VALIDATION AND VERIFICATION OF THE MTR_PC THERMOHYDRAULIC PACKAGE

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ABSTRACT

The MTR_PC v2.6 is a computational package developed for research reactor design and calculation. It covers three of the main aspects of a research reactor: neutronic, shielding and thermohydraulic. In this work only the thermohydraulic package will be covered, dealing with verification and validation aspects. The package consists of the following steady state programs: CAUDVAP 2.60 for the hydraulic calculus, estimates the velocity distribution through different parallel channels connected to a common inlet and outlet common plenum. TERMIC 1H v3.0, used for the thermal design of research reactors, provides information about heat flux for a given maximum wall temperature, onset of nucleate boiling, redistribution phenomena and departure from nucleate boiling. CONVEC V3.0 allows natural convection calculations, giving information on heat fluxes for onset of nucleate boiling, pulsed boiling and burn-out phenomena as well as total coolant flow. Results have been validated against experimental values and verified against theoretical and computational programmes results, showing a good agreement.

INTRODUCTION

Thermohydraulic design of research reactors is performed using the MTR_PC v2.6 – Thermohydraulic Package, which includes CAUDVAP 2.60, TERMIC 1H v3.0 and CONVEC V3.0 programs, [1]. In order to determine the applicability and sensitivity of these programs they were validated and verified comparing their results against those ones obtained with commercial and customized codes, theoretical values and experimental data.

The ***validation*** of a code is the comparison of the code results with experimental data while the ***verification*** is the same comparison against theoretical or other programs results, in order to determine code's applicability and sensitivity. A wide variety of cases has been calculated and comparisons were performed against:

- RETRAN results [2],
- Theoretical data from the IAEA TECDOC - 233, [3], values and
- Experimental data

Concerning the geometric and general data required for the analysed cases, all of them are included in Table 1. It is worth noticing that the present are ***"best-estimate"*** calculations, it means, no uncertainties are taken into account.

Geometric data	
Coolant channel width ***(W)***	3.8 / 7.1 cm
Coolant channel thickness ***(t)***	2.0 / 5.8 mm
Fuel plate width ***(w)***	3.5 / 6.9 cm
Fuel plate length ***(L)***	59.2 / 80. cm
Total plate length	62.35 /118 cm
Number of fuel plates	19 /23
General data	
Coolant Inlet temperature ***(Ti)***	17 / 70 °C
Pressure at channel exit ***(Po)***	1.566 / 1.86 bar
Coolant	water /air
Coolant velocity ***(v)***	2.0 / 6.0 m/s
Flow direction	Upward / Downward flow
Maximum heat flux ***(q")***	2. / 7.5 w/cm^2
Power distribution	Cosine shape / Uniform

Table 1: Main Data

CAUDVAP PROGRAM

CAUDVAP mode 2.60 was developed for the hydraulic design of research reactor cores. It calculates the velocity distribution in the steady state through different coolant channels connected, in a parallel array, between an inlet and an outlet common plenums. No power generation is supplied and it will be demonstrated that this consideration results in conservative velocity variations.

Given different channels geometry, coolant pressure, average temperature, flow direction, water column above core, coolant properties and uncertainty factors.

CAUDVAP gives:

- Either total pressure drop across the core for a given flow or total core flow for a given core pressure drop.
- Flow distribution in different channels
- Friction and form losses in each section

RESULTS COMPARISON

Comparison with RETRAN Results

RETRAN-02, is a one-dimensional, homogeneous equilibrium mixture thermal-hydraulic model for the reactor cooling system and it has become the industry standard with respect to nuclear power plant transient analysis. The following cases have been carried out:

- Hydraulic calculations at two different coolant temperatures; 30°C and 40°C, giving a percentage deviation within **-4.5%** in coolant velocity and , **–1%** in core pressure drop (***Dp***). Percentage deviation was calculated as (value$_{\text{CAUDVAP}}$ / value$_{\text{RETRAN}}$ x 100) - 100

- For this case, in which power is supplied (0.76 MW per fuel element), an increase in RETRAN velocity calculations was observed. This increase makes CAUDVAP results to be conservative as velocity remains the same and is lower. Percentage deviation is within **-4.4%** for both cases, with and without power.
- Another interesting aspect is that, while RETRAN uses cylindrical geometry for channel calculations, CAUDVAP presents two different correlations for both cylindrical and rectangular geometry. Percentage deviations for Dp are **12.6%** and **9.7%** for the case of rectangular and cylindrical geometries, respectively, while deviations reach **–5.2%** and **3.5%** for the same geometries.

Comparison against Theoretical Values

In this paragraph, CAUDVAP results are compared against theoretical values presented by the Argonne National Laboratory (ANL), for a generic MTR-type reactor of 10 MW, [3]. Values have been compared for a variable coolant velocity and different channel thicknesses resulting in percentage deviation for Dp between **– 4%** and **–10.5%,** probably due to the incomplete knowledge of geometry and mean temperatures.

Comparison with Experimental Results

Calculated results and experimental values obtained from the NUR Research Reactor, [4], are compared. It is an MTR type reactor with a variable number of fuel elements and comparison resulted in a maximum percentage deviation, in Dp, below **10.5%.**

TERMIC PROGRAM

TERMIC 1H v3.0 allows thermohydraulic design of reactor cores being where the main features are:

- Coolant in the single-phase state, in the low pressure range, either, light or heavy water, flowing in the upwards or downwards direction
- Heat correlations cover laminar, transition and turbulent regimes
- A statistical model accounts for engineering and manufacturing factors.

Given:

- A single channel geometry
- Coolant channel velocity range and inlet temperature
- Power shape/distribution along the fuel element
- Flow direction, water column above core and uncertainty factors.

It can be obtained:

- Heat flux for a user-defined maximum wall temperature, Onset of Nucleate Boiling, Redistribution Phenomena and Departure from Nucleate Boiling
- Axial distribution along the hot channel of coolant and wall temperatures and pressure.

RESULTS COMPARISON

RETRAN Comparison

To compare the code sensitivity a ***"reference case (4)"*** was defined and geometric data, as well as maximum heat flux (*q"*) and coolant velocity (*v*) have been varied, alternatively. **Cases 1 to 3** imply **v** variation, **cases 5 to 8**, **q"** variation, in **cases 9 and 10, L** was varied, in **cases 11 to**

13, w, and consequently W, were varied while **cases 14 to 16** imply **t** variation. A summary of percentage deviation is presented in Figure 1.

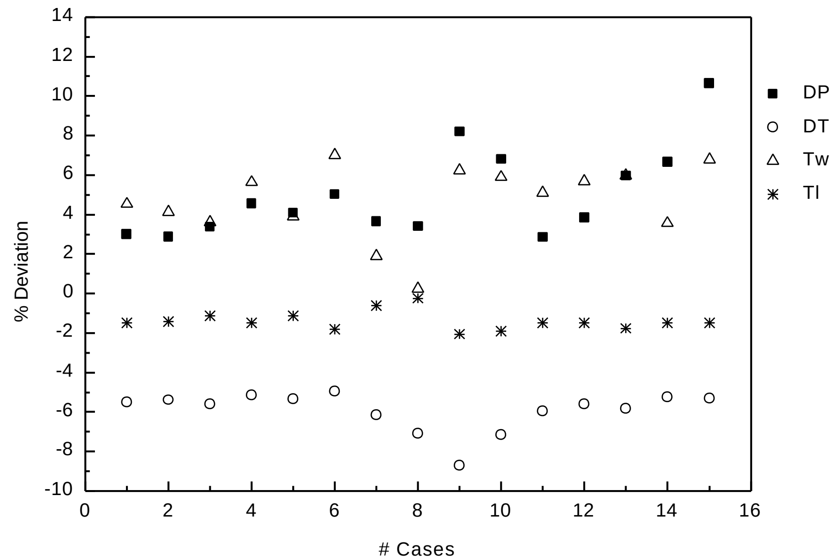


Figure 1: Core Pressure Drop Deviation between RETRAN and TERMIC Results.

Theoretical Comparison

TERMIC results are compared against theoretical values presented by the Argonne National Laboratory (ANL), [3], for different coolant velocities (Cases 1 to 4) and channel thickness (Cases 5 to 9). Figure 2 summarises percentage deviation for critical heat fluxes.

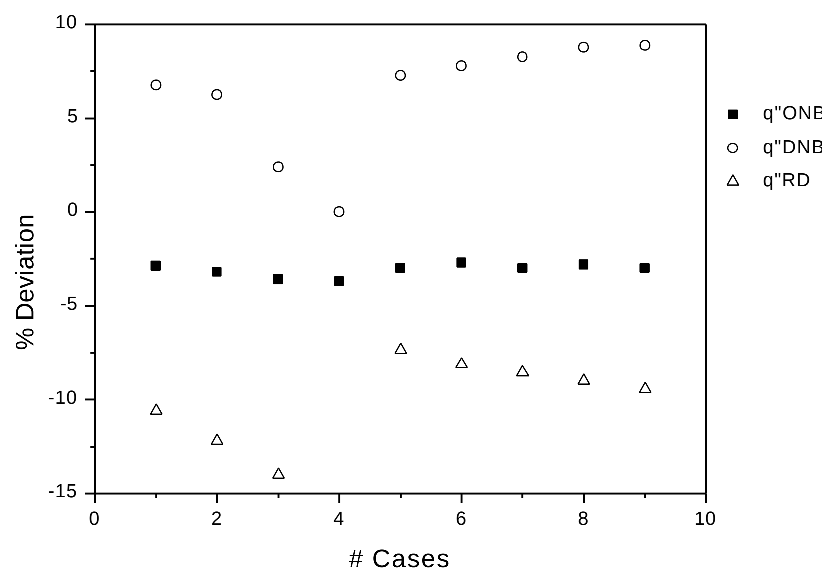


Figure 2: TERMIC Percentage Deviation of ONB, DNB and Redistribution Heat Fluxes.

These deviations may be explained due to:

- The way in which water properties are calculated for the Onset of Nuclaeate Boiling heat flux calculation and how the pressure is specified. Theoretical comparison gives a unique value for "pressure at channel exit", and in case of downward flow, outlet pressure would vary for varying velocities and/or channel thickness.
- Concerning the Redistribution phenomena, critical heat flux was estimated using different correlations besides the extra safety factor of 0.9, imposed in TERMIC calculations.

In both cases, q''_{ONB} and q''_{RD} calculated from TERMIC, result to be conservative when compared with ANL values.

Experimental Results

Calculated Redistribution heat flux was compared against experimental results from [5]. For a given coolant channel geometry, the coolant inlet temperature was varied as well as coolant velocity inside the channel. Results are presented in Figure 3.

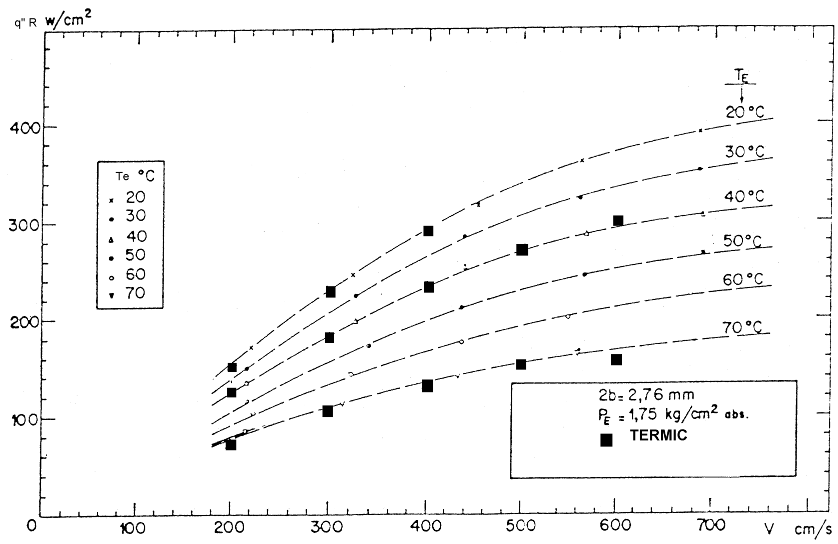


Figure 3: Redistribution Heat Flux Comparison for Coolant Velocities and Inlet Temperatures.

CONVEC PROGRAM

CONVEC v3.0 program is used in the thermohydraulic analysis of research reactors where natural circulation cooling is concerned and its main features are:

- Heat transfer correlations valid for laminar regime under forced and natural convection modes.
- Effect on the core behaviour of a chimney placed above the core as well as the inlet pipes connecting the flap valve and the core.

Given:

- Fuel element and natural convection loop geometries, including the chimney.
- Power generated in each fuel element.

It can be obtained:

- Heat flux for Burn-Out, Pulsed Boiling and Onset of Nucleate Boiling

- Axial distribution along the hot channel of coolant and wall temperatures
- Total coolant flow and average and maximum coolant velocity inside the core.

RESULTS COMPARISON

RETRAN Comparison

Figure 4 shows percentage deviation, calculated as [(values$_{CONVEC}$/values$_{RETRAN}$)x100]-100, for total core flowrate and coolant and wall temperatures as a function of maximum heat flux.

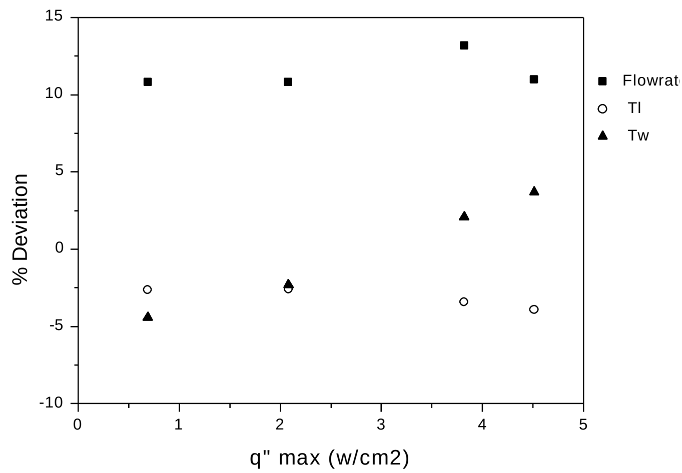
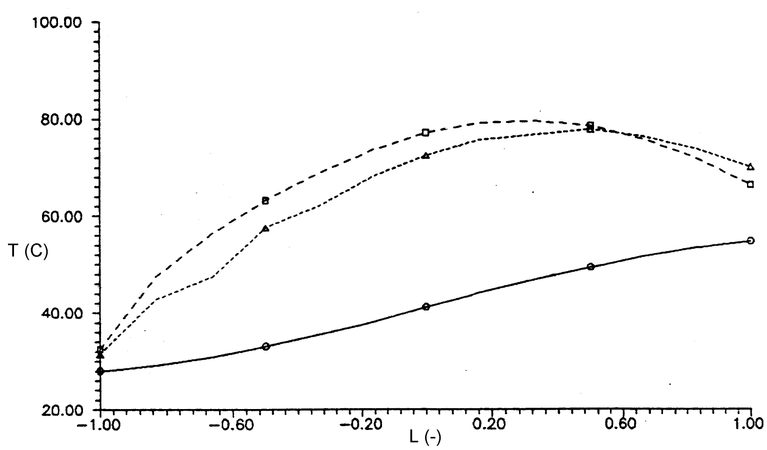


Figure 4: Flowrate and Temperatures Percentage Deviation

SPCONVEC Program

The SPCONVEC [6], is a finite element program developed for forced and natural convection regimes for incompressible flow inside cooling channels, in the transient state. It has been tested against analytical problems having known solution with accurate results, [7]. Comparison was done for a typical MTR fuel element and results are shown in Figure 5.



o - Liquid – CONVEC /   Wall – CONVEC / Δ Wall SPCONVEC

Figure 5: SPCONVEC Results Comparison (L: normalized length)

Theoretical Comparison

CONVEC results have been compared against theoretical values presented by Dalbert et al. [8]. In this case air flow, in a vertical channel in the laminar natural convection regime, is studied, for Grashof numbers between 10^{-1} and 3 x 10^{3}. Percentage deviation, in wall temperature for several heat fluxes between 0.02 w/cm^2 and 0.06 w/cm^2, does not exceed **±10%.**

RA-3 Experimental Results

Measurements at two different powers (200 and 400 KW), were performed in the RA-3 reactor in order to determine, among other things, the coolant temperature leaving the core, [9]. Percentage deviation in coolant temperatures are within **6%.**

OSIRIS Experimental Loop

Last but not least, CONVEC results have been compared with those from experimental measurements in the test loop E50CN, [10]. Here, one of the main features of CONVEC program is validated, the chimney above the core and the way it enhances natural convection. Three different chimney heights have been compared, 1, 2 and 3m height, for a range of maximum heat fluxes. Percentage deviations are presented in Figure 6.

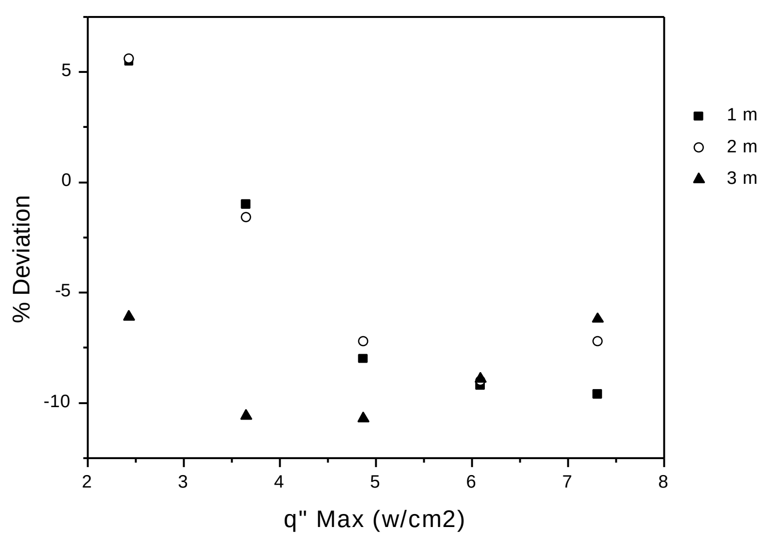


Figure 6: Percentage Deviation from OSIRIS Test Loop.

CONCLUSIONS

A good number of cases have been simulated using the Thermohydraulic Package of the MTR_PC v2.6 and results have been compared against commercial and customized programs results, theoretical values and experimental data. Almost in every case, percentage deviations are within 10%, only few cases in CAUDVAP and TERMIC comparisons present differences larger than 10% when compared with RETRAN or ANL values, however, results are conservative.

Referring to CONVEC program, the exceptional cases in which this value is exceeded are due to differences in geometry and it is worth remembering that natural convection is highly dependent on friction, i.e. geometry.

Considering these results comparison and the simple models adopted it can be concluded that the MTR_PC v2.6 – Thermohydraulic Package is an appropriate and simple tool for hydraulic and thermal design purposes of research reactor cores both, in forced and natural convection regimes.

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