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Pt. 1 and 2

Conversion Analyses for the VR-1 Reactor

RERTR Program
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Part I

Criticality Calculations for the VR-1 Reactor with IRT-3M HEU Fuel and IRT-4M LEU Fuel ANL Independent Verification Results

N. A. Hanan and J. E. Matos

Part II

Transient Analyses for the VR-1 Reactor with IRT-3M HEU Fuel and IRT-4M LEU Fuel

J.A. Stillman, A.P. Olson, P.L. Garner, and J.E. Matos

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Part I
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ANL Independent Verification Results

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Introduction

At the request of the Czech Technical University (CTU) in Prague, ANL has performed independent verification calculations using the MCNP Monte Carlo code for three core configurations of the VR-1 reactor: a current core configuration B1 with HEU (36%) IRT-3M fuel assemblies and planned core configurations C1 and C2 with LEU (19.7%) IRT-4M fuel assemblies. Details of these configurations were provided to ANL by CTU.

For core configuration B1, criticality calculations were performed for two sets of control rod positions provided to ANL by CTU. For core configurations C1 and C2, criticality calculations were done for cases with all control rods at the top positions, all control rods at the bottom positions, and two critical states of the reactor for different control rod positions. In addition, sensitivity studies for variation of the ^{235}U mass in each fuel assembly and variation of the fuel meat and cladding thicknesses in each of the fuel tubes were done for the C1 core configuration.

The reactivity worth of the individual control rods was calculated for the B1, C1, and C2 core configurations. Finally, the reactivity feedback coefficients, the prompt neutron lifetime, and the total effective delay neutron fraction were calculated for each of the three cores.

Core Configuration B1 With HEU (36%) IRT-3M Fuel Assemblies

A model for core configuration B1 of the VR-1 critical facility was received from CTU. This model was analyzed by ANL and found to be very well constructed.

Calculations were performed by ANL for the two sets of control rods positions provided by CTU that are required to obtain critical configurations for the Core B1, shown in Figure 1:

A) **Set #1**: Control Rods:
   B1/B2/B3=680 mm; E1=500 mm; R1=369 mm; R2=532 mm
B) **Set # 2**: Control Rods:
   B1/B2/B3=680 mm; E1=500 mm; R1=393 mm; R2=496 mm

The results of these calculations are provided in Table 1, followed by explanations for the different cases.

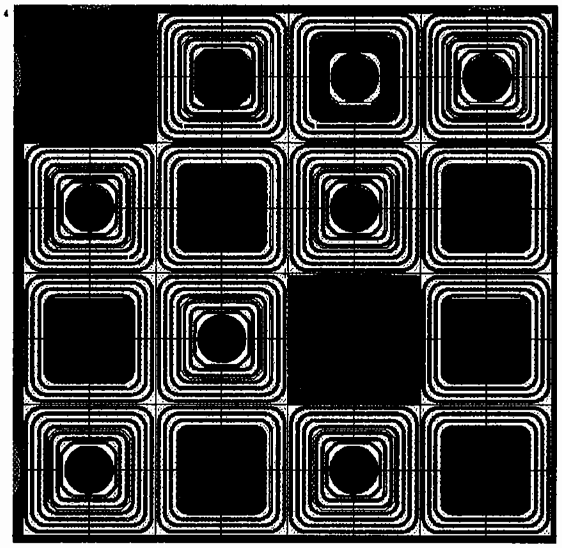


Figure 1. Core Configuration B1 with HEU (36%) IRT-3M FA.

Table 1. Results of Independent ANL MCNP Calculations Core Configuration B1 Using IRT-3M FA

	Set # 1 for CR **k-eff**	**Set # 2 for CR** **k-eff**
Base Case: Model of VR-1 as Received from CTU	1.00537 ± 0.00023	1.00496 ± .00021
Case 1: CR Modified to Include the SS below the Cd	1.00414 ± .00020 (-0.122%)*	1.00361 ± .00021 (-0.134%)*
Case 2: Added U-234 to Fuel Meat	1.00188 ± .00018 (-0.224%)*	1.00174 ± .00017 (-0.186%)*
Case 3: Changed Clad Material from Al to SAV-1 and Included 4 cm of SAV-1 tubes to bottom and top of fuel Assembly	1.00152 ± .00019 (-0.036%)*	1.00116 ± .00011 (-0.058%)*
Case 4: Changed radii of the corners of Fuel Elements Based on information received from the RF (fuel density modified to obtain same masses as in model received from VR-1)	0.99928 ± .00010 (-0.224%)*	0.99911 ± .00009 (-0.205%)*

* Difference in reactivity caused by specified change in model.

- **Base Case:** Model of the VR-1 reactor, as received from CTU.
- **Case 1:** Model of the control rods was modified to include the 4.35 cm of stainless steel below the Cd poison.
- **Case 2:** Model used in **Case 1** modified to include the ^{234}U isotope in the fuel meat. Based on information from references 1 and 2, the concentration of ^{234}U in Russian enriched uranium is assumed to be 1.1 w/o of the contained ^{235}U. The concentration of ^{234}U may be slightly different from that assumed but more precise information is not available.
- **Case 3:** Model used in **Case 2** modified to: a) change the fuel element clad from Al to SAV-1; b) increase the length of the fuel elements (58.0 cm) by adding 4.0 cm of SAV-1 tubes to the bottom and top of the fuel elements.
- **Case 4:** Model used in **Case 3** modified to change the corners of the fuel elements to agree with information provided to ANL by Russian colleagues. The outside corners of the fuel elements are provided below:

Outside Corner Radius

Tube #	VR-1 Model	ANL Information
1 (Outer)	1.12	0.92 (cm)
2	1.05	0.84
3	0.98	0.76
4	0.91	0.68
5	0.84	0.60
6	0.77	0.52
7	0.70	0.44

Fuel meat densities in the different fuel elements were changed to keep the masses equal to those used in the model of the VR-1 reactor that was received from the CTU.

The model used in **Case 4** is the model that has been used at ANL for calculations using the IRT-3M fuel assemblies.

Core Configuration C1 With LEU (19.7%) IRT-4M Fuel Assemblies

This configuration is defined as "the essential preliminary core mainly for education and training purposes." The core contains nine 6-tube assemblies and eight 8-tube assemblies. Figure 2 shows a cross section of the core. The input file for this configuration was provided to CTU by ANL. The nominal dimensions of the fuel elements (meat/clad = 0.7 / 0.45 mm) and the nominal ^{235}U masses (200.5 g for the 4-tubes FA, 263.8 g for the 6-tubes FA, and 300.0 g for the 8-tubes FA) were used. In all calculations presented in this study the clad used was SAV-1 and the ratio between the masses of ^{234}U and ^{235}U was assumed to be 1.1%.

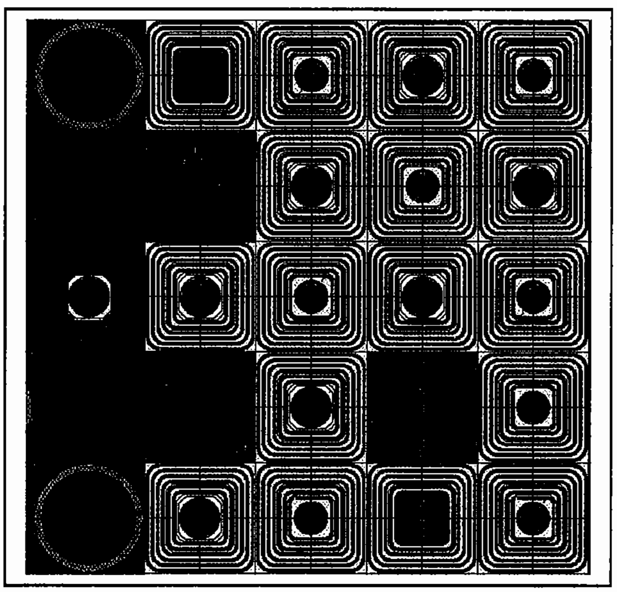


Figure 2. Core Configuration C1 with LEU IRT-4M FA.

The results for the requested calculations are presented below:

1. All control rods at the top positions (680 mm): k-eff = 1.00731 ± 0.00014

2. All control rods at the bottom positions (0 mm): k-eff = 0.92043 ± 0.00017

3. Finding two critical states (k-eff = 1.0) for different control rods positions:

   a) CR Positions:

      B1=B2=B3= 680 mm

      E1=E2=R1=R2= 500 mm

      k-eff = 0.99982 +/- 0.00015

   b) CR Positions:

      B1=B2=B3= 680 mm

      E1=E2= 500 mm

      R1= 425 mm

      R2= 575 mm

      k-eff = 1.00063 +/- 0.00013

4. Sensitivity Studies

   a) <u>Variation of the total ^{235}U mass in the FA</u>

   All the three cases below were analyzed for the CR at the top positions (680 mm), and using the nominal thicknesses for the fuel elements (meat/clad = 0.7 / 0.45 mm).

   1. Base Case: All nominal masses as provided above:
      k-eff = 1.00731 +/- 0.00014

   2. All FA with the maximum ^{235}U masses (210.5 g for the 4-tubes FA, 276.9 g for the 6-tubes FA, and 315.0 g for the 8-tubes FA)
      k-eff = 1.01392 +/- 0.00013

   3. All FA with the minimum ^{235}U masses (190.5 g for the 4-tubes FA, 250.7 g for the 6-tubes FA, and 285.0 g for the 8-tubes FA)
      k-eff = 1.00014 +/- 0.00014

   b) <u>Variation of the clad thickness</u>

   All the three cases below were analyzed for the CR at the top positions (680 mm), and using the nominal U-235 masses for the fuel assemblies.

   1. Base case: Nominal meat and clad thicknesses (meat/clad = 0.7 / 0.45 mm).
      k-eff = 1.00731 +/- 0.00014

   2. Meat thickness equal to 0.6 mm and clad thickness equal to 0.5 mm.
      k-eff = 1.00801 +/- 0.00013

3. Meat thickness equal to 1.0 mm and clad thickness equal to 0.3 mm (minimum clad thickness).

k-eff = 1.00771 +/- 0.00014

Core Configuration C2 With Leu (19.7%) IRT-4M Fuel Assemblies

This configuration is defined "mainly for R&D purposes". The core contains ten 6-tube assemblies and ten 8-tube assemblies. In the center of the core is located large graphite unit occupying four positions. Figure 3 shows a cross section of the core. The nominal dimensions of the fuel elements (meat/clad = 0.7 / 0.45 mm) and the nominal ^{235}U masses (200.5 g for the 4-tubes FA, 263.8 g for the 6-tubes FA, and 300.0 g for the 8-tubes FA) were used. In all calculations presented in this study the clad used was SAV-1 and the ratio between the masses of ^{234}U and ^{235}U was assumed to be 1.1%.

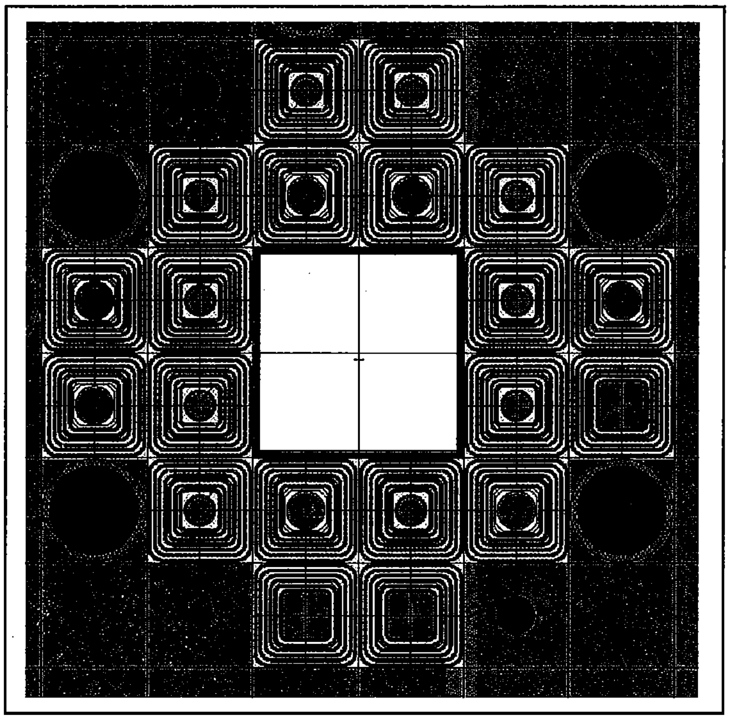


Figure 3. Core Configuration C2 with LEU IRT-4M FA

The results for the requested calculations are presented below:

1. All control rods at the top positions (680 mm): k-eff = 1.01380 ± 0.00013

2. All control rods at the bottom positions (0 mm): k-eff = 0.94615 ± 0.00017

3. Finding two critical states (k-eff = 1.0) for different control rods positions:

   a) CR Positions:

      B1=B2=B3= 680 mm

      E1= 500 mm

E2= 0 (bottom)
R1=R2= 375 mm
k-eff = 1.00042 +/- 0.00015

b) CR Positions:
B1=B2=B3= 680 mm
E1= 500 mm
E2= 0 (bottom)
R1= 300 mm
R2= 450 mm
k-eff = 1.00054 +/- 0.00014

Reactivity Worth For The Individual Control Rods

The reactivity worth of all the individual control rods was calculated by ANL for core configurations B1, C1, and C2. The results are shown in Table 2.

Table 2. Reactivity Worth (% Δk/k) of Control Rods

	IRT-3M FA Configuration B1	IRT-4M FA Configuration C1	IRT-4M FA Configuration C2
B1 rod	2.037 ± 0.028	1.468 ± 0.028	1.291 ± 0.023
B2 rod	2.073 ± 0.028	2.050 ± 0.028	1.227 ± 0.023
B3 rod	1.196 ± 0.027	1.568 ± 0.028	1.203 ± 0.023
R1 rod	0.659 ± 0.026	0.411 ± 0.026	0.542 ± 0.021
R2 rod	0.617 ± 0.025	0.887 ± 0.026	0.590 ± 0.020
E1 rod	1.015 ± 0.025	1.099 ± 0.027	0.861 ± 0.022
E2 rod	Not Applicable	0.955 ± 0.027	0.477 ± 0.023

The control rod worth was calculated by using the difference between the configuration with all control rods out and the configuration with the specific rod fully inserted. This was done because all these three core configurations have a small excess reactivity and operate with the rods B1/B2/B3 always withdrawn, and the other rods inserted only a small fraction of their poison.

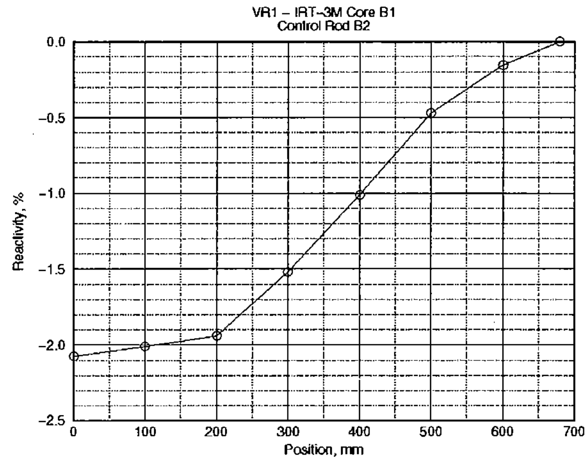
For the IRT-4M core configuration C1 calculations were also performed for a hypothetical case in which: a) Base case with control rods B1/B2/B3 withdrawn and all the other rods fully inserted; b) Same as case a) but the specific control rod for which the worth is to be determined is withdrawn. For all cases (control rods R1/R2/E1/E2) the worth of the rods increased (as expected) by less than 0.10%. Note that these calculations were performed just to show that the change in reactivity worth is very small. The results presented in Table 1 above are more appropriate and should be used.

Reactivity Worth Vs Position for Control Rod B2

Calculated reactivity worth versus rod position curves (S-Curves) for control rod B2 in core configurations B1, C1, and C2 are shown in Figure 4.

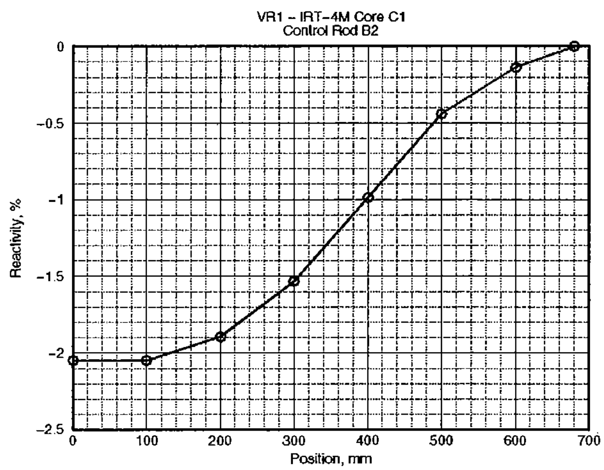
Core B1, Rod B2

Position	% Δk/k
680	0
600	-0.155
500	-0.471
400	-1.009
300	-1.517
200	-1.939
100	-2.010
0	-2.073



Core C1, Rod B2

Position	% Δk/k
680	0
600	-0.136
500	-0.439
400	-0.988
300	-1.533
200	-1.894
100	-2.050
0	-2.050



Core C2, Rod B2

Position	% Δk/k
680	0
600	-0.091
500	-0.284
400	-0.602
300	-0.894
200	-1.109
100	-1.201
0	-1.227

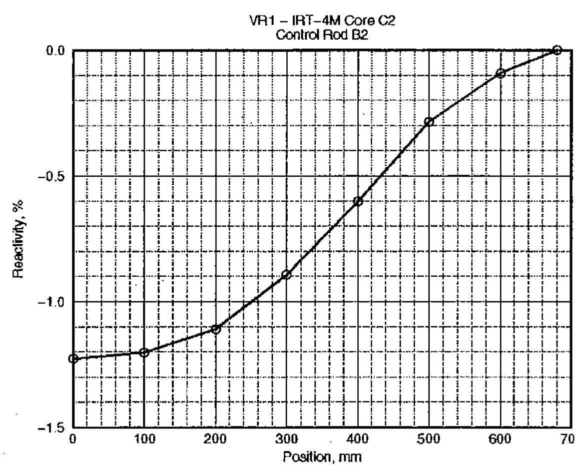


Figure 4. Reactivity Worth Vs Position for Control Rod B2 in Cores B1, C1, and C2

Kinetics Parameters And Reactivity Temperature Coefficients

The total effective delayed neutron fraction (β_{eff}), the prompt neutron lifetime (l_p), and the prompt neutron generation time (Λ) were calculated for the B1, C1, and C2 core configurations using the MCNP5 Monte Carlo code[3] and methods developed in Ref. 4. The results are shown in Table 3. Also shown in Table 3 are the moderator temperature reactivity coefficient, the fuel temperature (Doppler) coefficient, and the moderator density (void) coefficient, along with the uncertainties for each of the values. The coolant regions shown in pink in Figures 1-3 were included in the calculations of the reactivity temperature coefficients.

Table 3. Kinetics Parameters and Reactivity Temperature Coefficients for VR-1 HEU and LEU Cores Calculated Using the MCNP5 Code
(All shim and power rods inserted to the critical position)
(LANL XS provided with MCNP-5 used)

Parameter	HEU Core B1 Figure 1	LEU Core C1 Figure 2	LEU Core C2 Figure 3
Effective Delayed Neutron Fraction, β_{eff}	8.34e-3 [a]	8.20e-3 [e]	8.31e-3 [j]
Prompt Neutron Lifetime, l_p (μs)	44.5 [b]	41.0 [f]	53.4 [k]
Neutron Generation Time, Λ (μs)	44.5 [b]	41.0 [f]	53.4 [k]
Moderator Temperature Reactivity Coefficient, $\Delta\rho$(%)/°C:			
294 °K to 400 °K	-9.52e-3 [c]	-7.97e-3 [g]	-7.74e-3 [l]
400 °K to 600 °K	-1.002e-2 [c]	-8.76e-3 [g]	-8.06e-3 [l]
294 °K to 600 °K	-9.85e-3 [c]	-8.49e-3 [g]	-7.95e-3 [l]
Fuel Temperature (Doppler) Reactivity Coefficient, $\Delta\rho$(%)/°C:			
294 °K to 500 °K	-1.65e-3 [d]	-2.11e-3 [h]	-2.35e-3 [m]
500 °K to 600 °K	-8.16e-4 [d]	-1.68e-3 [i]	-1.84e-3 [n]
294 °K to 600 °K	-1.37e-3 [d]	-1.97e-3 [h]	-2.19e-3 [m]
Moderator Density (Void) reactivity Coefficient, $\Delta\rho$(%)/(% of Void):			
0 to 5%:	-2.41e-1 [c]	-2.94e-1 [g]	-3.28e-1 [l]
5% to 10%:	-2.57e-1 [c]	-3.29e-1 [g]	-3.50e-1 [l]
0 to 10%:	-2.49e-1 [c]	-3.11e-1 [g]	-3.39e-1 [l]

HEU Core B1 Uncertainties: a < 1.5%; b <3%; c <0.7%; d <3%
LEU Core C1 Uncertainties: e < 1.1%; f < 4%; g < 1%; h <2%; i < 4%
LEU Core C2 Uncertainties: j < 1%; k < 3%; l < 1%; m < 1.5%; n < 4%

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2. Yu. V. Petrov, A. N. Erykalov, and M. S. Onegin, "Accuracy of WWR-M Criticality Calculations with Code MCU-RFFI," Proceedings of the 1999 RERTR International Meeting, Budapest, Hungary, 3-8 October, 1999.
3. X-5 Monte Carlo Team, "MCNP — A General Monte Carlo N-Particle Transport Code, Version 5", LA-UR-03-1987, Los Alamos National Laboratory (April 2003).
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Part II

Transient Analyses for the VR-1 Reactor with IRT-3M HEU Fuel and IRT-4M LEU Fuel

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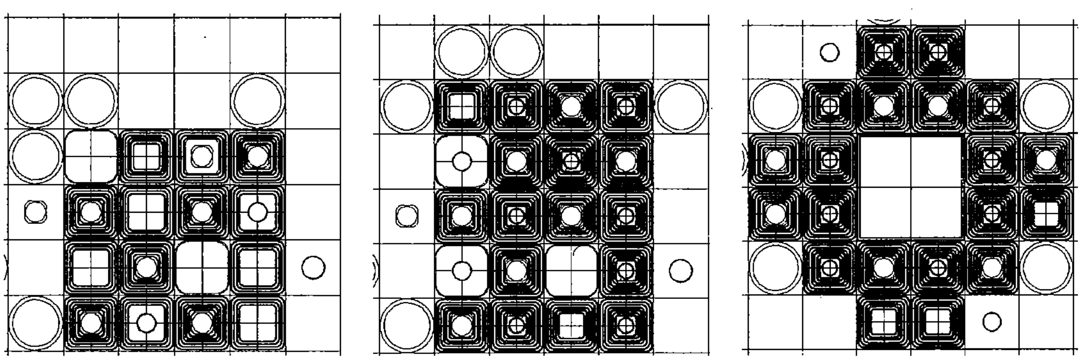
Introduction

The VR-1 is a 1 kW critical facility at the Czech Technical University (CTU), currently loaded with IRT-3M HEU fuel assemblies. The core may be converted to LEU IRT-4M fuel assemblies in 2005. As part of the conversion analyses, both unprotected and protected transient analyses have been performed, including step and ramp insertions of reactivity. The envelope of transient conditions that was used by CTU in previous safety analyses for use of HEU (36%) IRT-2M fuel assemblies and conversion to HEU (36%) IRT-3M fuel assemblies was used as a basis for the cases studied.[1,2]

In the current work, transient calculations were performed using the PARET v5 code[3]. One HEU and two LEU core configurations specified by CTU are illustrated in Figure 1. Table 1 shows the reactivity coefficients and kinetics parameters that were used in the transient calculations for the three VR-1 core configurations. The reactivity feedback coefficients, the prompt neutron lifetime, and the total delayed neutron fraction were derived from Table 3 in Part I of this report, and were calculated using the MCNP5 Monte Carlo code.[4] The six delayed neutron group data were calculated using the VARI3D diffusion theory code.[5]



B1 Core, HEU (36%)	C1 Core, LEU	C2 Core, LEU
14 IRT-3M fuel bundles	17 IRT-4M fuel bundles	20 IRT-4M fuel bundles
3.73 kg U-235	4.77 kg U-235	5.64 kg U-235

Figure 1. HEU and LEU Core Configurations for VR-1 Critical Facility.

A simple plate geometry was assumed in the transient calculations; peak and average channels in the core configuration were modeled. The axial power density profile for the peak channel in each of the three cores was derived from the MCNP analyses described in Part I of this report. Since the average channel will not be limiting with regard to fuel or clad temperature, the power density profile was assumed to have the same shape as the peak channel, but normalized to 1.0. The power density profiles for the HEU and the two LEU core configurations are shown in Figure 2.

Table 1. Reactivity Coefficients and Kinetics Parameters Used in the Transient Analyses. (Unless otherwise indicated, data were calculated with the MCNP5 code. See Part I, Table 3.)

		HEU Core, Configuration B1		**LEU Core, Configuration C1**		**LEU Core, Configuration C2**	
Average Power Density at 1 kW, W/cm^3		0.229		0.107		0.091	
Peak-to-Average Power Density		2.651		3.077		2.152	
Moderator Density (Void) Coefficient, % $\Delta\rho$/%void		-0.241[a]		-0.294[a]		-0.328[a]	
Moderator Temperature Coefficient, % $\Delta\rho$/°C		-9.52E-03[b]		-7.97E-03[b]		-7.74E-03[b]	
Fuel Temperature (Doppler) Coefficient, %$\Delta\rho$/°C		-1.65E-03[c]		-2.11E-03[c]		-2.35E-03[c]	
Prompt Neutron Generation Time (μs)		44.5		41.0		53.4	
Effective Delayed Neutron Fraction, β_{eff}		8.34E-03		8.20E-03		8.31E-03	
Six-Group Delayed Neutron Data	Group	β_i/β_{eff}[d]	$\lambda_i(s^{-1})$[d]	β_i/β_{eff}[d]	$\lambda_i(s^{-1})$[d]	β_i/β_{eff}[d]	$\lambda_i(s^{-1})$[d]
	1	0.0353	1.33E-02	0.0351	1.33E-02	0.0351	1.33E-02
	2	0.1810	3.27E-02	0.1805	3.27E-02	0.1805	3.27E-02
	3	0.1754	1.21E-01	0.1750	1.21E-01	0.1750	1.21E-01
	4	0.3830	3.03E-01	0.3832	3.03E-01	0.3831	3.03E-01
	5	0.1589	8.50E-01	0.1595	8.50E-01	0.1595	8.50E-01
	6	0.0664	2.85E+00	0.0667	2.86E+00	0.0667	2.86E+00

[a]MCNP5 with LANL σ's, Rods at critical position. Only the light-water coolant between fuel tubes is voided, from 0% to 5%.
[b]MCNP5 with LANL σ's, Rods at critical position. Temperature of coolant between fuel tubes increased from 294 to 400°K.
[c]MCNP5 with LANL σ's, Rods at critical position. Fuel temperature increased from 294 to 500°K.
[d]Delayed neutron group data calculated with VARI3D, using broad group data generated by WIMS-ANL, and ENDF-VI delayed neutron data.

SAV-1 Aluminum Alloy

The SAV-1 aluminum alloy used by the Novosibirsk Chemical Concentrates Plant to manufacture cladding for IRT-3M and IRT-4M fuel assemblies has the following composition (wt.%): Al: 98.55 – 97.6; Si: 0.7 – 1.2; Mg: 0.45 – 0.9; Fe: 0.2; Cu: 0.1. Based on comparison of this composition with Western aluminum alloys, we estimate that incipient melting of the SAV-1 aluminum alloy would occur at a temperature in the range of 550 – 580 °C.

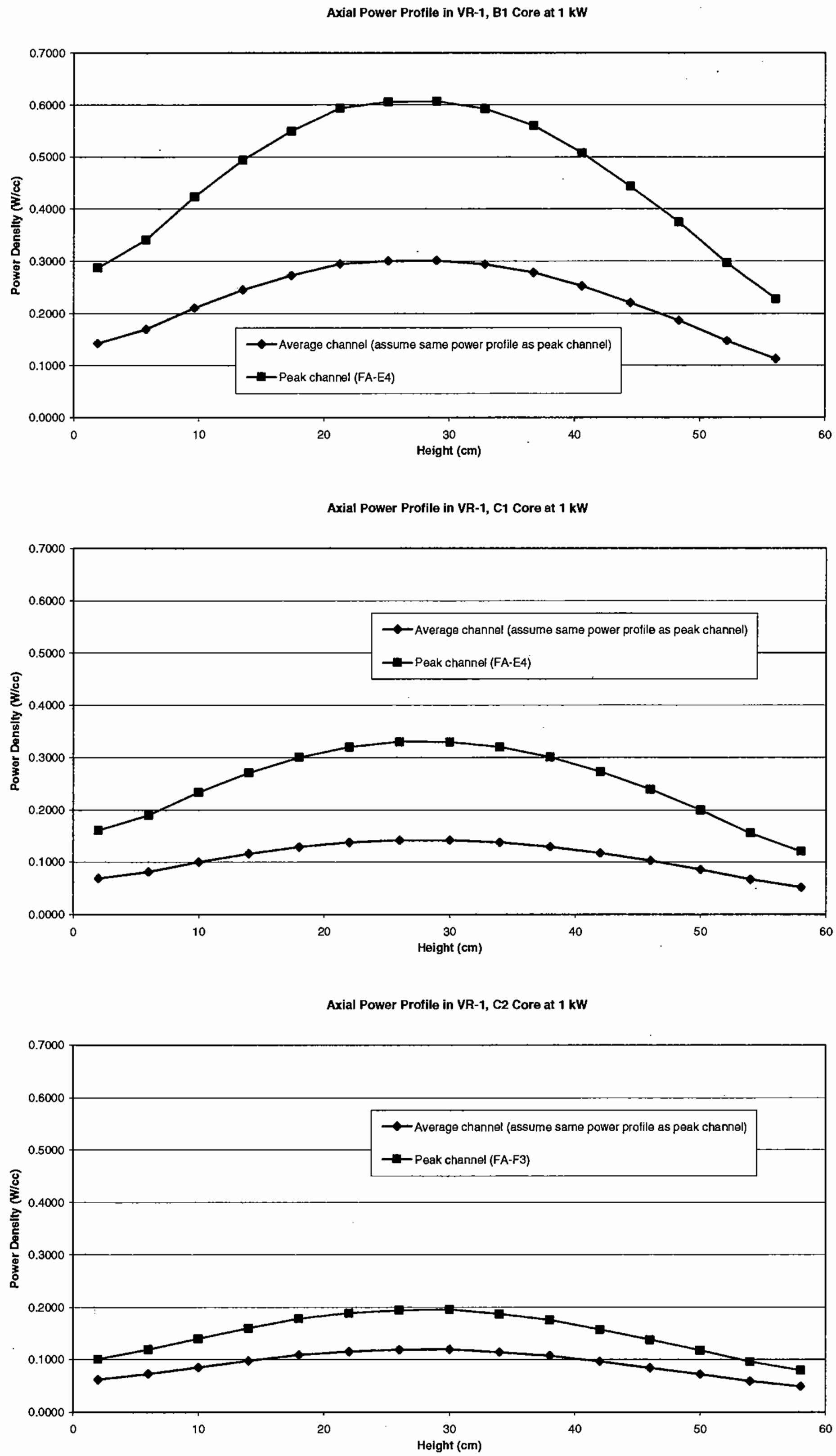


Figure 2. Axial Power Profile in VR-1 at 100 kW.

Transient Analyses

Calculations were performed for the HEU core configuration (B1) and the two LEU core configurations (C1 and C2) specified by CTU. The transient analyses performed by ANL adopted the same envelope of limiting events defined by CTU in Ref. 1 for analysis of core configurations consisting of IRT-2M and IRT-3M fuel assemblies containing HEU fuel. The transient identifier, the power level from which they were initiated, and the reactivity insertion for each transient are shown below.

Transient Identifier	Initial Power	Reactivity Insertion
S1	100 W	0.7 β_{eff} jump
L1	0.1 W	1.5 β_{eff} per 5 s
L2	0.1 W	2.5 β_{eff} per 5 s
L3	100 W	2.5 β_{eff} per 25 s
L4	100 W	0.7 β_{eff} per 7 s

The results are discussed in two parts because the S1, L1, and L4 transients were done without intervention of the control system, while the L2 and L3 transients require intervention of the control system to prevent melting of the fuel cladding.

Since the PARET code is capable of modeling steam generation and two-phase flow, the feedback mechanism associated with coolant voiding was included in the analyses. These conditions are difficult to model and the numerical computations can become unstable when there is large-scale boiling of the coolant that can occur in unprotected transients. The code failed to model the long-term transient behavior under these conditions and the calculations were terminated by a runtime error at some point well after the suppression of the initial power spike. Even so, PARET has been successfully benchmarked[6] against experimental results for unprotected transients (SPERT tests) through the time of the peak fuel temperature.

Additional preliminary calculations were done using the RELAP5/MOD3.2 code[7] to determine whether the initial power peak for each transient was totally suppressed by the temperature and void reactivity feedbacks after the initial peak power was reached or whether the reactivity insertion was so strong that additional power peaks occurred after suppression of the initial peak. If additional power peaks occurred, the temperature of the cladding could be driven above its melting temperature.

S1, L1, and L4 Transients

Results obtained using the PARET code for the S1, L1, and L4 transients for the three core configurations without intervention of the control system are shown in Table 2 and in Figures 3-5. The length of the transients that were modeled in each of the cases is due to limitations of the PARET code discussed above when there is massive voiding (> 99 % locally) in the hot channel. For each of these three transients, preliminary calculations using the RELAP5/MOD3.2 code showed that the initial power peak was totally suppressed by the temperature and void reactivity feedbacks and that no subsequent power peaks occurred.

For the S1, L1, and L4 cases in Table 2, the time to peak power in the B1, C1, and C2 cores are nearly the same. The peak power produced during the transient is consistently higher in the LEU cores before suppression of the power spike from temperature and void reactivity feedbacks, even though the reactivity coefficients in Table 1 are larger in magnitude for the LEU cores. The main reason is that the power density in the fuel is a factor of two or more lower in the C1 and C2 LEU cores relative to the B1 HEU core because the LEU cores contain a larger number of fuel assemblies and the fuel meat in IRT-4M fuel is thicker than in IRT-3M fuel.

The S1 and L4 transients are very similar in the three core configurations and result in peak cladding temperatures around 120 °C, which is far below the 550 - 580 °C temperature for incipient melting of the SAV-1 aluminum alloy cladding.

The L1 transient is more challenging because the reactivity insertion is more than one β_{eff}. However, the peak temperatures reached in the fuel and in the cladding are predicted to be less than about 290 °C, which is far below the 550 – 580 °C temperature for incipient melting of the SAV-1 aluminum alloy cladding.

L2 and L3 Transients

Preliminary calculations using the RELAP5/MOD3.2 code for the L2 and L3 transients provide strong indications that these large reactivity insertions of 2.5 β_{eff} would not be suppressed by the temperature and void reactivity feedback coefficients. Additional power peaks after the initial power peak would lead to melting of some number of fuel assemblies in the B1, C1, and C2 cores if there were no intervention of the control system. Consequently, the calculations performed by ANL for L2 and L3 transients included a scram of the safety rods, excluding the safety rod with maximum reactivity worth.

Based on measured data for the microprocessor control system[1], a delay time of 0.1 s was assumed between the overpower detection (microprocessor sends the scram signal) and the actual release of the safety rods. Reference 1 also indicates that the safety control rods require about 0.8 s to drop from their fully-withdrawn to their fully-inserted positions. Since no physical trip condition was mentioned in Ref. 1, two cases were analyzed – one in which the trip occurred at 1.2 kW (20% overpower) and the other in which the trip occurred at a power level of 100 kW.

The results obtained using the PARET code for the L2 and L3 transients are shown in Table 3. For the cases with scram at 1.2 kW, the peak temperature reached in the cladding is estimated to be less than 37 °C, essentially unchanged from the steady-state values. For the cases where the scram was assumed to occur at a power level of 100 kW, a peak temperature of about 133 °C was obtained in the L2 transient case and about 38 °C in the L3 transient case. For all cases, the temperatures reached are far below the 550 - 580 °C temperature at which incipient melting of the SAV-1 aluminum alloy cladding is estimated to occur.

Table 2. Results for Hypothetical Transients S1, L1, and L4 Without Intervention of the Control System for the HEU (B1) and LEU (C1 and C2) Core Configurations*

Case		S1	L1	L2	L3	L4
P_0 (MW)		$1.0x10^{-4}$	$1.0x10^{-7}$	$1.0x10^{-7}$	$1.0x10^{-4}$	$1.0x10^{-4}$
Reactivity Insertion		$0.70 step	$1.50/5 s	$2.50/5 s	$2.50/25 s	$0.70/7 s
Length of Transient Modeled (s)	B1	22.1	4.1	Results with scram		23.7
	C1	21.3	4.1	are shown in Table 3.		23.7
	C2	29.3	4.2			30.2
Time to Peak Power (s)	B1	13.9	4.0			19.3
	C1	14.8	4.0			20.0
	C2	14.9	4.1			20.2
Peak Power (MW)	B1	0.36	54.7			0.45
	C1	0.56	79.4			0.74
	C2	0.64	90.4			0.80
$T_{fuel,max}$ (°C)	B1	118	259			121
	C1	119	288			122
	C2	114	190			119
$T_{clad,max}$ (°C)	B1	118	253			121
	C1	119	282			122
	C2	114	180			118
$T_{cool,max}$ (°C)	B1	107	107			107
	C1	107	107			107
	C2	107	107			107

*Calculations performed using the following thermal-hydraulic data:
Reactor pressure = 129.1 kPa
Coolant temperature at inlet = 20 C
Coolant flow rate at inlet = 0.1 mm/s
Clad thermal conductivity = 180 W/mC
Fuel thermal conductivity = 138 W/mC (HEU), 80 W/mC (LEU)

Table 3. Results for Hypothetical Transients L2 and L3 With Intervention of the Control System for the HEU (B1) and the LEU (C1 and C2) Core Configurations

		SCRAM[1] at 1.2 kW		SCRAM[1] at 100 kW	
		L2	L3	L2	L3
		$2.50/5s Ramp from 0.1 W	$2.50/25s Ramp from 100 W	$2.50/5s Ramp from 0.1 W	$2.50/25s Ramp from 100 W
Peak Fuel Temperature at Steady State (°C)	B1	20.0	36.9	20.0	36.9
	C1	20.0	34.6	20.0	34.6
	C2	20.0	28.7	20.0	28.7
Time of Reactor Trip (s)	B1	2.25	6.88	2.39	9.14
	C1	2.25	6.87	2.38	9.13
	C2	2.29	6.87	2.44	9.15
Time of Peak Power (s)	B1	2.40	6.98	2.54	9.25
	C1	2.40	6.97	2.53	9.23
	C2	2.46	6.97	2.61	9.26
Peak Reactor Power (kW)	B1	60.9	1.33	21,759	149.8
	C1	78.0	1.33	28,043	150.9
	C2	69.5	1.33	25,308	150.0
$T_{fuel,max}$ (°C)	B1	20.5	36.9	135.8	38.2
	C1	20.4	34.6	125.7	35.4
	C2	20.3	28.7	95.4	29.3
$T_{clad,max}$ (°C)	B1	20.4	36.9	133.2	38.2
	C1	20.4	34.6	124.8	35.4
	C2	20.3	28.7	95.0	29.3
$T_{coolant,max}$ (°C)	B1	20.3	36.8	97.2	37.6
	C1	20.3	34.5	99.7	35.2
	C2	20.3	28.7	88.6	29.2

[1]Rod drop delay of 0.1 seconds, 0.8 second rod drop time, and most reactive worth rod stuck out. Rod worths, excluding the safety rod of maximum worth obtained from Table 2 in Part I of this report:

B1 Core = 3.88 β_{eff}
C1 Core = 3.70 β_{eff}
C2 Core = 2.92 β_{eff}

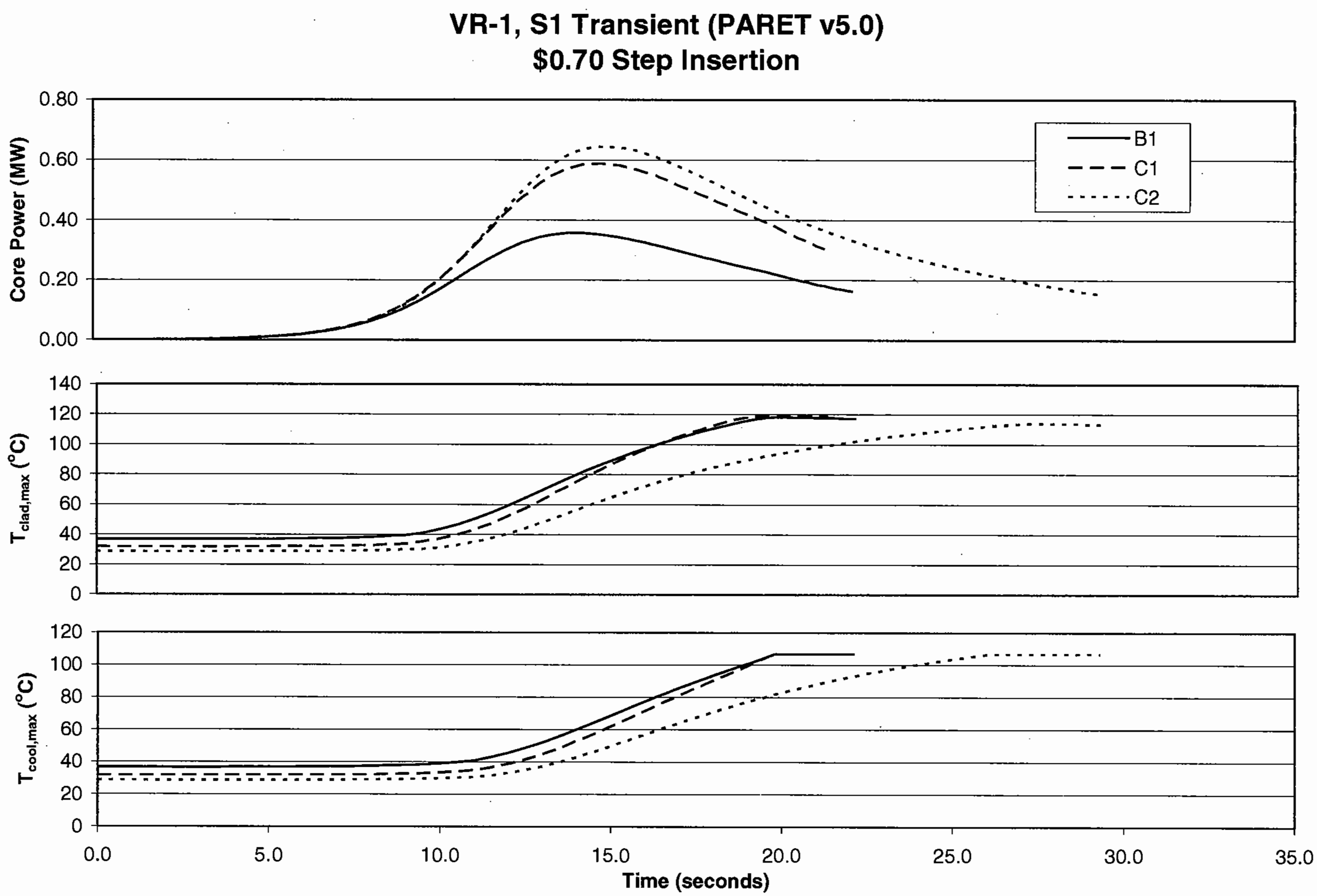


Figure 3. S1 Transient Results. Calculated with PARETv5.

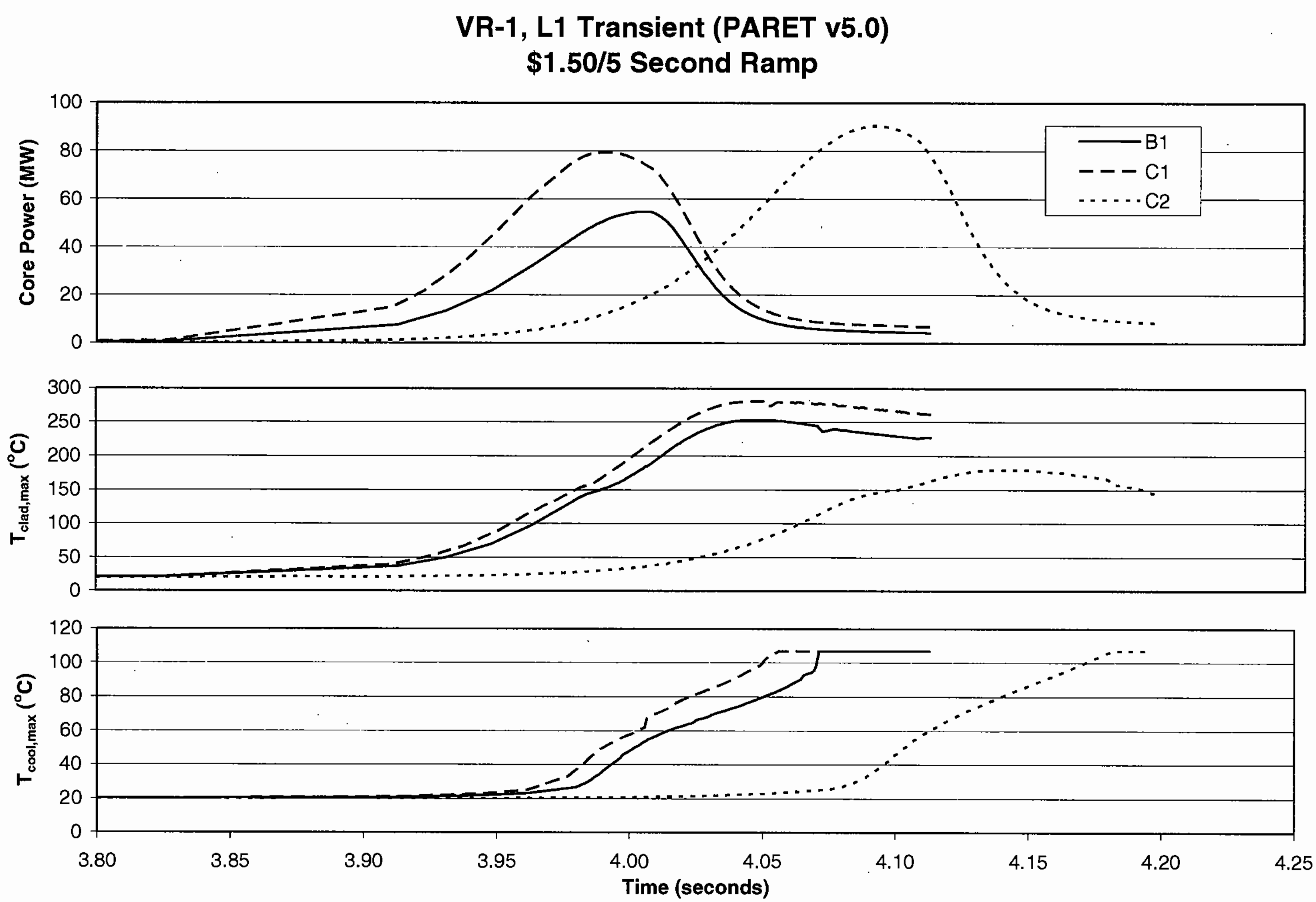


Figure 4. L1 Transient Results. Calculated with PARETv5.

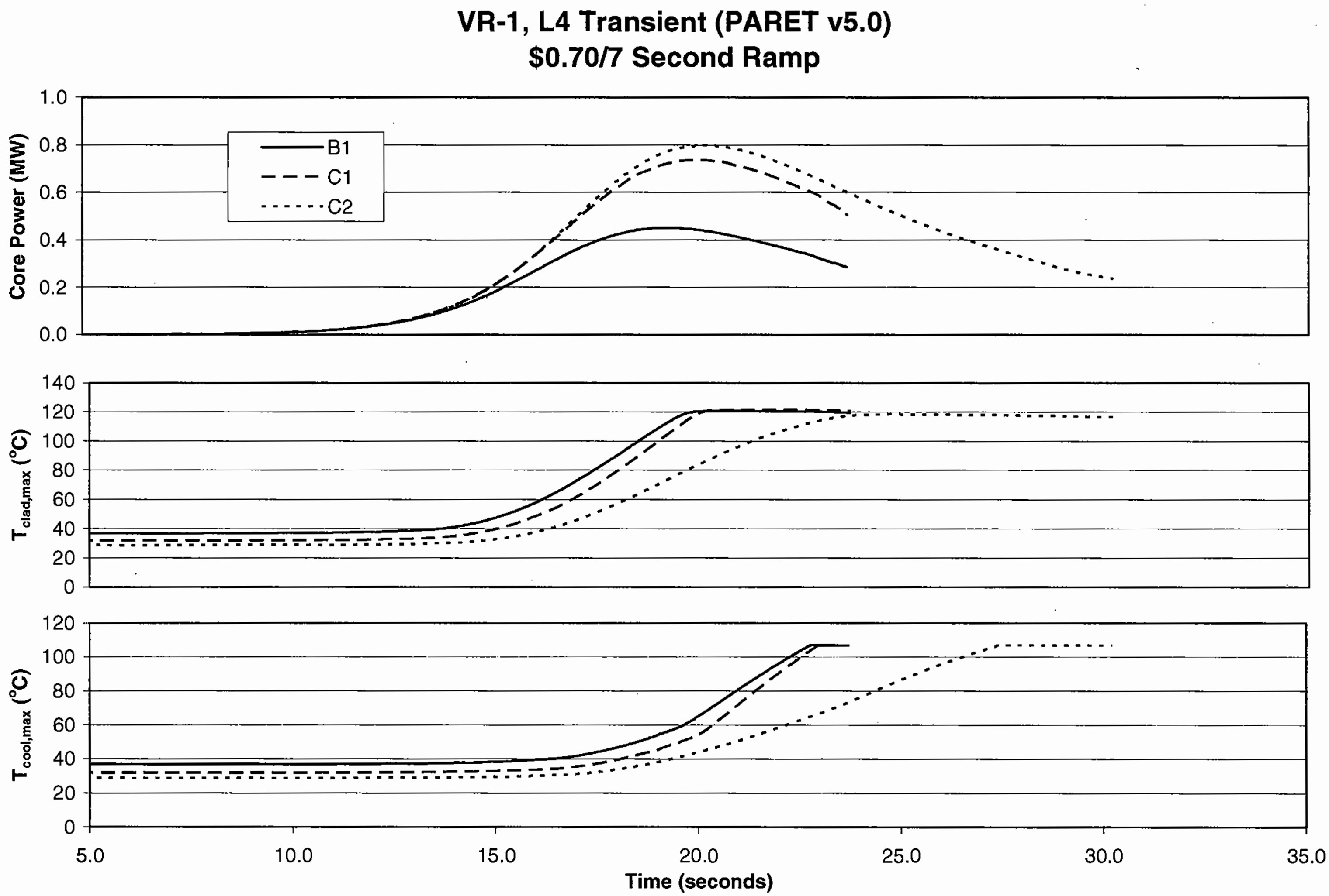


Figure 5. **L4 Transient Results.** Calculated with PARETv5.

Maximum Hypothetical Accident

A Maximum Hypothetical Accident (MHA) was analyzed by CTU in their Safety Analysis[1] for use of IRT-2M fuel in the VR-1. The source term for this accident was assumed to be meltdown of 10% of the core and release of 10% of the fission products. In the CTU accident analysis[2] for VR-1 using IRT-3M fuel, the MHA was not redone because "the results, as they were calculated, are not dependent on the fuel type (fuel melting temperature remains the same, volume of fission products remains on the minimal level."

This same reasoning is also valid for the C1 and C2 core configuration using LEU IRT-4M fuel. Consequently, the MHA was not redone here for the C1 and C2 cores.

Conclusions

An envelope of limiting transients specified by CTU in previous Safety Analyses of the VR-1 reactor was calculated by ANL using the PARET code for the B1 core configuration with HEU IRT-3M fuel assemblies and for the C1 and C2 core configurations with LEU IRT-4M fuel assemblies. Reactivity feedback coefficients and kinetics parameters were calculated using the MCNP5 Monte Carlo code, and the delayed neutron data were calculated using the diffusion theory code VARI3D.

The PARET analyses included void and temperature feedback effects, but terminated some time after peak power was reached due to numerical instabilities which occur when there is large-scale boiling of the coolant. Preliminary calculations were also run using the RELAP5/MOD3.2 code to determine whether the initial power peak of each transient would be totally suppressed by the temperature and void reactivity feedbacks or whether the reactivity insertion was strong enough that secondary power peaks would occur after the initial power peak was suppressed.

For the S1, L1, and L4 transient cases without intervention of the control system, the initial power peak was totally suppressed by the inherent feedback effects of increasing temperature and coolant voiding. The peak temperatures reached in the cladding in the PARET calculations for the B1, C1, and C2 core configurations were 119 C for the S1 transient, 282 °C for the L1 transient, and 122 °C for the L4 transient. These temperatures are far below the value of 550 - 580 °C needed to initiate melting of the SAV-1 aluminum alloy cladding.

For the L2 and L3 transients, preliminary calculations using the RELAP5/MOD3.2 code showed that the large reactivity insertions would lead to multiple power peaks which would drive the temperature of the cladding beyond its melting temperature. Consequently, the calculations that were done for these transients using the PARET code included intervention of the control system with scram of the safety rods (excluding the safety rod of maximum reactivity worth) when the power reached values of 1.2 kW (20% overpower) and 100 kW. For the L2 and L3 cases with scram at 1.2 kW, the maximum temperature of the cladding in the B1, C1, and C2 core configurations was calculated to be about 37 °C. For the L2 and L3 cases with scram at 100 kW, the peak temperatures of the cladding in the B1, C1, and C2 cases were calculated to be less than about 133 °C and 40 °C, respectively. These temperatures are far below the temperature of 550 - 580 °C needed to initiate melting of the SAV-1 aluminum alloy cladding.

A Maximum Hypothetical Accident (MHA) was analyzed by CTU in previous Safety Analyses of the VR-1 for use of IRT-2M and IRT-3M fuels. The source term for this hypothetical accident was assumed to be meltdown of 10% of the core and release of 10% of fission products. In the CTU accident analysis for use of IRT-3M fuel in the VR-1, the MHA was not redone because the results are not dependent on the fuel type. The fuel melting temperature remains the same and the volume of fission products remains on the minimal level. This same reasoning is also valid for the C1 and C2 core configurations using LEU IRT-4M fuel. Consequently, the MHA analysis was not redone here.

References

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