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SCK•CEN
Boeretang 200
BE-2400 Mol
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SCK•CEN
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BE-2400 Mol
Belgium

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Studiecentrum voor Kernenergie
Centre d’étude de l’énergie Nucléaire
Boeretang 200
BE-2400 Mol
Belgium

Phone +32 14 33 21 11
Fax +32 14 31 50 21

http://www.sckcen.be

Contact:
Knowledge Centre
library@sckcen.be

ABSTRACT

The objective of this work was to: (a) perform a steady state thermal hydraulic analysis of the BR2 reactor with the ANL code PLTEMP and compare its predictions with those of the code of S. Fabrega used in the BR2 safety analysis report, and (b) perform with PLTEMP the same analysis for a BR2 core with LEU fuel and compare its performance with that of the current HEU core. Analyses were performed at nominal operating conditions without and with hot channel factors. Safety margins to onset-of-nucleate-boiling, fully developed nucleate boiling and to flow instability were used as measures of performance. In the LEU (UMo) core, the geometrical parameters of the standard BR2 fuel element (assembly) were preserved. For the analyses with PLTEMP, power distributions were computed with the MCNPX code. The predictions of PLTEMP are in good agreement with those of the Fabrega code, and the limiting power and velocity in the LEU core are about 8% lower than those in the HEU core.

Key words: T/H safety margins, PLTEMP, HEU, LEU

1. Introduction

To support the conversion of a research reactor from HEU to LEU and establish the feasibility of a proposed LEU fuel element, a comparison of the steady-state thermal-hydraulic safety margins of the LEU and HEU cores is performed. It is important that a consistent comparison is made to ensure that only the changes in fuel design are reflected in the change of safety margins. It is therefore useful to perform the comparison using the same code and methodology.

The thermal-hydraulic steady-state safety margins of the current BR2 HEU core were determined with a code developed by S. Fabrega [1] while the ANL code PLTEMP [2] will be used for the steady-state thermal-hydraulics analyses related to the conversion from HEU to LEU. Because of this difference, it is important to demonstrate that the basis of the original analyses is well understood. One way to achieve this goal is to use PLTEMP to replicate some key results obtained with the Fabrega code using a similar approach and similar assumptions.

The main focus of this work is to use PLTEMP/ANL to: (1) recalculate the maximum heat flux allowed before Onset-of-Nucleate-Boiling (ONB), Full Nucleate Boiling (FNB) and Flow Instability (FI) occur using similar assumptions as in the analyses performed with the S. Fabrega code; and (2) perform a steady state thermal hydraulics analysis for the BR2 LEU core, and compare the performance of the LEU core with that of the HEU core.

More specifically, the purpose of the work presented in this report is: (1) to review the steady state thermal hydraulics analysis of the BR2 reactor presented in the safety analysis report, (2) repeat this analysis (HEU core) with the PLTEMP/ANL code at the same operating conditions as it was done in the 70's, (3) compare the PLTEMP predictions with those of the code of S. Fabrega, (4) perform a steady state thermal hydraulics analysis with the PLTEMP/ANL code for the BR2 LEU core, and (5) compare the performance of the LEU core with that of the HEU core.

A PLTEMP model of the standard six-tube BR2 fuel assembly was developed at Argonne National Laboratory. Hot channel factors and operating conditions (power, inlet mass flow, inlet pressure and temperature) with uncertainties were reviewed and introduced in the PLTEMP input deck. Then safety margins to ONB, FNB and to FI were calculated for the BR2 HEU core under forced flow conditions.

A preliminary feasibility study with U-Mo dispersion fuel has shown that the current fuel element design does not need to be changed in the conversion of the BR2 reactor from HEU to LEU fuel. Thus, in the analyses performed for the LEU fuel, the geometrical parameters of the standard BR2 fuel element are preserved. For the PLTEMP analyses, of both HEU and LEU cores, power peaking factors were computed with the MCNP code.

2. Review of the thermal-hydraulics analyses presented in the BR2 safety analysis report

This section presents a brief review of the steady state thermal hydraulics analysis of the BR2 reactor presented in the safety analysis report [3-5]. This analysis was performed with the code of S. Fabrega.

2.1 Calculation of the maximum heat flux at nominal operating conditions

This section describes the methodology used to obtain the maximum nominal heat flux to be used in the code developed by S. Fabrega to calculate the thermal-hydraulic safety margins.

To obtain the nominal power (P_i) in a given fuel element, it was assumed that the power sharing between elements could be approximated by the element fraction of the total core U-235 loading weighted by the relative thermal neutron flux (N_i) in the axis of the element. Therefore, for elements of the same type, P_i can be written as

$$P_i = P_{total} \times \frac{N_i M_i}{\sum_i N_i M_i}, \qquad (1)$$

where M_i is the U-235 mass in the fuel element i.

To obtain the heat flux associated with P_i, the methodology assumed that a fraction K of the total power will be deposited directly in the coolant and that heat diffusion in the axial and azimuthal directions within a plate will reduce the heat flux at the clad/water interface by a factor K_{th}. Therefore, the average heat flux (q_i) resulting from having a power P_i generated in the fuel element can be written as

$$q_i = \frac{P_i K_{th}}{SK}, \qquad (2)$$

where S is the total heat transfer area.

To obtain the fuel element highest heat flux, the following local peaking factors associated with each fuel element are applied: the axial peaking R_i and a combined radial and azimuthal peaking factor f_i. The maximum nominal heat flux in the BR2 core can therefore be obtained by

$$q_{\max} = \max\left\{\frac{P_i K_{th}}{SK} R_i f_i\right\}. \qquad (3)$$

The following values for the various parameters were considered: K=1.055 (i.e., 5.5% of element power as direct coolant heating), a K_{th} between 0.945 and 1.0, an axial peaking between 1.38 and 1.52, and a combined radial and azimuthal peaking between 1.3 and 1.48.

2.2 Hot Channel Factors (HCF) used in the S. Fabrega code

To estimate the hot channel factors (HCF) used to evaluate the thermal-hydraulics steady-state safety margins, the S. Fabrega code considered the following types of errors and uncertainties:

1- Error (ε_1) in the fuel element power P_i
2- Error (ε_2) in the measured axial peaking R_i
3- Error (ε_3) in the measured combined radial and azimuthal peaking f_i
4- Uncertainty (ε_4) in the total heat transfer area
5- Error (ε_5) in the estimated direct coolant heating fraction K
6- Uncertainty (ε_6) in heat flux due to non-uniform U-235 plate loading
7- Error (ε_7) in the calculated local burnup

For all these errors and uncertainties, values that reflect twice the standard deviation were assumed so that after the application of the HCF there is at most a 5% chance of exceeding the maximum heat flux.

For the calculations of the ONB and FNB margins at the hot spot, the errors/uncertainties are assumed independent and normally distributed. However, for the calculation of FI, it was assumed that all the errors/uncertainties in the hot stripe are going to be on their conservative side and therefore, the total error/uncertainty is simply the sum of all the considered errors/uncertainties. Eqs (4) and (5) give how the errors/uncertainties were combined for the HCFs on the hotspot ($\varepsilon^{hotspot}$) and the hotstripe ($\varepsilon^{hotstripe}$).

$$\varepsilon^{hotspot} = \sqrt{\sum_{i=1}^{7} \varepsilon_i^2} \ . \tag{4}$$

$$\varepsilon^{hotstripe} = \sum_{i=1}^{5} \varepsilon_i + \varepsilon_7 \ . \tag{5}$$

For the error of the fuel element power (ε_1), the methodology combined statistically the errors/uncertainties on the evaluation of relative thermal heat flux (N_i), the measured total power (P_{total}) and the uncertainty on the fresh fuel element U-235 loading (M_i). Consequently, ε_1 can be written as

$$\varepsilon_1^2 = \left(\frac{\delta N_i}{N_i}\right)^2 + \left(\frac{\delta P_{total}}{P_{total}}\right)^2 + \left(\frac{\delta M_i}{M_i}\right)^2 \tag{6}$$

For a fresh element, the value of ε_1 was estimated to be ±11.5%.

For the following four types of errors, these values were considered: ε_2=±7%, ε_3=±7%, ε_4=±4%, and ε_5=±1%.

Using an analysis on the impact of a non-uniform U-235 plate loading on the heat flux, the methodology assumed that uncertainties in loading of ±25% over 10mm^2 and ±20% over 1cm^2 will result in an uncertainty (ε_6) in heat flux of ±19%.

Using the total error/uncertainty given by Eq. (4), the maximum heat flux with HCF at the hot spot can be calculated by

$$q_{\max}^{hotspot} = q_{\max}\left(1 + \varepsilon^{hotspot}\right), \tag{7}$$

For a fresh fuel element, ε_7 is zero and Eq. (4) estimated the $\varepsilon^{hotspot}$ at 1.25.

Since the enthalpy rise is the dominant factor affecting the margin to FI, the methodology used the average heat flux over the hot stripe (q_{ave}) to calculate the “maximum” average heat flux using

$$q_{\max}^{hotsstripe} = q_{ave}\left(1 + \varepsilon^{hotsstripe}\right) = \frac{q_{\max}}{R_i}\left(1 + \varepsilon^{hotsstripe}\right), \tag{8}$$

For a fresh fuel element, ε_7 is zero and Eq. (5) estimated the $\varepsilon^{hotsstripe}$ at 1.305.

To obtain the error on the calculated burnup (ε_7), the error in the fuel element power P_i and the error in the measurement of the combined radial and azimuthal peaking f_i were combined. To further bound this error, a fuel element less likely to contain the hot spot (i.e., not in the central crown) for which the error in the relative thermal heat flux is larger (±18.6%) was considered. Using Eq. (6), a $\delta N_i/N_i$ of ±18.6%, and with other errors/uncertainties identical as for fresh fuel, ε_1 was calculated to be ±18.7%. The error on the calculated burnup can then be evaluated by

$$\varepsilon_7^2 = \left(\frac{\delta M_i^{depleted}}{M_i}\right)^2 + \left(\frac{\delta f_i}{f_i}\right)^2 = \left(\varepsilon_1^{'}\right)^2 + \left(\varepsilon_2\right)^2, \tag{9}$$

Eq. (9) gives ε_7 as ±20%.

Again using Eqs (4) and (5), $\varepsilon_{burn}^{hotspot}$ and $\varepsilon_{burn}^{hotsstripe}$ were calculated to be respectively, 1.31 and 1.405.

2.3 Determination of maximum admissible heat flux

The margin to the onset of nucleate boiling (ONB) was computed from the Bergles & Rohsenow correlation, that is

$$\Delta T_{ONB}\left[{}^\circ C\right] = 0.555\left\{\frac{q[W/cm^2]}{0.1053p^{1.156}[bar]}\right\}^{\frac{p^{0.0234}}{2.17}} \tag{10}$$

where ΔT_{onb}= Twall –Tsat, and Twall is the wall temperature at which nucleate boiling is initiated. For a maximum heat flux of 600 W/cm^2 and pressure at nominal operating conditions, a $\Delta T_{onb} = 9.5^\circ C$ was computed.

The margin to Full Nucleate Boiling (FNB) was computed with the Forster & Greif correlation.

$$\Delta T_{FNB}\left[{}^\circ C\right] = 4.57q^{0.35}[W/cm^2]p^{-0.23} \text{ for } 1 < p < 12 \text{ bar} \tag{11}$$

For a maximum heat flux of 600 W/cm^2 and pressure at nominal operating conditions, the correlation of Forster & Greif gives $\Delta T_{FNB} = 24^\circ C$.

The maximum admissible heat flux was determined by combining the most unfavourable uncertainties, that is:

minimum water gap of 0.27 cm;
loss of pressure at the inlet equal to 0.06 ata or 0.006 MPa (for a pressure drop on the fuel plates equal to 2.1 ata or 0.21 MPa**)**;
loss of pressure due to friction, by ~ 10%;
a reduction of ΔT_{FNB} by 15%;
and a heat transfer coefficient reduction by 15%;

2.4 Correlations used in the code of S.Fabrega

In the code of S. Fabrega, the correlation of Bergles and Rohsenow was used for ONB, the correlation of Forster and Greif for FNB and the correlation of Whittle and Forgan for flow instability were used.

For single phase flow the Nusselt number was computed from the Sieder-Tate corrrelation:

$$Nu = \frac{h}{\rho c v} = 0.023 Re^{-0.20} Pr^{-2/3}\left(\frac{\mu}{\mu_{plate}}\right)^{0.14} \tag{12}$$

The loss of pressure due to friction was computed from the Darcy equation:

$$\frac{\partial F}{\partial z} = \frac{\Lambda}{D}\rho\frac{v^2}{2} \tag{13}$$

where:

μ= dynamic viscosity;
v = coolant velocity;
ρ= density;
D = hydraulic diameter;
Pr= Prandtl number;
Re=Reynolds number;
and for turbulent flow and a rough wall, the Darcy's number (Λ) is

$$\Lambda = \Lambda_0 \left(\frac{\mu}{\mu_{\text{plate}}} \right)^c, \ c = 0.182 + \frac{8000}{\text{Re} + 18000}$$

where $\Lambda_0 \sim a.\text{Re}^{-b}$ ($\text{Re} < 49800, a = 0.316; b = 0.25; \text{Re} > 49800, a = 0{,}184; b = 0{,}2$).

2.5 Predictions of the code of S.Fabrega

For different coolant velocities, the predictions of the code of S. Fabrega for the maximum heat flux for flow instability (FI) at nominal conditions, and for the maximum heat fluxes at ONB and FNB with hot channel factors are summarized in Table I [1], [3-5]. The computations were performed for the channel B0 of core configuration 7B-MFBS (cycle 01/1971, operated at 70 MW), loaded with fresh fuel elements of type 6nG at the following reference conditions: inlet (reactor vessel) pressure(PRCA 4 -1302) = 1.20 MPa; a pressure drop from reactor vessel inlet to reactor vessel outlet (DPRCA 4 -1301) = 0.30 MPa (average fuel channel velocity: $V_{coolant\_\mathbf{average}}$ = 10.4 m/s, cold, for ΔP=2.1 ata or 0.21 MPa); Core inlet temperature (TRCA 4-1301) = 40°C .

Table I. Maximum heat flux at different average fuel channel coolant velocities for HEU fuel.

$V_{\mathbf{average}}$ [m/s]	ΔP [MPa]	Q_{max} [W/cm^2] (FI without HCF)	Q_{max} [W/cm^2] (FNB with HCF)	Q_{max} [W/cm^2] (ONB with HCF)
11	2.323	1396.1	705.2	626.7
10.4	2.100	1336.5	675.6	603.2
10	1.957	1296.1	655.7	585.2
9	1.619	1190.1	605.0	538.4
8	1.310	1078.9	548.5	490.5
7	1.030	963.3	490.4	441.3
6	0.780	841.0	430.8	389.8
5	0.564	718.1	369.1	334.4
4	0.382	594.0	305.2	278.2
3	0.231	459.5	238.9	219.3
2	0.114	311.3	169.3	157.1

3. PLTEMP Hot Channel Factors (HCF)

This section presents data that was used to determine the Hot Channel Factors (HCFs) to be provided as input to the PLTEMP code, as well as the derived HCFs. It also presents the approach

3.1 Determination of the HCF

Table II lists the nominal values and manufacturing tolerances for geometric data of the standard HEU BR2 fuel. Table III shows maximum values of power data based on the uncertainty of measurements, and the minimum value of the heat transfer coefficient. Table IV gives nominal values and uncertainties for the hydraulic data.

Table II. Nominal values and manufacturing tolerances for the standard HEU fuel.

	Nominal value	Tolerance
Aluminum cladding, mm	0.381	±0.077
Water gap, mm	3.00	±0.30
Fuel homogeneity on 10 mm^2	1.0	±20%
Fuel length, mm	970	+0.6/ -0.3
Outer diameter of the Al side plates	82.6	±0.2
External diameter of fuel plate six, mm (maximum)	77.7	
Total heated surface of a fuel element, cm^2	$1.33x10^4$	$\pm 0.08\ x10^4$

Table III. Maximum values of power data, and the minimum value of the heat transfer coefficient

	Nominal value	Maximum or minimum value
Total power	1.0	1.04
Channel power	1.0	1.06
Axial power (as function of CR position)	1.0	1.03
Power density (power distribution)	1.0	1.20
Heat transfer coefficient, *h*	1.0	0.85

Table IV. Nominal hydraulic data and uncertainties.

	Nominal value	Uncertainty
Inlet pressure, MPa	1.26	± 0.06
Pressure drop on the reactor, MPa	0.30	± 0.02
Pressure drop over fuel plates, MPa	0.21	± 0.014
Pressure drop between reactor vessel inlet and the inlet of fuel channel matrix, MPa	0.035	
Pressure drop between outlet of fuel channel matrix and reactor vessel outlet, MPa	0.055	
Average fuel channel coolant velocity (between fuel plates), m/s	10.4	± 0.90

The HCFs listed in Table V are the result of random and systematic sources of uncertainty. The random sources can affect any fuel plate or coolant channel. However, it is unlikely that all of the random sources can adversely affect the limiting location(s) in the reactor core simultaneously. The first four random sources relate to the distribution of power. The final two random sources relate to channel spacing and flow distribution.

The first two random uncertainties, which are caused by variations in the fuel meat thickness and ^{235}U homogeneity, are labelled "local" in that they are assumed to be hot-spot effects that affect the heat flux in only a local area with only minor perturbations in bulk coolant temperature. Since these sources of uncertainty affect the distribution of fuel rather than the total amount of it, the bulk coolant outlet temperature is not affected by these sources. However, the relocation of fuel so that it is closer to the coolant inlet can result in higher bulk coolant temperatures at locations upstream of the outlet. Where this is a concern, subcomponents for F_{bulk} from these sources should be included. When fuel meat thickness or the ^{235}U homogeneity subcomponents are included in F_{bulk}, it may not be appropriate to also include the ^{235}U loading per plate subcomponent in F_{bulk}.

The systematic errors can be directly included in the PLTEMP calculation by increasing the reactor power, decreasing the reactor flow and decreasing the Nusselt number, which provides the film coefficient, to reflect the systematic errors. Then only the combined random errors need be modelled as direct multiplicative factors applied to calculated temperature rises and heat fluxes. The approach used in PLTEMP to account for the random and systematic errors is discussed in section 3.2. More details about the HCF methodology in PLTEMP can be found in Refs. 8 and 9.

A line-by-line description of Table V follows:

Aluminum cladding
For this analysis, a 25% uncertainty on the cladding thickness was considered. This value is obtained by dividing the uncertainty in the thickness (0.077mm) by the minimum value (0.304mm).

Fuel meat thickness
This is a result of the manufacturing process. When the fuel plates are rolled to the desired size, the fuel meat thickness in some regions of the plate may be thicker by as much as a specified tolerance. Other regions of the fuel meat can be too thin and result in less than the nominal heat flux. The amount of ^{235}U in each plate is assumed to be measured separately so that the fuel meat thickness only affects the distribution of power within the plate. This uncertainty was determined in [4].

^{235}U homogeneity
This is a tolerance on how well the ^{235}U is mixed with the other ingredients that are in the fuel meat. The amount of ^{235}U in each plate is normally assumed to be measured separately so that the ^{235}U homogeneity only affects the distribution of power within the plate. The uncertainty 20% shown is the Table was determined in the 70's for the standard BR2 fuel and this value was used in the calculations by S. Fabrega [1], [3-7].

^{235}U plate loading
For this analysis, the effect of ^{235}U plate loading was not taken into account separately from the ^{235}U homogeneity and was folded into the tolerance on homogeneity.

Power density
This uncertainty is assumed to be a result of the physics calculations and can result in more power being in a particular plate than was predicted and used in the nominal thermal-hydraulic analysis. This uncertainty was reviewed again in 2009 by the BR2 staff and it was determined to be equal to 1.20.

Channel spacing
The TH studies in the 70's were performed varying the thickness of the water gap between 3.0 cm and 2.7 cm [4-6]. In the present PLTEMP calculations, this tolerance is obtained by dividing the nominal channel thickness by the minimum channel thickness allowed by the dimensional tolerances. In Table V, an 11% tolerance was obtained by dividing 3.0 cm by (3.0 – 0.3) cm. For plate geometry where the hydraulic diameter can be approximated as twice the channel thickness, the formulas for obtaining the F_{bulk} and F_h subcomponents for turbulent flow are described by

$$F_{bulk} = \left(t_{nc}/t_{hc}\right)^{3/(2-\alpha)} \tag{14}$$

$$F_h = \left(t_{nc}/t_{hc}\right)^{(0.4+\alpha)/(2-\alpha)} \tag{15}$$

where t_{nc} and t_{hc} are the nominal channel thickness and the minimum (or hot) channel thickness, respectively. α is the value of the Reynolds number exponent in the friction factor relationship. For both laminar and turbulent flow the F_w subcomponent is equal to the F_{bulk} one.

Flow distribution

This uncertainty is the result of the hydraulic analysis that is used to determine the distribution of flow through the reactor. This is a local effect that does not systematically affect all coolant channels. The determination of this uncertainty was based on the measurement of the flow velocities in the different coolant channels of the BR2 fuel elements [4].

Random errors combined

It is unlikely that all of the random errors and uncertainties will occur together at the most limiting location in the reactor and that each will adversely affect reactor performance. Therefore, the random subcomponents, F^i, of each hot channel factor, F, are combined statistically, i.e., $F = 1 + \sqrt{\sum_i \left(1 - F^i\right)^2}$.

Power measurement

This is a tolerance of the meter that is used to measure power and, if present, would affect all fuel plates essentially equally.

Flow measurement

This is a tolerance of the meter that is used to measure flow and, if present, would affect the flow in all flow channels essentially equally.

Heat transfer coefficient

This is due to uncertainties in the correlations for Nusselt number that are used to determine values of heat transfer coefficient, h. If the Nusselt number correlations that are used in the analysis predict values that are too large, then the predicted temperatures on all clad surfaces will be lower than would otherwise be experienced by the reactor. This is a core-wide effect rather than one that is random in location.

Systematic errors combined

Because systematic errors, such as an error in reactor power and flow measurement, affect all locations within the reactor at the same time, it is reasonable to expect that all of them could be present at the limiting location(s). Therefore, the systematic subcomponents are combined multiplicatively, i.e. $F = \prod_i F^i$.

The Hot Channel Factors for the HEU (93% U235) core derived, as explained above, from the data of Tables II to IV are given in Table V. The same HCFs were used to determine the power and flow limits in the LEU core (U-8Mo, 7.5 g/cc, cadmium wires, D=0.5 mm).

Table V. PLTEMP HCF for BR2 standard HEU fuel element and LEU fuel element

					Hot Channel factors				
Uncertainty	Type of tolerance	Effect on bulk ΔT, fraction	Value	Tolerance, fraction	Heat flux, F_q	Channel flow rate, F_w	Heat transf. coef., F_h	Chan. Temp. rise, F_{bulk}	Film temp. rise, F_{film}
Al cladding, [mm]	random		0.381 to 0.304	0.25	1.25				1.25
U5 homog.				0.20	1.20				1.20
U5 loading per plate				0.00	1.00			1.00	1.00
Power density		0.5		0.20	1.20			1.10	1.20
Water gap, [mm]		1.00	3.00 to 2.7	0.11		1.20	1.04	1.20	1.04
Flow distr. (water speed, m/s)		1.00	10.4 to 9.5	0.10		1.10	1.07	1.10	1.07
Random errors combined					**1.38**	**1.22**	**1.08**	**1.25**	**1.39**
Power measurement	systematic			0.06	1.06			1.06	1.06
Flow measurement				0.02		1.02	1.00	1.02	1.00
Heat transfer coef.				0.15			1.15		1.15
Systematic errors combined					**1.06**	**1.02**	**1.15**	**1.08**	**1.22**
Product of random and systematic errors					1.46	1.24	1.24	1.35	1.70

3.2 Application of HCFs in PLTEMP

The PLTEMP option used for the calculation of the margin to ONB proceeds as follows.

First a nominal or best estimate PLTEMP simulation is performed with input variables at their nominal values. The margin to ONB at a point on the cladding surface having a temperature $T_{w,op}$ can be defined by the temperature ratio r_{tr}

$$r_{tr} = (T_{onb} - T_{in})/(T_{w,op} - T_{in}) \qquad (16)$$

where T_{onb} is given by a correlation, e.g., Bergles & Rohsenow.

The margin to ONB at a given cladding location (node) can also be expressed in terms of the ratio of the reactor power that would give a cladding temperature at that node equal to T_{onb} to the operating reactor power, r_p. This margin is computed as follows. If the reactor power would increase at constant flow by a factor r_p, the convective heat transfer coefficient would remain practically constant and

$$q" = r_p\, q"_{op} \qquad (17)$$

$$T_b - T_{in} = r_p\,(T_{b,op} - T_{in}) \qquad (18)$$

$$T_w - T_b = r_p\,(T_{w,op} - T_{b,op}) \qquad (19)$$

Adding Eqs. (18) and (19), gives

$$T_w = T_{in} + r_p\,(T_{w,op} - T_{in}) \qquad (20)$$

Setting the nodal wall temperature of Eq. (20) equal to the ONB temperature corresponding to the heat flux $r_p\, q"_{op}$ (in W/m^2), one gets from the Bergles & Rohsenow correlation the following equation for r.

$$T_{in} + r_p\,(T_{w,op} - T_{in}) = T_{sat} + (5/9)[\, r_p\, q"_{op} / (1082.9\, P^{1.156})]**(P^{0.0234/2.16}) \qquad (21)$$

The value of r_p given by Eq. (21) is the margin to ONB at nominal operating conditions at the cladding location (axial node) under consideration.

After the margin at nominal conditions has been determined, a second PLTEMP simulation is performed where the reactor power, reactor flow and the heat transfer coefficient are set equal to F_qQ, Flow/F_f and h/F_h, where Q, Flow and h are the nominal values of reactor power, reactor flow and heat transfer coefficient, and F_q, F_f, and F_h are the corresponding HCFs. The cladding and coolant temperatures calculated at this step, step 2, and the three user-input local hot channel factors (F_{bulk} , F_{film} and F_{flux}) are used in the following equations:

$$T_{b,hc} - T_{in} = F_{bulk}\,(T_{b,2} - T_{in}) \tag{22}$$

$$T_{w,hc} - T_{b,hc} = F_{film}\,(T_{w,2} - T_{b,2}) \tag{23}$$

The addition of Eqs. (21) and (22), gives

$$T_{w,hc} = T_{in} + F_{bulk}\,(T_{b,2} - T_{in}) + F_{film}\,(T_{w,2} - T_{b,2}) \tag{24}$$

The heat flux in the hot channel is

$$q''_{hc} = F_{flux}\, q''_2 \tag{25}$$

If the reactor power would increase at constant flow by a factor r_{hc}, setting the nodal wall temperature of Eq. (24) equal to the ONB temperature corresponding to the heat flux given by $r_{hc}\, F_{flux}\, q''_2$ (in W/m^2), one gets from the Bergles & Rohsenow correlation the following equation for r_{hc}

$$T_{in} + r_{hc}\,\{F_{bulk}\,(T_{b,2} - T_{in}) + F_{film}\,(T_{w,2} - T_{b,2})\} =$$

$$T_{sat} + (5/9)[\, r_{hc}\, F_{flux}\, q''2 \,/\, (1082.9\, P^{1.156})]^{**}(P^{0.0234/2.16}) \tag{26}$$

The value of r_{hc} given by Eq. (26) is the margin to ONB at HFC-conditions at the cladding location (axial node) under consideration.

Finally, the user can request for the code (PLTEMP) to compute the reactor power under normal or HCF-conditions that in the most limiting cladding location (node) gives a cladding temperatures equal to T_{onb}, that is, at this location r_p, or r_{ch} is equal to one. This is the limiting reactor power value in terms of ONB under normal or HCF operating conditions.

A similar approach is used to compute the limiting reactor power in terms of FNB.

For the onset of excursive flow instability, FI, PLTEMP uses the correlation of Whittle and Forgan

$$(T_{out} - T_{in})/(T_{sat} - T_{in}) = R = \frac{1}{1+\eta D_H/L_H} \tag{27}$$

Where: η = constant = 32.5, D_H=channel heated diameter, and L_H is the channel heated length.

4.0 Thermal-hydraulic analysis with PLTEMP/ANL V3.6

PLTEMP/ANL V3.6 [2] can model an entire core, a number of fuel assemblies, or a series of fuel plates and coolant channels that form a portion of an assembly. The model includes heat transfer from one coolant channel to the next, through the thickness of the intervening fuel plate. However, the code does not include heat transfer from one assembly to its immediate neighbours. The code allows the user to specify the flow rate for each coolant channel. An incentive for modelling an entire core is that the code has a hydraulics model that can determine the flow distribution among core assemblies. The code allows only one axial power distribution per solution and applies it to all of the fuel plates being analyzed. Therefore, separate runs of the code need to be used to analyze potentially limiting portions of assemblies individually. In these runs, the relative axial power distribution of the limiting plate is used for the relative axial power shape of all plates.

A small portion of the reactor power that is deposited outside of the core is not included in the PLTEMP calculation. The remaining power is assumed to be deposited in the fuel meat.

The code geometry can accommodate fuel assemblies made from a series of flat (or slightly curved) plates, or from nested tubes. A variety of thermal-hydraulic correlations are available to determine safety margins such as Onset of Nucleate boiling (ONB), departure from nucleate boiling (DNB), and onset of flow instability (FI). Coolant properties for either light or heavy water are obtained from FORTRAN functions rather than from tables. The code is intended for thermal-hydraulic analyses in single phase flows. Correlations for both turbulent and laminar flows are available.

4.2 Results of the PLTEMP analysis

One of the objectives of this work was to determine power, heat flux and velocity limits in the BR2 core loaded with HEU standard BR2 fuel (93% U235) and to compare these limits with those determined in the 70's by the code of S. Fabrega. For this purpose, the same initial conditions as those used in the calculations of the 70's (see beginning of Sect. 2.5) were used in the PLTEMP analysis. Another objective was to perform a steady state thermal hydraulics analysis with the PLTEMP code for the BR2 LEU core, and compare the performance of the LEU core with that of the HEU core.

PLTEMP simulations were performed with different correlations for the heat transfer coefficient. These include the Sieder-Tate, Dittus-Boelter, Colburn, Petukhov & Popov and another Russian correlation [10]. For the margin to ONB the correlation of Bergles & Rohsenow was used, while the correlation of Forster & Greif was used for the margin to FNB. The margin to flow instability was computed with the correlation of Whittle & Forgan.

4.2.1 Axial power profiles for HEU and LEU cores

The power peaking factors were calculated with MCNPX (see Fig. 1a) using a load, similar to that of the BR2 cycle 04/2007A.5. For the HEU core the calculations were performed for a fresh 6NG standard BR2 fuel element: UAlx fuel, 93% enriched in U235; uranium density of 1.3 g/cc; and B_4C and Sm_2O_3 burnable absorbers homogeneously mixed in the fuel meat. For the LEU core the fuel was: UMo fuel, 19.8% enrichment in U235; uranium density 7.5 g/cc; cadmium wires with D=0.4 mm in the Al side plates.

Fig. 1b shows a comparison of the axial power profiles as calculated by MCNPX for the HEU and the LEU cores (the same profiles as in Fig. 1a, but normalized to unity) with the measured axial power profile used in the evaluation of the safety margins with the code of FABREGA (BR2 SAR [3-7]). The measurements were made in a fresh fuel element 6NG. Fig. 1b shows that the axial power profile used in the PLTEMP calculations for the HEU core is similar to that used in the calculations with the code of FABREGA.

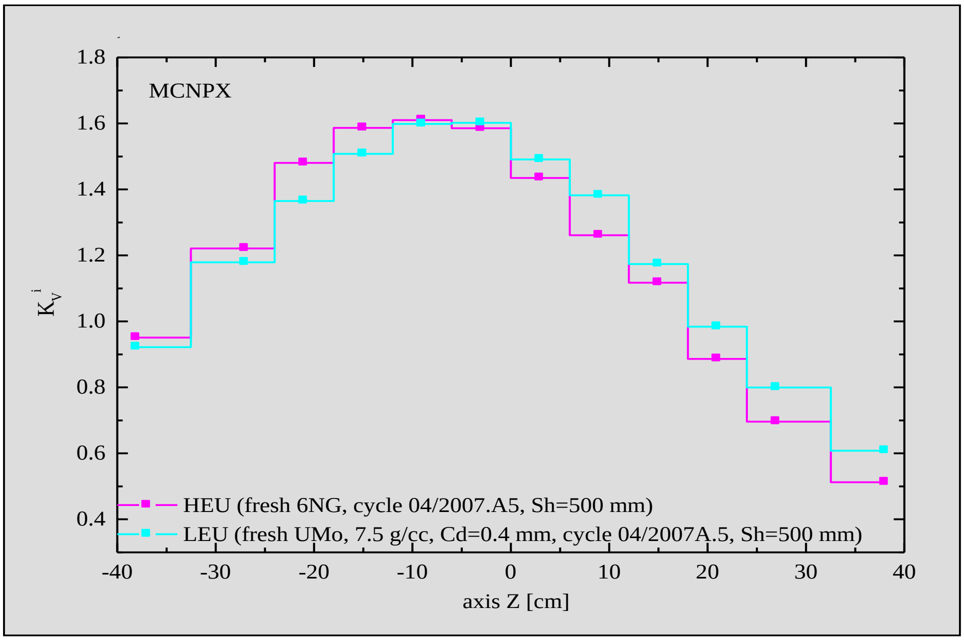

Figure 1a. Axial power distribution (local/average) in the outer (limiting) fuel plate of a standard fresh BR2 fuel element 6NG calculated with MCNPX in the HEU core and in the fully converted LEU core for the load of cycle 04/2007A.5.

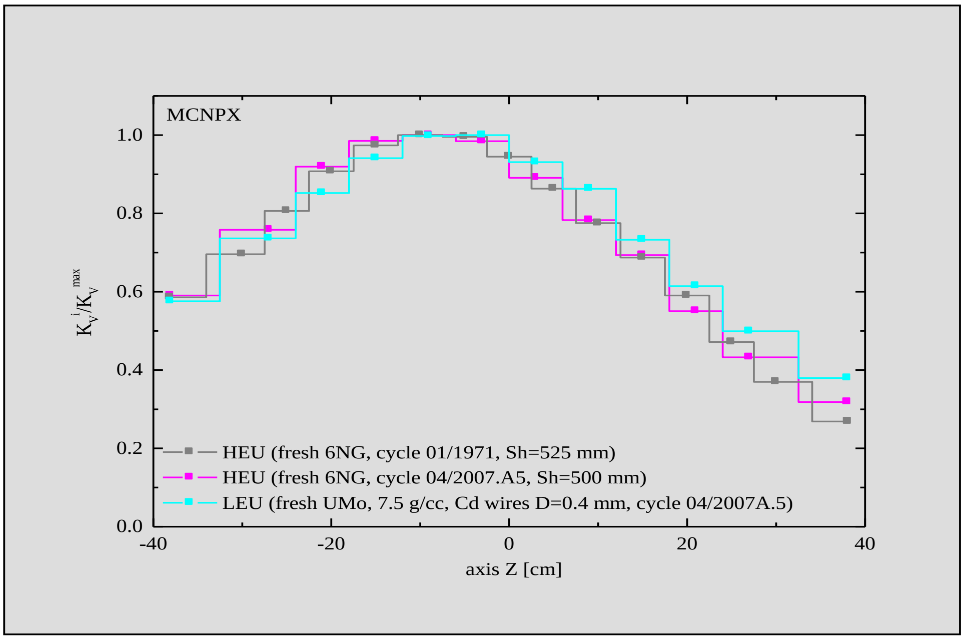

Figure 1b. Normalized axial power profiles in the outer (limiting) fuel plate of a standard fresh BR2 fuel element 6NG calculated with MCNPX for the HEU and LEU cores, and the measured [4] axial power profile in a fresh 6NG element in the load of cycle 01/1971.

4.2.2 Power and velocity limits in the HEU core

This section presents the power, heat flux and velocity limits for the HEU core as computed with PLTEMP. In the FABREGA analysis, the cycle 01/1971, which was operated at 70 MW, was used for the evaluations of the axial power distribution and the safety margins reported in SAR. For the evaluation of the total power and the peaking factors in a standard fresh BR2 fuel element, a load similar to that of cycle 04/2007A.5 was used. The initial total power in the fresh fuel element computed with MCNPX for this HEU core is equal to 2.02 MW at a total reactor power of 55 MW. To be consistent with the FABREGA and SAR analyses, the initial power of the fresh fuel element used in PLTEMP was therefore increased from 2.02 MW to 2.57 MW, which corresponds to a total reactor power of 70 MW. A thermal conductivity of 150 W/m-K° was used for the aluminium cladding, and 80 W/m-K° for the fuel meat in Ref. 11.

The results of calculations for coolant velocities of 10.4, 2.08, 21.0, and 0.104 m/s are summarized in Tables VI to XI. In the BR2 safety analysis report it is indicated that the reactor vessel inlet pressure is 1.26±0.06 MPa. The reference calculations performed with the code of S. Fabrega for the safety limits, which are reported in the BR2-SAR, used a reactor vessel inlet pressure of 1.20 MPa. For these reasons, each velocity calculation was performed at two values of the inlet pressure: 1.26 MPa and 1.20 MPa. The Tables show limiting nominal values as well as limiting values with HCFs taken into account. The latter values are shown in parentheses. The limiting values presented in the Tables include: heat flux, power in the limiting element (assembly), and corresponding cladding temperature.

Table VI. Maximum heat flux [W/cm^2] / Maximum fuel element power [MW] / Maximum wall (surface fuel-to-clad) temperature at which ONB, or FNB, or FI are reached at the following nominal conditions in HEU core: $\underline{V=10.4\ m/s;\ P_{in}=1.26\ MPa;\ \Delta P_{fuel\ plates}=0.21\ MPa}$; T_{in}=40° C.

Limiting Condition	Heat flux / Fuel element power without HCF (Heat flux / Fuel element power with HCF)				
	Single phase heat transfer correlation				
	Sieder-Tate	Dittus-Boelter	Colburn	Petukhov & Popov	Russian
Onset of Nucleate Boiling (Bergles & Rohsenow)	822.3/7.09 (678.2/4.07)	647.2/5.67 (536.2/3.25)	843.8/7.12	837.4/7.1	793.1/6.69
Fully Developed Nucleate Boiling (Forster & Greif)	926.0/8.0 (764.0/4.595)	715.6/6.35 (593.3/3.605)	972.8/8.2	936.9/7.95	882.5/7.45
Flow Instability (Whittle & Forgan)	1424.8/12.4 (1378.5/11.635)	1370.5/12.23 (1325.8/11.475)	1523.5/12.67	1475.8/12.55	1480.2/12.55

Table VII. Maximum heat flux [W/cm^2] / Maximum fuel element power [MW] / Maximum wall (surface fuel-to-clad) temperature at which ONB, or FNB, or FI are reached at the following nominal conditions in HEU core: V=10.4 m/s; P_{in}=1.20 MPa; $\Delta P_{fuel\ plates}$=0.21 MPa; T_{in}=40° C.

Limiting Condition	Heat flux / Fuel element power without HCF (Heat flux / Fuel element power with HCF) (Wall temperature at ONB or FNB = 1.0)				
	Single phase heat transfer correlation				
	Sieder-Tate	Dittus-Boelter	Colburn	Petukhov & Popov	Russian
Onset of Nucleate Boiling (Bergles & Rohsenow)	807.4/6.96 (666.7/4.00) (T_{clad}= 193.2)	636.5/5.57 (528.1/3.2) (T_{clad}=192.1)	827.1/6.98	822.2/6.97	779.0/6.57
Fully Developed Nucleate Boiling (Forster & Greif)	909.9/7.86 (752.6/4.525) (T_{clad}=209.8)	712.1/6.25 (585.3/3.555) (T_{clad}=207.5)	953.6/8.04	921.7/7.82	868.3/7.33
Flow Instability (Whittle & Forgan)	1399.9/12.18 (1353.4/11.42)	1346.6/12.01 (1302.3/11.27)	1494.4/12.4 3	1448.8/12.32	1454.3/12.33

Table VIII. Maximum heat flux [W/cm^2] / Maximum fuel element power [MW] / Maximum wall (surface fuel-to-clad) temperature at which ONB, or FNB, or FI are reached at the following nominal conditions in HEU core: V=2.08 m/s; P_{in}=1.26 MPa; $\Delta P_{fuel\ plates}$=0.12 MPa; T_{in}=40° C.

Limiting Condition	Heat flux / Fuel element power without HCF (Heat flux / Fuel element power with HCF)				
	Single phase heat transfer correlation				
	Sieder-Tate	Dittus-Boelter	Colburn	Petukhov & Popov	Russian
Onset of Nucleate Boiling (Bergles & Rohsenow)	215.0/1.80 (176.8/1.04)	170.8/1.47 (142.9/0.855)	217.2/1.775	209.3/1.73	206.0/1.685
Fully Developed Nucleate Boiling (Forster & Greif)	234.0/1.96 (192.7/1.135)	185.4/1.60 (153.8/0.923)	239.2/1.95	226.9/1.875	222.4/1.82
Flow Instability (Whittle & Forgan)	302.2/2.54 (289.0/1.372)	285.1/2.487 (274.2/2.328)	319.8/2.585	309.0/2.555	312.6/2.565

Table IX. Maximum heat flux [W/cm^2] and maximum assembly power [MW] at which ONB, or FNB, or FI are reached: values at nominal conditions and values, in parentheses, with HCF; V=2.08 m/s; P_{in}=1.20 MPa; $\Delta P_{fuel\ plates}$=0.12 MPa; T_{in}=40° C.

Limiting Condition	Heat flux / Fuel element power without HCF (Heat flux / Fuel element power with HCF) (Wall temperature at ONB or FNB = 1.0)				
	Single phase heat transfer correlation				
	Sieder-Tate	Dittus-Boelter	Colburn	Petukhov & Popov	Russian
Onset of Nucleate Boiling (Bergles & Rohsenow)	214.7/1.798 (176.6/1.039) (T_{clad}=188.2)	170.6/1.47 (142.8/0.855) (T_{clad}=188.3)	212.8/1.74	205.7/1.7	202.3/1.655
Fully Developed Nucleate Boiling (Forster & Greif)	233.7/1.958 (192.5/1.134) (T_{clad}=199.2)	185.3/1.6 (153.9/0.924) (T_{clad}=198.8)	235.4/1.92	222.6/1.84	218.6/1.79
Flow Instability (Whittle & Forgan)	303.2/2.49 (290.6/2.385)	286.5/2.5 (275.6/2.342)	313.3/2.53	303.6/2.51	307.0/2.52

Table X. Maximum heat flux [W/cm^2] / Maximum fuel element power [MW] / Maximum wall (surface fuel-to-clad) temperature at which ONB, or FNB, or FI are reached at the following nominal conditions in HEU core: V=21 m/s; $\Delta P_{fuel\ plates}$=0.21 MPa; T_{in}=40° C.

Limiting Condition	Heat flux / Fuel element power without HCF (Heat flux / Fuel element power with HCF)			
	P_{in}=1.26 MPa		P_{in}=1.20 MPa	
	Sieder-Tate	Dittus-Boelter	Sieder-Tate	Dittus-Boelter
Onset of Nucleate Boiling (Bergles & Rohsenow)	1490.4/12.95 (1222.8/7.385)	1159.7/10.186 (960.0/5.84)	1465.4/12.73 (1202.3/7.26) (T_{clad}=196.6)	1143.4/10.04 (946.8/5.758) (T_{clad}=195.2)
Fully Developed Nucleate Boiling (Forster & Greif)	1707.7/14.87 (1401.5/8.48)	1314.6/11.58 (1077.5/6.57)	1681.5/14.64 (1380.4/8.35) (T_{clad}=216.0)	1296.8/11.42 (1063.9/6.485) (T_{clad}=213.2)
Flow Instability (Whittle & Forgan)	2836.3/24.89 (2751.0/23.37)	2747.7/24.61 (2663.3/23.104)	2784.6/24.43 (2701.2/22.94)	2699.3/24.165 (2615.8/22.68)

Table XI. Maximum heat flux [W/cm^2] / Maximum fuel element power [MW] / Maximum wall (surface fuel-to-clad) temperature at which ONB, or FNB, or FI are reached at the following nominal conditions in HEU core: V=0.104 m/s; $\Delta P_{fuel\ plates}$=0.01 MPa; T_{in}=40° C.

Limiting Condition	Heat flux / Fuel element power without HCF (Heat flux / Fuel element power with HCF) (Wall temperature at ONB or FNB = 1.0)			
	P_{in}=1.26 MPa		P_{in}=1.20 MPa	
	Sieder-Tate	Dittus-Boelter	Sieder-Tate	Dittus-Boelter
Onset of Nucleate Boiling (Bergles & Rohsenow)	17.0/0.1293 (14.9/0.0812)	14.7/0.117 (12.3/0.0696)	17.5/0.133 (15.2/0.071) (T_{clad}=185.2)	14.8/0.12 (12.5/0.071) (T_{clad}=185.2)
Fully DevelopedNucleate Boiling (Forster & Greif)	17.7/0.1342 (15.55/0.0848)	15.2/0.1213 (12.8/0.0723)	18.2/0.138 (15.9/0.087) (T_{clad}=190.9)	15.7/0.125 (13.0/0.074) (T_{clad}=190.5)
Flow Instability (Whittle & Forgan)	17.6/0.134 (16.67/0.1247)	16.37/0.1306 (15.5/0.1216)	18.3/0.139 (17.2/0.129)	16.9/0.135 (16.0/0.126)

The power versus flow velocity diagrams that determine the regimes at nominal operation are shown Figs. 2 and 3. Fig. 3 also shows comparisons of the limiting values obtained by PLTEMP with those of FABREGA. The code of S. FABREGA used the Sieder-Tate correlation [3-5]. Fig. 3 shows that at nominal conditions (P_{in}=1.20 MPa, V=10.4 m/s, $\Delta P_{fuel\ plates}$=0.21 MPa; T_{in}=40° C) and for the Sieder-Tate correlation the FABREGA predictions are in good agreement with those of PLTEMP.

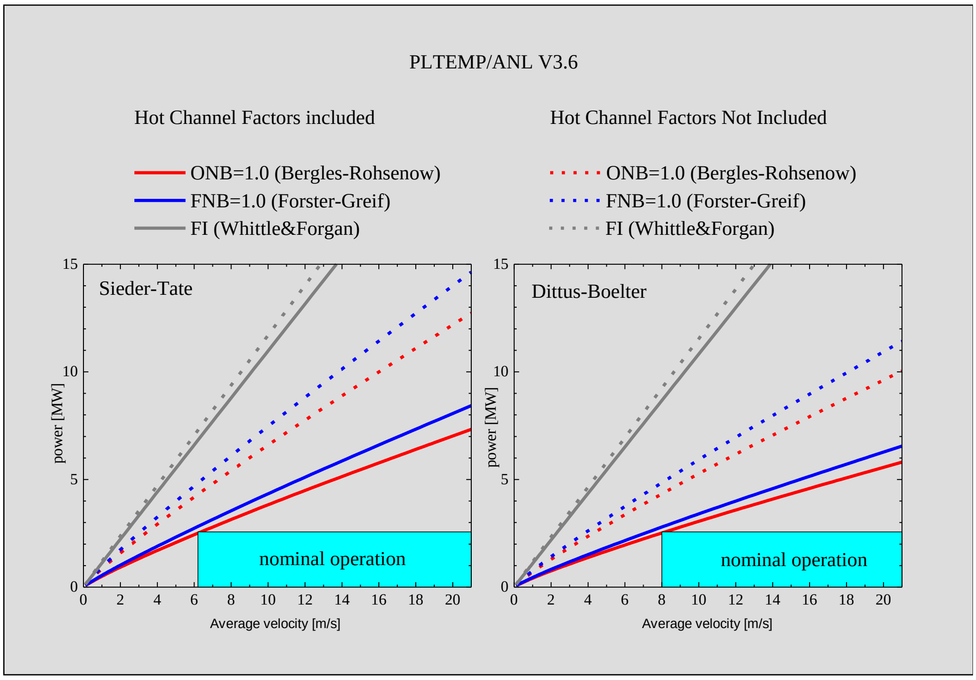

Figure 2. Fuel element power vs flow velocity curves with and without HCF for ONB, FNB and FI in HEU core.

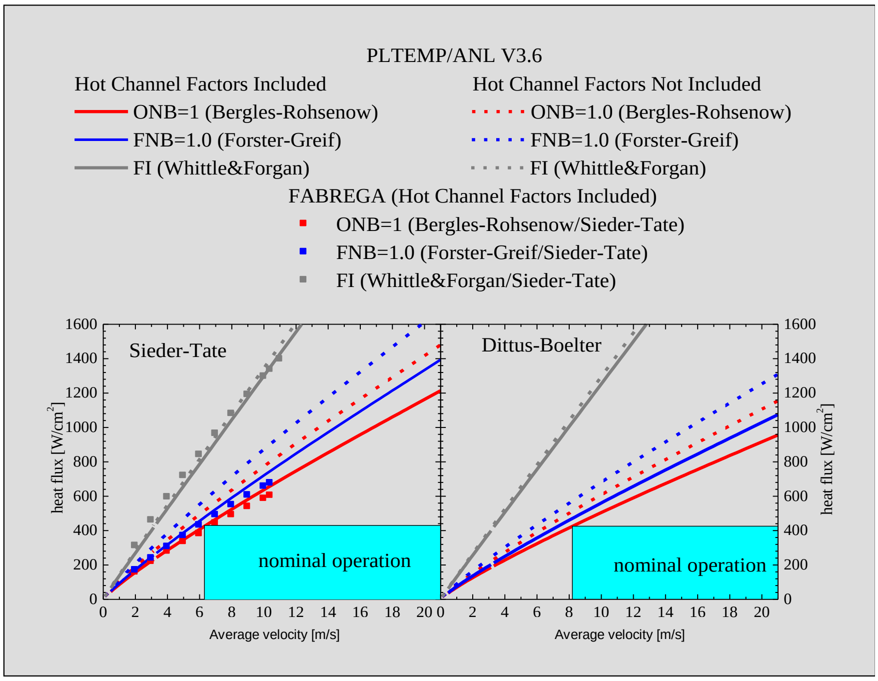

Figure 3. Maximum heat flux vs flow velocity curves with and without HCF for ONB, FNB and FI in HEU core.

Fig. 4 shows the variation of the maximum wall temperature, T_{onb} and T_{fnb}, with maximum heat flux in HEU core. Comparison of these figures with similar figures generated with results [4] of the S. Fabrega code show very good agreement of the S. Fabrega predictions with those of PLTEMP. For example, at the same Q_{max}=600 W/cm2, ΔT_{onb} = 9.2°C calculated with PLTEMP vs. ΔT_{onb} =9.5°C obtained by Fabrega; ΔT_{fnb} = 23.85°C, calculated with PLTEMP, vs. ΔT_{fnb} =24° C obtained by Fabrega.

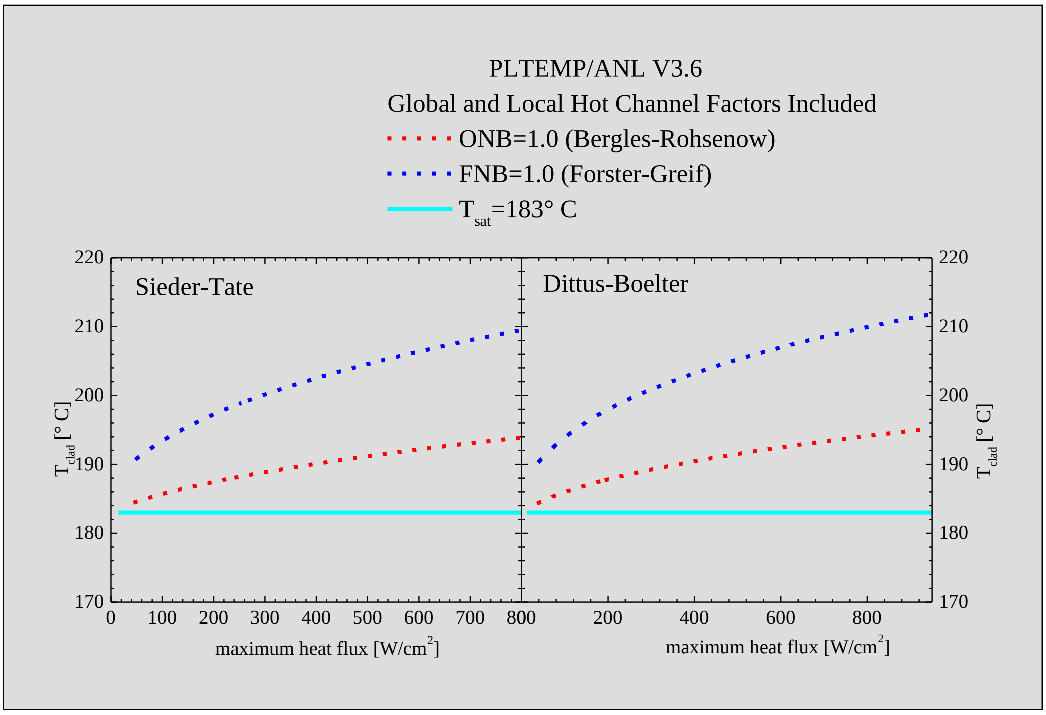

Figure 4. T_{onb} and T_{fnb} vs maximum heat flux in HEU core.

4.2.2 Power and velocity limits in the LEU core

This section presents the power, heat flux and velocity limits for the LEU core as computed with PLTEMP. Fig. 1 shows the normalized axial power profile for the U-8Mo fuel: 20% U235 enrichment, 7.5 g/cc uranium density and cadmium wires of 0.5 mm diameter. The total power in the LEU hot fuel element (assembly) is equal to 2.75 MW for a total reactor power of 70 MW. A thermal conductivity of 180 W/m-K° was used for the aluminium cladding, and 14 W/m-K° for the fuel meat [6].

The results of calculations for coolant velocities of 10.4, 2.08, 21.0, and 0.104 m/s are summarized in Tables XII to XIV. For each velocity calculations were performed for a reactor vessel inlet pressure of 1.20 MPa. The Tables show limiting nominal values as well as limiting values with HCFs taken into account. The latter values are shown in parentheses. The limiting values presented in the Tables include: heat flux, power in the limiting assembly, and corresponding cladding temperature. The calculations were performed with the Sieder-Tate and the Dittus-Boelter heat transfer correlations. The HCFs were the same as those used for the HEU core, listed in Table V.

The power versus flow velocity diagrams and the region of nominal operation are shown Figs. 5 and 6. Fig. 7 presents a comparison of the power, heat flux and flow velocity curves in the HEU and LEU cores.

Table XII. Maximum heat flux [W/cm^2] / Maximum fuel element power [MW] / Maximum wall (surface fuel-to-clad) temperature at which ONB, or FNB, or FI are reached at the following nominal conditions in LEU core: <u>V=10.4 m/s; P_{in}=1.20 MPa; $\Delta P_{fuel\ plates}$=0.21 MPa</u>; T_{in}=40° C.

Limiting Condition	Heat flux / Fuel element power without HCF (Heat flux / Fuel element power with HCF) (Wall temperature at ONB or FNB = 1.0)	
	Sieder-Tate heat transfer corr.	Dittus-Boelter heat transfer corr.
Onset of Nucleate Boiling (Bergles & Rohsenow)	802.04/6.765 (657.1/3.785) (T_{clad}= 193.0)	634.47/5.4 (527.7/3.057) (T_{clad}=192.2)
Fully Developed Nucleate Boiling (Forster & Greif)	905.78/7.65 (744.2/4.293) (T_{clad}=208.6)	712.3/6.074 (585.2/3.396) (T_{clad}=207.6)
Flow Instability (Whittle & Forgan)	1387.9/11.773/288 (1350/11.0651)	1355.7/11.673/332 (1316.7/10.9658)

Table XIII. Maximum heat flux [W/cm^2] / Maximum fuel element power [MW] / Maximum wall (surface fuel-to-clad) temperature at which ONB, or FNB, or FI are reached at the following nominal conditions in LEU core: V=2.08 m/s; P_{in}=1.20 MPa; $\Delta P_{fuel\ plates}$=0.12 MPa; T_{in}=40° C.

Limiting Condition	Heat flux / Fuel element power without HCF (Heat flux / Fuel element power with HCF) (Wall temperature at ONB or FNB = 1.0)	
	Sieder-Tate heat transfer corr.	Dittus-Boelter heat transfer corr.
Onset of Nucleate Boiling (Bergles & Rohsenow)	214.4/1.76 (177.3/1.019) (T_{clad}=189.9)	170.8/1.432 (142.8/0.832) (T_{clad}=189.0)
Fully Developed Nucleate (Forster & Greif)	233.4/1.918 (192.9/1.11) (T_{clad}=200.8)	185.4/1.557 (152.8/0.892) (T_{clad}=198.4)
Flow Instability (Whittle & Forgan)	299.6/2.47 (288.4/2.31)	286.7/2.43 (275.7/2.27)

Table XIV. Maximum heat flux [W/cm^2] / Maximum fuel element power [MW] / Maximum wall (surface fuel-to-clad) temperature at which ONB, or FNB, or FI are reached at the following nominal conditions in LEU core: P_{in}=1.20 MPa; T_{in}=40° C.

Limiting Condition	Heat flux / Fuel element power without HCF (Heat flux / Fuel element power with HCF) (Wall temperature at ONB or FNB = 1.0)			
	V=21.0 m/s		**V=0.104 m/s $\Delta P_{fuel\ plates}$=0.0 MPa**	
	Sieder-Tate heat transfer corr.	Dittus-Boelter heat transfer corr.	Sieder-Tate heat transfer corr.	Dittus-Boelter heat transfer corr.
Onset of Nucleate Boiling (Bergles & Rohsenow)	1457.9/12.371 (1187.2/6.875) (T_{clad}=195.1)	1145.0/9.76 (946.7/5.498) (T_{clad}=195.2)	17.58/0.132 (15.48/0.083) (T_{clad}=186.3)	15.17/0.119 (12.69/0.0704) (T_{clad}=186.5)
Fully Developed Nucleate Boiling (Forster & Greif)	1673.9/14.22 (1368.0/7.93) (T_{clad}=214.9)	1296.8/11.075 (1065.2/6.195) (T_{clad}=213.5)	18.381/0.138 (16.15/0.0866) (T_{clad}=191.5)	15.8/0.124 (13.2/0.073) (T_{clad}=191.2)
Flow Instability (Whittle & Forgan)	2775.2/23.665 (2706.4/21.846)	2731.0/23.5288 (2660.1/21.706)	18.04/0.1355 (17.13/0.1262)	16.816/0.132 (15.98/0.123)

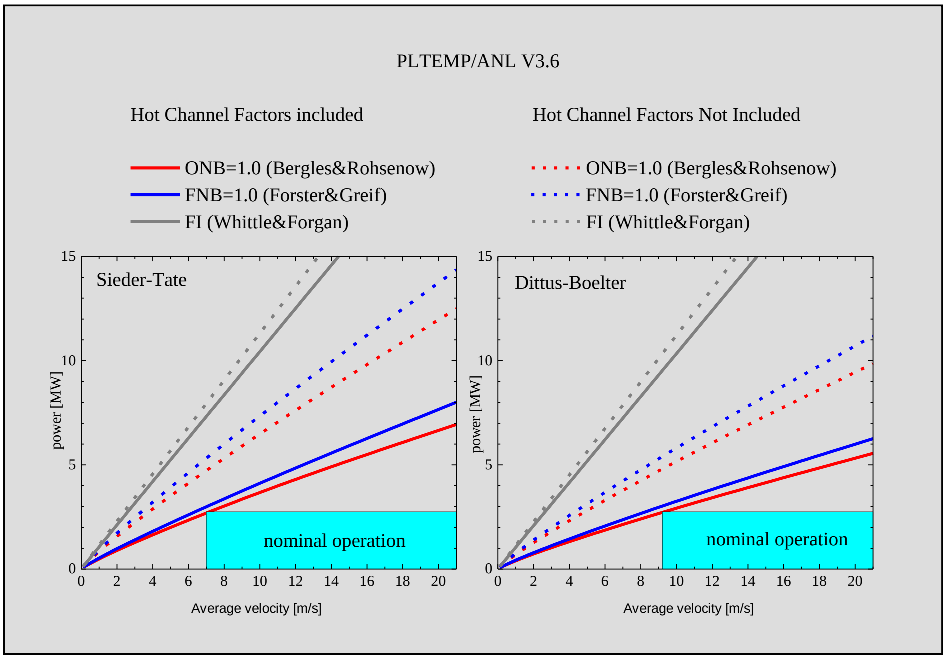

Figure 5. Fuel element power vs flow velocity curves with and without HCF for ONB, FNB and FI in LEU core.

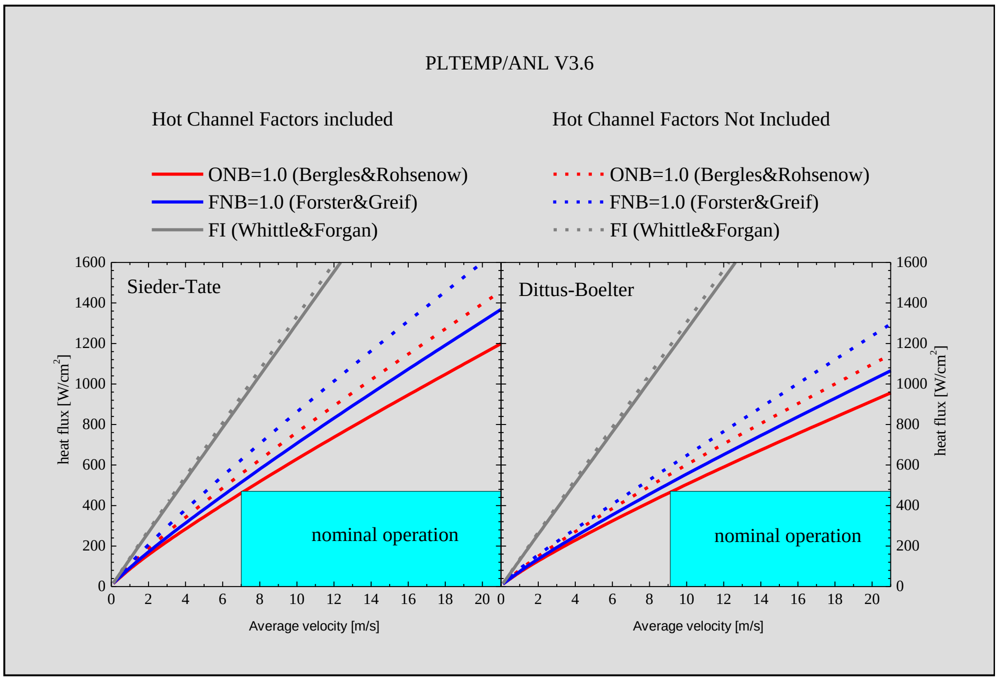

Figure 6. Maximum heat flux vs flow velocity curves with and without HCF for ONB, FNB and FI in LEU core.

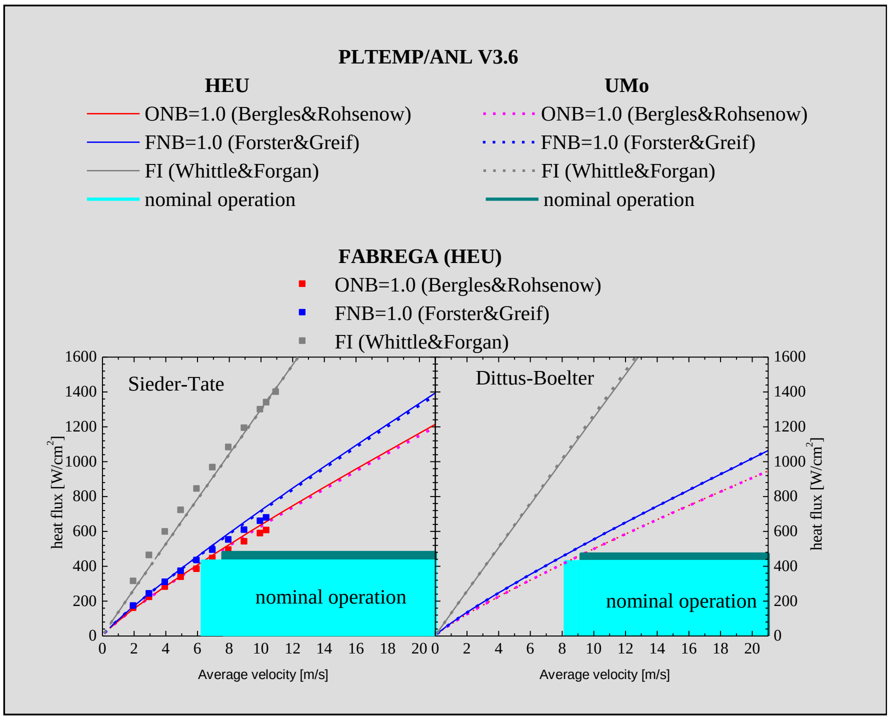

Figure 7. Comparison of HEU (PLTEMP), HEU (FABREGA) and LEU (PLTEMP) maximum heat flux vs flow velocity curves for ONB, FNB and FI with HCFs

The HEU and LEU fuel element can also be compared by the looking at minimum flow velocity allowed at nominal operating conditions, i.e., by reducing the flow velocity at the nominal heat flux until ONB and FNB ratios are equal to one. From Fig. 7, the nominal operating heat flux used the HEU and LEU cores are 432 W/cm^2 (hot element power of 2.57 MW) and 470 W/cm^2 (hot element power of 2.75 MW), respectively.

Table XV. **PLTEMP** results for the maximum heat flux [W/cm^2] / maximum T_{clad} [° C] in a fresh HEU fuel element (P_{FE}=2.57 MW, computed by MCNPX for the HEU core at P_{BR2}=70 MW) and in a fresh LEU fuel element (P_{FE}=2.75 MW, computed by MCNPX for the LEU core at P_{BR2}=70 MW) at nominal initial operation conditions: P_{in}=1.20 MPa, $\Delta P_{fuel\ plates}$=0.21 MPa; T_{in}=40° C.

Heat transfer correlation	ONB or FNB correlation	Clad temperature (ONB or FNB =1) (Allowed minimum velocity)	
		HEU (P_{FE}=2.57 MW)	LEU (P_{FE}=2.75W)
Sieder-Tate	Bergles & Rohsenow	146 (7.0)	155 (7.6)
	Forster & Greif	147 (7.0)	155 (7.6)
Dittus-Boelter	Bergles & Rohsenow	164 (9.1)	176 (9.9)
	Forster & Greif	164 (9.1)	176 (9.9)

The main results for the maximum heat flux, maximum wall temperature and minimum velocity, which can be maintained in the HEU and LEU cores at the same initial operation conditions, are summarized in Table XV. This table shows that ***<u>at the same total BR2 reactor power:</u>***

- The power in the fresh LEU fuel element (2.75 MW, computed by MCNPX) is about 7% higher than the power in the fresh HEU fuel element (2.57 MW, computed by MCNPX).

- The maximum heat flux, computed with PLTEMP in the fresh LEU fuel element is about 9% higher (due to the higher fuel element power) than in the fresh HEU fuel element.

- The maximum wall temperature in the fresh LEU fuel element will be higher (due to the higher fuel element power and heat flux) than in the fresh HEU fuel element.

- The minimum velocity which can be maintained in the LEU fuel element is higher (due to the higher power and heat flux) than in the HEU fuel element.

However, at a same fuel element power to which corresponds different total reactor power in the HEU and LEU cores, the maximum heat flux, and maximum wall temperature and allowed minimum velocity will be similar for the HEU and LEU cores.

5. Conclusions

The objective of this work was to: (1) review the steady state thermal hydraulics analysis of the BR2 reactor presented in the Safety Analysis Report, (2) repeat this analysis (HEU core) with the PLTEMP code, (3) compare the PLTEMP predictions with those of the code of S. Fabrega, (4) perform a steady state thermal hydraulics analysis with the PLTEMP code for the BR2 LEU core, and (5) compare the performance of the LEU core with that of the HEU core.

For the HEU core, with an average core coolant velocity of 10.4 m/s which was used in the evaluations of the safety margins in the SAR by the code of S. Fabrega [3-7], and with HCFs (see Table VII), the following results were obtained:

- The correlation of Bergles & Rohsenow with the heat transfer correlation of Sieder-Tate predicts a hot assembly power of 4.0 MW at ONB, while with the Dittus Boelter correlation predicts 3.2 MW. For FNB the corresponding predictions of the correlation of Forster and Greif are 4.5 MW (Sieder-Tate) and 3.6 MW (Dittus-Boelter).
- The power level at which FI (Whittle & Forgan) would occur is predicted to be 11.4 MW if the correlation of Sieder-Tate is used and 11.3 MW if the correlation of Dittus-Boelter is used.
- The heat flux at ONB (Bergles & Rohsenow) is equal to 667 W/cm^2 if the correlation of Sieder-Tate is used and 528 W/cm^2 if the correlation of Dittus-Boelter is used.
- The heat flux at FNB (Forster & Greif) is equal to 753 W/cm^2 if the correlation of Sieder-Tate is used and 585 W/cm^2 if the correlation of Dittus-Boelter is used.
- The maximum heat flux at which FI (Whittle & Forgan) would occur is equal to 1353 W/cm^2 if the correlation of Sieder-Tate is used , and 1302 W/cm^2 if the correlation of Dittus-Boelter is used.
- The code of S. FABREGA uses the correlation of Sieder-Tate and predicts the following heat flux limits: for ONB (Bergles & Rohsenow) 603 W/cm^2; for FNB 676 W/cm^2 (Forster & Greif) and 1336 W/cm^2 for FI.

For the LEU core, with an average core coolant velocity of 10.4 m/s and HCFs (see Table XII), the following preliminary conclusions have been obtained:

- The correlation of Bergles & Rohsenow with the heat transfer correlation of Sieder-Tate predicts a hot assembly power of 3.9 MW at ONB, while with the Dittus Boelter correlation predicts 3.1 MW. For FNB the corresponding predictions of the correlation of Forster and Greif are 4.4 MW (Sieder-Tate) and 3.5 MW (Dittus-Boelter).
- The power level at which FI (Whittle & Forgan) would occur is predicted to be 11.0 MW if the correlation of Sieder-Tate is used and 10.97 MW if the correlation of Dittus-Boelter is used.
- The heat flux at ONB (Bergles & Rohsenow) is equal to 658 W/cm^2 if the correlation of Sieder-Tate is used and 528 W/cm^2 if the correlation of Dittus-Boelter is used.
- The heat flux at FNB (Forster & Greif) is equal to 744 W/cm^2 if the correlation of Sieder-Tate is used and 585 W/cm^2 if the correlation of Dittus-Boelter is used.
- The maximum heat flux at which FI (Whittle & Forgan) would occur is equal to 1350 W/cm^2 if the correlation of Sieder-Tate is used , and 1317 W/cm^2 if the correlation of Dittus-Boelter is used.

For the HEU core, a comparison of the predicted power and velocity limits by PLTEMP with the predictions of the code of S. Fabrega shows that the Fabrega predictions are lower by~ 8 %.

For HEU fuel, margins to ONB (T_{onb}- T_{sat}, T_{sat} = 183°C) and FNB (T_{fnb}- T_{sat}) were computed with PLTEMP and compared with values computed with the code of Fabrega for a maximum heat flux of 600 W/cm2 [4]. These margins are:

- PLTEMP predicted ΔT_{onb} = 9.2°C vs. ΔT_{onb} =9.5^0 C predicted by FABREGA.
- PLTEMP predicted ΔT_{fnb} = 23.85°C vs. ΔT_{fnb} =24^0 C predicted by FABREGA.

The predictions of PLTEMP are nearly identical to those obtained in the 70's with the code of S.Fabrega.

Table XVI summarizes the safety margins (maximum heat flux and maximum fuel element power at ONB, FNB and FI) predicted with the code of Fabrega [3-7] for the HEU core, and with PLTEMP for the HEU and LEU cores. The values in parenthesis are predictions of ΔT_{onb} and ΔT_{fnb}. This table shows:

- In the HEU core, the predictions of PLTEMP for the maximum heat flux at ONB (667 W/cm2) and at FNB (753 W/cm2) are about 10% higher than those of the Fabrega code (603 W/cm2 and 676 W/cm2, respectively).
- PLTEMP predicts that the maximum fuel element power and the maximum heat flux at ONB, FNB and FI for the LEU core are similar to those for the HEU core.

Table XVI. Maximum heat flux [W/cm^2] and maximum fuel element power [MW] at which ONB, or FNB, or FI are reached, with HCFs and nominal conditions of: V=10.4 m/s; P_{in}=1.20 MPa; $\Delta P_{fuel\ plates}$=0.21 MPa; T_{in}=40° C.

Single phase heat transfer correlation	Correlations for margins	Heat flux / Fuel element power (ΔT_{wall} at ONB or FNB = 1.0)		
		HEU		LEU
		FABREGA	PLTEMP	PLTEMP
Sieder-Tate	Bergles & Rohsenow(ONB)	603 (ΔT_{wall}=9.4°C)	666.7 / 4.00 (ΔT_{wall}=10.2°C)	657.5 / 3.86 (ΔT_{wall}=10.0°C)
	Forster & Greif (FNB)	675.6 (ΔT_{wall}=24°C)	752.6 / 4.52 (ΔT_{wall}=26.8°C)	744.0 / 4.37 (ΔT_{wall}=25.6°C)
	Whittle & Forgan(FI)	1336.5	1353.4 / 11.42	1350.0 / 11.06
Dittus-Boelter	Bergles & Rohsenow(ONB)	–	528.1 / 3.20 (ΔT_{wall}=9.1° C)	527.5 / 3.12 (ΔT_{wall}=9.2°C)
	Forster & Greif (FNB)	–	585.3 / 3.56 (ΔT_{wall}=24.5°	585.0 / 3.46 (ΔT_{wall}=24.6°C)
	Whittle & Forgan (FI)	–	1302.3 / 11.26	1316.7 / 10.97

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