MNSR TRANSIENT ANALYSES AND THERMAL-HYDRAULIC SAFETY MARGINS FOR HEU AND LEU CORES USING PARET

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ABSTRACT

Thermal-hydraulic performance characteristics of Miniature Neutron Source Reactors under long-term steady-state and transient conditions are investigated. Safety margins and limiting conditions attained during these events are determined. Modeling extensions are presented that enable the PARET/ANL code to realistically track primary loop heatup, heat exchange to the pool, and heat loss from the pool to air over the pool. Comparisons are made of temperature predictions for HEU and LEU fueled cores under transient conditions. Results are obtained using three different natural convection heat transfer correlations: the original (PARET/ANL version 5), Churchill-Chu, and an experiment-based correlation from the China Institute of Atomic Energy (CIAE). The MNSR, either fueled by HEU or by LEU, satisfies the design limits for long-term transient operation.

1. Introduction

Miniature Neutron Source Reactors have been studied by the RERTR Program for over a year [1]. The methodology for both neutronic and thermal-hydraulic analysis has been improved to account for the transient characteristics of these reactors. A companion paper [2] describes an approach using the RELAP5-3D code. This paper addresses MNSR transient analyses based upon extensions to the modeling capabilities of the PARET/ANL code. PARET/ANL version 7.3 is intended for the analysis of research and test reactors using both plate-type (flat) and pin-type

fuel elements. PARET is an acronym for: "**P**rogram for the **A**nalysis of **RE**actor **T**ransients." The original code from 1969 has been under periodic modification, extension, and improvement since the 1980's at ANL [3-7]. It has been used for many research reactor conversion studies in support of reactor licensing. The key change in version 7.3 is the addition of an external primary loop and pool. In prior versions of PARET, coolant discharged from a channel was essentially lost, and had no impact on subsequent coolant inlet temperature. MNSR's operate in a closed-loop mode inside a vessel that in turn is located inside a pool. It is essential, when simulating many-hour transients, to account for the primary coolant heatup and temperature feedback effects. Results will be shown for transient analysis against experiment, obtained in the NIRR-1 MNSR in Nigeria, for a limiting reactivity insertion event of 3.77 mk in which the inlet temperature rise is about 13 C. Similar results will also be presented for 7-hour operation at 15 kW, in which the inlet temperature rises by about 8 C. Similar results for two LEU cores will also be presented.

2. Description of the MNSR

Figures 1 and 2 show the arrangement of the NIRR-1. This reactor was designed by the China Institute of Atomic Energy (CIAE). The reactor uses rod-type fuel (347 fuel pins and 3 dummies) with aluminum clad. A typical fuel rod has a diameter of 5.5 mm, containing HEU in the form of Al-26.1wt % U at 90.2 % enrichment. The fuel meat is 4.3 mm o.d. The ^{235}U loading in the core is 999 g. The LEU fuel rod proposed for conversion contains 12.45% enriched uranium as UO_2 pellets in zircaloy-4 cladding of the same dimensions as the HEU rod. Reactor control is provided by one central control rod consisting of a cadmium rod clad in stainless steel. The core is contained in an aluminum tank of light water surrounded by an annular Be reflector. It is cooled and moderated by light water, with beryllium reflectors on the side and bottom. The radial beryllium reflector, in blue, is heated by the primary loop flow as well as by direct heating. The lower beryllium reflector is expected to remain relatively cool. The water in light blue in the lowest part of the vessel will not participate significantly in the coolant circulation, and is therefore decoupled from transient events. Heated primary loop water is shown in pink. Temperature feedback coefficients were obtained to represent the heated water and the fuel Doppler effect. A small, but important, positive feedback correction was used to represent the consequences of heating the radial beryllium and the water above the top of the core. Natural convection flow removes heat from the core and deposits some of it through the tank wall to the surrounding pool, and to the radial beryllium reflector. As reactor operation continues, the primary loop water returning to the core gradually warms.

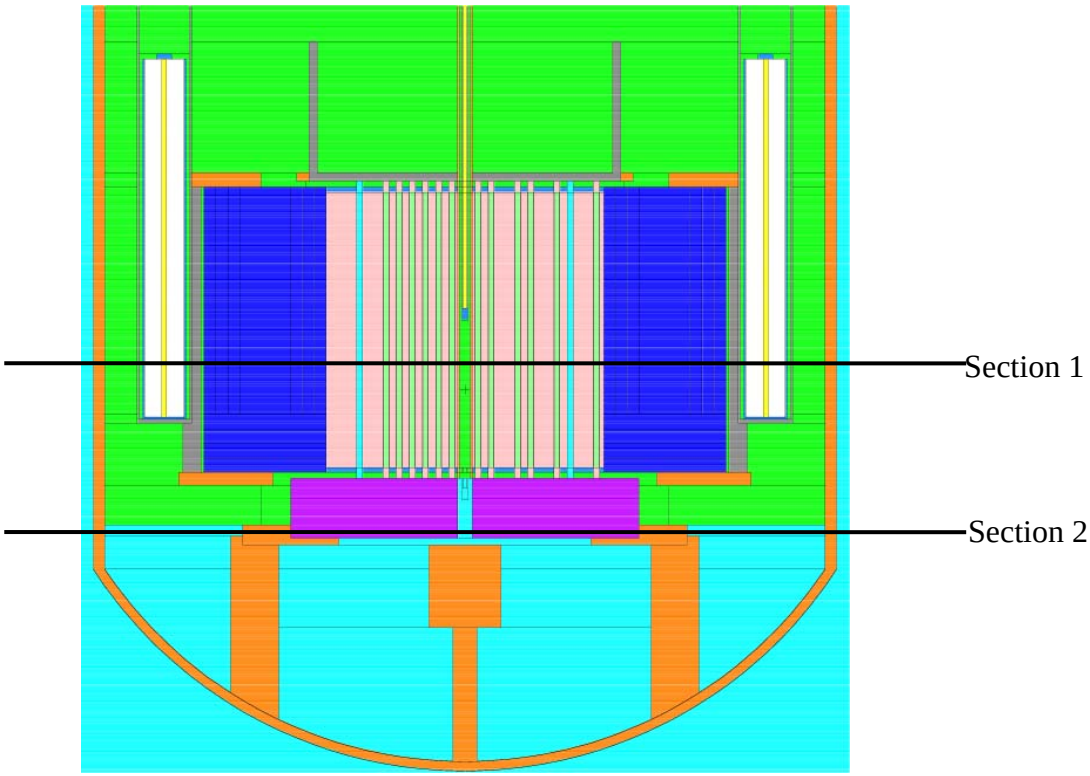

Figure 1. MNSR Layout: Side View

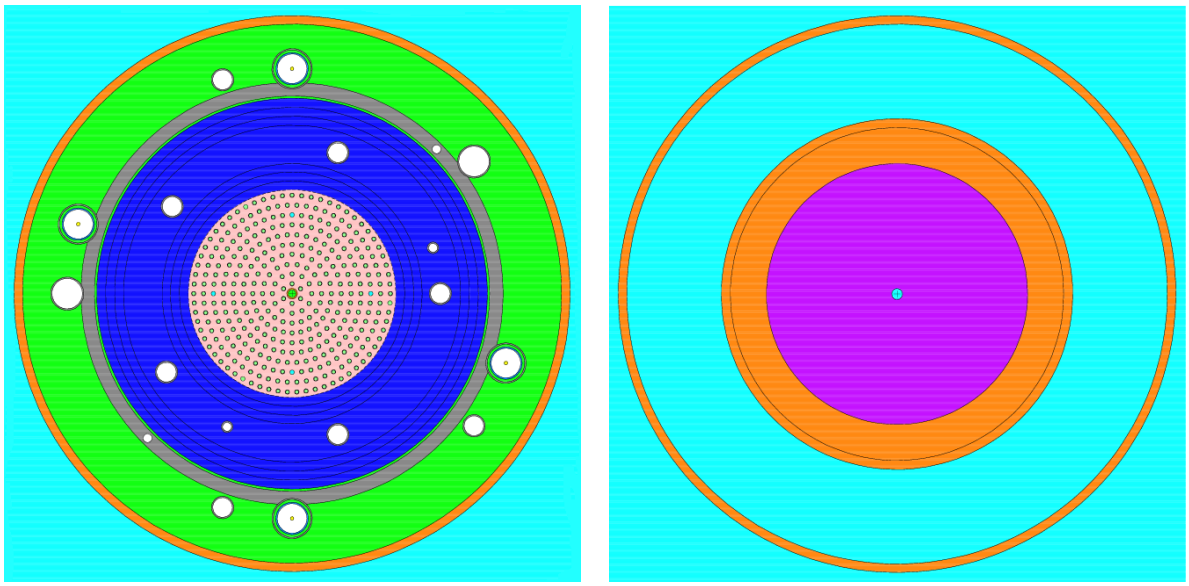

Figure 2. MNSR Layout: Top View (Section 1 on left; Section 2 on right)

3. PARET Thermal-Hydraulic Model

The water volume that is actively involved in natural convection cooling in the core tank (primary loop) is treated as passing through an inlet pipe section, the core, an outlet pipe section, and a lump volume. The pipe sections are assumed to be 10% of the active primary loop volume each, and are subdivided into 40 nodes. Coolant is tracked through these nodes and the lump based upon volumetric flow rate discharged from the core. By “tracking” is meant that the coolant in each node is subjected to heat and mass balance equations which are integrated over a time step typically much longer than that needed for the reactor kinetics calculations. A one-second step size was used in this work. Heat is exchanged through a tank surface area between the primary loop and the pool, using a specified heat-transfer coefficient. This heat transfer coefficient can be obtained by other means, such as from a RELAP model, or simply fitted to the experimental results described below. The pool acts as a secondary loop. Pool heatup is modeled as a lump receiving heat from the primary loop through the tank surface, and at the same time being cooled by air flow over the pool surface causing evaporation. The pool air temperature, humidity, and velocity control the evaporation rate of the water at the pool surface and affect its bulk temperature. These parameters can be specified along with the pool surface area, volume, and initial temperature. Since the pool temperature changes only a few degrees over many hours, it is not necessary to model it in greater detail.

Evaporative cooling from the pool surface is computed using an ASHRAE model [8]. The evaporation rate is w_p (kg/s) = $A(p_w-p_a)(0.089 + 0.0782\ V)/Y$ where

A =area of pool surface, m^2

p_w =saturation vapor pressure taken at surface water temperature, kPa

p_a =saturation vapor pressure taken at surface water temperature, kPa

V =air velocity over water surface, m/s

Y =latent heat required to change liquid to vapor at surface water temperature, kJ/kg.

The original PARET model for natural convection was based on fitting SPERT III transient tests in the “E-core” under no forced flow conditions [10]. These tests used pin-type fuel clad in stainless steel (11.83 mm o.d. by 1.036 m length, 5x5 rods per assembly in a 76.2mm x 76.2 mm box with holes in the sides). The coolant passed up through the core components and then reversed direction to flow downward through the thermal shields surrounding the core. This flow path was quite similar to that of the MNSR. But the active height of SPERT III E-core fuel was 97.3 cm, which is substantially larger than the 23 cm active length of a typical MNSR. Also, the coolant passages were partially isolated in a square lattice, with a hydraulic diameter of about

11.9 mm, compared to an open rod lattice (neither square, nor triangular, but in 10 rings) of 2.25 mm for a MNSR. The pitch/diameter ratios were 1.258 and ≈ 2.0 for the E-core, and for the MNSR. These differences have large impacts on the development of the natural convection flow during reactor operation as they affect the driving head, the heat transfer coefficient, and the friction losses. They make this model problematic for application to MNSR transients. In principle, the resulting heat transfer coefficient could be scaled by user input. But in fact, the code logic of PARET prevented this. In PARET 7.3, this logic was revised to make the default correlation scalable.

Recent work on extending PARET options in various heat transfer and flow regimes has resulted in adding many of the RELAP5-3D Athena correlations for vertical pipe flow. The Churchill-Chu correlation [9] for natural convection was developed for a vertical flat plate, but is used in RELAP5-3D for round pipes as well. It is:

$Nu = [0.825 + B/C]^2$ where

$B=0.387(Ra)_L)^{1/6}$ and $C=[1+0.492/Pr)^{9/16}]^{8/27}$; Gr =Grashoff number = $\rho^2 g\beta(T_w - T_b)L^3/\mu^3$

Pr = Prandtl number = $\mu c_p/k$; L is the natural convection length

T_w refers to wall temperature; T_b refers to bulk coolant temperature

For the MNSR's, the Chinese Institute of Atomic Energy carried out measurements of heat transfer on a full scale simulation facility. Chimney heights ranging from 20-100 cm were studied. The effects of changing the inlet and outlet orifices were also studied. CIAE determined the following simple correlation, applicable to the explicit geometry and flow conditions of the MNSR's:

$Nu=n\ (Gr\cdot Pr)^m$

where n = 0.68, and m = ¼, when $Gr\cdot Pr < 6 \times 10^6$

where n = 0.174, and m = 1/3, when $Gr\cdot Pr \geq 6 \times 10^6$

This is a new option that compares very well against experimental data for the Nigeria MNSR (NIRR-1), and when compared to the other two natural convection heat transfer options in PARET 7.3.

Two key parameters arise when simulating the natural convection flow. They are: 1. the measured temperature rise across the core; and 2. the peak clad temperature. Friction losses in the heated section of the core are small compared to K-losses in the inlet and outlet orifices. Hence the choice of laminar friction factor correlation for the core lattice is not very important. We used a laminar flow friction factor of (64/1.6)/Re for this work, with an auxiliary factor of 1.6 to correct for the lattice pitch-to-diameter ratio. Friction losses overall largely control the primary

loop flow rate, which by energy balance determines the temperature rise across the core. Under natural convection, the flow is in balance when the gravitational driving head from density differences equals the total pressure drop from friction losses. The chosen heat transfer correlation will define the peak fuel and clad temperatures that apply for a given flow solution.

4. Steady-State Long-Time Operation

MNSR's slowly heat up over time, once they reach operating steady-state power. For the NIRR-1, the maximum licensed power is 30 kW, but the normal operating power is 15 kW. A comprehensive series of temperature measurements were made in the NIRR-1 over a period of about 6 hours that was used as a key benchmark to fine-tune the PARET 7.3 external loop model. Flow resistance was varied to establish the correct core temperature rise. Heat transfer from the primary vessel to the pool was then adjusted to achieve the desired pool heatup rate. Finally, a small positive reactivity insertion was calculated and included in order to account for the effect of the heating of the radial Be reflector and water above the core. In this model it is assumed to be maintained at the temperature of the primary loop outlet. Fig. 3 shows how the model performs *vs*. experiment.

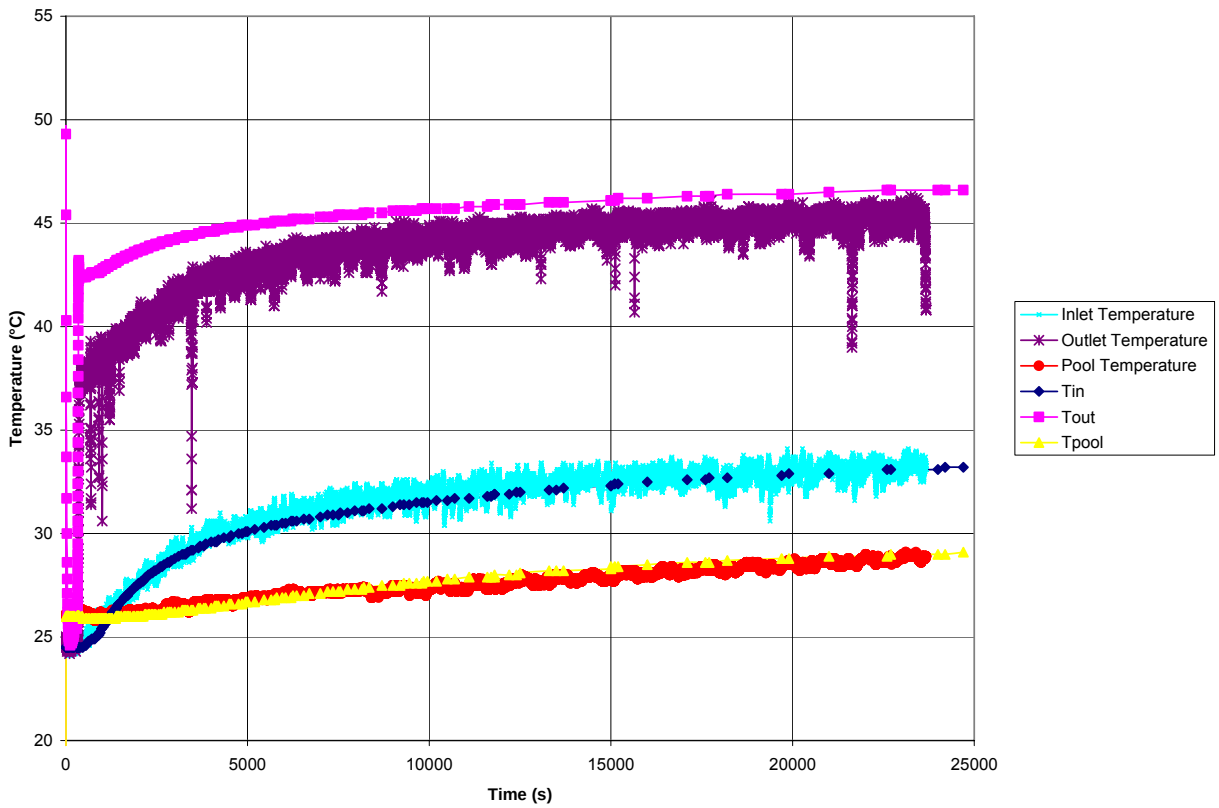

Fig.3. NIRR-1, HEU FUEL: 15 kW Steady-State Operation

5. Transient Analysis: HEU

An experiment was performed in the NIRR-1 using a slow reactivity ramp of 3.77 mk. The maximum allowed excess reactivity is 4 mk. This event, shown in Figs. 4-5, was studied with PARET 7.3 using the same model parameters as were developed for the steady-state operation. Differences between the three natural convection models (CIAE, RELAP5-3D/Athena, and original PARET 5) were relatively small. The CIAE correlation predicted no onset of nucleate boiling, as did the original PARET natural convection model. In contrast, the Athena (Churchill-Chu correlation as in RELAP5-3D) case predicted very localized nucleate boiling from about 470-690 seconds, while the power exceeded 70 kW. Because the CIAE correlation was experimentally determined for the specific fuel geometry of the NIRR-1, it should be the most reliable method.

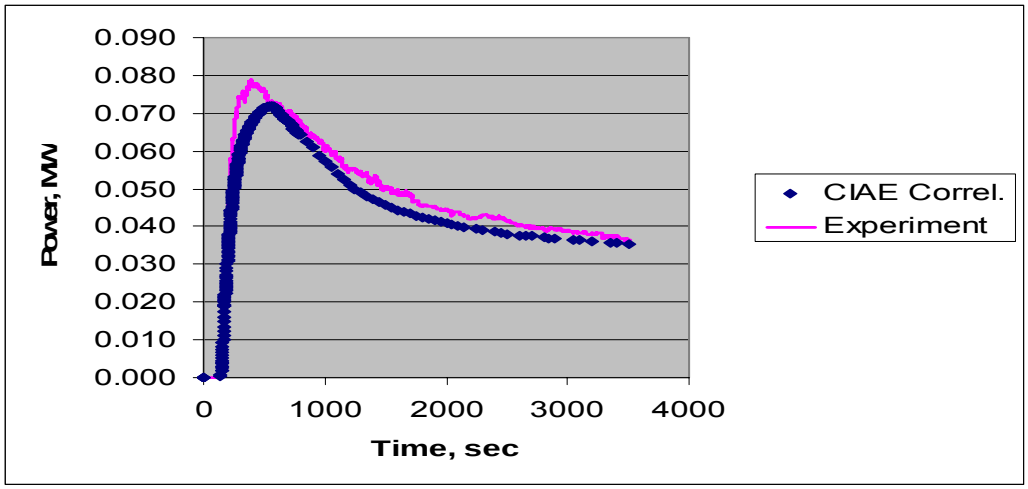

Fig. 4. HEU, 3.77 mk Transient: Power

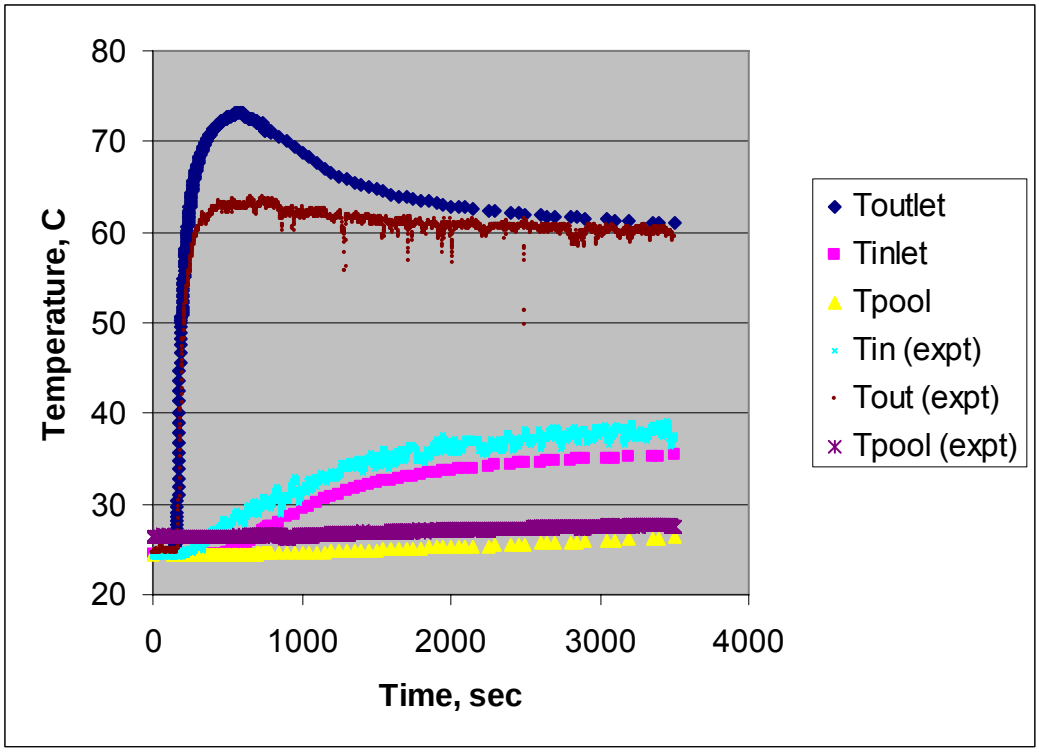

Fig. 5. HEU, 3.77 mk Transient: Temperatures

6. Transient Analysis: LEU

For LEU fuel, the results shown in Figs. 6-7 are very similar, whether using 12.45% or 12% enriched fuel. A helium-filled 50 micron gap between the pellet and clad was assumed. Depending upon the choice of natural convection model, peak power ranges between 84-92 kW for the 12 % case, and between 94-102 kW for the 12.45 % case. Nucleate boiling begins at 108 kW. The PARET model predicts numerically unstable flow conditions to occur near 205 kW. Using mass flow rate data from RELAP5-3D calculations, the PARET model predicts numerically stable coolant flow up to 245 kW. A definitive power limit is currently being investigated.

As was found for the HEU case, there is no onset of nucleate boiling predicted by either the CIAE or the original PARET natural convection models. The Athena model predicts very localized nucleate boiling from about 470-690 seconds, during which time the power exceeded 70 kW. Peak fuel centerline temperature attained ranged over 185-196 C, while peak clad surface temperature was 113-121 C. Peak water coolant temperature ranged over 82-84 C. For

UO_2 fuel clad in Zr-4, these temperatures are not at all limiting since the UO_2 melting temperature is about 2800 C while that of Zr-4 is about 1850 C.

In steady state conditions with a fixed inlet temperature, unstable coolant flow occurs at a few kW above 205 kW when using the CIAE correlation. DNB is well beyond that, but is never reached in these calculations. For DNB, the 2006 Groeneveld CHF lookup table [11] was used. The prediction of CHF for an MNSR from that table is reliable, but conditions at CHF were not attainable in either PARET 7.3 or in RELAP5-3D computations. CHF is not a limiting safety issue because it will occur at very high power. The DNBR is at least 7, and could be as high as 10 based upon currently available analyses.

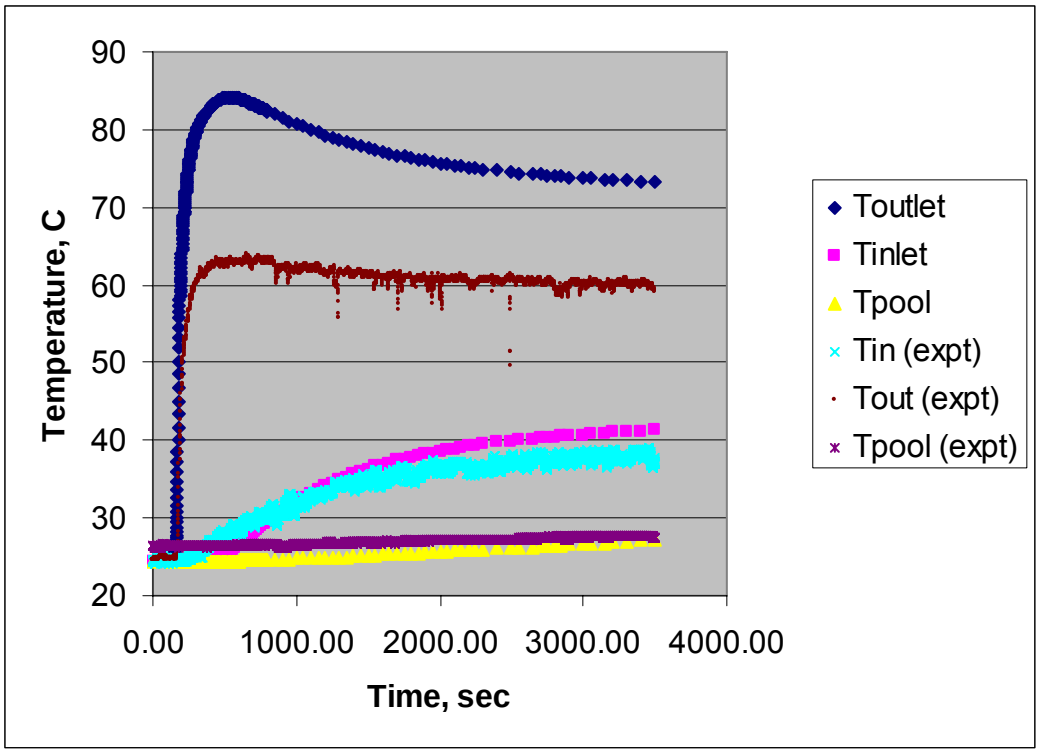

Fig. 6. NIRR-1, LEU, UO_2 12.45% Enriched, 3.77 mk Transient

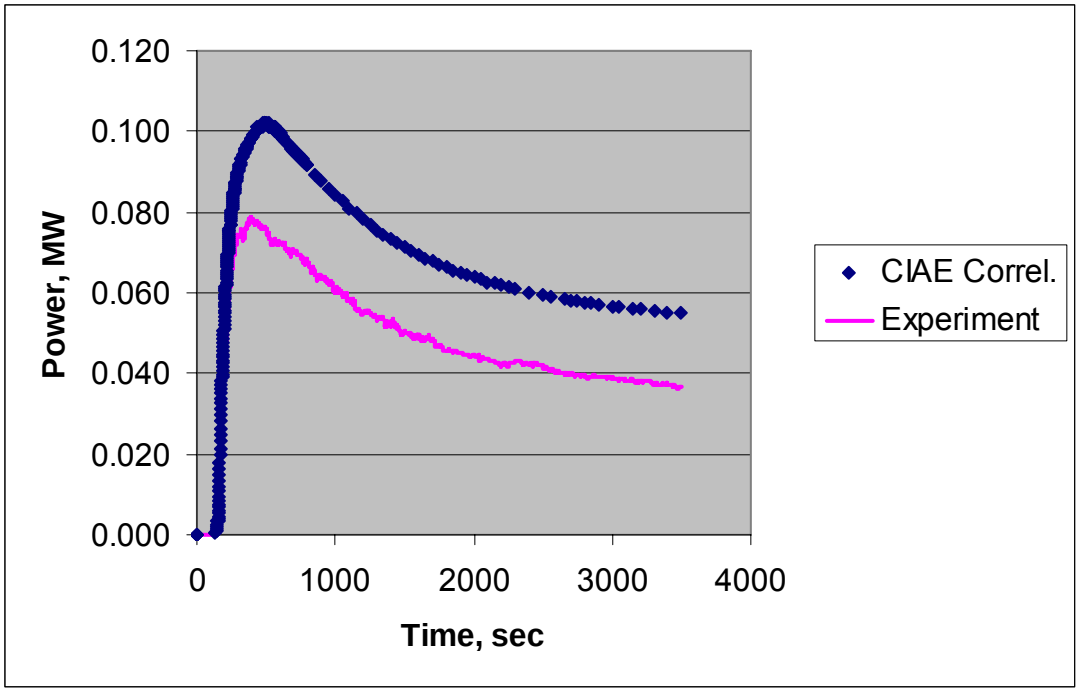

Fig. 7. NIRR-1, LEU, UO_2 12.45% Enriched, 3.77 mk Transient

For the 12% enriched case, we also studied a more severe event caused by a slow reactivity ramp of 6.00 mk. Even this event (Figs. 8-9) was of little safety consequence because the predicted power peak was 160 kW, while the coolant flow became numerically unstable at about 200 kW, from the CIAE model. Onset of nucleate boiling occurred over a time range from about 178-800 seconds. The peak fuel centerline temperature attained was 168 C, while peak clad surface temperature was 135 C. Also, the coolant outlet temperature remained at least 10 C below the saturation temperature. For UO_2 fuel clad in Zr-4, these temperatures are not at all limiting since the UO_2 melting temperature is about 2800 C while that of Zr-4 is about 1850 C. For comparison, the aluminum clad of the original HEU fuel has a melting temperature of about 650 C.

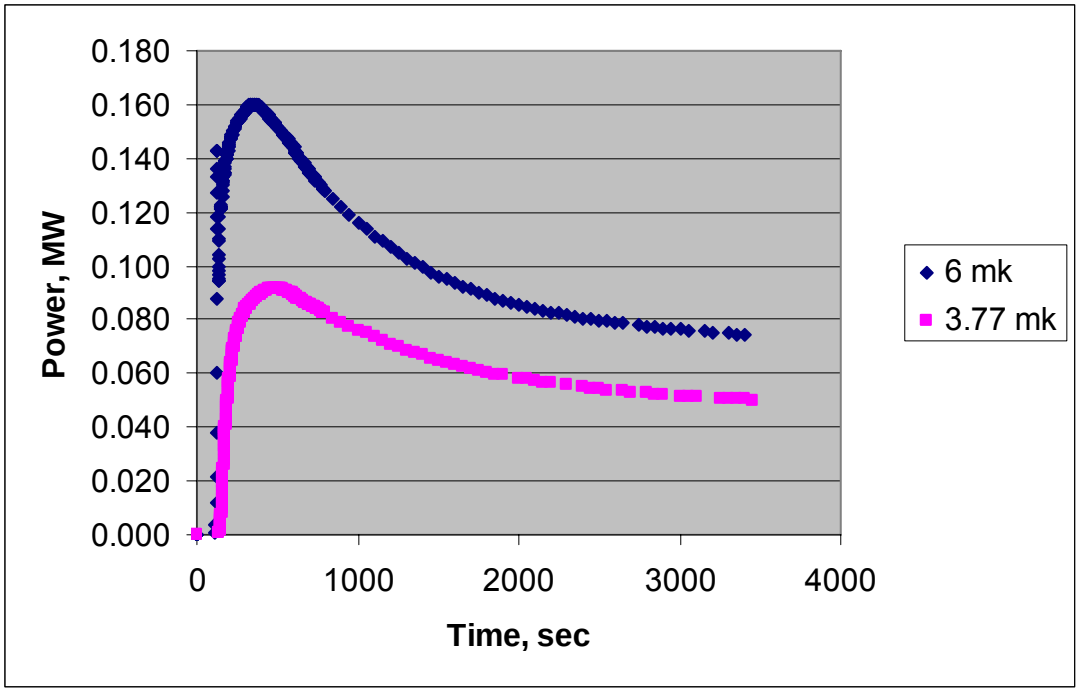

Fig. 8. LEU, UO_2 12.00% Enriched, 6 and 3.77 mk Transients

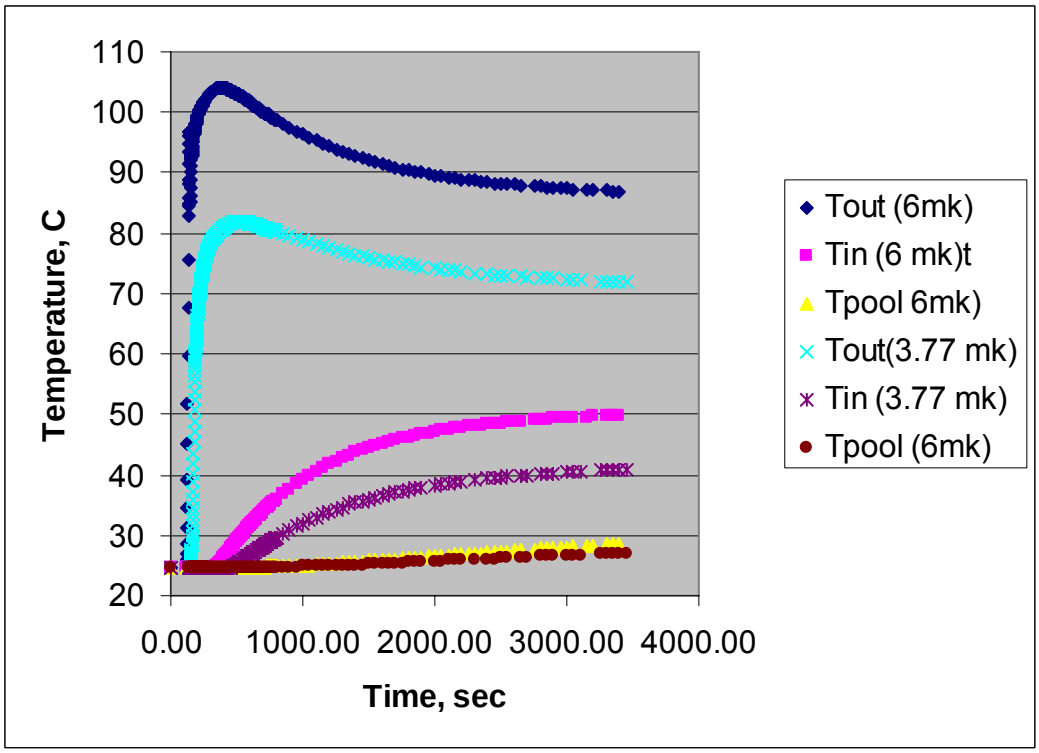

Fig. 9. LEU, UO_2 12.00% Enriched, 6 and 3.77 mk Transients

7. Conclusions

Significant improvements have been made to the PARET 7.3 code to model natural convection cooling conditions in the NIRR-1 MNSR. The code performs well against measured transients in the existing HEU core. All three natural convection heat transfer correlations available in PARET 7.3 do a reasonable analysis of the overall performance on the MNSR in long-time transients. But the CIAE heat transfer correlation appears to provide the best fit to the experimentally measured HEU core peak clad temperature during the long-time steady state operation event, and during the 3.77 mk insertion reactivity transient event. It predicts a significant margin to boiling of about 21 C (T_{sat} =115.6 C at 1.72 bar). We find that the Churchill-Chu correlation predicts peak fuel temperatures very close to the coolant saturation temperature of about 115 C. The original PARET heat transfer model for natural convection, when scaled by an input parameter, does about as well as the Churchill-Chu correlation. The design basis for the MNSR includes the requirement that there be no boiling. The CIAE correlation predictions of peak clad temperature clearly stay well away from boiling. We find that the MNSR fueled by HEU has a very large margin to either fuel or clad melting, and also has a significant margin to onset of nucleate boiling, to numerically unstable flow, and to DNB. The LEU core will have larger safety margins than the original HEU core.

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