TRACE Code Application to Low Operating Pressure Research Reactor Safety Analysis

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Abstract. The TRACE code developed by the U.S. NRC is a best estimate thermal hydraulic system code for nuclear power plant (NPP) safety analysis. The need for a standard methodology for the safety analysis of research reactors encourages the transfer of nuclear power plant safety technologies to research reactors. A fundamental difference between many research reactors and power reactors is in the operating pressures and flows. Hence, a thorough investigation on the potential application of TRACE code to safety analysis of McMaster University Research Reactor has been performed. This work is structured in two parts: the first one dealing with the assessment of TRACE code against subcooled boiling experiments at low pressures, and the second one with the IAEA 10 MW reactor benchmark.

As a first stage validation and code-to-code comparison exercise is performed using TRACE V5.0P3 and RELAP5/M3.3 for the reproduction of subcooled flow boiling experiments in a vertical annulus. The numerical predictions from these two codes were compared with experimental void fraction profiles. As a second stage in this assessment process, TRACE was used to model the idealized IAEA 10 MW reactor and to run a set of benchmark transients specified in the IAEA research reactor core conversion guidebook (IAEA-TECDOC-643). TRACE's predictions were later compared to those obtained by six other codes. Results concerning the boiling experiments indicate a potential bias when TRACE is used to predict void fraction at low-flow and low-pressure conditions. Comparison of RELAP5 to the same experiments revealed better performance. The IAEA safety-related benchmark also indicated potential issues with computations of two-phase flow at low-pressure conditions, in particular in some of the constitutive relationships. Also, in both investigations, RELAP5 showed better performances than TRACE for problems involving two-phase flow at low-pressure conditions, mainly as a result of the extended range of the constitutive relations available in RELAP.

1. Introduction

A large variety of research reactors have been designed and operated during the last 50 years. These reactors are primarily designed for research purposes, yet they are widely applied for education and training, materials testing and isotope production. Due to the diversity of research reactor designs and operating conditions, there is a wide variety of computational tools used in the safety analysis and nowadays, it is desired to adopt a standard approach for safety analysis of these research reactors [1]. The development of high power research reactors and small modular reactors, together with the extended and intensive utilization of research reactors and the increased safety requirements of nuclear installations after the Fukushima accident [2] encourages the adoption of nuclear power plant (NPP) technology and methods to research reactor safety analysis. However, the use of this technology for research reactors is not straightforward as there are important differences in operating pressure, coolant flow, size and power.

A number of codes used in the nuclear power plant industry fall into this category but each needs to be validated for the range of parameters that characterize the transient analyses in research reactors. TRACE (TRAC/RELAP Advanced Computational Engine) is a best-estimate thermal-hydraulic system code developed by the U.S. NRC (United States Nuclear Regulatory Commission) for the steady state and transient behavior of light water reactors. TRACE has been developed for NPP safety analysis and it has been so far validated for a wide range of accident conditions using experiments, nuclear power plant data or code-to-code comparison [3-8]. However, it is still in developmental stage and still requires additional detailed verification and validation (V&V) activities. Since for research reactors the range of operational conditions are significantly different from those for NPPs, the present paper focuses on the TRACE code performance under low pressure and low flow conditions, which are typically found in the majority of research reactors.

This paper is structured in two main parts, a first one dealing with TRACE code assessment against low-pressure subcooled boiling experiments, and a second one with TRACE results for the IAEA 10MW reactor benchmark.

2. TRACE assessment against low-pressure void fraction experiments

Predicting the reactor core void fraction levels with reasonable accuracy is very important due to its effects on reactor kinetics, two-phase friction losses, and phenomena defining the acceptance criteria of research reactors such as onset of significant void (OSV) and onset of flow instability (OFI). It is well known that at near atmospheric pressures, the rate of change of void fraction with quality is far more significant than at high pressures due to the high-density ratio between water and vapor; thus, models and correlations developed (and validated) for high pressures may not be valid at low pressures. This section presents the results of the TRACE code assessment against experimental low-pressure void fraction data obtained by Zeitoun and Shoukri in 1997 [9].

The experimental setup consisted of a *holding tank* in which the water temperature is controlled by an immersed electric heater and a cooling coil, a *circulating pump*, a *pre-heater* and a *test section*. The measurements were carried out in a vertical, concentric annular test section as sketched in Fig. 1. The outer tube was a 25.4 mm inner diameter (D_o) plexi-glass tube that allowed visual observation. The inner tube, which had an outside diameter (D_i) of 12.7 mm, was made of three axial sections. The middle section of the inner tube was a 306 mm long, thin-walled stainless steel tube (0.25 mm thickness), which was electrically heated. This heated section was preceded and followed by 340 and 500 mm long thick-walled copper tubes, respectively. The entire inner tube was electrically heated resulting in a uniform heat generation in the middle section where the subcooled boiling takes place, and relatively no heat generation in the upstream and downstream copper tubes as a result of the low resistance in these sections. The vapor formed in the boiling section was condensed in the adiabatic downstream section. The experimental setup allowed control of the inlet mass flux (G), wall heat flux (q'') and the inlet subcooling (θ_{in}). The axial void fraction (area-averaged) profile in the boiling section of the annulus was measured using a gamma attenuation technique by a single beam gamma densitometer. For the range of $0.02 < \alpha < 0.3$, the average uncertainty was within ±4% of the calibrated range. The maximum absolute error was estimated to be 0.015. A more detailed description of the experimental setup can be obtained from the authors' works [9,10].

In addition to a TRACE V5.0P3 model, a RELAP5/MOD3.3 nodalization of the experimental setup has been developed. Figure 1 shows the test section schematic and the TRACE and RELAP input nodalizations. Only the heated section has been modeled here since this is where subcooled boiling takes place. The boundary conditions, geometries and heat structure inputs for both codes were developed to match the experimental conditions. More details on the TRACE and RELAP5 models can be found in Ref. [11].

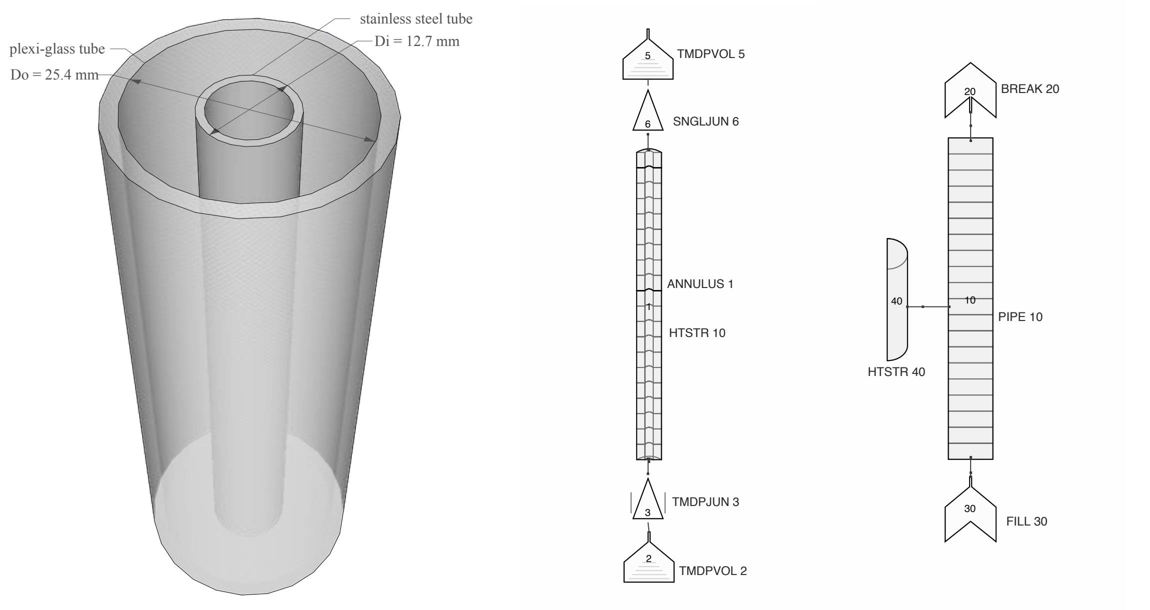

FIG. 1. Sketch of the heated section (left), and nodalizations for RELAP5 (center) and TRACE (right)

2.1 Results

The void fraction profiles predicted by TRACE V5.0P3 and RELAP5/MOD3.3 were compared against the experimental results. From the available experimental data, only six cases have been selected and simulated with the two codes. These cases cover approximately the ranges of operating pressures of a typical pool-type research reactor, with relatively low mass fluxes (G = 200-400 kg/m^2 s). Results for four of these simulations are shown in Fig. 2 and 3. Plots have been generated using equilibrium quality $x_e=(h_l-h_{l,sat})/(h_{v,sat}-h_{l,sat})$ as the independent variable. The measured quantity in the experiment is the cross-sectional average void fraction, and it is defined as

$$\langle \alpha \rangle = \frac{A_g}{A_g - A_l} \tag{1}$$

Where A_g and A_l are the cross sectional areas of the gas and liquid respectively. Both equilibrium quality (x_e) and void fraction (α) are dimensionless quantities.

All TRACE simulations indicate that void fraction is being overestimated for each flow condition and quality. The magnitude of this overestimation becomes important for (actual) void fractions above 0.1, where the error falls outside the ±0.1 band as shown in Fig. 4. On the other hand, RELAP5 predictions are, overall, in better agreement with the experimental data. Previous studies also showed that RELAP5 void fraction predictions at low pressures are reasonable [12]. It should also be noted that under the highest heat flux and mass flux, while the absolute value of the void is predicted within 0.1 void fraction, there is a qualitiative difference in the trend predicted by RELAP5.

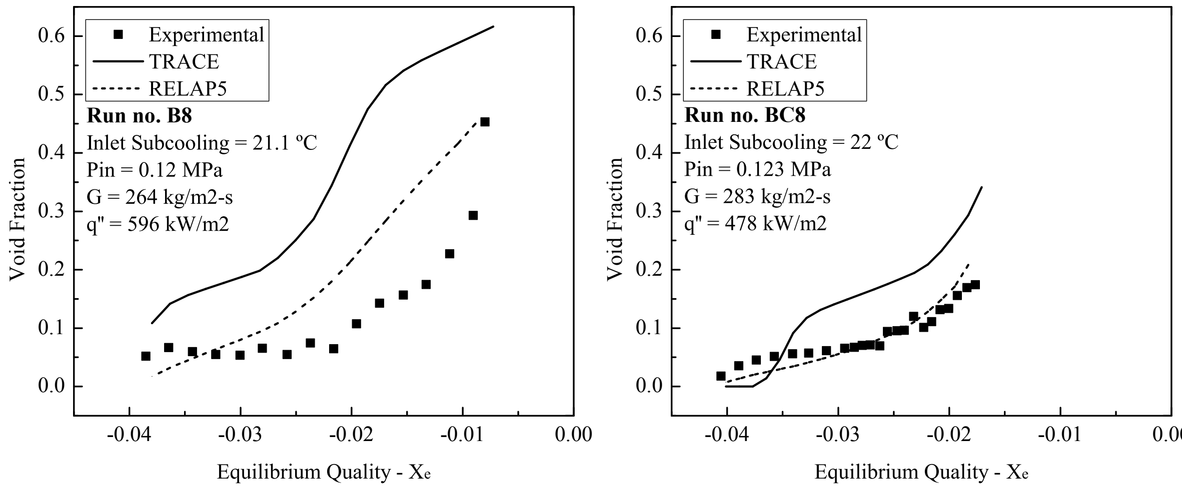

FIG. 2. Results from runs B8 and BC8

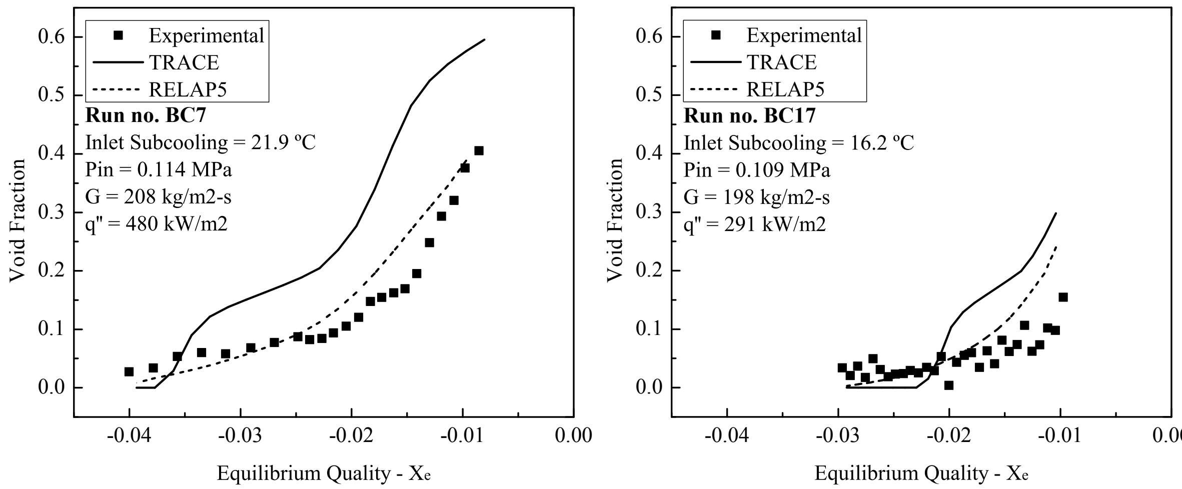

FIG. 3. Results runs BC7 and BC17

Since at low pressures TRACE is not able to predict void fraction with the same accuracy as RELAP5, subcooled nucleate boiling at medium and high pressures was also investigated in a code-to-code comparison. These second set of simulations were carried out using the same geometry as that one for the experimental setup described earlier, except that the conditions were extrapolated in both RELAP and TRACE to cover a much wider range of subcooled boiling. The boundary conditions for these simulations are: constant inlet subcooling θ_{HS} = 15 K, mass flux G = 205 kg/m^2 s and wall heat flux q'' = 480 kW/m^2. The main purpose of this code-to-code benchmark is to determine whether the cause of the deviation is related to the low-pressure conditions.

Results for this code-to-code comparison are shown in Fig. 5 and suggest that the divergence between codes increases as the system pressure decreases. At relatively high pressures (15 MPa) both codes predict similar trends and values. At lower pressures, the differences in void fraction increase with TRACE always predicting high voids than RELAP. The experiment simulations and the code-to-code comparison indicate that TRACE deviations are present at low-pressure conditions, but not at intermediate or high pressures.

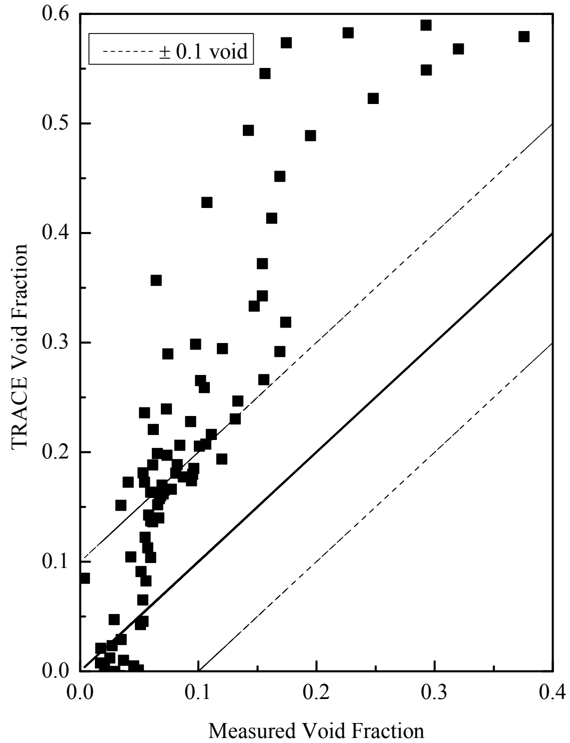

FIG. 4 Void fraction measurements vs. TRACE predictions

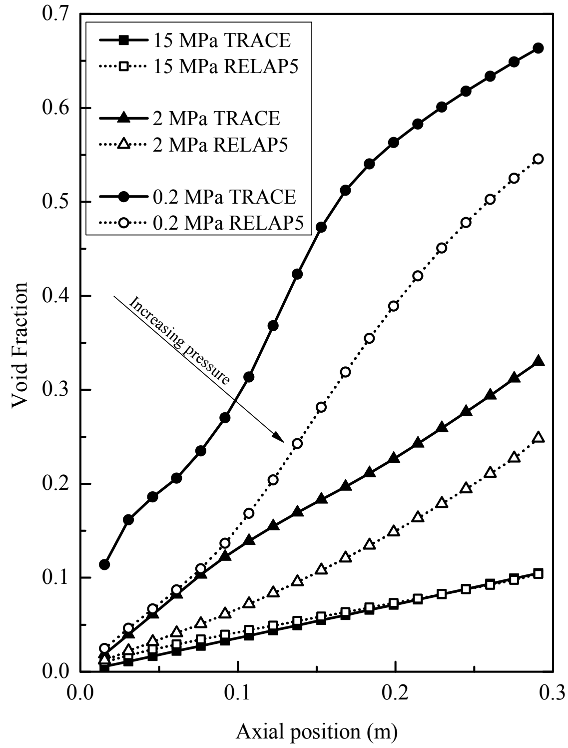

FIG. 5. RELAP5 vs. TRACE void fraction predictions at various pressures

2.2 Discussion on TRACE void fraction predictions

Both TRACE and RELAP5 are thermal-hydraulics system codes based on the two-fluid model formulation, and compute the void fraction from the interaction of three constitutive models. These models are for: (1) the source term for voids, which comes from the model for wall heat flux partitioning (giving the fraction of wall heat flux that goes to vapor generation), (2) the sink term, given by the interfacial heat and mass transfer from the condensation of the vapor (bubbles lifting off) in the subcooled fluid, and (3) interfacial friction. For mass transfer between phases, RELAP5 and TRACE use the same approach, which is based on a mechanistic concept of evaporation and condensation. Particularly, for subcooled boiling, this can be represented as a contribution from the wall evaporation (caused by subcooled boiling) and the condensation in the liquid-vapor interface.

$$\Gamma = \Gamma_i + \Gamma_{sub} \tag{2}$$

Where the subindex '*i*' represents the liquid-vapor interface mass exchange and '*sub*' subcooled boiling component. The Γ_{sub} accounts for the source term for void and through investigation it is believed to be a potential cause for TRACE's overprediction. In both codes, the models to calculate Γ_{sub} are given as a function of wall heat flux according to a heat flux partitioning submodel (i.e., models that separate the wall heat flux into a component that raises the sensible energy of the continuous liquid medium and a component that contributes to phase-change through boiling). The latter is one of the most important submodels for the determination of Γ_{sub}. A comparison on the wall heat flux partitioning models used by RELAP5 and TRACE is shown below.

RELAP5: $$q''_{ev} = q''_w \cdot \left(\frac{T_l - T_{l,OSV}}{T_{l,sat} - T_{l,OSV}} \right) \cdot \frac{1}{1+\epsilon} \tag{3}$$

TRACE: $$q''_{ev} = (q''_w - q''_{fc,2\phi}) \cdot \left(\frac{T_l - T_{l,OSV}}{T_{l,sat} - T_{l,OSV}} \right) \tag{4}$$

Where q''_{ev} is the evaporation heat flux, q''_w the wall heat flux, ε the so-called pumping factor, $T_{l,OSV}$ the bulk liquid temperature at the Onset of Significant Void, and $q''_{fc,2\varphi}$ the two-phase forced convection heat flux. A more detailed description of these wall heat flux partitioning models can be found in TRACE and RELAP5 theory manual [13,14] and Ref. [11].

Both expressions in Equations 3 and 4 are similar, with only a few noticeable differences: RELAP5's model contains the so-called pumping factor (ε), and TRACE's contains a $q''_{fc,2\varphi}$ term. Both terms act to suppress the wall evaporation rate (q''_{ev}), however, the pumping factor (ε) is density ratios dependent (see Eq. 5) while the wall evaporation rate in TRACE is not. Lahey formulation for the pumping factor [15] is given by

$$\epsilon = \frac{\rho_l}{\rho_v} \cdot \left(\frac{h_{l,sat} - h_l}{h_{fg}} \right) \tag{5}$$

Given the strong sensitivity of the pumping factor to densities ratio, and thus to pressure, it is beilieved that this results in the differing predictions as a function of pressure between TRACE and RELAP. Further investigation is needed to confirm the TRACE models applicability to low pressure or to include a stronger tie between the wall flux partitioning model and the pressure.

3. IAEA 10 MW reactor benchmark

As a complementary assessment of the code performance for research reactors conditions, TRACE was benchmarked against other codes using an integral effects benchmark. A safety-related transient problem for research reactors was provided in the IAEA Research Reactor Core Conversion Guidebook (IAEA-TECDOC-233) [16] for a 10 MW pool type reactor. The benchmark problems based on this idealized reactor were used in the framework of core conversion from Highly Enriched Uranium (HEU) to Low Enriched Uranium (LEU) loading. Within this framework, a comparative assessment was performed on computational tools and methodologies used by participating teams around the world. The results from several groups were reported in IAEA technical documents in 1980 and 1992.

The IAEA 10 MW reactor benchmark consists of a set of problems involving (1) static neutronic calculations and (2) transient calculations. Since neutronic calculations are out of the scope of this work, only the transient problems were considered and will be presented in the following order,

a. Fast loss of flow (FLOF). This transient is characterized by an exponential flow reduction ($e^{-t/\tau}$) with a time constant of $\tau = 1$ second. The trip for the reactor SCRAM is set at 85% of nominal flow, with a 200 ms delay before control rod insertion. The latter is accomplished with a linear reactivity insertion of –$10 in 0.5 seconds.

b. Slow loss of flow (SLOF). This event is also initiated by a flow exponential decay ($e^{-t/\tau}$) but with a time constant of $\tau = 25$ second, which makes the decay more gradual. The SCRAM conditions are the same as for the FLOF transient.

c. Slow reactivity insertion accident (SRIA). In this transient, a $0.1/s (for HEU) and a $0.09/s (for LEU) reactivity ramps are inserted in a critical reactor at an initial power of 1 W. The safety system trip point is set at $1.2 \cdot P_0$ (12 MW) with a time delay of 25 ms before a linear reactivity insertion (absorber blade insertion) of –$10 in 0.5 seconds. For the hot channel, the 'Radial', 'Local × Axial' and 'Engineering' power factors were considered (see Table 1). No overpower uncertainty was included since the safety system trip point is already set at $1.2 \cdot P_0$ = 12 MW.

d. Fast reactivity insertion accident (FRIA). This transient is initiated by a positive reactivity addition into the core of $1.5 in 0.5 seconds for both LEU and HEU cores. The safety trip point is identical to that one specified for the SRIA.

Even though this work does not cover the static neutronics calculation, they are needed to provide the transient problems with the core kinetic parameters. Thus, these parameters were taken from results obtained by the Argonne National Laboratory (ANL), USA [Ref. 17, Vol.3 p. 15–64].

The benchmark specifications correspond to a 10 MW, 6x5 fuel element assembly core reflected by a graphite row on two opposite sides, and surrounded by water as sketched in Fig. 6. The standard fuel elements (SFE) contain 23 fuel plates, whereas the control fuel elements (CFE) contain 17 fuel plates and four aluminum guide plates to accomodate the absorber rods. Figure 6 also gives a simple representation of the fuel element configuration. The benchmark specifies two different core loadings, one with HEU fuel and the other one with LEU. The reactor specifications are summarized in Table I.

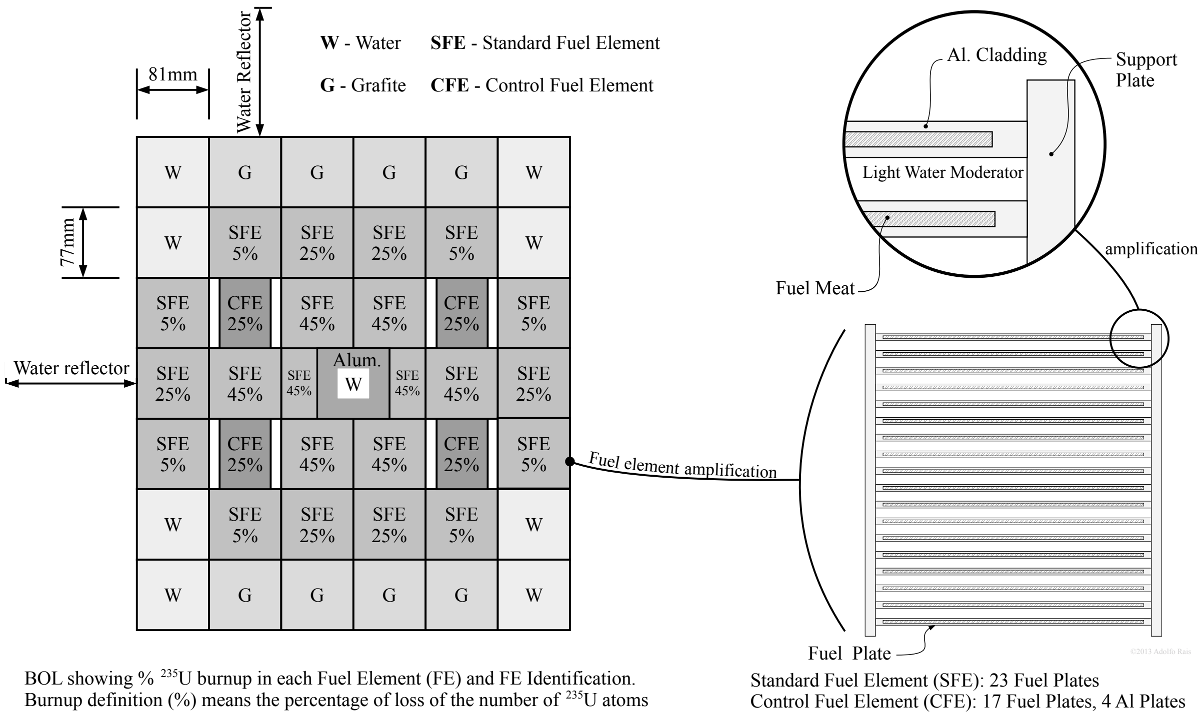

FIG. 6. Core and fuel elements schematic for the IAEA benchmark problem

A TRACE model of the reactor was developed based upon extensive testing utilizing different nodalizations, boundary conditions and thermal-hydraulic parameters. More details on this model can be found in Ref. [11].

TABLE I. IAEA 10 MW reactor core specifications

Reactor characteristics	
Reactor type	Light-water pool-type MTR
Fuel element	Plate-type. Al cladding
Coolant and Moderator	Light water
Flow mode	Downward forced flow
Reflector	Graphite & Light water
Core nominal power (MW)	10
Cores considered	HEU (93%) and LEU (20%)
Core parameters	
Radial x Local peaking factor	1.4
Axial power peaking factor	1.5
Engineering factor	1.2
Overpower factor	1.2
Nominal Flow Rate (m^3/h)	1000
Coolant inlet temperature (◦C)	38
Coolant inlet pressure (bara)	1.7
Fuel thermal conductivity (W/m-K)	158 (HEU) and 50 (LEU)
Cladding thermal conductivity (W/m-K)	180
Dimensions	
Grid cross section (mm^2)	77.0 x 81.0
Fuel Element cross section (mm^2)	76.0 x 80.5
Fuel Element height (mm)	600
Nr. of plates per fuel element	23 (SFE) and 17 (CFE)
Fuel Meat dimensions LxWxH (mm)	63 x 0.51 x 600
Cladding thickness (mm)	0.38
Water channel between plates (mm)	2.23

3.2 Results and discussion for the IAEA 10 MW reactor benchmark

TRACE V5.0P3 was used to simulate the four transients specified in the benchmark. The objective was to compare TRACE results to those obtained by other codes results as documented in the IAEA reports. Table II summarizes TRACE's most important results for the four transients, and also compares them with those obtained by six other codes: RELAP5/MOD3.2 (University of Pisa, Italy), PARET (Argonne National Laboratory, US), RETRAC-PC (Laboratoire d'Analyse de Sûreté, Algeria), COSTAX-BOIL (Junta de Energia Nuclear, Spain), EUREKA-PT & RELAP4/MOD5 (Japan Atomic Energy Research Institute) and COBRA III-C (Interatom, Germany).

TABLE II. Code intercomparison for the IAEA 10 MW transients

		McMaster TRACEv5.0p3	PISA R5/Mod3.2	ANL PARET	LAS RETRAC-PC	JEN COSTAX-BOIL	JAERI R4/Mod5	INTERATOM COBRA III-C
Fast loss of flow transient								
Power level	HEU	11.54 (0.363)	11.87	11.9 (0.316)	11.75	11.8 (0.36)	11.7	11.54 (0.363)
at SCRAM (MW)	LEU	11.48 (0.363)	11.83	11.9 (0.316)	11.72	11.7 (0.36)	11.7	11.40 (0.363)
Peak fuel centerline	HEU	84.33 (0.370)	-	89.2 (0.371)	-	94.5 (0.37)	99.4	91 (0.363)
temp. (°C)	LEU	85.23 (0.368)	-	90.3 (0.371)	-	95.4 (0.37)	98.7	91.9 (0.363)
Peak clad surface	HEU	82.56 (0.371)	91.28 (0.408)	87.5 (0.376)	91.74 (0.385)	94.0(0.38)	98.4 (0.40)	89.5 (0.380)
temp. (°C	LEU	82.50 (0.371)	95.58 (0.400)	87.5 (0.371)	87.92 (0.400)	93.9 (0.37)	97.1 (0.40)	89.3 (0.363)
Peak coolant outlet	HEU	59.40 (0.446)	59.53 (0.503)	60.3 (0.451)	60.04 (0.481)	59.4 (0.43)	58.4 (0.48)	56.5 (0.460)
temp. (°C)	LEU	59.37 (0.443)	59.50 (0.504)	60.3 (0.446)	59.92 (0.465)	59.3 (0.43)	58.1 (0.48)	56.4 (0.460)
Slow loss of flow transient								
Power level	HEU	11.60 (4.264)	11.62 (4.298)	11.6 (4.28)	11.61	11.8 (4.26)	11.6	11.55 (4.263)
at SCRAM (MW)	LEU	11.54 (4.263)	11.56	11.6 (4.28)	11.56 (4.050)	11.7 (4.06)	11.6	11.46 (4.263)
Peak fuel centerline	HEU	82.63 (4.264)	-	85.5 (4.29)	-	91.2 (4.27)	97.7	87.4 (4.263)
temp. (°C)	LEU	83.43 (4.264)	-	86.8 (4.29)	-	91.9 (4.27)	97.7	88.2 (4.263)
Peak clad surface	HEU	80.75 (4.265)	88.67 (4.305)	83.9 (4.29)	84.69 (4.160)	90.7 (4.27)	96.4 (4.2)	85.8 (4.263)
temp. (°C)	LEU	80.55 (4.265)	88.41 (4.299)	83.7 (4.29)	84.63 (4.240)	90.3 (4.27)	96.1 (4.2)	85.5 (4.263)
Peak coolant outlet	HEU	57.87 (4.282)	58.78 (4.305)	58.9 (4.29)	58.83 (4.160)	58.3 (4.27)	57.7 (4.3)	55.6 (4.263)
temp. (°C)	LEU	57.77 (4.280)	57.97 (4.300)	58.8 (4.29)	58.82 (4.272)	58.1 (4.27)	57.5 (4.3)	55.4 (4.263)
Slow Reactivity Insertion Transient								
Trip time	HEU	10.63	10.62	10.62	10.62	10.61	10.64	10.6
at 12 MW	LEU	11.94	11.92	11.87	11.8	11.68	11.9	12.03
Peak	HEU	13.49 (10.65)	13.69 (10.65)	14.1	14.05 (10.64)	14.93 (10.64)	13.75 (10.67)	14.36 (10.59)
power (MW)	LEU	12.33 (11.96)	12.34 (11.94)	12.4 (11.89)	12.29 (11.82)	13.01 (11.71)	12.35 (11.923)	12.18 (12.053)
Energy release	HEU	1.83	-	1.74	-	1.63	1.75	1.53
to peak power (MJ)	LEU	4.98	-	4.55	-	1.63	4.69	5.94
Peak fuel centerline	HEU	70.44 (10.67)	-	70.6	-	69.9 (10.66)	70.5 (10.67)	70.5 (10.61)
temp. (°C)	LEU	77.97 (11.97)	-	80.6 (11.90)	-	73.2 (11.72)	81.2 (11.933	80.8 (12.06)
Peak clad surface	HEU	68.84 (10.67)	71.70 (10.66)	69.9	75.01 (10.66)	69.5 (10.66)	69.2 (10.69)	69.2 (10.62)
temp. (°C)	LEU	75.0 (11.97)	81.12 (11.95)	77.59 (11.90)	78.52 (11.83)	71.9 (11.73)	78.5 (11.933)	78.1 (12.06)
Peak coolant	HEU	48.31 (10.75)	47.98 (10.74)	48.1	48.05 (10.73)	47.5 (10.73)	47.7 (10.77)	45.2 (10.70)
outlet temp. (°C)	LEU	53.20 (12.01)	53.15 (11.99)	53.9 (11.93)	53.52 (11.83)	48.8 (11.78)	52.8 (11.978)	51.1 (12.10)
Fast Reactivity Insertion Transient								
Trip time	HEU	0.607	0.609	0.609	0.608	0.611	0.619	0.605
at 12 MW	LEU	0.571	0.572	0.573	0.572	0.597	0.576	0.569
Peak	HEU	131.2 (0.653)	131.17 (0.655)	132 (0.655)	128.4 (0.655)	132.7 (0.659)	114.8 (0.664)	135.1 (0.650)
power (MW)	LEU	134.7 (0.610)	150.37 (0.612)	147.7 (0.613)	141.14 (0.612)	116.1 (0.638)	143.8 (0.616)	143.9 (0.608)
Energy release	HEU	3.23	-	3.26	-	3.47	2.86	3.14
to peak power (MJ)	LEU	2.72	-	2.95	-	2.62	2.95	2.83
Peak fuel centerline	HEU	193.7 (0.675)	175.5 (0.673)	171 (0.670)	-	167.1 (0.672)	155.4 (0.678)	173.4 (0.665)
temp. (°C)	LEU	187.2 (0.628)	-	183.4 (0.626)	-	166.4(0.654)	171.0 (0.625)	185.8 (0.625)
Peak clad surface	HEU	185.2 (0.677)	163.3 (0.673)	156 (0.672)	158.6 (0.668)	162.3 (0.675)	147.3 (0.678)	160 (0.665)
temp. (°C)	LEU	174.2 (0.631)	166.55 (0.629)	156.7 (0.628)	155.94 (0.626)	156.6 (0.654)	149.2 (0.627)	168.2 (0.625)
Peak coolant	HEU	82.6 (0.81)	78.9 (0.770)	84.3 (0.780)	83 (0.745)	108.7 (0.747)	62.3 (0.820)	70.7 (0.783)
outlet temp. (°C)	LEU	72.5 (0.74)	78.01 (0.728)	82.0 (0.735)	79.42 (0.706)	80.4 (0.711)	62.7 (0.762)	63.2 (0.740)

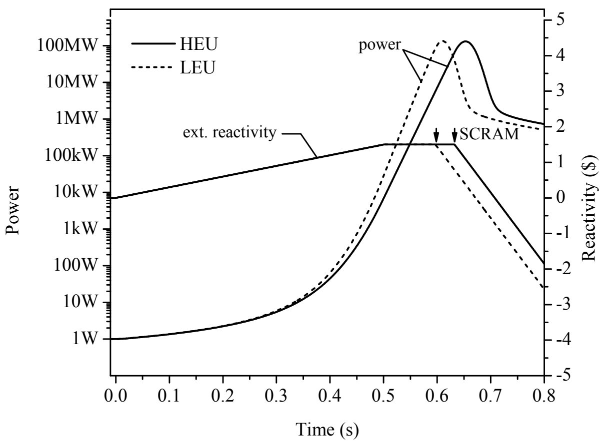

FIG. 7. FRIA - Power and external reactivity.

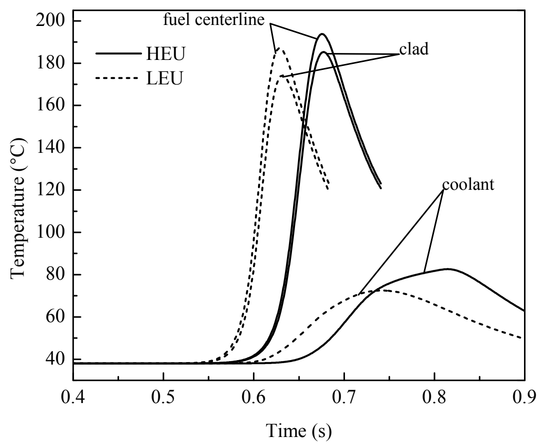

FIG. 8. FRIA - Temperatures.

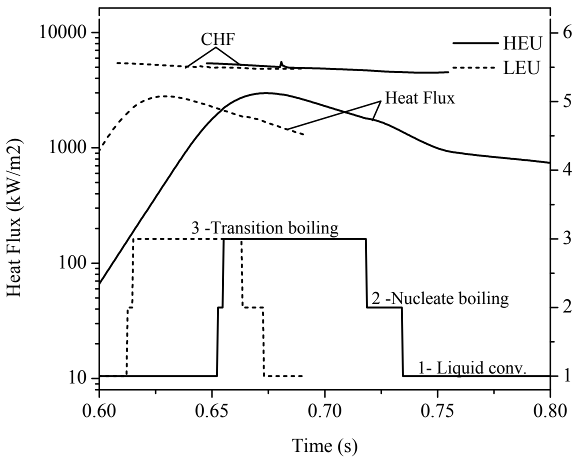

FIG. 9. FRIA - Actual heat fluxes, CHF and heat transfer regime

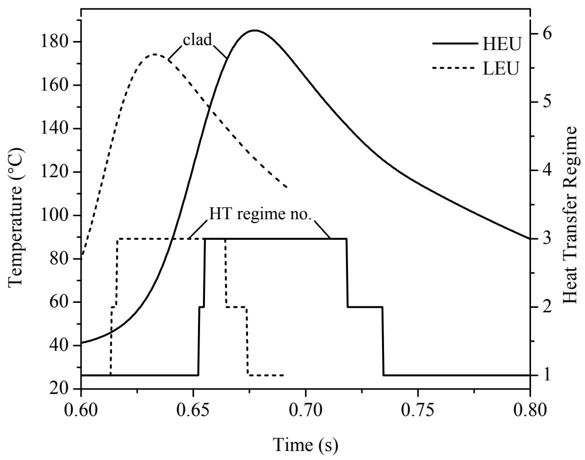

FIG. 10. Heat transfer regime and clad temperatures during FRIA

The discussion will be mainly focused on the fast reactivity insertion transient (FRIA) since it was found to be the most limiting case in terms of temperatures and also the largest deviations between code predictions. More information about TRACE's results for the three other transients can be found in Ref. [11].

The fast reactivity insertion accident (FRIA) is the most limiting transient among all four since power increases eight orders of magnitude in less than one second and also because clad temperatures become high enough to trigger two-phase flow regimes (see Fig. 7 and 8). On the contrary, no two-phase flow was observed in any of the other three transients (FLOF, SLOF and SRIA). From Figures 7, 8 and Table II it can be observed that the peak power is reached earlier in the LEU core due to the shorter prompt neutron generation time (Λ). The table indicates that there is a considerable time lag between peak power and peak temperatures, which reduces the effect of reactivity feedback. In particular, for the HEU core, the magnitude of the Doppler feedback is small but it is more prompt than the coolant feedback effect. On the other hand, for the LEU core, the Doppler feedback is much more significant than the coolant temp. TRACE predicts large amounts of void fraction only in the hot channel while the average channel conditions show little or no void. This has an almost negligible effect upon the core kinetics as the hot channel accounts only for 1/551 of the core, and also because void fraction appears after the peak power is reached.

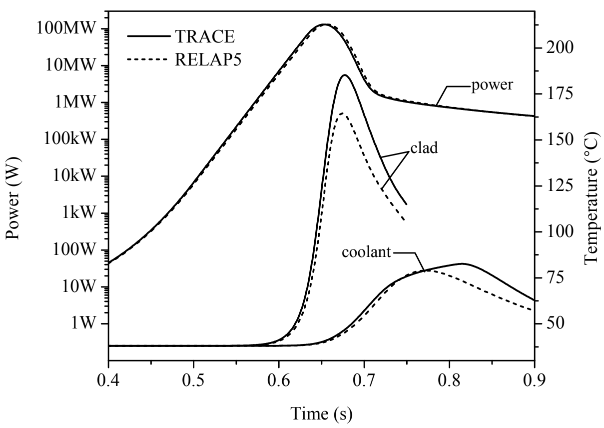

FIG. 11. R5 vs TRACE for FRIA (HEU case)

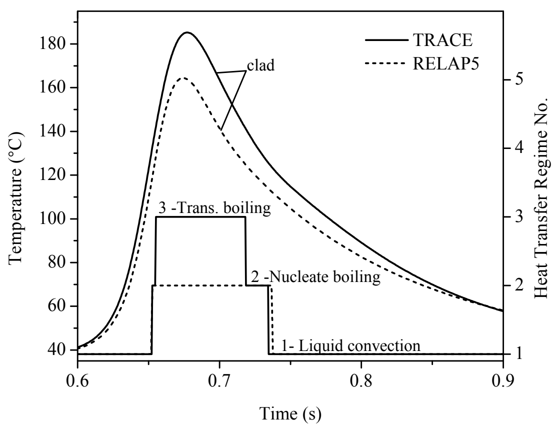

FIG. 12. R5 vs TRACE HT regimes during FRIA (HEU case)

It is worth noting that TRACE predicts *transition boiling* heat transfer regime right after reaching the peak power, as it can be observed in Fig. 9 and 10. This condition was unexpected since the logic used by TRACE to initiate this heat transfer regime involves reaching the critical heat flux wall temperature ($T_{w,CHF}$). Conversely, the actual heat flux remains far below CHF as shown in Fig 9. In this figure it can also be appreciated the heat transfer regime number, indicating liquid forced convection, nucleate boiling and transition boiling. Fig. 10 shows that peak clad temperatures are taking place during the *transition boiling* regime, which causes higher predictions of clad and fuel centerline temperatures.

Finally, a RELAP5 model of the IAEA 10 MW reactor [18], was used to simulate the same benchmark problems, and particularly to compare its predictions for the FRIA with TRACE. Results for this comparison are shown in Fig. 11 and 12. While the core kinetics shows comparable behavior, the difference in clad temperatures and boiling heat transfer regime are evident. Figure 12 shows the clad surface temperature along with the heat transfer regime, which suggest that the '*transition boiling*' regime is the reason for TRACE's temperature overprediction during the FRIA transient. Further investigation on the unexpected change in heat transfer regime to transition boiling in TRACE is needed.

4. Conclusions

The objective of this work is to study TRACE applicability to low operating pressure research reactors. This was achieved by performing code evaluations against subcooled boiling experiments and also by means of code-to-code comparision based on the IAEA benchmark exercise. The following main conclusions can be derived from both exercises:

TRACE code significantly overpredicts void fraction level at a given quality under low pressure conditions while RELAP5/MOD3.3 provided better agreement with experimental data. The overprediction is presumed to occur only at low pressures. It is caused by TRACE's heat flux partitioning model that overestimates the portion of heat flux contributing to vapor generation. Further investigation under low pressure conditions is recomended.

The code-to-code comparisons for the IAEA 10 MW reactor benchmark revealed that TRACE is the only code predicting transition boiling regime during the fast reactivity insertion transient (FRIA). The reasons for this unique behavior are still unclear and require deeper investigation.

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