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Neutronics, Steady-State, and Transient Analyses for the Kazakhstan VVR-K Reactor with LEU Fuel

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ABSTRACT

Calculations have been performed for steady state and postulated transients in the VVR-K reactor at the Institute of Nuclear Physics (INP) in Alatau, Kazakhstan. These calculations have been performed at the request of staff of the INP who have performed similar calculations. Calculations were performed for the fresh low-enriched uranium (LEU) core and for four subsequent cores as beryllium is added as a radial reflector to maintain criticality during the first 15 cycles of operation. The calculations include neutronics parameters, steady-state power and temperature distributions, and response to transients. These calculations have been performed independently from those being performed by INP and serve as one step in the verification process.

1. Reactor Description

The VVR-K reactor is a 6 MW pool type reactor preparing for conversion to LEU fuel. The core is located in a cylindrical reactor vessel having diameter 2.3 m and water level height of 5.3 m. Cooling water enters and leaves the reactor vessel through pipes in bottom of reactor vessel.

There are two types of LEU FA denoted VVR-KN. As shown in Figure 1, type FA-1 has 7 concentric tubular fuel elements (FE) of hexagonal cross section and an 8-th central cylindrical FE. There is a cylindrical structural tube interior to the 8-th FE. Type FA-2 has the same outermost 5 concentric tubular FE as in FA-1; interior to the FEs is a cylindrical guide tube (32 mm outer diameter and 1 mm thick) for Control Rod (CR). Corner rounding is 6.9 mm radius for outside of outermost FE,

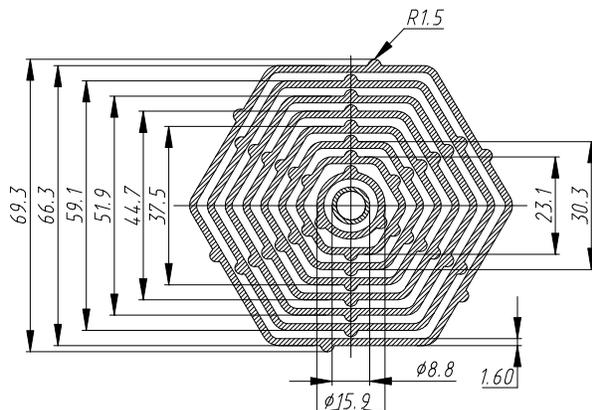


Figure 1 LEU Type FA-1 Fuel Assembly Geometry [1]

decreases by 0.4 mm for each tube moving inward; inner corner rounding is 1.6 mm less than outer corner rounding for each FE. The ribs are actually trapezoid shape rather than the half circle implied by dimension “R1.5” in figure.

The FEs are 1.6 mm thick, consisting of 0.7 mm of fuel meat and 0.45 mm of cladding on each side. The fuel meat is UO₂-Al, enriched to 19.75% in U-235. The U-235 masses are 248.2 g in FA-1 and 197.6 g in FA-2; this yields a mean fuel density of about 2.8 g/cm³ of uranium. Cladding and other structural items are made of the aluminum-alloy SAV-1. Ribs of height 1.5 mm provide stiffening of FE and help maintain 2 mm water gap between adjacent FE. The design of fuel meat is 0.6 m in length with a standard deviation of 0.002 m. In the analyses presented in this paper the nominal dimensions and masses of the fuel were used.

The initial critical configuration is achieved by loading a few LEU FA then performing measurements to assess the neutron multiplication factor (k_{eff}). Criticality is achieved when multiplication factor reaches 1.0. INP has provided ANL with the intended order for loading FA [2]. The FA-2 are loaded first, since each contains a CR. During the early stages of loading more than one FA can be loaded between measurement points. Calculations using MCNP [3] predict that criticality will be achieved with 10 FA-2 and 11 FA-1. All neutronics calculations in this paper were performed using MCNP6 and ENDF/B-VII.0 cross sections. INP will continue loading FA-1 to reach “work load” core having 10 FA-2 and 17 FA-1.

Figure 2 shows the initial work load (i.e., cycle 1) LEU core configuration. Legend for hexagons: grey-shaded is FA-1; grey-shaded with white interior circle is FA-2 with CR identifier (nKO, nAZ, or AR) noted; white is water displacer; white with interior circle is irradiation position. After three cycles of operation calculations using MC-REBUS [4-5] and MCNP indicate that the excess reactivity will have dropped sufficiently low as to be unable to sustain criticality. At that time INP will replace some water-displacer blocks with beryllium blocks (Be) outside of the FA locations to reflect neutrons back into the core. Additional Be will be added after cycles 9, 11, and 14. The order in which Be are added was provided by INP [2]. The necessity of adding the Be is shown in Figure 3, which plots excess reactivity versus full power days; additional Be must be added if the excess reactivity is projected to be between 0 and 1 at the end of the defined 20 day cycle length.

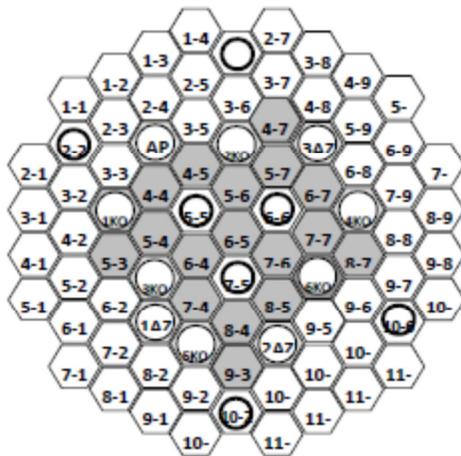


Figure 2 LEU Core Configuration for Cycle 1

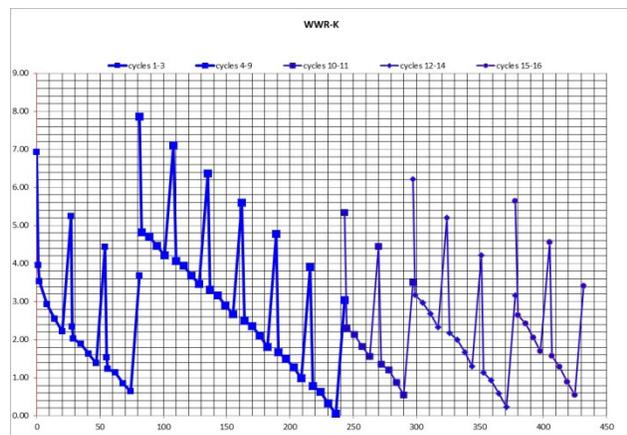


Figure 3 Excess Reactivity [%] vs. Time [full power days]

This paper concentrates on presenting results calculated at ANL for cycle 1. Results have also been calculated [6] for the cores at cycles 4, 10, 12, and 15; these are presented in summary manner in this paper.

2. Neutronic Parameters

In order to prepare for transient analysis, various contributions to reactivity change and other neutronics kinetics parameters must be calculated for each core.

The delayed neutron fraction [$\beta(i)$] and decay constant [$\lambda(i)$] for 6 groups are shown in Table 1 for cycle 1. The effective delayed neutron fraction (β_{eff}) is 0.00768 and the prompt neutron generation time (Λ) is 46.7 μs . There is little shift in the group-wise delayed neutron fractions and decay constants during the first 15 cycles. The effective delayed neutron fraction decreases by about 10%. The prompt neutron generation time increases by about 50% due to addition of the Be reflector.

Table 1 Delayed Neutron Fractions and Decay Constants for Cycle 1 Core

Group	1	2	3	4	5	6
$\beta(i)$	0.03385	0.16276	0.16016	0.45703	0.13802	0.04818
$\lambda(i)$ [1/s]	0.01249	0.03181	0.10946	0.31739	1.35298	8.66508

There are three types of control rods (CR): (1) There are 3 safety rods (AZ1 through AZ3, alternately shown as 1AZ through 3AZ); these are fully withdrawn from the core during reactor operation; they fall into the core due to gravity in response to a scram signal to terminate the nuclear chain reaction. (2) There are 6 shim rods (KO1 through KO6 (or 1KO-6KO)); they are partially withdrawn from the core during normal operation and are adjusted during operation to maintain criticality; these rods also fall into core due to gravity in response to a scram signal. (3) There is 1 automatic rod (AR); it is partially withdrawn from the core during normal operation and its drive motor is attached to a logic circuit used to maintain (or make programmed adjustments to) power; it does participate in scram (but this small additional worth is ignored in the ANL transient calculations). The reactivity worths of AZ and KO CR for cycle 1 core are shown in Table 2; the AR rod is worth 0.35 \$; shutdown margin (i.e., KO plus AR) is 2.51 \$; worth of all rods is 6.65 \$. With some exceptions, individual control rod worths increase as cycle number increases.

Table 2 Control Rod Worths [\$] for Cycle 1 Core

Rod #i	1	2	3	4	5	6
AZi	1.16	1.34	1.26	---	---	---
KOi	1.38	2.57	2.81	1.42	1.74	2.55

There are detection, processing, and action circuits in order to automatically respond to various emergency conditions. Upon detection of a low-level variance the operator is given a warning signal. If the variance exceeds a certain level the systems will take automatic actions intended to bring the reactor to a safe, shut down condition. One of these actions is denoted “scram”, which involves insertion of all CR into the core. In analysis of emergency situations, regulations require that one assumes that the highest worth safety rod is stuck and does not fall into the core.

Additionally, one transient which must be analyzed involves the spontaneous withdrawal of the highest worth shim rod; the scram for this transient must consider that the withdrawing shim rod does not participate in the scram; therefore there are two scram reactivity insertions associated with each core. As a reminder, when adding or removing a CR from the calculation, one must redo the neutronics calculation rather than merely deducting the worth of the CR from the total – the individual worths are not directly additive because of the shadowing factor.

Figure 4 shows the worth of shim rod KO3 in cycle 1 core as a function of withdrawal position. The initial CR movement has the highest worth per centimeter and decreases as the CR is withdrawn further from the core. The highest worth shim rod varies during operation, being KO3 for cycles 1, 12, and 15; KO6 for cycle 4; and KO2 for cycle 10. An illustration of reactivity due to CR insertion is shown in Figure 5 for cycle 1; curve “AZ1+AZ3+KO” in Figure 5 is used for most transients; it omits the highest worth safety rod and includes all shim rods; curve “AZ1+AZ3+KO-KO3” in Figure 5 is used for rod withdrawal transient; it is like the other scram curve but omits the highest worth shim rod. CR insertion of only 5 to 10 cm is required to insert more than 1 \$ of reactivity, and, thus, lead to termination of the nuclear chain reaction in all transients to be analyzed later in this report.

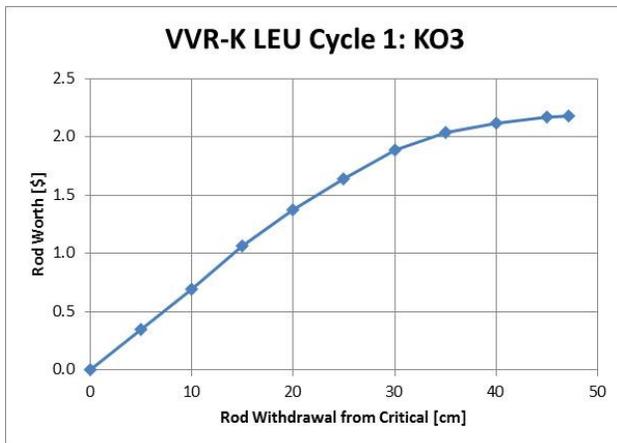


Figure 4 KO3 Rod Worth as Function of Withdrawal

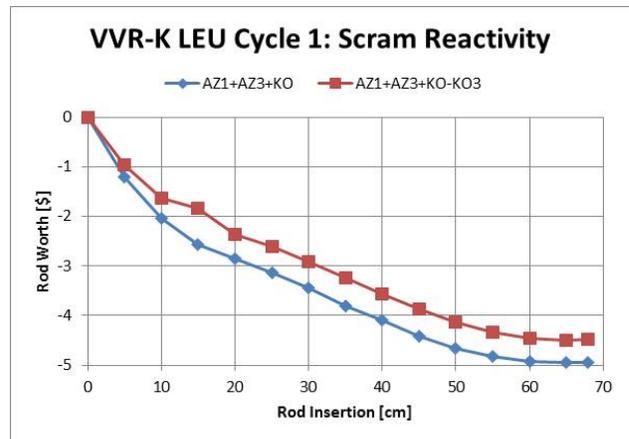


Figure 5 Scram Reactivity as Function of Rod Insertion

Table 3 Temperature and Density Reactivity Feedback Coefficients for Cycle 1

Range [K]	294<T<350	350<T<400	294<T<400	
Coolant Temperature [\$/K]	-1.2212E-2	-1.3180E-2	-1.2667E-2	
Range [%]	0—5%	5—10%	0—10%	
Coolant Density [\$/%]	-0.41245	-0.44670	-0.42958	
Range [K]	294<T<400	400<T<500	500<T<600	294<T<600
Fuel Temperature [K]	-2.9287E-3	-2.5861E-3	-2.3591E-3	-2.6280E-3

The reactivity feedback coefficients associated with coolant and fuel temperature changes and coolant density changes are shown in Table 3 for cycle 1 core. The temperature-based

coefficients increase and the density-based coefficient decreases as the cycle number increases. All of the reactivity coefficients are negative for the cycle in each of the different core configurations. Note that for the cores, the lateral reflector temperature (either water or Be) was considered to be equal to room temperature; repeating the analysis with the Be reflector at 350 K shows that the changes in the reactivity coefficients is negligible.

3. Core Power

The FA having peak power is at core position 6-5 (see Fig. 2). For cycle 1 with total core power of 6 MW, the FA in this position has power of 0.409 MW; the peak FA power decreases to 0.389 MW at cycle 4, 0.348 MW at cycle 10, 0.335 MW at cycle 12, and 0.308 MW at cycle 15. There is little variation in average power in all FA-1 (decreasing from 0.256 to 0.246 MW) or FA-2 (increasing from 0.164 to 0.182 MW) across the cycles analyzed.

Within each FA the power density varies from FE to FE, as well as variations in axial and azimuthal directions. An example is shown in Figure 6 for the peak power FA at cycle 1. Tube 1 is the outer tube and tube number increases moving inward. In general, power density decreases moving inward from Tube 1 due to self shielding. The peak power density occurs 23 to 25 cm above bottom end of fuel meat for each tube, corresponding to 5 to 7 cm below core mid-plane; this is determined by the position of the control rods at the critical position for each of the cores. It is important to note that at the critical state the automatic rod (AR) is inserted to the core centerline and the KO rods are all inserted to the same height. Power density is peaked toward bottom of core due to CR being inserted from the core top.

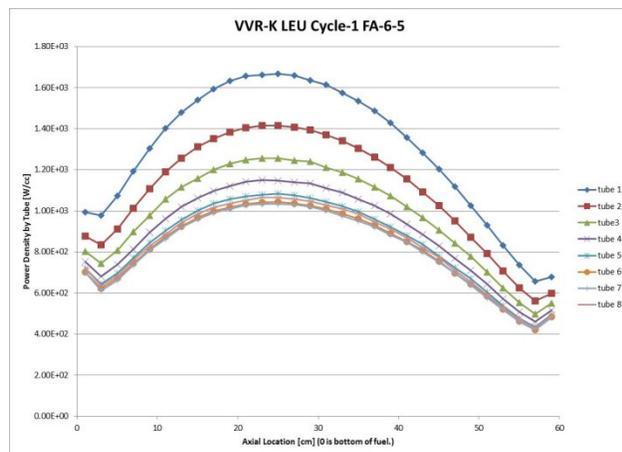


Figure 6 Axial Variation of Power Density by Tube for Peak Power FA in Cycle 1

Table 4 Power Density [W/cc] Summary for BOC in Each Cycle Analyzed

Cycle	1	4	10	12	15
Peak	1667	1535	1394	1351	1279
Avg in Peak FA	912	867	777	746	686
Avg in FA-1	571	573	559	557	548
Avg in FA-2	458	455	484	488	508
Avg in FA-all	535	535	535	535	535

Peak and average power densities are shown for each core in Table 4. The maximum peak power density is usually in the peak power FA, which is at core location 6-5; for cycle 15 the peak power density is in core location 7-7, although location 6-5 has peak FA power; the difference in the peak power density in those two FAs is only about 3%. The peak power

density of 1667 W/cc occurs in cycle 1 and decreases in later cycles, to 1279 W/cc (a 24% decrease) by cycle 15. The average power density in FA-1 is higher than the average in FA-2. In general, the average power density in FA-1 is decreasing as irradiation proceeds, while the average in FA-2 is increasing. Since all cores produce 6 MW and have the same fuel volume the average power density for all cores stays constant at 535 W/cc.

4. Steady-State Thermal-Hydraulics Analysis

ANL has performed steady-state thermal-hydraulics analysis for the VVR-K cores using two computer codes: RELAP5 [7] and PLTEMP [8]. The primary ANL results are those from RELAP5, in order to be a method different from INP, which has done their analysis using PLTEMP. ANL has also performed the analysis using PLTEMP, in order to illustrate the types of code-to-code differences which can occur even when using the same input assumptions. Even though ANL and INP are both performing calculations using PLTEMP there may be some differences since ANL is using the 60-cm design value for fuel meat length and INP is using 61-cm value for the as-delivered fuel.

The model extent is the same in both RELAP5 and PLTEMP. Each code can model a nested-tube FA as a set of material layers, alternating among water and clad-fuel-clad. The water gaps are described by flow area and hydraulic diameter. The FE solids are modeled as plates described by thickness of the three material layers and a perimeter (or “width” in the context of PLTEMP). The axial extent of the model covers the fuel height; all regions have the same axial mesh divisions. The Petukhov-Popov correlation for clad to coolant heat transfer was chosen in both codes.

Coolant flow is from top to bottom. Coolant flow rate is specified for the FA, which is 18 m³/h for FA-1. The apportionment of flow among the gaps is known from measurements. In RELAP5 the entrance loss coefficient for each gap was adjusted to obtain the measured flow distribution [9]. In PLTEMP the measured flow rate distribution is directly specified. Although accomplished in different ways, the ANL analyses using RELAP5 and PLTEMP have the same flow rate distribution. The inlet coolant temperature is chosen as 45°C (318.15 K), which is the maximum allowed during operation. The inlet pressure of 0.135 MPa is specified at the top of the fuel element.

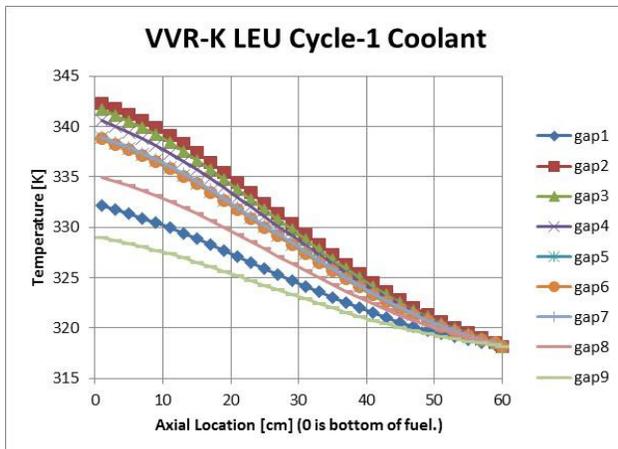


Figure 7 Coolant Temperature Calculated using RELAP5

There are several criteria to be satisfied. The maximum clad surface temperature must be lower than 98°C and the minimum value for the Onset of Nucleate Boiling Ratio (ONBR) using the Bergles-Rohsenow correlation must be above 1.3. These criteria are to be met without using hot-channel factors. The satisfaction of these criteria can be assessed for each core by performing calculations for only the FA having peak power density.

The coolant temperatures calculated using

RELAP5 for cycle 1 are shown in Figure 7. The hottest coolant is in Gap 2, which is between Tube 1 (outermost) and Tube 2; the temperature is 342 K at the exit, which is well below boiling. Coolant temperatures get cooler as one moves from Gap 2 toward the interior. For the hottest FA, Gap 1, which is outside of the FA, is cooler than Gaps 2 through 8; this gap is larger than the interior gaps and there is heat from only one fuel plate surface. Although a figure is not shown here, there is good agreement between coolant temperatures calculated using PLTEMP with those calculated using RELAP5.

The temperatures calculated using RELAP5 for the outer and inner surface of each tube in FA having peak power in cycle 1 are shown in Figures 8 and 9, respectively. The hottest temperature of 361 K occurs on the inner surface of Tube 1 a few centimeters below location of peak power density; this peak temperature is well below the damage criteria. As a reminder, Tube 1 has the peak power density. There is almost no difference in temperature between the outer and inner surface of each tube as calculated using RELAP5. Although a figure is not shown here, there is good agreement between surface temperatures calculated using PLTEMP with those calculated using RELAP5.

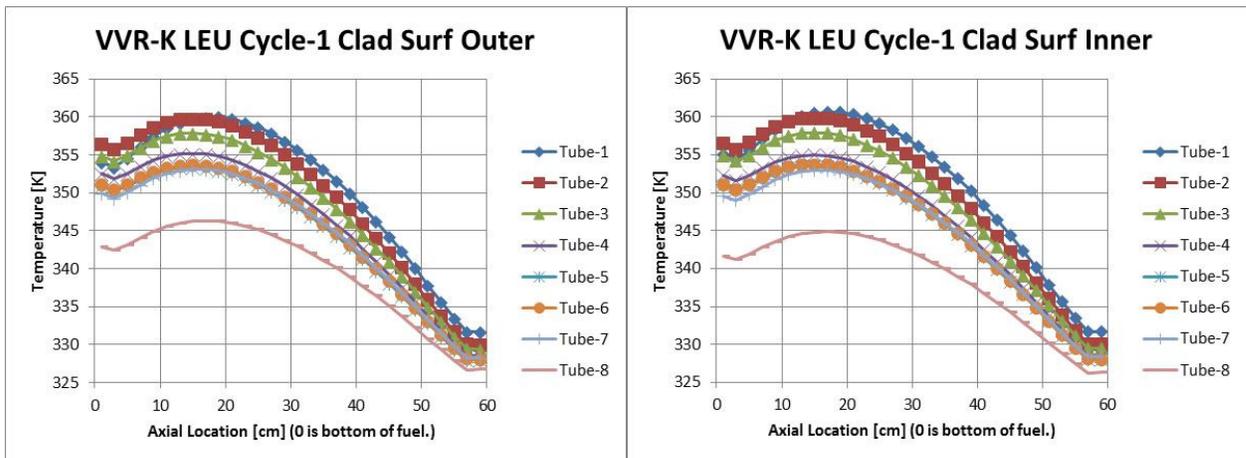


Figure 8 Outer Clad Surface Temperature Calculated using RELAP5

Figure 9 Inner Clad Surface Temperature Calculated using RELAP5

Calculations were also performed for cycles 4, 10, 12, and 15. The power densities are the only input values which change from cycle to cycle. In general, the trends follow those seen for cycle 1. The peak temperatures decrease for higher cycle numbers since the peak power per FA decreases; peak coolant temperature decreases from 342 K at cycle 1 to 336 K for cycle 15. Peak clad temperature is less than 98°C (371 K) design limit for all cycles analyzed, decreasing from 361 K in cycle 1 to 352 K in cycle 15. The minimum ONBR calculated for all cycles is above the 1.3 design limit, increasing from 1.62 in cycle 1 to 1.98 in cycle 15.

5. Transient Analysis

After discussion with INP ANL agreed to perform detailed transient analysis for four initiating events: inadvertent shim rod withdrawal, primary pump seizure, loss of offsite electric supply, and impact of experiments. ANL has performed the analysis of VVR-K transients using two

computer codes: RELAP5 and PARET [10]. The primary ANL results are those from RELAP5, in order to be a method different from INP, which has done their analysis using PARET. ANL has also performed all of its analyses using PARET in order to illustrate the types of code-to-code differences which can occur even when using the same input assumptions.

The model extent is the same in both RELAP5 and PARET. Rather than the multi-channel nested-tube approach used for the steady-state calculation in Section 4, the transient model consists of two channels: hot and average. The hot channel isolates the portion of the core having the peak power density; this is treated as a half thickness of fuel meat separated by a full thickness cladding from a half thickness of water gap. The other channel also has layers of fuel, cladding, and water and represents the entire core on an average basis. In our experience this two-channel approach is sufficient for transient reactor analysis. The average channel gives appropriate reactivity feedbacks when temperatures and densities change, while the hot channel provides knowledge of the peak temperatures.

Within each channel, the water gap is described by flow area and hydraulic diameter and the FE solids are described by thickness of the material layers and a perimeter. The axial extent of the model covers the fuel height; all regions have the same axial mesh divisions.

5.1. Common Conditions

The maximum normal operating power for the reactor is 6 MW. The core is assumed to have been operating at full power for full 20 day cycle. Forced convection cooling of the core is provided by two pumps operating in parallel; collectively these pumps provide 700 m³/h coolant flow during normal operation. Only a portion of this coolant flow goes through the fuel assemblies. For the cycle 1 LEU core the flow rates (provided by INP) are as follows: 18.0 m³/h in each of 17 type FA-1, 15.34 m³/h in each of 10 type FA-2, 4.19 m³/h in each of 51 water displacer tubes, and 4.319 m³/h in each of 7 irradiation tubes.

A number of parameters are monitored by the VVR-K operators and systems. Going above or below set point values generates an emergency protection action or scram to shut down the power production. Examples of these set points which are significant in the present analysis are as follows: power exceeding 7.2 MW (i.e., 20% overpower); period less than 10 s; total primary coolant flow less than 80% of nominal; and pool level dropping from 5.3 to 4.9 m. There is a delay time of 0.3 s between crossing any of these set points and start of control rod motion for scram. The action of the 3 types of CR during scram was described in Section 2; as reminders, the KO rod critical positions differ for each cycle analyzed and the highest worth AZ rod is assumed stuck for all analyses. Coolant pumps continue to operate following scram unless the transient initiator is loss of pumping. When water level drops below 3.3 m the core emergency cooling system is activated, providing 10.4 and 10.0 m³/h water flow to 2 sets of spray headers.

Coolant flow is from top to bottom. Coolant flow rate is divided between the two channels based on the results obtained during the steady-state analysis. For VVR-K the peak clad temperature at steady state is always the interior of Tube 1 and, thus, the associated water is the gap between Tubes 1 and 2. The coolant velocity in this gap at steady state is 2.44 m/s. This same value of velocity is input to the calculations for the average channel. The inlet coolant

temperature and pressure are the same as used in the steady-state analyses (i.e., 45°C (318.15 K) and 1.34 bar).

The initial 1 s of each transient calculated using PARET has constant boundary conditions in order to illustrate that the calculation is, indeed, at a steady state before the transient is imposed. The varying boundary conditions are imposed starting at 1 s. This constancy is shown in the plots of results. In contrast, the RELAP5 runs start with a steady-state initialization which will be hidden from view; the RELAP5 results are plotted with a 1 s offset when comparing with PARET.

5.2. Shim Rod Withdrawal (KO)

The transient is initiated by assuming an inadvertent withdrawal of the highest worth shim rod with a speed of 0.4 cm/s. The specific shim rod and worth versus rod withdrawal position are different for each cycle analyzed. The values are converted to worth versus time using a constant rod speed of 0.4 cm/s. Inlet coolant temperature and flow rate are assumed constant during the transient. When scram occurs, the reactivity inserted by the falling CRs is reduced due to the assumed continued upward motion of the highest worth shim rod.

Key results calculated for this transient in the cycle-1 core are shown in Figure 10; the left graph shows net reactivity, the middle graph shows reactor power, and the right graph shows peak cladding and coolant temperatures. Results are shown for RELAP5 and PARET. The reactivity insertion from KO rod motion starts at 1 s, at which time power and temperatures start to increase. The reactor power reaches 7.2 MW in 6.1 s after start of KO rod motion in RELAP5, which satisfies 20% overpower trip condition; after 0.3 s system delay for scram, the control rods start their insertion, which terminates the power increase. Peak cladding temperature is 100°C and peak coolant temperature is 78°C, and these occur at time of peak power; after that time the temperatures decrease to values below those for steady state at full power. The increases in temperature are sufficiently small that no cladding damage would occur due to this transient. As shown in Figure 10, the results calculated using PARET are essentially the same as those calculated using RELAP5.

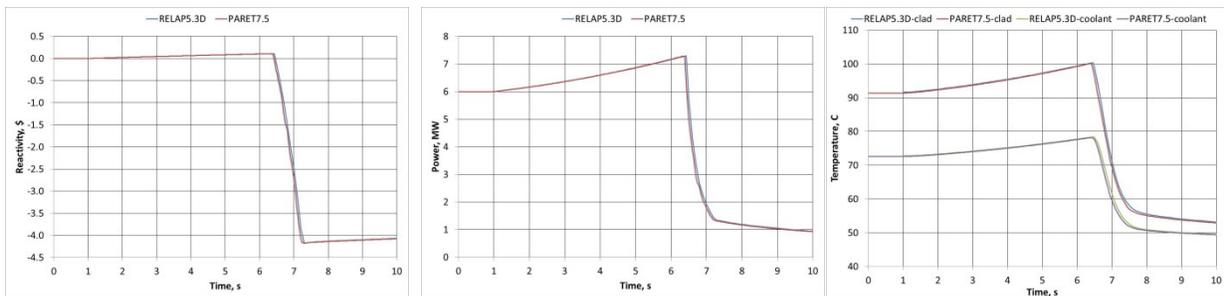


Figure 10 Results Calculated for KO Transient

Although results figures are not shown here, the results for the other cycles are qualitatively the same as for cycle 1. The 20% overpower trip condition of 7.2 MW is reached in 4.4 to 5.1 s after start of KO withdrawal, and scram occurs 0.3 s later for all cycles analyzed. The peak

temperatures decrease for the higher cycle numbers analyzed since the peak power density is lower.

5.3. Primary Pump Seizure (BLADE)

The transient is initiated by assuming a blade in one of the two primary pump breaks, stopping coolant flow through this pump; the coolant flow through this pump is assumed to decrease linearly to zero over 1 s; the other primary pump continues to provide coolant flow at half of the total core nominal value. Inlet coolant temperature is assumed constant during the transient.

Key results calculated for this transient in the cycle-1 core are shown in Figure 11; the left graph shows coolant flow rate (normalized to 1 at steady state), the middle graph shows reactor power, and the right graph shows peak cladding and coolant temperatures. Results are shown for RELAP5 and PARET. The coolant flow rate decrease starts at 1 s, at which time temperatures start to increase. Temperature increase leads to a power decrease due to the negative values for reactivity feedback coefficients. The coolant flow rate crosses 80% at 0.2 s after the start of flow decrease in RELAP5, which satisfies low flow trip condition; after 0.3 s system delay for scram, the control rods start their insertion, which causes the power to decrease even faster. Peak cladding temperature is 102°C and peak coolant temperature is 80°C, and these occur 0.3 s after start of scram (or 0.8 s after start of flow decrease); after that time the temperatures decrease to values below those for steady state at full power and flow. The increases in temperature are sufficiently small that no cladding damage would occur due to this transient. As shown in Figure 11, the results calculated using PARET are essentially the same as those calculated using RELAP5.

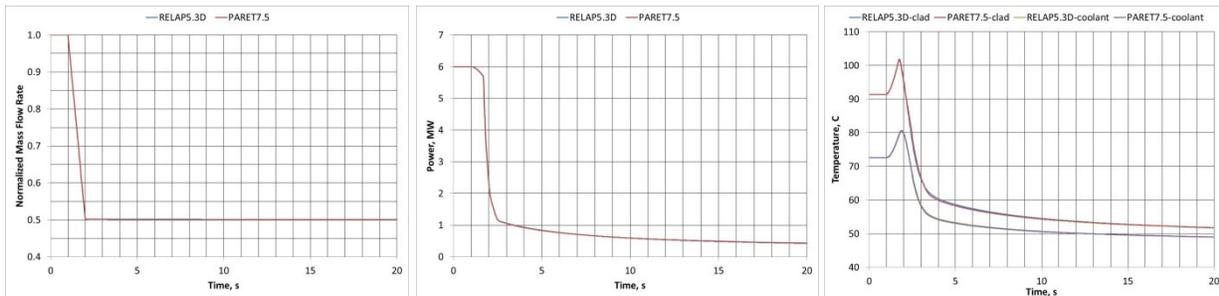


Figure 11 Results Calculated for BLADE Transient

Although results figures are not shown here, the results for the other cycles are qualitatively the same as for cycle 1. Coolant flow rate is the same for all cycles; the flow rate drops below 80% in 0.2 s and scram occurs 0.3 s later for all cycles analyzed. The power history is essentially the same for all cycles analyzed. The peak temperatures decrease for the higher cycle numbers analyzed since the peak power density is lower.

5.4. Loss of Offsite Electric Supply (LOOP)

The transient is initiated by assuming loss of offsite power. Loss of power means that both primary coolant pumps will stop; this is assumed to occur over 43.5 s, as provided by INP [2].

Loss of offsite power will send a signal to automatically start one (or the other of two) diesel generators (DG) to supply emergency power to critical systems. Additionally there is an auxiliary primary coolant pump connected to uninterruptable power source (UPS) (as well as to DG). Therefore, primary coolant flow decreases from 700 m³/hr provided by two main primary coolant pumps to the 45 m³/hr provided by the auxiliary primary pump. The LOOP condition can be detected by the instrumentation and control system (ICS); it does not, however, cause immediate loss of power to CR drive magnets, since they are supplied by separate UPS; therefore, scram is not instantaneous. Although ICS can detect LOOP, scram will be due to decrease in primary coolant flow rate below 80%; there is the normal 0.3 s delay between sensing an emergency condition and start of CR motion into the core. Inlet coolant temperature is assumed constant during the transient.

Key results calculated for this transient in the cycle-1 core are shown in Figure 12; the left graph shows coolant flow rate (normalized to 1 at steady state), the middle graph shows reactor power, and the right graph shows peak cladding and coolant temperatures. Results are shown for RELAP5 and PARET. The coolant flow rate decrease starts at 1 s, at which time temperatures start to increase, which leads to power decrease due to the negative values for reactivity feedback coefficients. The coolant flow rate decrease below 80% at 6.42 s after start of flow decrease; CR start their movement into the core 0.3 s later. The temperatures reach maximum at time of scram - 99°C for peak cladding and 78°C for peak coolant. Thereafter, since power and coolant flow rate are decreasing at different rates, the temperatures will initially decrease, having minima of 54°C for peak cladding and 51°C for peak coolant at 13 s after start of transient. Subsequently, coolant flow rate is decreasing faster than the decrease in power; therefore, the temperatures start rising, having secondary maxima of 76°C for peak cladding and 68°C for peak coolant at 47 s after start of transient as calculated using RELAP5; these maxima are below steady-state values at full power and flow rate. Since coolant flow rate is now constant and power is continuing to decrease, the temperatures decrease from this time onward. The increases in temperature are sufficiently small that no cladding damage would occur due to this transient. As shown in Figure 12, the results calculated using PARET are essentially the same as those calculated using RELAP5; the secondary maximum in temperature calculated using PARET is 1 to 2°C lower than the values calculated using RELAP5; we consider this different to have no safety significance.

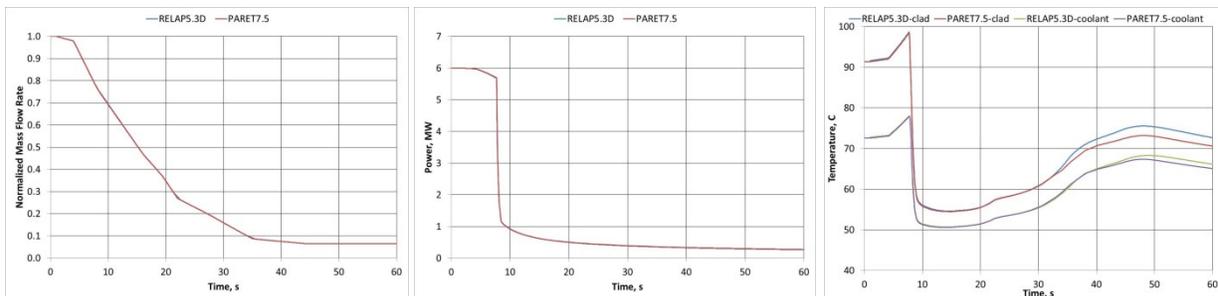


Figure 12 Results Calculated for LOOP Transient

Although results figures are not shown here, the results for the other cycles are qualitatively the same as for cycle 1. Coolant flow rate is the same for all cycles. Scram occurs 0.3 s after

sensing coolant flow less than 80% for all cycles analyzed. The power history is essentially the same for all cycles analyzed. The peak temperatures decrease for the higher cycle numbers analyzed since the peak power density is lower.

5.5. Impact of Experiments (EXP)

The transient is initiated by assuming an unspecific failure of the mounting for an experiment in the reactor that inserts reactivity of 1.5% in 1 s. The reactivity inserted in percent is the same for all cores; however, the value must be converted to dollars for input to the RELAP5 and PARET codes; since β_{eff} is different for each core, the reactivity inserted in dollars is different for each core, ranging from 1.95 \$ for cycle 1 to 2.17 \$ for cycle 15. Inlet coolant temperature and flow rate are assumed constant during the transient.

Key results calculated for this transient in the cycle-1 core are shown in Figure 13; the left graph shows reactor power and the right graph shows peak cladding and coolant temperatures. Results are shown for RELAP5 and PARET; the RELAP5 results are discussed first. The reactivity insertion from EXP motion starts at 1 s, at which time power and temperatures start to increase. Reactor period is below the 10 s trip condition almost immediately; scram occurs 0.3 s later, at which time the reactor power reaches 12.6 MW; power then rapidly decreases due insertion of the control rods. Peak cladding surface temperature is 112°C and peak coolant temperature is 87°C, and these occur slightly after peak reactor power; after that time the temperatures decrease to values below those for steady state at full power. The increases in temperature are sufficiently small that no cladding damage would occur due to this transient. As shown in Figure 13, the results calculated using PARET are essentially the same as those calculated using RELAP5; the only noticeable difference is that the peak cladding temperature is slightly (2°C) higher using PARET than using RELAP5.

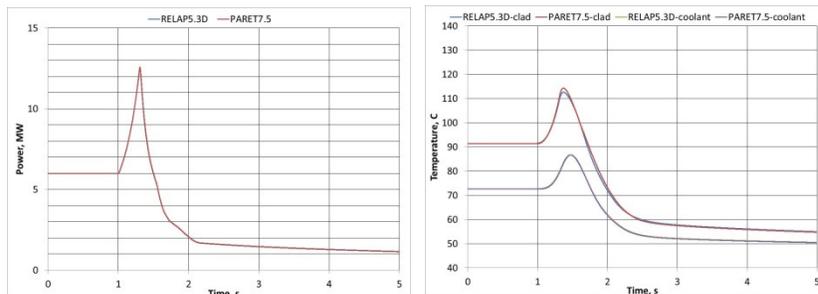


Figure 13 Results Calculated for EXP Transient

Although results figures are not shown here, the results for the other cycles are qualitatively the same as for cycle 1. Peak power varies from 12.8 to 13.6 MW (since inserted reactivity in dollars increases), peak cladding temperature varies from 110 to 103°C, and peak coolant temperature varies from 85 to 76°C as the cycle number increases from 4 to 15.

5.6. Transients Summary

Analysis of four transients (shim rod withdrawal, primary pump seizure, loss of offsite electric supply, and impact of experiments) using the RELAP5 code indicates that built-in safety

systems automatically act quickly on high power, low coolant flow, or low reactor period signals to shut down the VVR-K reactor before there can be any significant cladding damage. In general, the cycle 1 core presents the highest peak FA power and highest material temperatures of the cycles analyzed. Essentially identical results were obtained for these transients using the PARET code.

6. Conclusions

At the request of INP, ANL personnel have performed calculations for INP's VVR-K reactor during its intended initial operation with LEU fuel and for the subsequent cores using beryllium reflector instead of water reflector. These calculations include nuclear characteristics, steady-state operation, and various transients. The various cores (i.e., fresh LEU water reflected core at cycle 1 plus beryllium reflected cores at cycles 4, 10, 12, and 15) satisfy the stated steady-state limits for peak cladding temperature and minimum ONBR. The four postulated transients (shim rod withdrawal, primary pump seizure, loss of offsite electric supply, and impact of experiments) for which detailed analyses were performed are automatically terminated by safety systems before there can be any cladding damage. These calculations have been performed independently from those being performed by INP and serve as one step in the verification process.

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