25,26]		
760	5.3	5.1	5.0
1625	10.6	11.0	10.6

Thermal-Hydraulic Data with Min. Coolant Flow of 1625 GPM and Max. Inlet Temp. of 123°F.

	GTRR-HEU	ANL-HEU	ANL-LEU
Coolant Velocity, m/s	2.44	2.44	2.61
Friction Pressure Drop[1], kPa	10.9	11.0	15.0
Power/Plate[2], kW	21.2	21.2	18.8
Outlet Temperature of Hottest Channel, °F	157	154	156
Peak Clad Surface Temperature, °F	219	229	224
Minimum DNBR[3]	2.29	2.37	2.17
Limiting Power Based on Min. DNBR, MW	11.5	11.9	10.8
Flow Instability Ratio (FIR)[4]	2.12	2.19	2.11
Limiting Power Based on FIR, MW	10.6	11.0	10.6

[1] Pressure drop across active fuel only.

[2] Assuming 95% of power deposited in fuel.

[3] Using modified Weatherhead Correlation[23,24] for DNB.

[4] Using Whittle-Forgan Correlation[25,26] with η = 25.

Safety Limits on Reactor Inlet and Outlet Temperatures.

	GTRR-HEU	ANL-HEU		ANL-LEU	
Parameter	DNB	DNB	Flow Inst.	DNB	Flow Inst.
Limiting Reactor Inlet Temp., °F	172	175	172	171	170
Ave. Coolant Temp. Rise across Core, °F	16	17	17	17	17
Limiting Reactor Outlet Temp., °F	188	192	189	188	187

Margins to D_2O Saturation Temperature and ONB

Parameter	GTRR-HEU	ANL-HEU	ANL-LEU
Thermal Power, MW	5.0	5.0	5.0
Reactor Coolant Flow, gpm	1800	1800	1800
Reactor Inlet Temp., °F	114	114	114
Æ T_{sub}, °F	8	5	11
Margin to ONB[1]	-	1.34	1.44
Limiting Power Based on ONB, MW	-	6.7	7.2

[1] Using the Bergles and Rohsenow correlation[27].

Power Levels and Inlet Temperatures for Zero Subcooling at a Coolant Flow of 1800 GPM

Parameter	GTRR-HEU	ANL-HEU	ANL-LEU
Thermal Power, MW	5.45	5.35	5.6
Reactor Inlet Temp., °F	114	114	114

Thermal Power, MW	5.0	5.0	5.0
Reactor Inlet Temp., °F	123	122	128