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A HEAT TRANSFER STUDY FOR VERTICAL STRAIGHT-TUBE STEAM GENERATORS HEATED BY LIQUID METAL

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RESUME :

Une maquette monotubulaire de générateur de vapeur à tube droit vertical chauffée par du NaK (alliage sodium-potassium) a fait l'objet d'essais de performances thermiques et hydrauliques dans des conditions représentatives du fonctionnement d'une chaudière nucléaire à neutrons rapides.

[redacted]

A single-tube mockup of a vertical straight-tube steam generator heated by sodium-potassium alloy NaK was submitted to thermal and hydraulic testing in conditions representative of fast breeder reactor operation.
The mockup consisted of a 10mm I.D. ferritic steel heat exchange tube centered inside a cylindrical stainless steel shell. The complete assembly was 20.9 meters long. Water flowed upward inside the exchange tube, and NaK flowed downward in the annular gap between the tube and the shell. The steam outlet pressure ranged from 90 to 195 bars, while the liquid metal temperature at the mockup inlet was between 480 and 580°C. The water flowrate in the tube ranged from 153 to 2460 $kg.m^{-2}.s^{-1}$. During the tests the fluid inlet and outlet temperatures, flowrate and pressures were measured, as was the NaK temperature profile over the full length of the device.
The test results were subsequently compared with heat exchange and pressure drop values calculated using the standard formulas for straight-tube heat exchangers. The heat exchange coefficients predicted by these correlations in the boiling zone were found to be largely overestimated, while the calculated pressure drop values proved satisfactory.
A set of modified correlations is proposed to account for the observed phenomena, and for use in designing commercial units, provided the sodium flow in the tube bundle is adequately distributed.

NOMENCLATURE.

Physical Units

- Cl Laplace's coefficient =
- Cp Specific heat at constant pressure
- D Tube inside diameter
- g Acceleration of gravity
- h Heat exchange coefficient
- L Vaporization heat
- P Water-steam pressure
- Q Mass flowrate
- T Temperature
- V Mean fluid cross-sectional flow velocity
- x Steam mass quality at thermal equilibrium
- λ Thermal conductivity
- μ Dynamic viscosity
- ρ Density
- σ Surface tension

Subscripts and Superscripts

- l Liquid
- v Steam
- w Fluid wall conditions
- ' Saturated liquid
- " Saturated steam

Dimensionless Number

- Re Reynolds number
- Nu Nusselt number
- Pr Prandtl number
- Pe Peclet number

I. INTRODUCTION.

Along with the development of the coiled tube steam generator design adopted for Superphenix ., the CEA also continued design studies for vertical straight-tube steam generators. The thermal dimensioning of a steam generator for a fast breeder reactor depends to a large extent on a good assessment of heat exchange phenomena and pressure drop characteristics on the water-steam side.

Published data on boiling in vertical straight tubes shows widely scattered values. Moreover, the heat source (electric or liquid metal) significantly affects the results, especially with respect to the onset of dryout at the tube walls.

The CEA's Fast Breeder Reactor Department therefore decided to build and experiment at the Grand Quevilly Thermal Test Center a vertical single-tube mockup heated by a sodium-potassium alloy and specially designed for thermal and hydraulic studies of the water-steam flow. The purpose of these tests was to establish the correlations required for accurate thermal dimensioning and to assess the overall and local behavior of the device under partial loading in order to obtain data for the mechanical design of the unit.

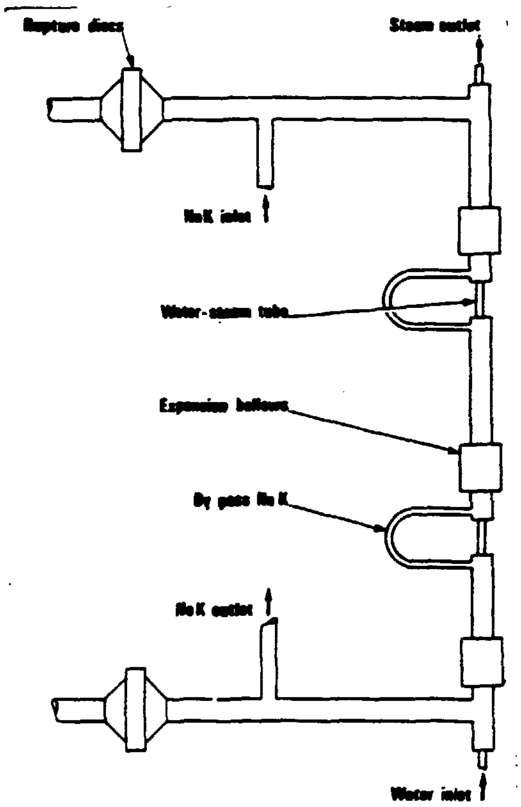

Fig: 1 OVERALL VIEW OF SINGLE STRAIGHT TUBE MOCK UP

2. MOCKUP GEOMETRY AND INSTRUMENTATION.

The steam generator mockup consisted of a 15 mm O.D. x 10 mm I.D. ferritic steel tube in which pressurized water-steam flowed from bottom to top, located at the center of a 42.2 mm O.D. x 32.5 mm I.D. stainless steel shell in which the NaK alloy flowed from top to bottom.

The shell consisted of three stacked cylindrical sections measuring 5.095 m, 9.295 m and 6.010 m long. The base of each section included a bellows unit to compensate for the differential expansion rates of the water tube and the shell (refer to Figure 1).

The NaK alloy flowed from one section to the next through a horizontal U-shaped tube, in order to permit instrumentation of the water-steam tube. In the lower two sections the inner tube was made of Chromesco III steel (2.25% Cr, 1% Mo), while EM 12 steel (9% Cr) was used for the upper section.

The mockup instrumentation was designed to determine the heat exchange and pressure drop laws on the water-steam side. Thimble-mounted thermocouples were located on the water-steam tube at the mockup inlet and outlet as well as between the shell sections, and pressure taps were provided at the same locations. On the NaK side, thermocouples were placed in thimbles at the inlet and outlet of each section, and every 900 mm along the full length of the exchange tube.

The NaK flowrate was measured by an electromagnetic flowmeter. Two calibrated orifice plates for low and high flowrates were inserted in the mockup feedwater line.

The mockup was completely heat-insulated, and thermocouples were placed outside the insulation to estimate thermal losses.

3. TEST DESCRIPTION.

The tests were designed to cover the widest possible range of parameters, including the water-steam pressure, fluid flowrates, heat exchange power, water-NaK temperature differential and heat flux density, while remaining close to the probable operating conditions of a commercial steam generator.

The relative differences between the NaK exchange power and the water exchange power generally did not exceed ± 2 or 3%.

The following parameter ranges were examined :

. Heat exchange power	30 - 450 kW
. Water-steam flowrate	12 - 193 $g.s^{-1}$
. NaK flowrate	0.15 - 2.3 $kg.s^{-1}$
. Water-steam mass flowrate	153 - 2460 $kg.m^{-2}.s^{-1}$
. Steam pressure	90 - 195 bar
. Steam temperature	saturation point - 505°C
. NaK inlet temperature	480 - 580°C

4. TEST RESULTS.

The tests were conducted for the purpose of developing a calculation system capable of predicting the overall thermal and hydraulic performance characteristics of a steam generator and, if possible, the local performance values : i.e. an accurate temperature and pressure map of the entire heat exchange surface.

The code could subsequently be used for the thermal dimensioning of a similar commercial unit, or for low-power operation studies. The results could also be applied to mechanical design studies.

The following data are required for thermal calculations for a single-tube heat exchanger of the type tested :

- the physical properties of the primary and secondary fluids,
- the heat exchanger geometry,
- the physical properties of the heat exchange surface,
- the thermohydraulic parameters (flowrates, pressures, temperatures),
- the relevant fluid-wall heat exchange laws.

The physical properties of the water-steam were taken from international tables [1]. The physical properties of the NaK alloy containing 52% potassium were interpolated from published data on 44% -and 56%- potassium NaK alloys. Simple interpolation was warranted by the virtually linear nature of the physical properties as a function of the composition in the range studied, and by the 6 - 7% maximum difference between the extreme values.

The mockup geometry was described in section 2, above.

The physic :l properties of Chromesco III and EM 12 steels required for thermal calculations are their thermal conductivity. The following formulas were used [2]:

- Chromesco III $\lambda = 39.75 - 0.75 \frac{T}{100}$

- EM 12 $\lambda = 25.05 - 0.825 \frac{T}{100}$

where λ is expressed in $W.m^{-1}\,°C^{-1}$ and T in °C.

The test data relevant to heat exchange correlations were entered in the analysis code. This one-dimensional code calculates the temperature and pressure evolution along the exchange surface for elementary exchange power increments, and the computed values can then be compared with the actual measured results. A set of heat exchange correlations is then determined which minimizes the discrepancy between the theoretical and experimental values.

The exchange phenomena on the liquid metal side is considered to be well established in annular geometry. The correlation published by DUCHATELLE and de NUCHEZE [3] is used :

$Nu = 0.02\ Pe^{0.8} + 6.15$. The problem is thus one of determining the heat exchange law for the water-steam side.

The conventional analysis of a once-through steam generator discriminates between 4 heat exchange zones along the direction of the water-steam flow :

(1) single-phase liquid zone,
(2) nucleate boiling zone,
(3) film-boiling or post-dryout zone,
(4) superheated steam zone.

The limit between zones 1 and 2 is in fact rather vague, since within the liquid zone wall-boiling conditions can be distinguished while the fluid remains subsaturated on the whole.

Similarly, between zones 3 and 4 water droplets can coexist with superheated steam in conditions of thermal disequilibrium.

A sudden transition occurs, on the other hand, between zones 2 and 3, especially in the upward boiling conditions of a vertical straight tube. The transition point is designated the dryout point, critical heat flux (CHF) point, or point of departure from nucleate boiling (DNB). This transition point between two very different heat exchange regions is subject to thermal stresses, including fluctuations due to boiling irregularities, liable to result in damage to the heat exchange surface.

The purpose of the analysis was therefore to determine the heat exchange correlation in each zone that is best suited to the configuration studied, and to ascertain the conditions in which dryout occurs. The water-steam side pressure drop calculations were also compared with the measured values.

The initial calculations were done with an arbitrarily assigned dryout point, determined in most cases by the rather sharp inflection of the temperature profile measured on the liquid metal side. The analysis procedure is thus limited to finding a set of correlations for the water-steam side.

A bibliographical review was carried out to find heat exchange laws determined under conditions similar to those of the experimental study described here. A wide selection is available in the literature, but with highly scattered results : the extreme differences between the heat exchange coefficients calculated under identical conditions substantially exceeds 100%.

The first calculations used the following laws :

In the single-phase flow zones (both for water and for steam), Mikheev's formule [4] was used :

$$Nu = 0.021\ Re^{0.8}\ Pr^{0.43} \left(\frac{Pr}{Pr_w}\right)^{0.25}.$$

The Reynolds and Prandtl numbers were adopted for the mean fluid conditions except for Pr_w, which was calculated at the temperature of the wall in contact with the fluid.

In the nucleate boiling zone Rosenhow's formulas [5] were used in conjunction with the formula $Nu = 0.19\ Re^{0.8}\ Pr^{1/3}$ for single-phase conditions.

$$h = hc + hb$$

$$hc = \frac{Nu\,\lambda}{d}$$

$$hb = \frac{\mu'}{C1} \left(\frac{C_p'}{0.013\ Pr^{1.7}}\right)^3 \left(\frac{T_w - Tsat}{T_w - T}\right)^3$$

In the film boiling zone, Mikheev's formula was corrected by Miropolsky's law [6] :

$$Nu = Nu_{Mikheev} \left(1 - 0.1\left(\frac{\rho'}{\rho''} - 1\right)^{0.4} (1-x)^{0.4}\right)\left(x + \frac{\rho''}{\rho'}(1-x)\right)^{0.8}$$

With these laws it was shown that the heat exchange was on the whole largely overestimated in the nucleate boiling and film boiling zones with a number of differences depending on the pressure values.

The other available correlations for the nucleate boiling zone gave values higher than Rosenhow's. In the film boiling zone the available correlations gave highly scattered and thus unreliable results. They were obtained on the basis of imposed flux tests representative of light water reactor core conditions.

It was therefore decided to retain the previously used correlations and to apply a suitably determined correction factor.

A similar decision was made for the liquid zone, where the heat exchange was also overestimated. In fact, in the liquid zone where the steam quality is negative (i.e. where the mean fluid enthalpy is less than the saturation enthalpy) two regions can be distinguished. The first is the single-phase region, and the second is where boiling appears at the exchange wall when the surface temperature exceeds the saturation temperature while the water flow core temperature remains subsaturated. The heat exchange in this zone was calculated using Rosenhow's method.

In order to avoid a flow discontinuity that has no physical existence in the portion surrounding the transition from film boiling to superheated steam, the correction factor was modified in a linear manner according to the steam quality between x = 1 and w = 1.3. This variation also expresses the fact that the flow is in thermal disequilibrium even beyond the point x = 1 (where the mean fluid enthalpy is equal to the saturated steam enthalpy).

The following correction factors were adopted :

. liquid zone	0.7
. nucleate boiling zone	0.25
. film boiling zone	0.4
. superheated zone	1.1

These values were adjusted to minimize the discrepancy between the measured and calculated NaK temperature, at specified fluid flowrates and inlet temperature. The measured and calculated results can also be compared for the following parameters : intermediate water temperatures, steam outlet temperature, NaK outlet temperature.

The pressure drop values were calculated according to the following method. The pressure drop can be broken down into three terms : acceleration, gravity and friction. Each of these terms was computed for each mesh. Two-phase pressure drop values were computed using Thom's method [7]. The friction coefficient was evaluated by means of Colebrook's formula with a surface roughness of 2 μm.

4.1. Analysis of test results.
The calculations were carried out using the correction factors indicated above. The tests were classified according to the steam outlet pressure, and four pressures were studied : 195 bars, 140 bars, 125 bars and 90 bars. No low-power tests were carried out at 195 bars.

The lower pressure tests were intended to assess the steam generator behavior under conditions approximating startup, without anticipating on the stability criteria that will be adopted for power plants.

A pressure of 90 bars would no doubt be too low for stable operation. The correction factors were therefore adjusted with greater weight on the high-pressure tests and especially for conditions approximating the planned rating, i.e. 195 bars and a high exchange power value of 200 - 300 kW per tube.

Table 1 indicates the significant parameter values (flowrate, pressure, flux and steam gravity) for dryout conditions.

The diagrams show the calculated values with solid or broken lines, and the measured values with points.

On the flux curve a discontinuity can be seen at the beginning of the boiling zone. This does not correspond to any physical reality, but rather to the large difference between the correction factors for the liquid and nucleate boiling zones. This discontinuity had only a slight effect on the analysis results.

4.2. Notes on the 195 bar tests.
On the whole the calculations very satisfactorily corresponded to the actual results, but the discrepancy increased as the flowrate diminished. The pressure drop values appear to be slightly underestimated. The dryout steam quality was approximately 0.25 under the test conditions, but this value had little effect on the calculations (Table 1). A typical calculation is shown in Figure 2.

4.3. Notes on the 140 bar tests.
These tests were also suitably represented by the calculations. The calculated pressure drop values were underestimated by 10 - 20%. The critical steam quality was between 0.3 and 0.45, and decreased as the exchange power rose from 125 to 440 kW and the flux from 50 to 100 $W.cm^{-2}$ (Table 1). As in the 195-bar test series, the water and NaK flowrates were well balanced and the boiling zone was limited to the mockup intermediate section. A typical calculation is shown in Figure 3.

4.4. Notes on the 125 bar tests.
Here again the conclusions are practically the same. The mockup heat exchange was well accounted for in the high flowrate and high

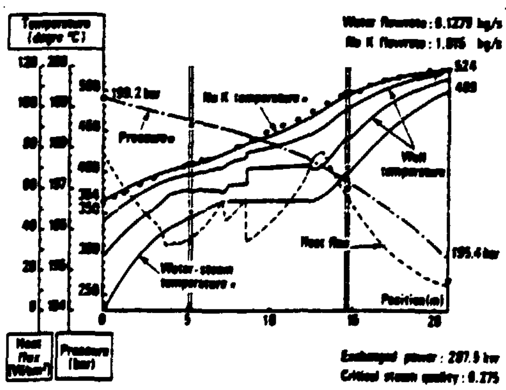

Fig: 2 TEST 201

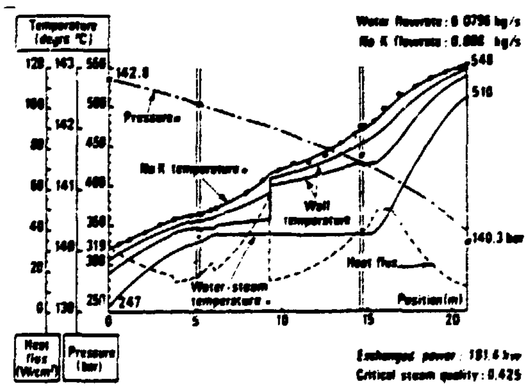

Fig: 3 TEST 210

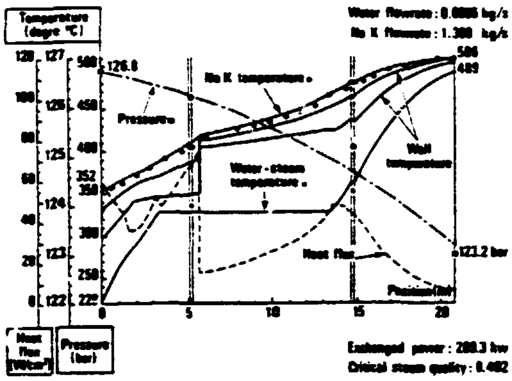

Fig: 4 TEST 206

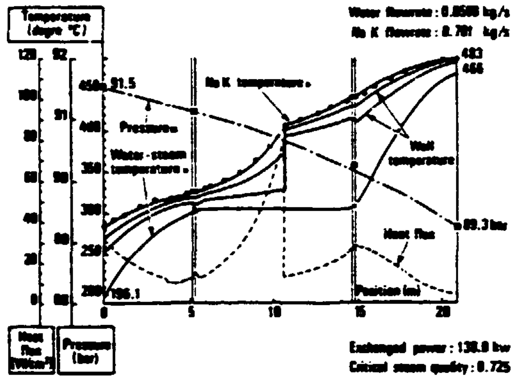

Fig: 5 TEST 182

power (130-250 kW) tests. The critical steam quality was approximately 0.45 for the medium and high power tests (Table 1). The computed and experimental pressure drop values were identical within measurement tolerances. A typical calculation is shown in Figure 4.

4.5. Notes on the 90 bar tests.
The power ranged from 33 to 320 kW for these tests. The results of this series showed the greatest discrepancies between calculated and measured values, especially for the NaK temperature profile. As for the other series, the calculated results were relatively close to the measured results at high power, but were increasingly different at lower power ratings.

The critical steam quality ranged from 0.45 to 0.9 according to the power and the flux between 80 and 120 $W.cm^{-2}$ (Table 1).

A closer analysis of the dryout conditions in these tests shows that for high flowrates dryout occurred at $x = 0.46$ with a flux of 112 $W.cm^{-2}$. Another group of tests at high flowrates (72-110 $g.s^{-1}$) with critical flux values of about 90 $W.cm^{-2}$ showed a critical steam quality of approximately 0.55. Still another group of tests indicated critical steam qualities between 0.65 and 0.9 for flowrates of 34-75 $g.s^{-1}$ and flux values of 80-103 $W.cm^{-2}$.

In fact, the following trends were noted. Regardless of the flowrate, dryout occurred at very high flux values (at least 80 $W.cm^{-2}$) and, paradoxically when compared with the imposed-flux test results, at steam qualities inversely proportional to the flowrate. This was attenuated by the fact that as the flowrate increased the dryout flux also increased, and thus naturally tended to reduce the critical steam quality value.

The conclusions for low flowrates are based on an inadequate number of interpretable test results, and would require experimental confirmation if such operating conditions were adopted.

A typical calculation is shown in Figure 5.

4.6. General remarks.
For all of the tests, the experimental dryout conditions were compared with the results given by Duchatelle's formula, previously used by the CEA, which gives the dryout steam quality as a function of the water mass flowrate, the wall heat exchange flux and the water-steam pressure :

$$x = 0.4166\ \phi^{0.3236}\ G^{-0.6098}\ e^{-0.0011098\ P}$$

with ϕ in $W.m^{-2}$

G in $kg.m^{-2}\ .\ s^{-1}$

P in bars.

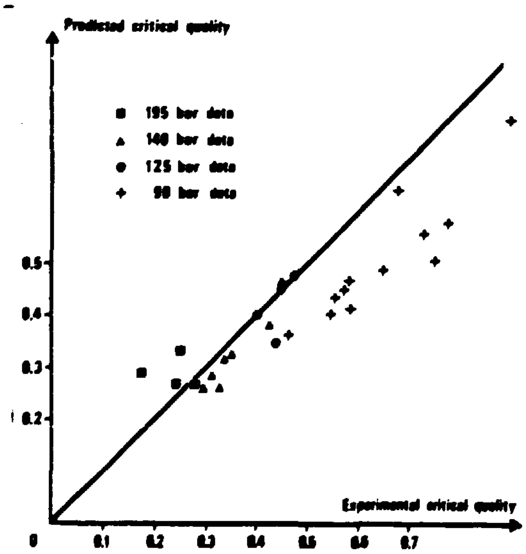

Fig: 6 COMPARISON OF CRITICAL STEAM QUALITY PREDICTIONS TO OUR EXPERIMENTAL DATA

TABLE 1
DRYOUT CONDITIONS

Test N°	Flowrate ($g.s^{-1}$)	Pressure (bars)	Flux ($W.cm^{-2}$)	Critical steam quality	Critical steam quality from Duchatelle's formula	Remarks
101	83	93	87	0.570	0.450	
102	56	91	80	0.725	0.558	
103	14	89	86	1	1.334	Unreliable critical quality value
104	59	126	60	0.477	0.474	
105	57	126	50	0.450	0.456	
201	126	198	55	0.275	0.266	) Calculated NaK temperature
202	110	198	43	0.237	0.269	) is slightly low in dryout
203	85	197	33	0.175	0.289	) zone
204	62	196	28	0.250	0.333	)
205	111	129	76	0.440	0.347	
206	89	126	75	0.402	0.397	
207	12	125	56	0.800	1.226	Unreliable critical quality value.
208	55	141	50	0.450	0.459	
209	55	141	50	0.450	0.459	
210	80	142	58	0.425	0.383	
211	115	144	69	0.330	0.324	
212	117	144	68	0.337	0.319	
213	155	147	83	0.312	0.295	
214	194	151	98	0.299	0.262	
215	194	151	100	0.323	0.263	
216	109	97	103	0.543	0.401	
217	86	94	84	0.553	0.435	
218	133	99	112	0.465	0.364	
301	34	90	86	0.900	0.775	
302	41	90	68	0.676	0.641	Calculated NaK temperatre is too low.
303	32	90	89	0.950	0.814	Large discrepancy for intermediate temperatures
304	72	91	86	0.645	0.490	
305	54	92	83	0.775	0.577	
306	75	94	103	0.752	0.505	
307	94	94	85	0.582	0.414	
308	79	94	91	0.581	0.470	
309	18	90	75	0.752	1.093	Unreliable critical quality value

The results are shown in Table 1, which indicates good agreement under nominal conditions and, more generally, for the tests at 195 bars, 140 bars and 125 bars, with differences in the quality value not exceeding 0.05. The discrepancy appears to be greater for two tests at 195 bars, but the uncertainty on the steam quality determined from the measurement results was very great for these two tests since there was no inflection of the NaK temperature curve. This implies that the thermal calculation is, in fact, not sensitive to the critical quality value (Figure 6).

The proposed formula for calculating the critical steam quality thus appears to be well suited to steam generators with 10 mm I.D. vertical straight tubes at pressures above 125 bars, provided dryout occurs under conditions similar to the test conditions.

5. CONCLUSION.

It has been shown that the heat exchange correlations generally used in the boiling zone are largely overestimated and must be corrected by factors of 0.25 for the nucleate boiling zone in order to accurately account for the results measured on the test mockup. The critical steam quality is about 0.25 under nominal-conditions. Duchatelle's critical steam quality correlation as used previously is suitable for the heat exchange calculations. The pressure drop values were also well accounted for by the analysis.

The vertical single-tube mockup tests thus resulted in the development of an analysis method for designing vertical straight-tube steam generators.

REFERENCES.

[1] SCHMIDT E., Properties of water and steam in SI-units.
Springer-Verlag Berlin 1979

[2] CEA internal report.

[3] DUCHATELLE L., DE NUCHEZE L.
Détermination des coefficients de convection d'un alliage sodium potassium circulant à contre courant dans un échangeur monotubulaire.
Entropie n° 17, pp. 51 - 58, 1967.

[4] MIKHEEV M.A.
Convection forcée en écoulement longitudinal.
Izvestia Akad. Vank Tome 10 (1952) pp. 1448 - 1454.

[5] ROSENHOW W.M., CLARK J.A.
Heat transfer and fluid mechanics Inst., pp. 193 - 207, Stanford University Press 1951.

[6] MIROPOLSKY Z.L.
Heat transfer in film boiling of a steam water mixture in steam generating tubes.
Teploenergetka pp. 49 - 52, vol. 10, n° 5 (1963).

[7] THOM J.R.S.
Prediction of pressure drop during forced circulation boiling of water.
Int. J. Heat Mass Transfer, 7, pp. 709 - 724 (1964).