THERMAL-HYDRAULIC STUDY OF TRR CORE IN FUEL CONVERSION FROM HIGHLY ENRICHED URANIUM TO LOW ENRICHED URANIUM FUEL

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Abstract The Atomic Energy Organization of Iran (AEOI) planned to refuel the HEU fueled core by LEU fuel in 1992*. In the fuel conversion, neutronics analysis and kinetic study for HEU and LEU cores have been carried out in order to be able to adjust LEU core parameters with those of HEU. In the thermal-hydraulic study of the LEU core, the diversity of combinations of number of plates/SFE (standard fuel element), fuel-loading/SFE and water channel gap were considered. In the calculational study, temperature distribution of clad surface along the hottest channel was determined and T(ONB), T(Sat), CHF (critical heat flux), and DNBR (departure from nucleate boiling) were calculated. No major technical problem is encountered in the fuel conversion. MDNBR was calculated in the hottest channel to be about 3.55 in normal operation with fuel loading varying between 255-300 gU-235/SFE

Key Words Thermal-hydraulic, F uel-conversion, HEU/LEU Fuel, MDNBR, COBRA-3C, U_3O_8-Al Fuel, Flux Ratio, T(ONB), Rectangular Channel, Downward Forced Convection, Subchannel.

چکیده سازمان انرژی اتمی ایران برای تعویض سوخت راکتور پژوهشی تهران از اورانیوم بسیار غنی (٪۹۳) به اورانیوم باغنای کم (٪۲۰) لازم دید تا قبل از اقدام به تعویض سوخت، محاسبات پایاهای قلب را انجام دهد. در تعویض سوخت، چون تغییرات فیزیکی درخواص نوترونی و حرارتی و نیز تغییراتی در ابعاد قلب و مجاری خنک کننده به وجود می آید، لازم است که محاسبات ترمو هیدرولیکی انجام گیرد، تا مشخص شود که تغییرات در پایاهای اصلی قلب درتحت چه شرایطی در حاشیه ایمنی قرار می گیرد. برای انجام چنین محاسباتی لازم است که با حفظ یکپارچگی میله های سوخت استاندارد، بعضی از مشخصات میله سوخت مثل تعداد صفحات سوخت، چگالی سوخت، ضخامت صفحات سوخت و پهنای مجرای خنک کننده را تغییر داد تا پایاهای نوترونی و حرارتی قلب جدید نزدیک به پایاهای قلب قدیم باشد. با توجه به این موضوع، محاسبات حرارتی قلب برای آرایش های مختلفی از میله سوخت استاندارد انجام گرفت و کمیت های زیر محاسبه شد: توزیع دما در طول میله سوخت، بیشترین دمای خنک کننده، دمای غلیان جنینی (T_{ONB}) دمای اشباع (T_{sat}) شاربحرانی (CHF) و MDNBR. نتایج محاسبات نشان داد که میتوان با ترکیب مناسب بین تعداد صفحات سوخت، پهنای مجرای خنک کننده و چگالی سوخت، بدون اشکالات حاد فنی قلب جدیدی را طراحی کرد که با همان قدرت ۵ مگاوات و شرایط هیدرولیکی و دستگاه کنترل موجود کار کند. دراین محاسبات از کد کامپیوتری COBRA-3C باتغییرات مناسبی برای مطابقت با شرایط قلب راکتور پژوهشی تهران استفاده شد.

INTRODUCTION

In view of unavailability of HEU (Highly Enriched Uranium fuel) for research and test reactors due to non-proliferation policy, substantial research has been conducted over the last decade to develop high den-

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sity LEU (Low-Enriched Uranium fuel) to replace HEU in research and test reactors with the minimal impact on reactor operation. While neutronic calculations are essential in fuel conversion, still thermal-hydraulic core design plays an important role in the safety aspect of the reactor operation. The purpose of this investigation is to describe the thermal-hydraulic behaviour of the LEU fueled core in comparison with that of HEU fueled core. The heat transfer properties of the HEU-core were calculated in a variety of combinations of number of plates/SFE, water gap thickness, fuel thickness and fuel loading assuming U_3O_8-Al fuel meat. The calculations were performed based on the procedures and data outlined in the IAEA-Guide-Book[1].

MAJOR FEATURES OF TRR CORE

Tehran Research Reactor (TRR) is a pool type core moderated and cooled by ordinary water. The core configuration of the HEU/LEU is schematically shown in Figure 1 with the exception of no graphite reflector for HEU core. Each HEU standard fuel element(SFE) consists of 18 curved plates of which two end plates are dummy. The HEU control fuel element(CFE) has only 9 curved plates with a water gap in the middle to allow axial movement of oval control rod. In these calculations, the LEU control rod was assumed to be of oval type. LEU standard fuel element comprises 19 flat plates. In the thermal-hydraulic calculations, two size cores of 28 and 24 fuel elements were considered. The HEU core is surrounded from all sides by water except on thermal column side. In LEU core, two rows of graphite were assumed on the opposite sides of the core to enhance the core reactivity. The core is cooled by downward forced convection with flow rate of 500 m^3/h.

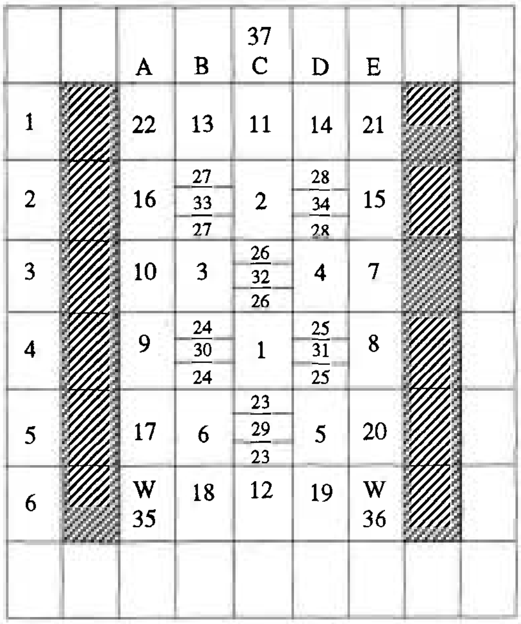

Figure 1. Core configuration.

CORE DESIGN CRITERIA AND CALCULATIONAL PROCEDURE

Objectives

The main objective in the LEU fueled core design was to reach an equilibrium core with thermal flux (ϕth) not less than that of the HEU core to meet experimental beam port requirements. In the fuel conversion from HEU to LEU fuel flux ratios [ϕ(20%)/ϕ(93%)] decreases as fuel loading/SFE increases [2]. The decrease in thermal flux ratio is much greater for resonance and fast fluxes. In order to increase the thermal flux in LEU core at one of the beam tubes, a partial fuel element and/or a dummy element was considered near the beam entry. However to achieve the ϕ_{th}/ϕ_f ratio high enough for beam applications, a criterion was set that the MDNBR value in the hottest channel of LEU core should not exceed the DNB ratio of HEU core (MDNBR= 2.69) which is consistent with the present safety margin. Moreover, the inherent LEU core reactivity has been adjusted with the reactor operational procedure (~12%Δk/k).

To reach the above objectives a variety of combinations of number of plates/SFE, fuel loading per SFE, water channel gap, and a number of fuel elements in the core were considered. In each case thermal-hydraulic properties of the core were investigated. In fact there are two competetive parameters, flux ratio ϕth(20%)/ϕth(93%)-and ONB temperature of clad surface that have to be compromised. The latter is mainly due to axial flux inflation in the lower part of the core because of water trap below the control rods and flux depression in upper part of the core, shown in Figure 2. The interesting point is that the thermal flux ratio in uncontrolled (lower) part of the core has opposite trend as compared to the controlled part, shown in Figure 3.

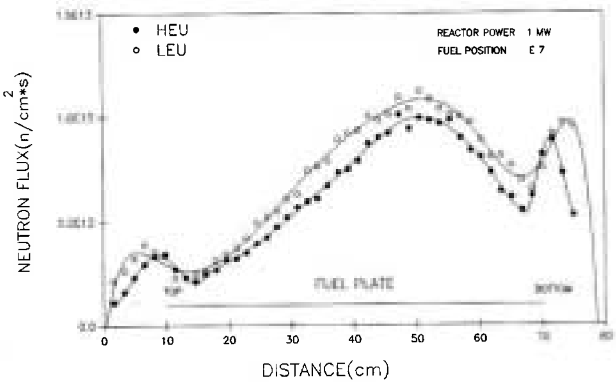

Figure 2. Measured thermal flux in SFE(1).

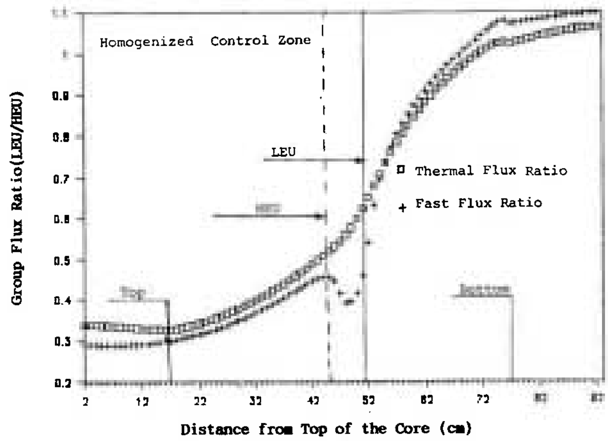

Figure 3. Calculated axial flux ratio in TRR core.

Heat-Treatment Concept in the code

In the present heat transfer analysis describing fluid and energy transport, basic equations are derived using general equations of continuity of mass, conservation of energy and momentum. This approach has been adopted in the code COBRA-3C [3] and in the calculations in a more general form of time dependent for slanted subchannels. COBRA-3C assumes several neighbouring subchannels with identical or different geometry and different fluid flow velocity. Accordingly, the standard fuel element was divided into several computational subchannels. The use of the computational subchannels in the computations allows the analysis of individual subchannels. From the solution of the relevent equations by finite difference method enthalpy, massflow, cross-flow, and pressure long each subchannel were obtained.

Thermal Channel Parameters

The temperature distributions along the hottest channel, in which the thermal neutron flux peaking occurs for fuel, clad surface and coolant were calculated by Fourier's equation. In the case of HEU fuel, the measured flux distribution Figure 3 was used. The thermal neutron flux in the hottest channel was measured in fuel element in position-1. For the LEU fuel, thermal parameters of the hot-test subchannels were calculated using the calculated neutron flux distribution shown in Figure 4. The temperature distributions in the hottest channel for HEU fueled core are shown in Figure 5. The temperature of onset of nucleate boiling T(ONB), saturation temperature T(Sat), heat flux (HF), q", critical heat flux (CHF), q", and DNBR were calculated, shown in Figures 6 and 7. The cladding surface temperature at ONB was calculated from Bergeles-Rohsenow's correlation [4] given

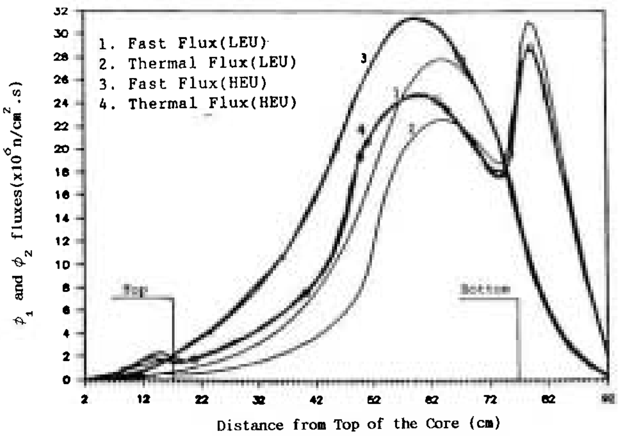

Figure 4. Calculated Two group fluxes in LEU fuel element

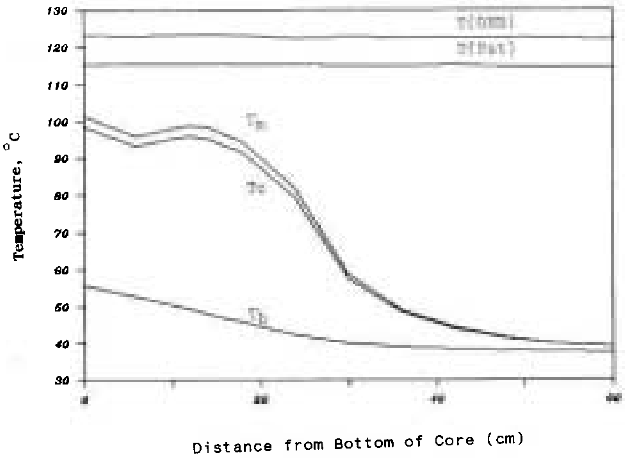

Figure 6. Temperature distributions along the hottest channel of LEU fuel element (using calculated thermal flux).

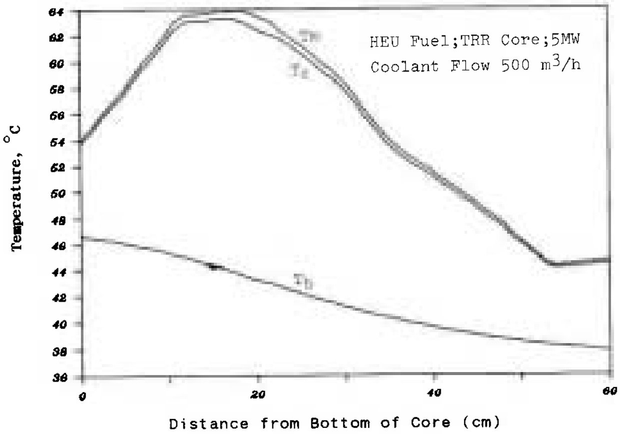

Figure 5. Tm, Tc and Tb along the hottest channel of HEU fuel.

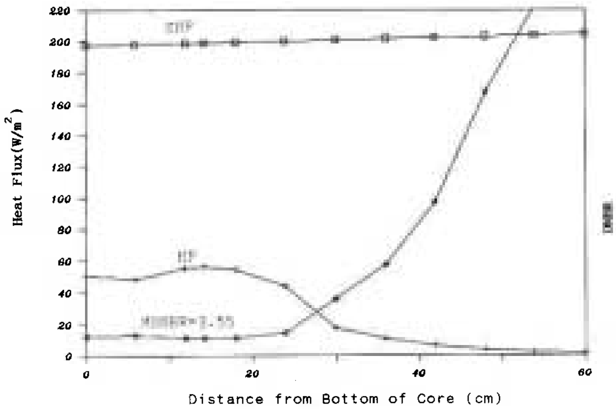

Figure 7. Heat fulx, CHF (Critical Heat Flux), and DNBR (depar-departure from nucleate boiling) along the LEU fuel element using calculated thermal flux.

below:

$$T(ONB) = T(Sat)\left[\frac{q''\wedge(p\wedge 0.0234/2.3)}{(15.6p\wedge 1.156)}\right] \quad (1)$$

The equation of Newton's law of cooling at T(ONB) with saturation temperature of the fluid can be written as:

$$q''(ONB)= h(T(ONB)-T(Sat)) \quad (2)$$

The solution of two simultaneous aforementioned equations yields T(ONB) and q"(ONB) shape. In the case of LEU fuel element at the same position a calculated flux distribution was used. The group flux distributions along the coolant channel were obtained by solving the two energy group diffusion equation using COSTANZA code [5], see Figure 4. In COSTANZA the core is divided axially into several zones in which the control absorbers, control rods, and fission products were treated as uniformly distributed in respective zones. The CHF was calculated from the correlation developed by Sudo [6] given below:

$$q''=0.005\, h_{f\sigma}\, G^{0.611}\left[\left(\frac{\sigma}{\gamma_l}-\right)^{0.5} g\,(\gamma_l-\gamma_\sigma)\,\gamma_\sigma\right]^{0.1945} \quad (3)$$

Figure 7 shows the calculated HF, CHF, and DNBR for LEU fuel element. In the numerical treatment of the heat transfer equations, the SFE was taken as an assembly with half the fuel element being divided into ten identical subchannels except the first one. In pressure drop calculation friction coefficient, f, was obtained from the correlation for smooth surface [7]:

$$f= 0.185\ Re^{-2} \quad Re>2 \times 10^{4} \tag{4}$$

The surface temperature was determined by Newton's law of cooling in which the heat transfer coefficient was also obtained from McAdam's correlation [8]:

$$h= k/D\ (0.023) Re^{m}\ Pr^{n} \tag{5}$$

where m= 0.8 and n= 0.4 for heating and 0.3 for cooling. The values of Re and Pr numbers and L/D in these calculations are in the range of applicability of the correlation [7] i.e.,

$$Re>10^{4} \qquad 0.7>Pr>160 \qquad L/D>10 \tag{6}$$

Time-dependent Thermal Behaviour

With respect to the safety aspect of the reactor operation with LEU fuel elements, the MDNBR which is a major factor in the safety analysis was computed. In normal operation, the MDNBR in the hottest LEU fuel element was calculated to be 3.55 which is in the safety margin as compared to the MDNBR of HEU fuel-core. To estimate the effect of low-energy removal from the hottest channel on the MDNBR value in the case of coolant flow irregularity, the flow rate was decreased to a minimum flow of 20 m^3/h.

Two cases of coolant drainage system malfunction resulted in slow and fast decreases in flow rate were considered. In the case of slow decrease in coolant flow-rate from normal flow rate of 500 m^3/h to flow rate of 0.75, 0.5, 0.25, 0.1 and 0.04 times of normal flow rate for HEU fueled core, clad surface temperature was calculated, see Figure 8. In the case of fast decrease in coolant flow rate clad surface temperature distribution was calculated see Figure 9. In the study of slow drop in coolant flow rate from 100% to 4% of normal flow rate, it was assumed that it occured in 5s while in fast drop it occured in 0.5s. To investigate the effect of positive reactivity insertion into the core resulting in a power surge, on temperature distributions, reactor power was increased from 5 to 25 MWt in 1.5s and back to 10 MWt in 0.5s. The calculated temperature distributions of coolant and fuel meat along the subchannel are shown in Figures 10 and 11. This power rise assimilates the fast withdrawal of the control rods. The highest clad surface temperature for HEU fuel element, were calculated in normal condition. The results of calculations are shown in Figure 12. This illustrates that in the cases studied, LEU core thermal behaviour is more reliable than that of HEU core.

CONCLUSIONS

The computed temperature distribution of fuel meat, clad surface, and average bulk temperature along the

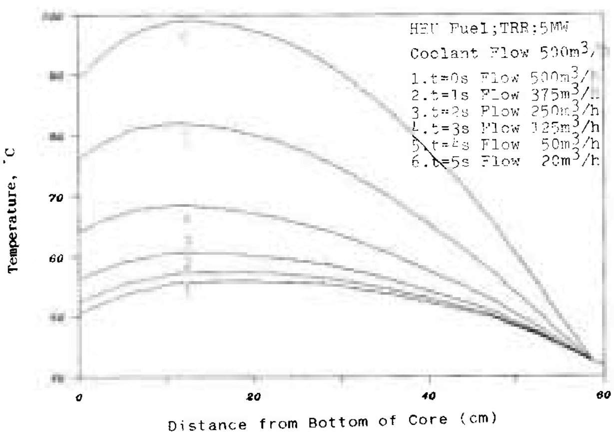

Figure 8. Clad surface temperature in slow drop in flow rate in HEU fueled core.

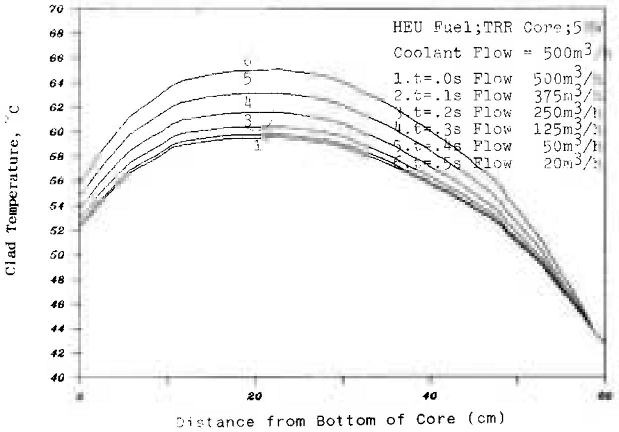

Figure 9. Clad surface temperature in fast drop in flow rate in HEU fueled croe.

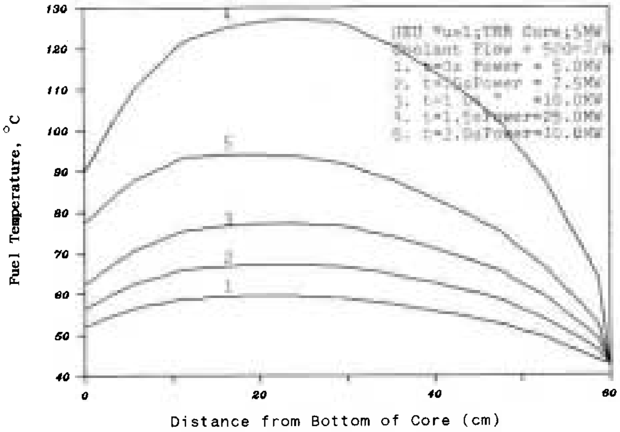

Figure 11. Fuel temperature in power surge in HEU fueled core.

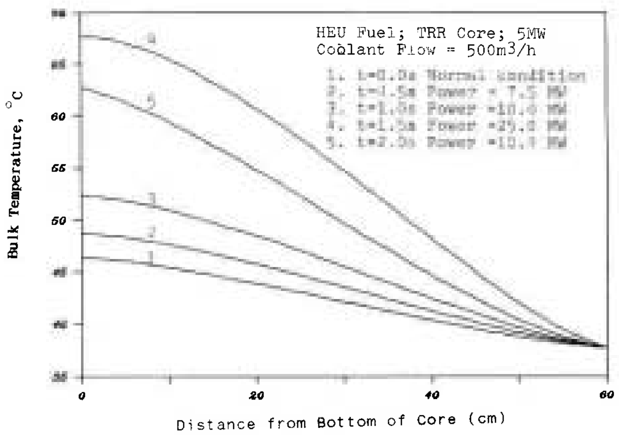

Figure 10. Bulk Temperature in water channel in power surge in HEU fueled core.

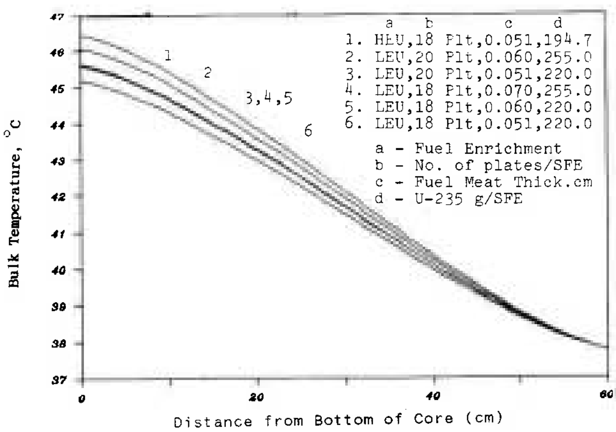

Figure 12. Fuel temperature for HEU and different LEU fuel elements.

central subchannel for HEU fuels in normal and transients was performed using a measured thermal flux distribution. The calculated clad temperature in normal reactor operation is 25°C lower than T(ONB) for HEU fuel and 16°C lower for LEU fuel. The values for HF, CHF, and DNBR along the indicated subchannels were calculated. In the case of tube rupture or water drainage malfunction leading to lower coolant flow rate of one-fourth of normal flow rate through the core, as the calculations show clad surface temperature taking into account the power peaking of 1.70 barely reaches the T(ONB). Further decrease in coolant flow rate by virtue of local power peaking and lower energy removal, clad surface temperature exceeds the T(ONB).

ABBREVIATIONS

AEOI	Atomic Energy Organization of Iran
CFE	Control Fuel Element
CHF	Critical Heat Flux
DNBR	Departure from Nucleate Boiling Ratio
HEU	Highly Enriched Uranium
HF	Heat Flux
LEU	Low Enriched Uranium
MDNBR	Minimum DNBR

ONB	Onset of Nucleate Boiling
SFE	Standard Fuel Element
Sat	Saturation
TRR	Tehran Research Reactor
G	Coolant Flow Rate (kg/s)
h	Heat Transfer Coefficient (W/m^2. K)
k	Thermal Conductivity (W/m. K)
q"	Heat Flux (W/m^2)
P	Pressure (N/m^2)
γ	Specific Weight (kg/m^3)
σ	Surface Tension (kg/m)
l	liquid state
g	gas state

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