THERMAL-HYDRAULIC CALCULATIONS FOR THE CONVERSION TO LEU OF A RESEARCH REACTOR CORE

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ABSTRACT

The thermal-hydraulic analysis performed for the needs of the conversion of the open pool 5MW Greek Research Reactor (GRR-1) to a pure Low Enrichment (LEU) configuration is presented. The methodology was based on a complete set of neutronic calculations performed for the new core configuration, in compliance with pre-defined Operation Limiting Conditions. The hottest channel analysis approach was adopted, and peaking factors were used to account for fabrication or measuring uncertainties. Calculations were carried out using the numerical codes NATCON, PLTEMP and PARET provided by Argonne National Laboratory (ANL). Two main different classes of conditions were considered, namely i) steady state normal operating conditions and ii) transient cases related to accidental events including reactivity feedback effects. For steady state operating conditions the behaviour of the new configuration was examined both for forced and natural convection cooling modes. Transient calculations considered several initiating events including reactivity insertion accidents (slow or fast reactivity insertion) and total or partial loss-of-flow accidents, i.e. in accordance to guidelines provided by the IAEA for research Reactors.

1. Introduction

The GRR-1 conversion to LEU entailed an up-dated safety assessment and thermal-hydraulic analysis. The thermal-hydraulic analysis of the original HEU core has been performed with numerical codes purposely developed in house in mid-eighties. The results of this early assessment have been accommodated in the first Safety Analysis Report made for a pure HEU core [1]. In response to the thermal hydraulic calculational needs stemming from the core conversion, new capabilities have been built, tested and implemented at the Research Reactor Laboratory [2].

The conceptual pure LEU core consists of 21 standard LEU fuel assemblies (18 fuel plates each), 1 special (10 fuel plates) and 5 control assemblies (special + control rod) with a Be reflector on two opposite faces (Figure 1). Although the maximum power is delivered by the fuel assembly at location D3, the hottest plates for the current configuration are those located within the special assembly D4 which delivers 157.4 kW i.e. 15.7 kW in average from each plate. Therefore, for the analysis to follow, the hottest channel was assumed to be located at D4, delivering 15.7 kW to a single water channel of dimensions 2.9x66.6x625.5 mm. When compared to the previous mixed LEU/HEU GRR-1 core configuration, the power of the hottest assembly is increased by a factor of

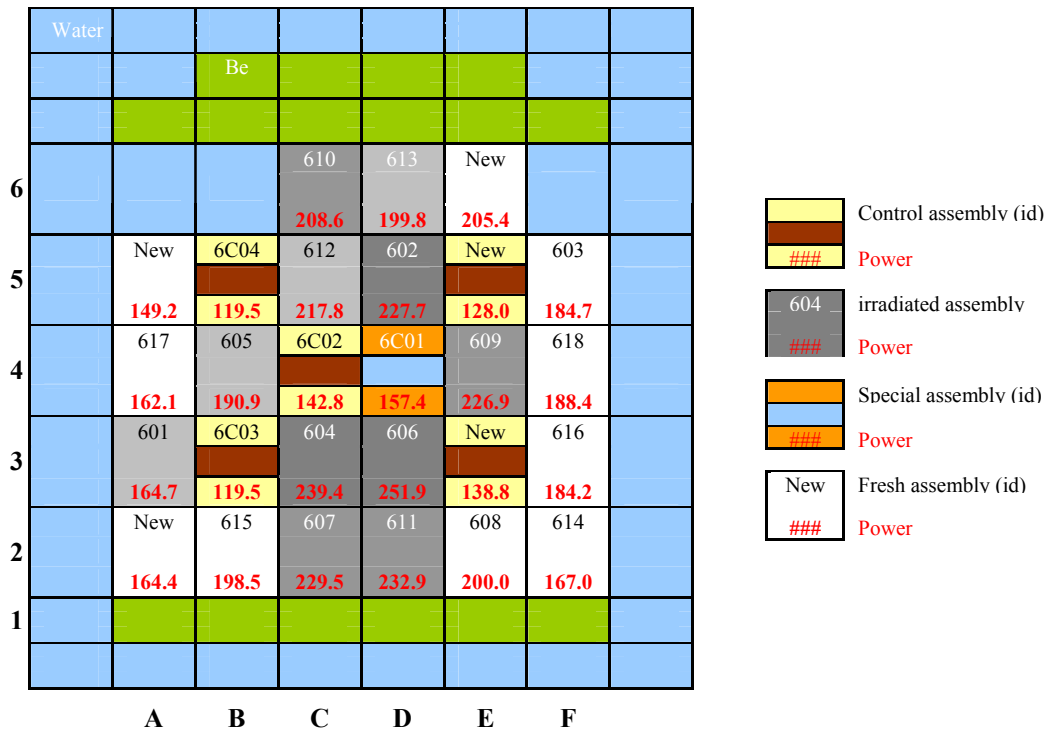

Figure 1. Operational GRR-1 core configuration of conceptual pure LEU core. Average power delivered by each assembly [3]

1.27, since the pure LEU core considered is smaller and the power of 5 MW is delivered by fewer fuel plates.

The objective of the present thermal-hydraulic study is to present the methodology and the results of the various calculations performed for the needs of the core conversion to pure Low Enrichment Fuel. Two different sets of calculations have been performed, using numerical codes provided by Argonne National Laboratory (ANL) based on the hot-channel analysis. First, steady state operational conditions were examined using the numerical codes PLTEMP and NATCON, presented in section 2. The investigation of transient cases was carried-out using the code PARET and it is discussed in section 3.

2. Steady state calculations, calculation of operational limits

Calculations have been performed by considering a single hottest channel and specifying appropriate engineering hot-channel factors to account for fuel fabrication tolerances, uncertainties in calculated parameters and in the ability to measure certain variables such as the Reactor power. The engineering hot-channel factors are applied as three separate components corresponding to a) uncertainties F_{qs} affecting the heat flux, such that $Q_{hc}=Q_{nc}F_{qs}$ b) uncertainties F_{bs} in the flow or enthalpy rise in the channel, such that $m_{hc}=m_{nc}/F_{bs\ c)}$ c) uncertainties F_{hs} in the heat transfer to the coolant, such that $h_{hc}=h_{nc}/F_{hs,}$ where the subscripts *hc* refer to the hot channel and *nc* to the nominal channel values for

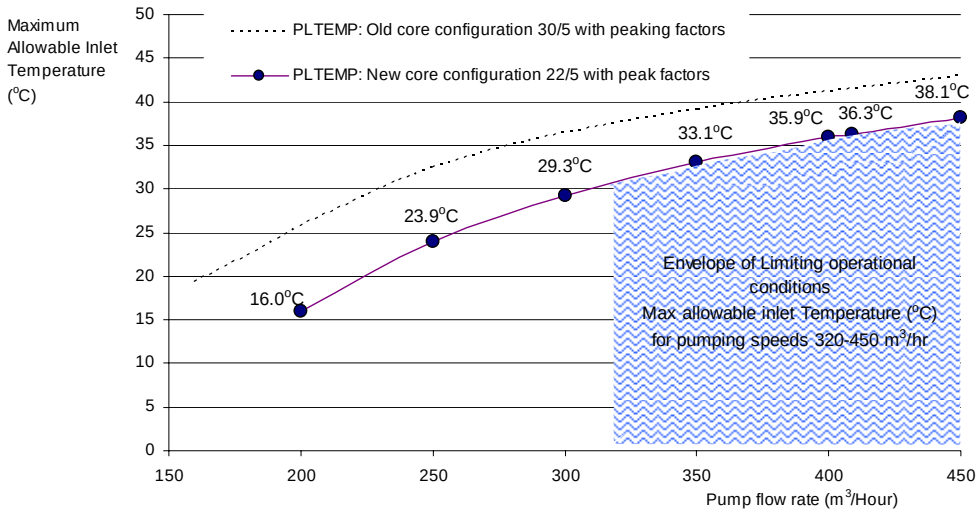

Figure 2. Envelope of limiting operational conditions for forced convection mode. Maximum allowable inlet temperature vs flow rate through the core

the heat flux (Q), mass flow rate (m), and the heat transfer coefficient (h). The engineering hot-channel factors have been determined as F_{qs}=1.21, F_{bs}=1.20 and F_{hs}=1.30 as derived from uncertainties allowed in the LEU fuel assembly specifications and the Safety Analysis Report (SAR). The calculations of the operational and safety limits are derived according to the GRR-1 SAR on the following parameters for each heat removal regime, i.e.:

Forced convection mode cooling

1. Reactor Power < 6.5 MW
2. Reactor Coolant Flow Through the Core > 160 m^3/h
3. Reactor Coolant Outlet Temperature (flow direction downwards) < 56 °C
4. Height of Water in the Pool > 8.5m

Natural convection mode cooling

1. Reactor Thermal Power < 400 kW
2. Reactor Coolant Inlet Temperature (flow direction upwards) < 56 °C
3. Height of Water in the Pool Natural convection mode cooling > 8.5m

2.1 Forced convection mode

Using the code PLTEMP, steady-state forced convection conditions were made, providing the flow velocity, the axial distributions of fuel plate temperature, coolant temperature, as well as the margins to boiling crisis and flow instability. The Reactor is assumed to operate at the power trip level of 6.5 MW with the coolant's outlet temperature at the maximum permissible value specified by the safety limit of 56°C. The operational limit

in the forced convection mode is derived as an operating envelope as a function of the flow rate in the range 200-450 m^3/h as shown in Figure 2. When the analysis does not include hot channel engineering peaking factors, the predicted values are in very close agreement with the original specifications for the mixed LEU/HEU core. With peaking factors included though, a lower operation envelop for the inlet temperature is predicted. The allowable bulk temperature of the water inlet was predicted from 8.6 °C to 4.8 °C lower when compared to the existing mixed core. For the lowest pumping speed of 200 m^3/h examined, the bulk coolant temperature enters at 16.0 ^oC and gains 40.0 ^oC through the core. For that marginal pumping speed, ONB was predicted on the clad surface from 25% to 52% of the active plate height. However, nucleate boiling was not predicted once a higher pumping speed of 250 m^3/h was maintained, since the maximum clad surface temperature of the hottest plate did not exceed 116.5 ^oC.

For all pumping speeds an increase of heat transfer coefficients of the order of 12-30% was noticed from the inlet of the plate to its outlet. For the normal operational conditions of 450 m^3/h, the actual heat transfer coefficient varied from 7120 $W/m^2/^oC$ to 7998 $W/m^2/^oC$. The heat flux to the water channel followed the shape of the delivered power and varied from q''=103.5 kW/m^2 to q''=373 kW/m^2. The maximum heat flux was predicted at 20.9 mm after the bottom of the plate's active length, i.e. at 35% of its length.

2.2 Natural convection mode

For the natural convection mode of cooling, the Reactor core is immersed within a pool of water that is assumed to be at a constant temperature. According to the original specifications and early calculations [1] for natural convection mode operation, the power limit should be such that "cladding temperature shall not reach the boiling point of the water coolant". Here, this limit is taken to mean the Onset of Nucleate Boiling (ONB) point rather than the bulk boiling point of the coolant. Using the engineering hot-channel factors of the previous section, the code NATCON was repeatedly run by gradually increasing the Reactor power, until an indication of ONB was obtained in the hot channel, according to the correlation of Bergles-Rohsenow [4]. Figure 3 shows the clad temperature along the core height, obtained at maximum power conditions. As can be seen, nucleate boiling was effectively predicted in the upper half of the core, close to the core mid-plane. For the new pure LEU configuration, the maximum permissible power for the occurrence of ONB was 659 kW. Although that value is lower than the original power prediction of 814 kW for the previous mixed core, it is far above the maximum operational allowable power safety limit allowed with natural convection of 100 kW [5].

3. Transient calculations

The transient calculations performed considered several types of Reactor accidents like maximum start-up accident (slow reactivity insertion), maximum reactivity insertion accident (fast reactivity insertion) as well as total and partial loss-of-flow accidents. These cases have been selected on the basis of the suggestions on the initiating events, typically recommended for research Reactors [6] and are precisely the cases considered in the IAEA benchmark exercise on Reactor computational methods [7, 8].

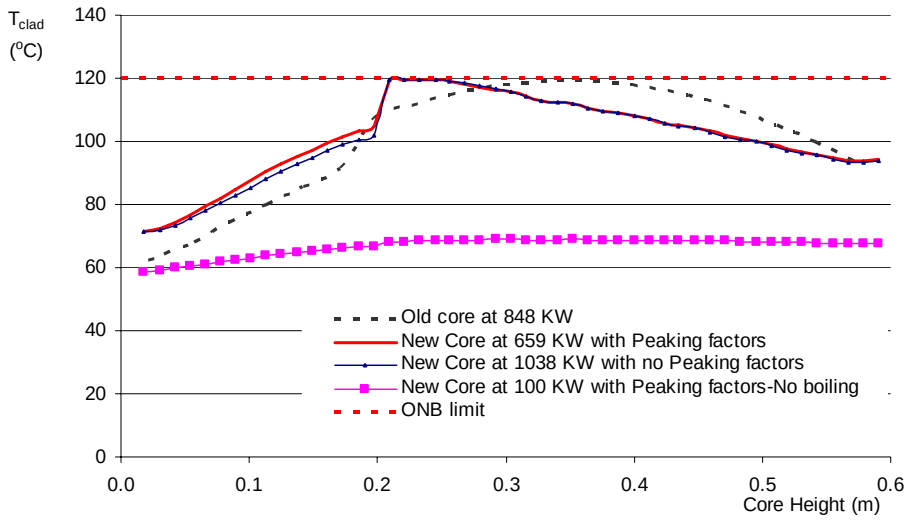

Figure 3. Clad surface temperature along the core and core's power during boiling crisis for natural convection mode cooling

In the simulations, a two channel model was utilized in the PARET code. One channel represented the hottest flow channel, and the second represented the rest of the core, i.e. average core conditions. Each channel contributes to the reactivity feedback through an externally specified weight coefficient which was determined in a simple volume-weight sense. The results presented here correspond to the hottest channel only.

Originally, the PARET code assumed a constant core inlet temperature (or, equivalently, pool water temperature). The in-house version has been extended to accommodate pool water heat-up, and thus provide a capability for analyzing very long transients in which significant pool temperature variations may be encountered. For the analysis to follow, the safety limits of the previous core configuration have been used. For all scenarios, cases with and without scram were considered.

3.1 Basic Reactor Model

The main neutronic parameters and the related reactivity coefficients have been inferred from the Reactor analyses performed with the conceptual GRR-1 core configuration [3]. Since the values of the feedback coefficients vary across the temperature range, their mean values have been used for the transient computations.

3.1.1 Radial and local peaking factors.

The vertical flux profile predicted by the neutronic calculations is assumed had a peak value such that, the ratio of (*Peak local Power in plate / Average Power along the plate*) was F_p=1.457, while the total peaking factor (*Maximum plate power / Average plate power*) was F_e= 16.3/11.04 =1.36.

3.1.2 Kinetic parameters

The basic Reactor's kinetic parameters are summarized in Table I. The obtained values are close to those typically employed in such type of Reactors [9, 10].

Core	Enrichment (%)	Uranium density ($g/cm^{3)}$	keff	β_{eff}	Delayed neutron fraction (μs)	Mean Neutron (λ/keff, μ)
pure HEU	93%	0.58	1.057870	0.006728	71.2	67.31
New LEU	19.75%	2.2	1.000046	0.008520	37.0	37.00

Table I: Reactor's kinetic parameters [3].

3.1.3 Feedback coefficients

Since strong inherent feedback mechanisms (Doppler effects, voids, etc.) are expected for the case of a LEU core [9, 10], the feedback coefficients of Table II have been used for the present analysis.

Reactivity coefficient	Value	
fuel temperature α_{TF}	$-1.74x10^{-3}$ $/^oC$	or $-1.483x10^{-5}/^oC$
water temperature α_{Tw}	$-9.996x10^{-3}$ $/^oC$	or $-8.516x10^{-5}$ $/^oC$
void water α_{vw}	-0.515 $/%	or $-4.388x10^{-3}$ /%

Table II: Feedback coefficients as evaluated from the neutronic calculations [3].

3.1.4 Control rod worth

The total control rod reactivity worth was estimated to be 14.64% of $\delta k/k$ or -17.19$ with respect to the fully withdrawn position.

3.2 Maximum start-up accident

A maximum start-up accident was considered, by assuming that the five control rods are all withdrawn at their constant speed. The period trips are assumed to fail, and the trip is taken on the maximum power limit of 6.5 MW (130% of full power). An initial Reactor power of 1 W was assumed [2], and an excess reactivity of ρ_0=9.07x10^{-3} ($\delta k/k$), or 1.06 $ was considered. In the calculations, the insertion of that reactivity was idealized by two different methods: i) by a step reactivity insertion of 1.06 $, ii) by a gradual increase of reactivity that follows the rod withdrawal with speed 1 mm/s, i.e. with a rate of reactivity insertion of 0.02889 $/s until a total reactivity of 17.19 $ was inserted in the core.

The time evolution of clad temperature is shown in Figure 4a. In both cases with and without scram, the peak clad temperature did not exceed 150°C i.e. well below the cladding melting point. The power excursion could be terminated even without scram, at

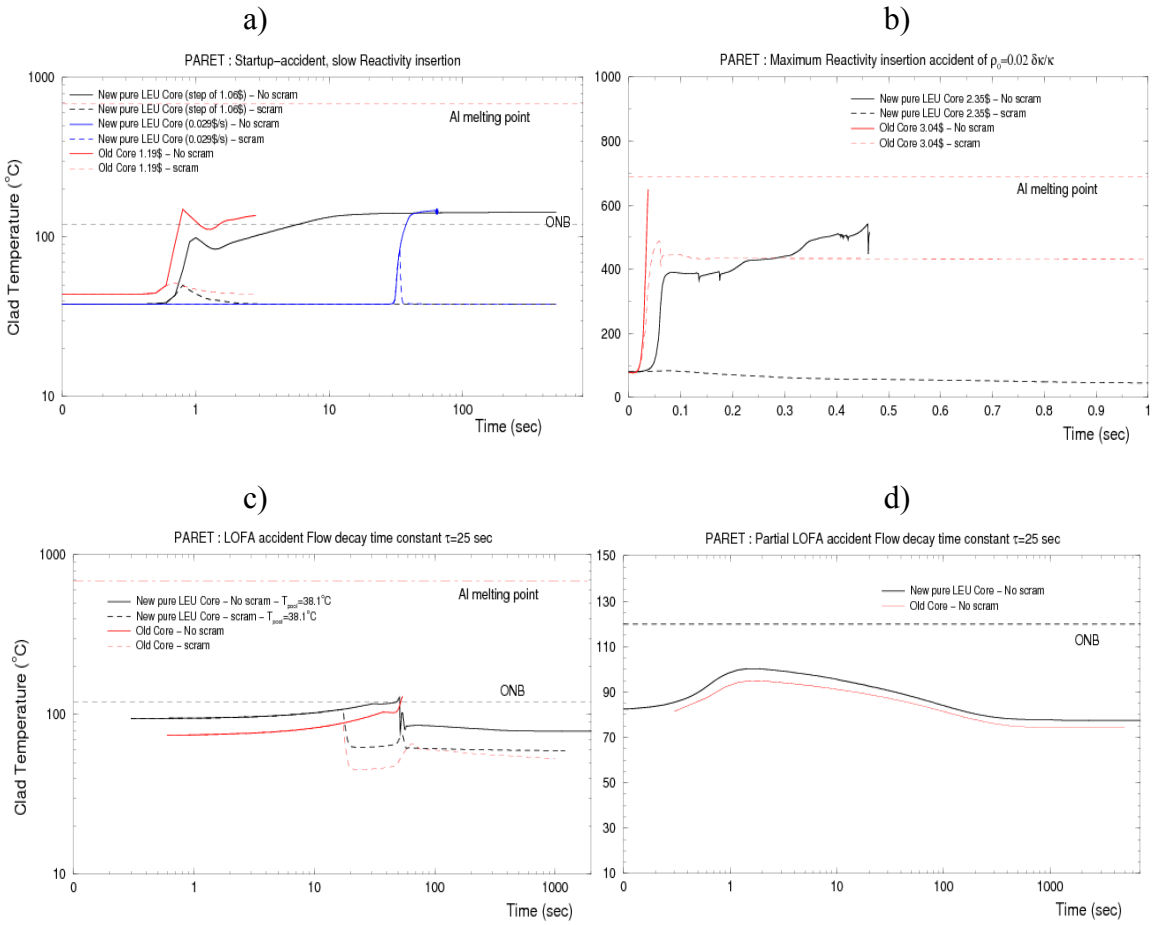

Figure 4. Clad surface temperature evolution in time for various conditions. a) Start-up accident, b) maximum reactivity insertion, c) LOFA, d) partial LOFA

a much higher peak value. This is a demonstration of the strong feedback mechanisms which are expected to be stronger with LEU fuel because of enhanced Doppler effects.

Using a constant rate of reactivity insertion resulted in a delayed increase of power and clad temperatures in the core, which started to increase rapidly after 35 s, while for thestep reactivity insertion case, that event happened at 700 ms. For cases without scram, bulk boiling occurred after 35 s, which was also predicted for the old core case. For cases with scram and gradual reactivity insertion, the maximum predicted clad and peak water temperature were 79.8 °C and 56.2 °C respectively.

3.3 Maximum reactivity insertion

The assumed event consists of a step reactivity insertion of ρ_0=0.02 ($\delta k/k$) while the Reactor operates normally at 5 MW. This amount of reactivity corresponds to the event of

a sudden addition of an extra fuel assembly in the matrix, assumed to have dropped inadvertently during a crane manipulation. The analysis of such a reactivity accident is instructive because it envelops all other conceivable reactivity accidents, such as start-up accidents, loading accidents, influences from experimental facilities etc. For the current configuration, the postulated accident corresponds to a large reactivity insertion of 2.35 $.

In the simulations, the Reactor power increased rapidly, and the trip level of 6.5 MW was attained at 6.7 ms. Power continued to rise well above the trip level, reaching a peak of 9.05 MW at 65 ms. With the Reactor scram "on", that power peak was much lower than the value of 1084 MW obtained with the old core. With the scram disabled, the obtained values showed a sharp peak 1067 MW at 58 ms, before the feedback mechanisms started to act leading to an unstable behaviour of the core.

The evolution of clad temperature is plotted in Figure 4b. With the scram enabled, the maximum temperature is obtained at 75 ms and corresponds to clad temperatures of 83.5°C for the LEU case. If the Reactor protection system fails, the clad temperature approaches the clad melting point at 0.5 s. Due to the Reactor's instability it is thus reasonable to assume an accident initiation. Thus, in this case, core damage occurs.

3.3 Maximum loss of flow accident

It is assumed that the Reactor cooling system is operating at its nominal flow rate of 450 m^3/h when a 100% loss-of-flow occurs. The flow rate decay was approximated by an exponential function as, $G = G_0 \exp(-t/\tau)$ where G_0 is the initial flow rate and τ a time constant describing the rapidity of the decay. For the present analysis, the time constant τ was assumed to be 25 s corresponding to a slow transient (e.g. pump failure with large flow inertia). At $t=2\tau$, i.e. when flow-rate G reduces to ≈15% of the initial value G_0, the flapper valve sealing the lower plenum is assumed to open, hence, downward forced flow stops and natural circulation develops through flow reversal. This is the most adverse part of the transient because the hot coolant of the lower plenum will first undertake natural cooling, until cool water from the pool expels progressively the hot lower plenum volume. In flow reversal conditions, low-pressure systems like open-pool type research Reactors are susceptible to a form of instability known as flow excursion or Ledinegg instability [11, 12]. The threshold conditions for flow instability were determined using the correlation of Whittle and Forgan [13] for the Onset of Flow Instability (OFI) which was found to be appropriate for Research Reactor applications [11, 7, 14].

As shown in Figure 4c, in all cases with scram the peak clad temperatures computed during the transient was 108.2°C, thus, far below the melting temperature of the cladding. With reduced initial power and initial pool temperature, no significant bulk boiling takes place during flow reversal, and stable natural cooling conditions are ultimately established. Instead, starting with elevated initial power or pool temperature, vigorous boiling is predicted, inducing flow instabilities. For the case without scram, clad temperature exceeds the ONB point, vigorous boiling initiates and the system is ultimately found under unstable conditions (OFI point encountered).

3.4 Partial loss of flow accident

The case is concerned with the loss of one of the two primary pumps and simultaneous loss of secondary cooling. Such a sequence may happen, for instance, in a loss of off-site power event (the emergency unit provides power only for one primary pump). An initial power level of 5 MW and a maximum initial pool temperature of 38.1°C were assumed. All safety trips were assumed to fail; hence, the Reactor was not let to scram.

The clad temperature and peak water temperatures are shown in Figure 4d. The 50% flow reduction gives rise to an initial overshoot of both clad and coolant temperatures prior to the establishment of the reduced-flow cooling conditions. The peak clad and water temperatures obtained during the transient were 100.0°C and 66.5°C respectively, thus far below any safety concern. The temperature spikes induce quite strong feedback mechanisms, and as a result the power drops sharply at the beginning of the transient. Since secondary cooling was not available, pool temperature increases gently, introducing negative reactivity, and causing a gradual power decrease at later times. After 7200 s (2 h) safe temperature and power levels (2.6 MW) are prevailing in the core.

4. Conclusions

The numerical codes PLTEMP, NATCON and PARET have been used to assess the safety issues of the GRR-1 core conversion from a mixed core to a pure LEU fuel core. As it was shown, these numerical tools fulfil satisfactorily the operational needs of GRR-1, as required for performing thermal-hydraulic safety assessments in house on a routine basis. For steady state operational conditions, the increased power density in the new LEU core, lead to lower operational limits for the allowable bulk temperature of the water inlet. For forced convection mode cooling, the allowable inlet temperature was predicted from 8.6 °C to 4.8 °C, lower when compared to the previous mixed HEU/LEU operational core. For natural convection mode, the maximum permissible power for the occurrence of ONB was 659 kW. In both cases though, regardless of the increased heat flux of the new configuration, the future operation of the pure LEU core complies with the current safety limits and procedures of the GRR-1 Reactor.

The same conclusion was drawn from the analysis of the transient cases which did not have any safety implication when compared to the previous core configuration. For the transient cases considered, safety is promoted with the use of pure LEU fuel. This is mainly attributed to the Doppler effect, which is significant in LEU fuel because of the abundance of the contained U-238. In all the above transient simulations, strong feedback responses were predicted, contributing significantly to inherent Reactor power drop.

5. References

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