MNSR Transient Analyses and Thermal Hydraulic Safety Margins for HEU and LEU Cores Using the RELAP5-3D Code

F.E. Dunn, J. Thomas, J. Liaw and J.E. Matos

RERTR Program
Argonne National Laboratory
Argonne, IL 60439 USA

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ABSTRACT

For safety analyses to support conversion of MNSR reactors from HEU fuel to LEU fuel, a RELAP5-3D model was set up to simulate the entire MNSR system. This model includes the core, the beryllium reflectors, the water in the tank and the water in the surrounding pool. The MCNP code was used to obtain the power distributions in the core and to obtain reactivity feedback coefficients for the transient analyses. The RELAP5-3D model was validated by comparing measured and calculated data for the NIRR-1 reactor in Nigeria. Comparisons include normal operation at constant power and a 3.77 mk rod withdrawal transient. Excellent agreement was obtained for core coolant inlet and outlet temperatures for operation at constant power, and for power level, coolant inlet temperature, and coolant outlet temperature for the rod withdrawal transient. In addition to the negative reactivity feedbacks from increasing core moderator and fuel temperatures, it was necessary to calculate and include positive reactivity feedback from temperature changes in the radial beryllium reflector and changes in the temperature and density of the water in the tank above the core and at the side of the core. The validated RELAP5-3D model was then used to analyze 3.77 mk rod withdrawal transients for LEU cores with two UO_2 fuel pin designs. The impact of cracking of oxide LEU fuel is discussed. In addition, steady-state power operation at elevated power levels was evaluated to determine steady-state safety margins for onset of nucleate boiling and for onset of significant voiding.

OVERVIEW

The maximum allowed excess reactivity in MNSR reactors is 4 mk (0.4% $\Delta k/k$). Insertion of this maximum allowed reactivity by withdrawal of the control rod is an important part of the commissioning process for each MNSR and is a key measurement in defining the reactor safety basis.

Experimental data[1] from the NIRR-1 reactor was used to validate a RELAP5-3D[2] model of the MNSR reactor. The validated model was then used to analyze LEU cores with two proposed UO_2 fuel pin designs. With each LEU core, a 3.77 mk rod withdrawal transient was run. Steady-state thermal hydraulic margins were evaluated for one LEU core. In addition, two cases that are not expected to occur were evaluated in order to determine the potential behavior of one of the LEU cores. One of these cases was a 3.77 mk rod withdrawal with cracked oxide fuel. Because of the low power level in the MNSR, fuel cracking is not expected to occur. The second case that is not expected is a 6 mk rod withdrawal. With the current MNSR design, the maximum rod withdrawal worth allowed by the Operating Limits and Conditions is 4 mk. The 6 mk case provides an indication of the margin of safety.

Because the MNSR design contains somewhat irregular flow orifices near the core inlet and outlet, measured data for steady-state operation of the NIRR-1 reactor was used to determine the best values to use for the orifice coefficients.

MNSR DESCRIPTION

Schematic diagrams of the NIRR-1 MNSR reactor are shown in Figure 1. The reactor core contains 347 active fuel pins plus 3 dummy pins. The pin dimensions for the current HEU core and for the two LEU cores considered in this work are given in Table 1.

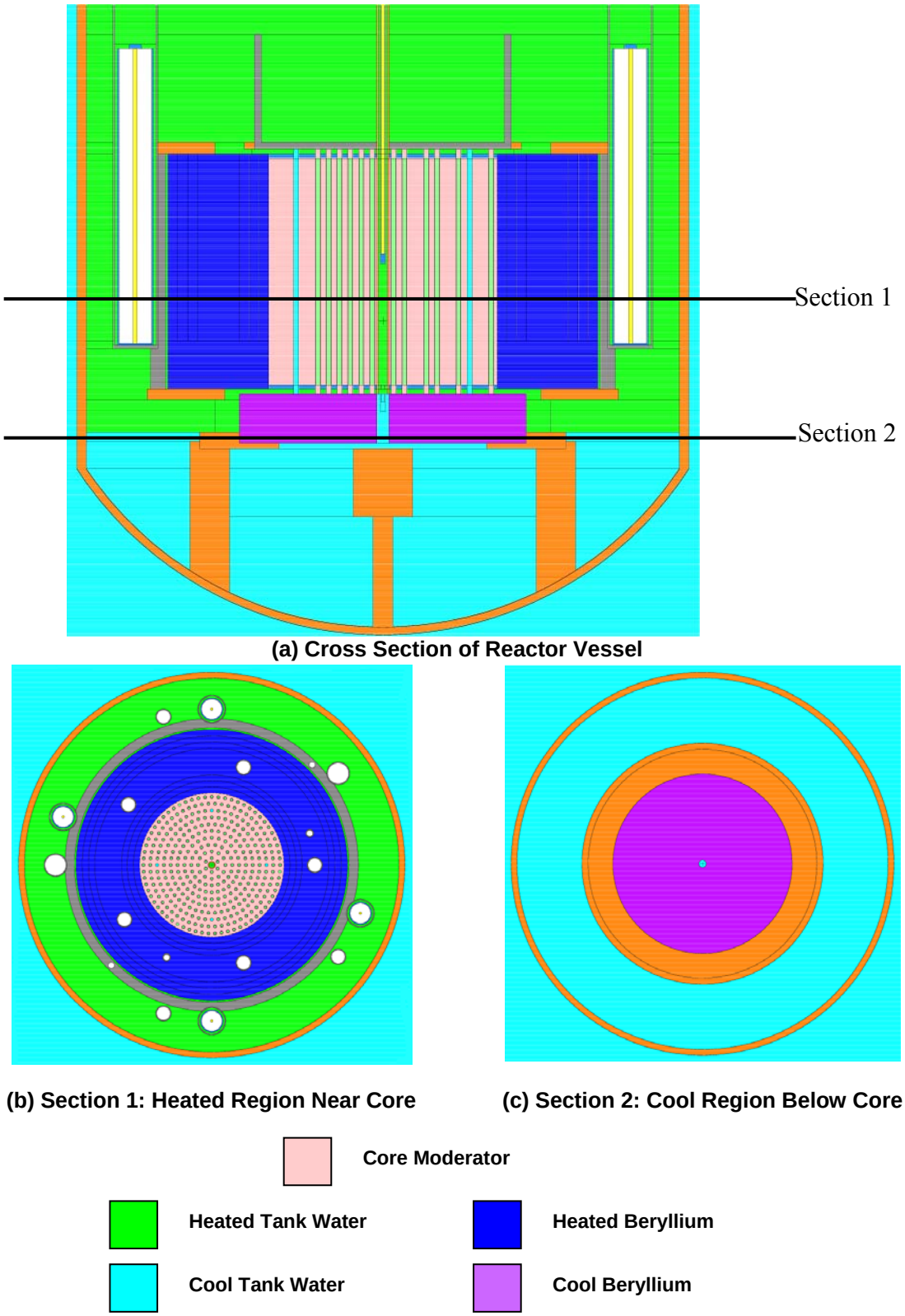

Figure 1. Schematics of the NIRR-1 MNSR Core. Regions Used to Calculate Reactivity Coefficients Are Highlighted.

Table 1. MNSR HEU and LEU Fuel Pin Dimensions

	HEU	LEU-12.45%	LEU-12.0%
Pin OD (mm)	5.5	5.5	5.1
Fuel OD (mm)	4.3	4.3	4.2
Gap size*(mm)	0.02	0.050	0.050
Fuel length (mm)	230	230	230
Total pin length (mm)	248	248	248

* Estimated

A radial Be reflector and bottom Be reflector surround the core. Beryllium shim plates are added on top of the core to compensate for reactivity losses due to burnup. Gaps between the bottom Be reflector and the radial Be reflector create flow orifices at the core inlet and outlet.

The core is installed a tank containing 1.5 m^3 of water. The tank is located in a pool containing 30 m^3 of water. There are no pumps or heat exchangers in the system. All coolant flow is due to natural circulation. Heat removal from the tank is by conduction through the tank wall to the pool water. Heat removal from the pool is by evaporation and conduction to the air, and by conduction through the wall of the pool.

RELAP5-3D MODEL

The RELAP5-3D thermal-hydraulic model used for these calculations includes the whole MNSR system, including core, water in the vessel and the pool. Perfect mixing of the coolant water was assumed as it emerges from the top of the core. Additional information on the model can be found in Ref. 3.

For the HEU core, the pin with peak power has 19% more power per pin than the average pin. For the LEU cores, the peak pin has 21% more power than the average pin. The peak pin coolant and the average pin coolant have the same coolant flow area per pin. In this model, the radial Be reflector transfers heat to the average pin coolant channel and the coolant down-flow.

MATERIAL PROPERTIES USED IN RELAP5-3D CALCULATIONS

Table 2 lists the material properties used in the HEU fuel cases and Tables 3-5 list the material properties used in the LEU fuel cases. The density x heat capacity used for the helium-filled gap in the LEU cases was 10,200 J/m^3/K. Also, at higher temperatures a small addition was made to the helium thermal conductivity to account for radiation heat transfer. The Be properties were the same in the HEU and LEU cases. The zircaloy-4 properties and the UO_2 properties were obtained from the International Nuclear Safety Center database (www.insc.anl.gov/matprop).

Table 2. Thermal Properties Used for HEU U-Al Alloy Fuel Pins

	Fuel	Gap	Al clad	Be
Thermal conductivity, W/m/K	167.6	0.0282	199.7	200
Density x heat capacity, J/m^3/K	2.24×10^6	10,200	2.42×10^6	3.38×10^6

Table 3. Thermal Properties Used for Uranium Dioxide with 95% Theoretical Density

Temperature, K	Thermal conductivity, W/m/K	Density x heat capacity, J/m^3/K
296.15	7.63	2.35×10^6
500	5.78	2.83×10^6
700	4.61	3.00×10^6

Table 4. Thermal Properties of Zircaloy-4

Temperature, K	Thermal conductivity, W/m/K	Density x heat capacity, $J/m^3/K$
295	13.383	1.9×10^6
400	13.987	1.9×10^6
500	14.741	1.9×10^6

Table 5. Thermal Conductivity of Helium-Filled Gap

Temperature, K	Thermal conductivity, W/m/K	Temperature, K	Thermal conductivity, W/m/K
280	0.1373	440	0.1916
340	0.1620	500	0.2116
400	0.1796	800	0.3121

REACTIVITY FEEDBACK

Extensive calculations using the MCNP-4C code[4] were performed to obtain power shapes and reactivity feedback coefficients for the HEU core and the two LEU cores. Figure 1 shows the geometry used for these calculations. Tables 6-9 show the kinetics parameters and reactivity feedback coefficients obtained from these calculations.

Table 6. Kinetics Parameters for the NIRR-1 MNSR Core

Parameter	HEU	LEU (12.45%)	LEU (12.0%)
Prompt Neutron Lifetime, μs	57.9 ± 4.8	47.0 ± 0.7	50.5 ± 4.4
β_{eff}, %	0.857 ± 0.0086[1]	0.845 ± 0.008[2]	0.832 ± 0.008[2]

[1]Rod Out, [2]Rod Inserted 15 cm

Table 7. Core Moderator Reactivity Changes

		Reactivity Changes for Moderator Temperature and Density, Δρ ($)		
T (°C)	Density, kg/m^3	HEU	LEU (12.45%)	LEU (12.0%)
19.85	998.34	0.0	0.0	0.0
30	995.67	-.1657 ± 0.009	-0.1467 ± 0.007	-0.138 ± 0.011
50	988.07	-.6324 ± 0.008	-0.5775 ± 0.007	-0.566 ± 0.011
60	983.24	-.9183 ± 0.008	-0.8272 ± 0.007	-0.8389 ± 0.011
100	958.58	-2.2859 ± 0.009	-2.1787 ± 0.007	-2.1827 ± 0.011

Table 8. Fuel Reactivity Feedback Coefficients

	HEU	LEU (12.45%)	LEU (12.0%)
Doppler Coefficient, $/°C	-0.00029 ± 0.00010[1]	-0.00092[2]	-0.00109[2]

[1]Rod Out, [2]Rod Inserted 15 cm

Table 9. Positive Reactivity Feedback Coefficients for Heated Tank Water and Radial Be Reflector

	HEU	LEU (12.45%)	LEU (12.0%)
Heated Tank Water Outside Core $/°C	+0.00591 ± 0.00023	-	+0.00647
Radial Be Reflector, $/°C	+0.00223 ± 0.00028	-	+0.00270

In addition to the negative reactivity feedbacks due to increasing the temperatures of the core water and fuel, positive reactivity feedback effects were calculated for the heated tank water outside the core and for the radial Be reflector. The water and Be reflector below the core do not heat up much during a rod withdrawal transient and were neglected in the feedback calculations.

Two issues related to the feedback coefficients are statistical uncertainties and the small difference between the coefficients obtained with the control rod at its critical position and the control rod fully-withdrawn. Since the neutronics calculations were done with the MCNP code, there is a statistical uncertainty associated with them. Also, most of the neutronics feedback calculations were done with the control rod near the critical position, whereas the control rod was fully-withdrawn for most of the rod withdrawal transient. Several calculations were done for the HEU core with the control rod fully-withdrawn to investigate this issue. It was found that for this case there was a small, but statistically significant, difference between the value of beta effective with the control rod at its critical position and with the rod fully-withdrawn. The rod-out value was used for the HEU calculations reported here. Also, the rod-out value for the Doppler feedback coefficient was barely statistically different from the rod-critical value; so the rod-out value was used. For the moderator density coefficient and the moderator temperature coefficient, the differences between rod-out and rod-critical values were small and not statistically significant. The rod-critical values were obtained with more neutron histories, giving better statistics. Therefore, the rod-critical values for the moderator feedback coefficients were used here.

HEU MODEL VALIDATION USING MEASURED DATA FROM NIRR-1

Measured data from the NIRR-1 reactor, both from operation at constant power and from a rod withdrawal transient was provided[1] by the Center for Energy Research and Training at Ahmadu Bello University for use in these analyses.

Operation at Constant Power

Figure 2 shows the measured and calculated inlet and outlet coolant temperatures and pool temperatures for a run in the NIRR-1 at a constant power of 15 kW.

The calculations shown in Figure 2 were done for a number of values of the orifice coefficients for the core inlet and outlet orifices, with the inlet and outlet orifice coefficients set to the same value. Results for orifice coefficients of 1.35, 1.40 and 1.45 are shown. A value of 1.40 was used in the rod withdrawal cases described below. In the later parts of the transient, heat losses not included in the RELAP5-3D model probably have some impact on the measurements.

In general, the calculated inlet and outlet temperatures agree well with the measurements, except possibly near the end of the transient. This indicates that for most of the transient the RELAP3-3D model accurately accounts for all of the significant heat transfer in the system. In the early part of the transient, from 500 to 3,000 seconds, the rise in the inlet and outlet temperatures is determined mainly by the heat capacity in the water in the vessel. Later in the transient, from 7,500 to 17,500 seconds, the heat generated in the core is transferred through the vessel wall to the pool about as fast as it is generated. In this time span, the heat capacity of the water in the pool dominates the temperature rise. After 17,500 seconds (4.9 hours), the calculated inlet and outlet temperatures rise toward the top of the experimental data band. This is probably due to heat transfer from the pool to the air above and to the pool wall. These heat transfer paths are not included in the RELAP5-3D model, but they could be included in future models.

At the beginning of operation, when all of the temperatures in the system might be expected to be the same, the measured pool temperature was about 1.5 ^{o}C above the core inlet and outlet temperatures. During operation, the measured pool temperature was consistently 1.5 – 2.0 degrees above the calculated pool temperature. The 1.5 – 2.0 degree temperature difference may have occurred because the pool temperature and the vessel temperature were not in equilibrium at the start of operation.

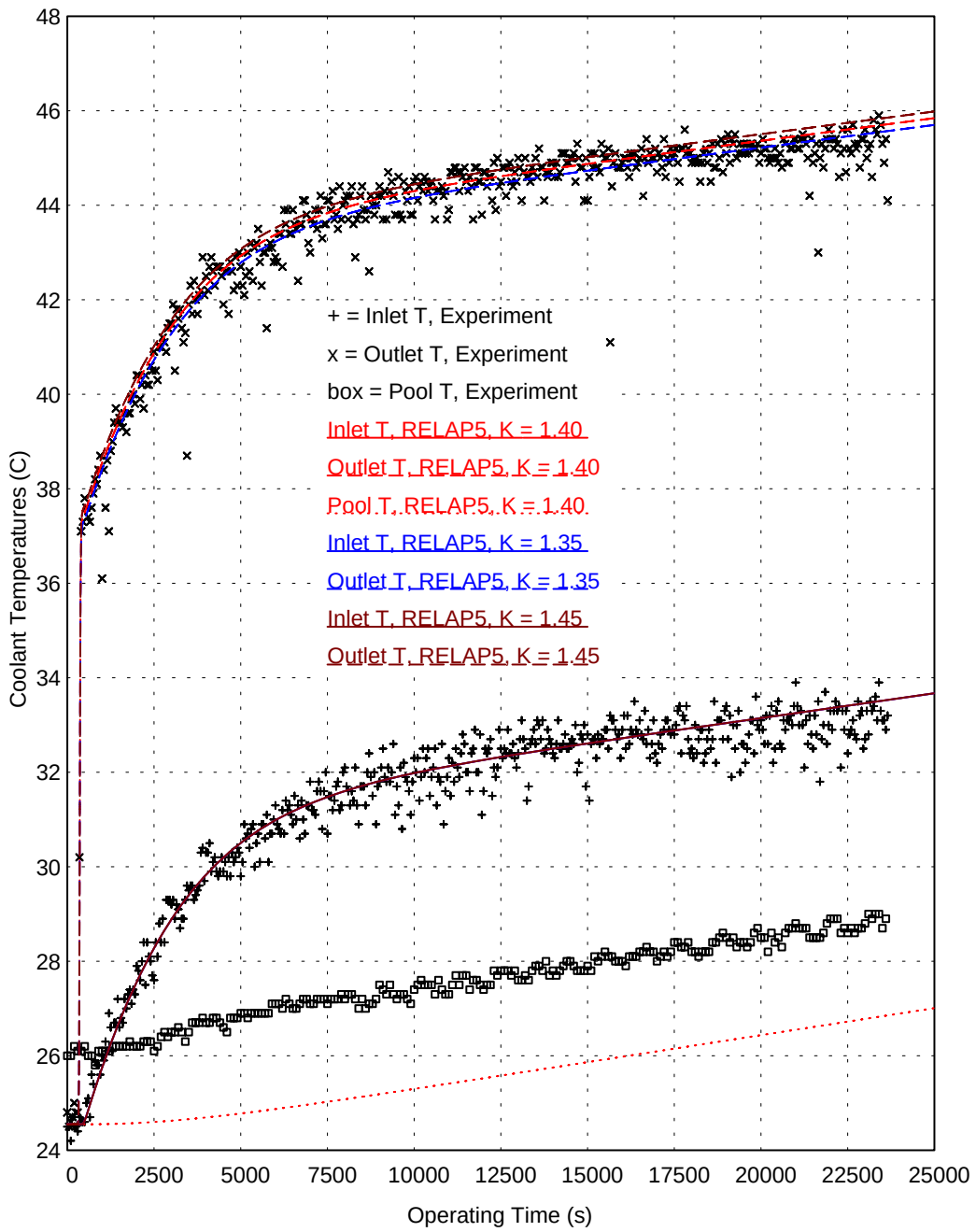

Figure 2. Core Inlet and Outlet Coolant Temperatures for NIRR Operation at Constant Power, Extended Time Scale

Rod Withdrawal Transient, HEU Fuel

Figures 3 and 4 compare experimental measurements and calculated results for the 3.77 mk reactivity insertion with HEU fuel. The calculated results agree almost exactly with the measured data. This shows that the RELAP5-3D thermal-hydraulic model and the reactivity feedback coefficients accurately model the NIRR-1 MNSR reactor with HEU fuel.

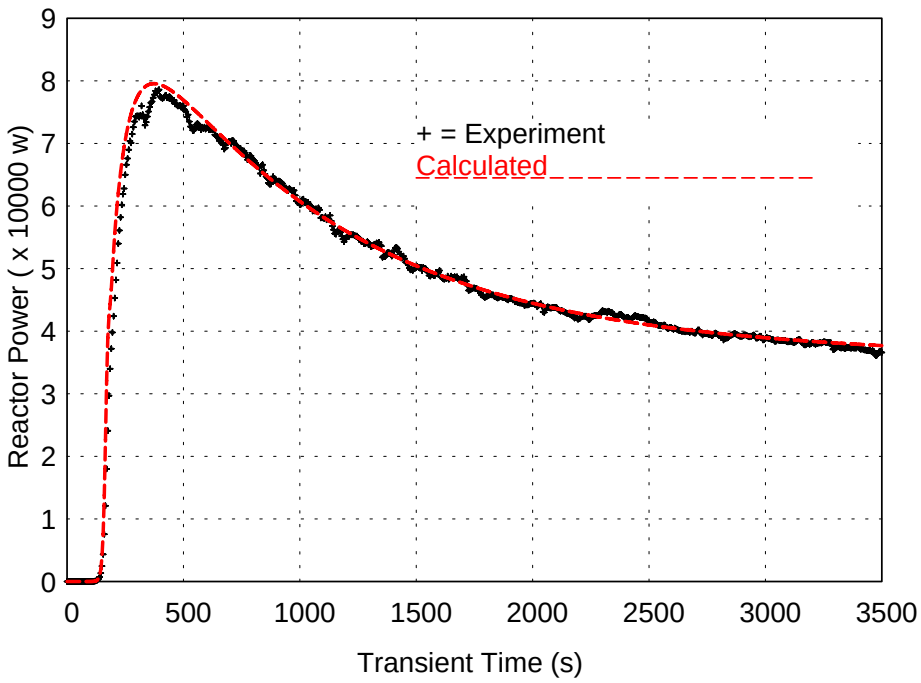

Figure 3. Reactor Power for MNSR 3.77 mk Reactivity Insertion, HEU Core

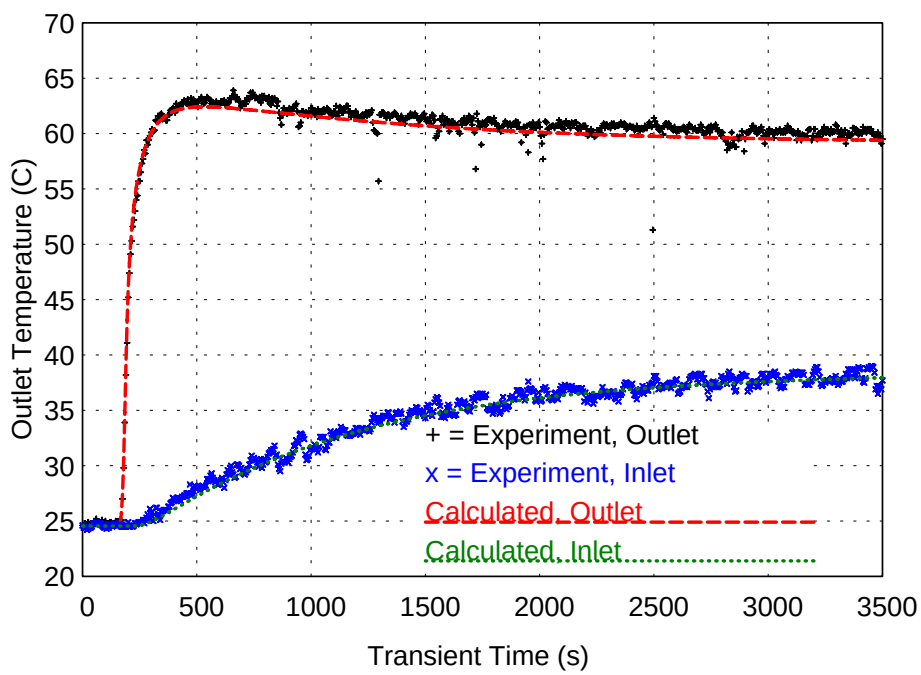

Figure 4. Inlet and Outlet Temperatures for MNSR 3.77 mk Reactivity Insertion, HEU Core

Transient calculations for research reactors are usually made using reactivity feedback for core moderator temperature, core moderator density and core fuel temperature. Feedback from beyond the core is usually neglected. Positive reactivity from heated water above the core and from the radial beryllium reflector are included in the transient calculations described above.

LEU FUEL ROD WITHDRAWAL RESULTS

The validated RELAP5-3D model used for analysis of the HEU core was modified appropriately for LEU cores with the two proposed UO_2 fuel pin designs.

Nominal 3.77 mk Reactivity Insertion Cases

Figures 5 and 6 show the results of a 3.77 mk reactivity insertion for the HEU case and for the two LEU cases. The powers and peak clad temperatures with LEU fuel are lower than those with HEU fuel. The peak fuel temperatures are significantly higher for the LEU cases because of the lower thermal conductivity of the LEU oxide fuel.

Since the melting temperatures of the fuel and cladding for the LEU fuel are much higher than those for the HEU fuel, the safety margins for this transient are significantly larger for the LEU fueled cores. The Al cladding of the HEU fuel begins to melt at about 650 ^{o}C, whereas the Zr-4 cladding of the LEU fuel begins to melt at about 1850 ^{o}C. The melting temperature of HEU U-Al alloy is about 650 ^{o}C. The melting temperature of the UO_2 of the LEU fuel is about 2800 ^{o}C.

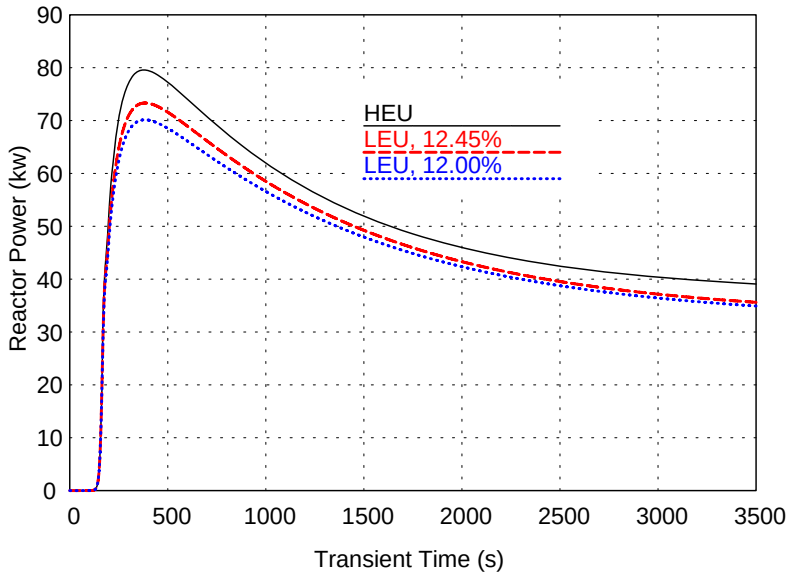

Figure 5. Power vs. Time for a 3.77 mk Reactivity Insertion with HEU and LEU Fuel

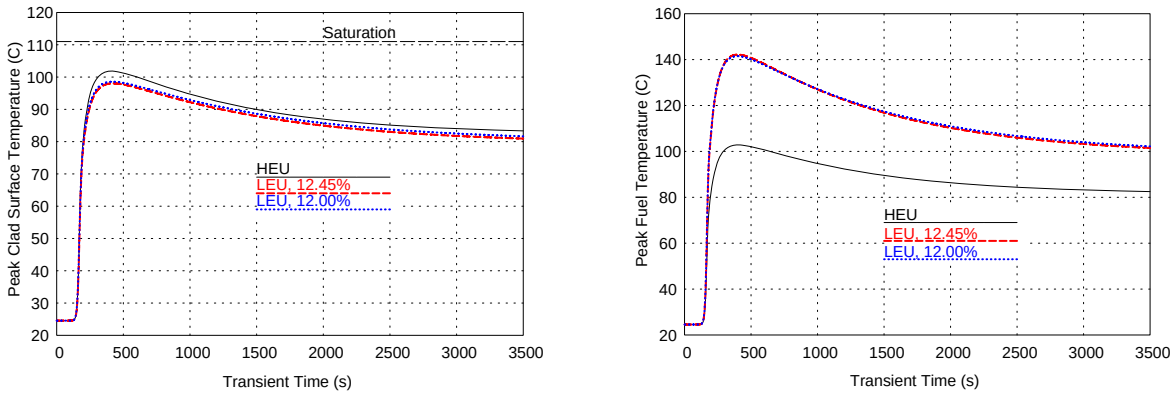

Figure 6. Peak Clad Surface Temperature and Peak Fuel Temperature for a 3.77 mk Reactivity Insertion with HEU and LEU Fuel

Influence of Gap Size in the LEU Fuel Pins

In order to assess the impact of a variation in the gap between the as-fabricated LEU fuel pellets and the zircaloy-4 cladding, 3.77 mk reactivity insertion cases were run with gap sizes of both 50 microns and 100 microns. Figures 7 and 8 show the results of these analyses. As expected, a larger gap size results in a higher fuel temperature, larger Doppler reactivity feedback, and lower peak transient power.

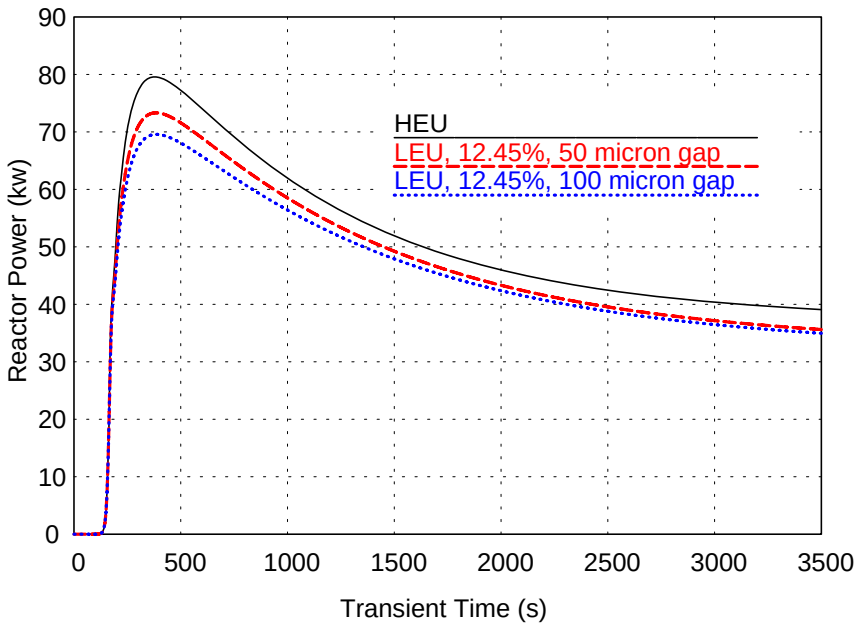

Figure 7. Influence of Gap Size on 3.77 mk Reactivity Insertion, Powers

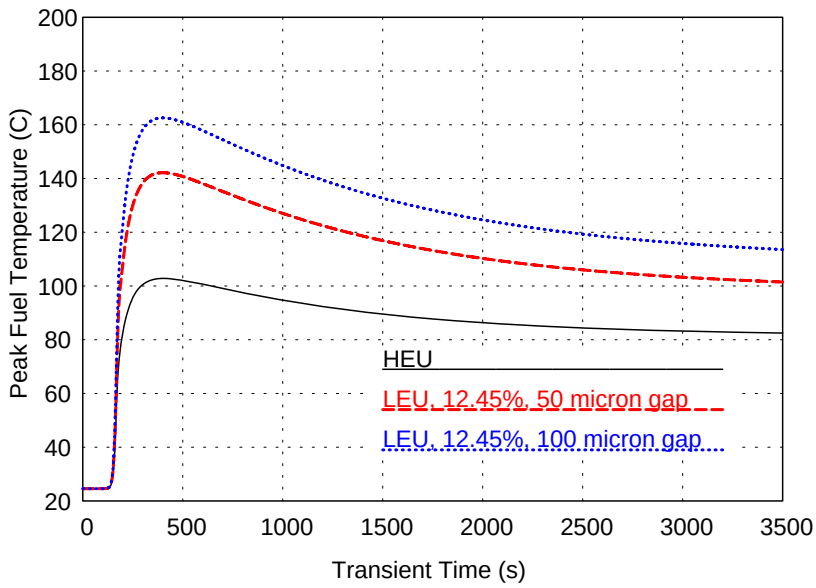

Figure 8. Influence of Gap Size on 3.77 mk Reactivity Insertion, Fuel Temperatures

Fuel Cracking

Cracking of uranium dioxide fuel during irradiation has been addressed by many authors[5,6]. The maximum thermal stress at the time of cracking is set equal to the fracture strength. The maximum thermal stress is proportional to the pin power rating (kW/m/pin), but it is independent of the pin diameter. The fracture strengths listed[6] for various UO_2 samples range from 8,000 to 20,000 psi. This approach gives a pin power rating at fracture of 3.4 – 8.8 kw/m/pin. The average pin power rating for an MNSR operating at a reactor power of 30 kW is 0.4 kW/m/pin. Therefore, there should be no cracking the LEU fuel pellets in an MNSR unless the power gets much higher than nominal. Also, the fuel temperatures and burnups in MNSR reactors are low enough that there should be very little release of fission product gas from the fuel.

If for some reason, the MNSR LEU fuel did crack, it would probably have little or no impact. Power reactors routinely operate with cracked oxide fuel. If radial cracks occur in the fuel, the fuel tends to move out until it comes into contact with the cladding. Thus the main consequence of cracking fuel is an increase in the bond gap conductance. A typical value for the bond gap conductance with cracked fuel is 1 W/cm^2-K, as compared to a value of 0.3 W/cm^2-K for uncracked fuel with a 50 micron helium-filled gap. To determine the impact of increasing the bond gap conductance, the same 3.77 mk reactivity insertion transient was run with uncracked and cracked fuel bond gap conductances. The results are shown in Figures 9 and 10. The cracked fuel case gives somewhat lower fuel temperatures and slightly higher powers than the uncracked case. There would be no safety issues related to the integrity of the fuel cladding.

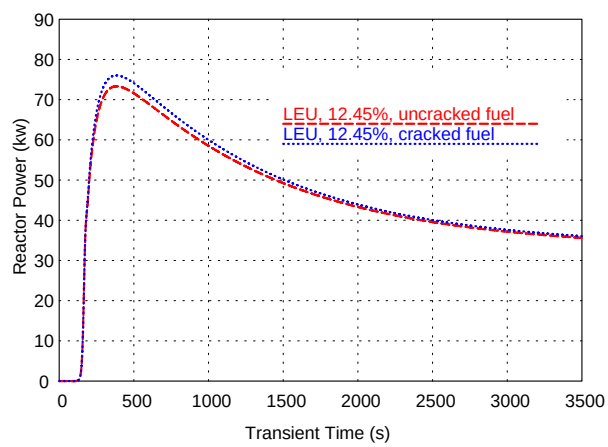

Figure 9. Impact of LEU UO_2 Fuel Cracking, Powers for a 3.77 mk Reactivity Insertion

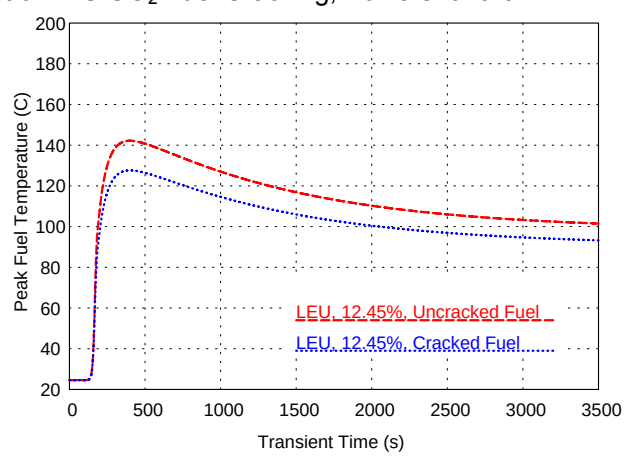

Figure 10. Impact of LEU UO_2 Fuel Cracking, Fuel Temperatures for a 3.77 mk Reactivity Insertion

Reactivity Insertion of 6 mk

For the current MNSR design, the maximum allowed reactivity insertion in the Operating Limits and Conditions due to full withdrawal of the control rod is 4 mk. To provide an indication of the margin of safety inherent in the LEU fueled core, a case with a reactivity insertion of 6 mk was also done for the core with 12.5% enriched fuel, a 50 micron gap, and uncracked fuel pellets. The results are shown in Figures 11 and 12. For the 6 mk case, the power goes to an initial sharp spike of 119 kW during the reactivity insertion and then drops after the insertion ends. The power then goes up more gradually to 132 kW before dropping gradually. The clad surface temperature exceeds the water saturation temperature, and sub-cooled boiling occurs for over 1,000 seconds. The peak fuel temperatures for the 6 mk case rises to about 200 ^{o}C, about 60 ^{o}C higher than for the 3.77 mk case.

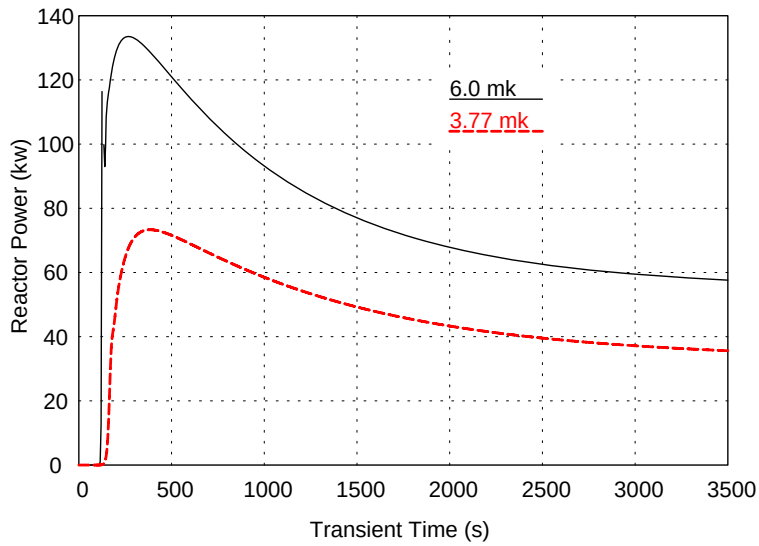

Figure 11. Reactor Power for 6 mk Reactivity Insertion, LEU UO_2 Fuel, 12.5% Enrichment, 50 Micron Gap, Uncracked Fuel Pellets

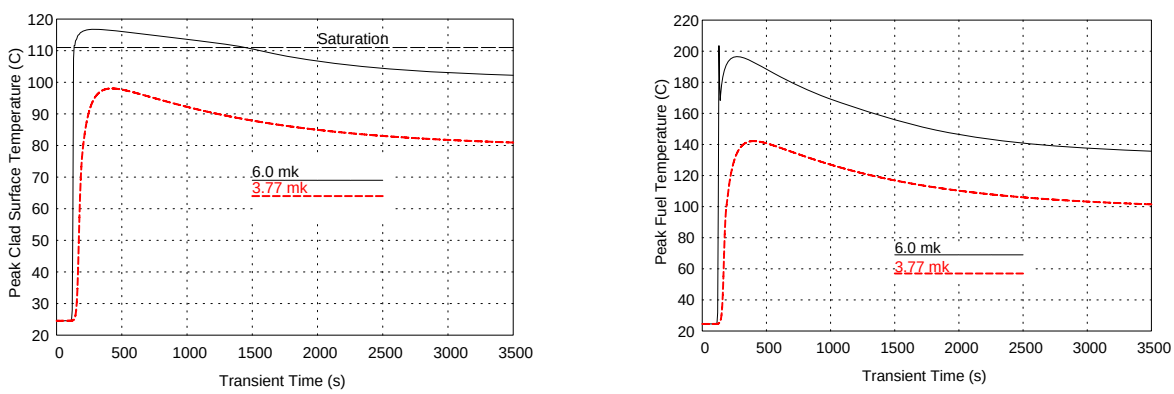

Figure 12. Peak Clad Temperatures and Peak Fuel Temperatures for 6 mk Reactivity Insertion, LEU UO_2 Fuel, 12.5%, Enrichment, 50 Micron Gap, Uncracked Fuel Pellets

LEU STEADY-STATE THERMAL HYDRAULIC SAFETY MARGINS

Steady-state thermal hydraulic safety margins were calculated for the 12.45% LEU case. The results are shown in Figures 13 and 14. The peak clad surface temperature rises almost linearly with power to a reactor power of about 90 kW. At this point sub-cooled boiling begins. The boiling reduces the rate of rise of the clad temperatures. Onset of significant voiding occurs at a power of about 350 kW.

The coolant flow rate rises approximately linearly with power up to the onset of significant voiding. At 300 kW and above, flow oscillations occur in RELAP3-3D, so there is not a single steady-state flow rate for each power level. In this range, the flow rates shown in Figure 14 are approximate averages of the oscillating values.

The peak fuel temperature rises linearly with increasing power to the onset of sub-cooled boiling at a power level of 90 kW. At that point, the slope of the curve changes due to the increase in the coolant heat transfer coefficient after the start of sub-cooled boiling.

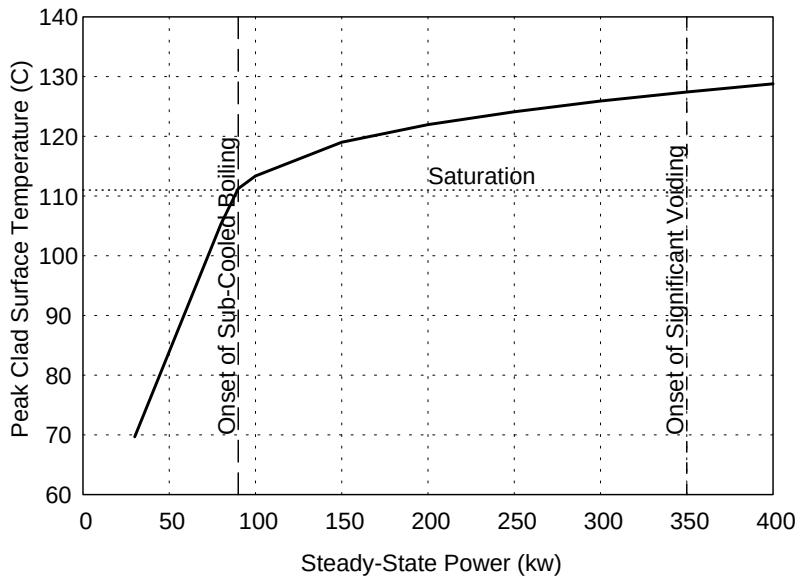

Figure 13. Steady-State Operation of the 12.45% LEU Core, Clad Temperatures

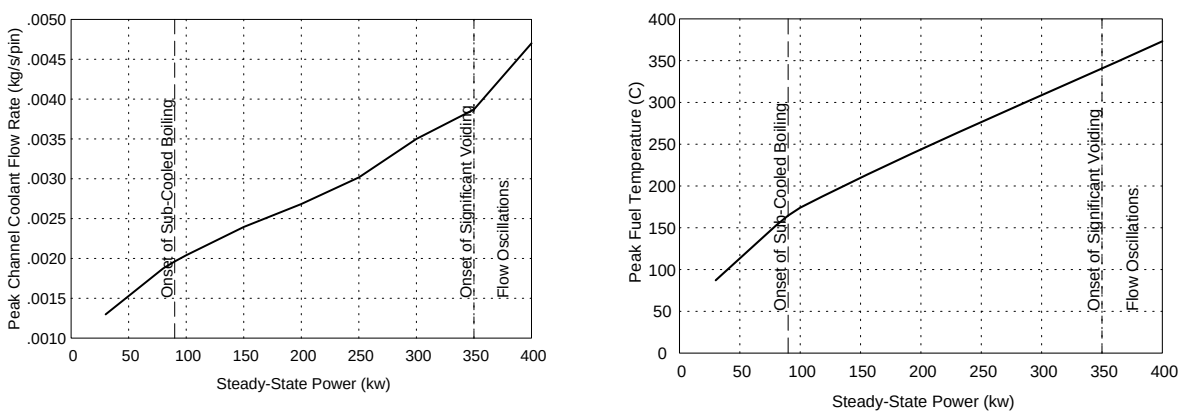

Figure 14. Steady-State Operation of the 12.45% LEU Core, Coolant Flow Rates and Fuel Temperatures

CONCLUSIONS

Comparison with experimental data taken during commissioning of the NIRR-1 MNSR reactor in Nigeria shows that the steady-state and transient behavior of the HEU core is accurately predicted by the RELAP5-3D calculations. The same validated models and methodology were used for similar transients for LEU cores with two UO_2 fuel pin designs.

Because of much higher melting temperatures of the LEU UO_2 fuel pellets and the zircaloy-4 cladding, the LEU cores will have significantly larger safety margins than the current aluminum-based fuel of the HEU core.

Moderate uncertainties in the size of the as-fabricated gap between LEU fuel pellets and the zircaloy-4 cladding are not important.

Oxide fuel pellets should not crack at the low MNSR power levels. If cracking did occur, it would not be important.

The steady-state thermal-hydraulic safety margins in the HEU and LEU cores are very large. Onset of nucleate boiling occurs at a power level of about 90 kW – approximately three times the maximum licensed power level. Onset of significant voiding was calculated to occur at about 350 kW – more than ten times the maximum licensed power level. Departure from Nucleate Boiling (DNB) would occur at a still higher power level.

An LEU conversion demonstration of an MNSR reactor and a transient run with a reactivity insertion of ~ 4 mk is needed to validate the transient results shown in this paper.

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