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Validation of the 1D+2D thermal-hydraulic module of the FRENETIC code

R. Bonifetto, L. Savoldi Richard, R. Zanino, A. Del Nevo

CONSORZIO INTERUNIVERSITARIO PER LA RICERCA TECNOLOGICA NUCLEARE - CIRTEN -

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VALIDATION OF THE 1D+2D THERMAL-HYDRAULIC MODULE OF THE FRENETIC CODE

R. Bonifetto, L. Savoldi Richard, R. Zanino (POLITO), A. Del Nevo (ENEA)

Settembre 2013

Report Ricerca di Sistema Elettrico

Accordo di Programma Ministero dello Sviluppo Economico - ENEA
Piano Annuale di Realizzazione 2012
Area: Produzione di energia elettrica e protezione dell'ambiente
Progetto: Sviluppo competenze scientifiche nel campo della sicurezza nucleare e collaborazione ai programmi internazionali per il nucleare di IV Generazione
Obiettivo: Sviluppo competenze scientifiche nel campo della sicurezza nucleare
Responsabile del Progetto: Mariano Tarantino, ENEA

Il presente documento descrive le attività di ricerca svolte all'interno dell'Accordo di collaborazione "Sviluppo competenze scientifiche nel campo della sicurezza nucleare e collaborazione ai programmi internazionali per il nucleare di IV generazione"
Responsabile scientifico ENEA: Mariano Tarantino
Responsabile scientifico CIRTEN: Giuseppe Forasassi

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Termoidraulica del nocciolo
Reattori nucleari veloci
Generation IV reactors

Sommario

The FRENETIC code for the dynamic simulation of LFR cores with closed hexagonal fuel elements at a reduced computational cost has been recently developed at Politecnico di Torino by the research group of nuclear engineering. The tool is composed by two modules, the neutronic module and the thermal-hydraulic (TH) module, that can be run together to solve the coupled neutronic and thermal-hydraulic model equations, or separately to analyze only the thermal-hydraulic or neutronic behavior. The TH module has been successfully validated against experimental data from the CIRCE facility at ENEA Brasimone, as far as the 1D TH analysis along each fuel assembly (FA) is concerned. The results of a first full core TH benchmark against another computational tools (RELAP5-3D®) are reported here, to check the horizontal 2D coupling model in FRENETIC.

Note:

Lavoro eseguito in collaborazione con ENEA

Autori:

R. Bonifetto, L. Savoldi Richard, R. Zanino (POLITO)
A. Del Nevo (ENEA)

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Validation of the 1D+2D thermal-hydraulic module of the FRENETIC code

Autori

Roberto Bonifetto, Laura Savoldi Richard, Roberto Zanino (PoliTo)

Alessandro Del Nevo (ENEA)

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Abstract

The FRENETIC code for the dynamic simulation of LFR cores with closed hexagonal fuel elements at a reduced computational cost has been recently developed at Politecnico di Torino by the research group of nuclear engineering. The tool is composed by two modules, the neutronic module and the thermal-hydraulic (TH) module, that can be run together to solve the coupled neutronic and thermal-hydraulic model equations, or separately to analyze only the thermal-hydraulic or neutronic behavior.

The TH module has been successfully validated against experimental data from the CIRCE facility at ENEA Brasimone, as far as the 1D TH analysis along each fuel assembly (FA) is concerned.

The results of a first full core TH benchmark against another computational tools (RELAP5-3D®) are reported here, to check the horizontal 2D coupling model in FRENETIC.

Introduction

The FRENETIC code has been recently developed for the simulation of coupled neutronic/thermal-hydraulic transients in lead-cooled fast reactors (LFR), with the core arranged in closed hexagonal fuel assemblies (FA) [1].
The neutronic module in FRENETIC has evolved from point kinetics [2] to a full 3D time dependent multi-group diffusion solver [3].
The quasi-3D thermal hydraulic (TH) module of FRENETIC solves the 1D (axial) mass momentum and energy conservations laws of the coolant, together with the 1D (axial) heat conduction equation in the fuel pins, in each FA. The FAs are thermally coupled to their neighbors in the other two directions, at selected axial locations.
The TH module of FRENETIC was already successfully validated in the case of a single FA against data from the Lead-Bismuth Eutectic (LBE) CIRCE experiment at ENEA Brasimone [4].

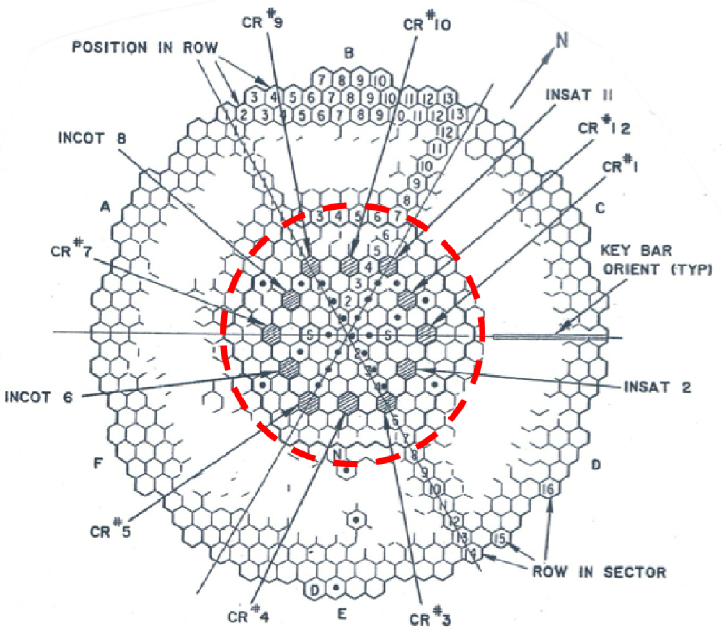

Fig. 1. Sketch of the cross section of the EBR-II core [6]. The red dashed circumference shows the approximate location of the radial boundary of the computational domain.

In order to validate the multi-FA capabilities of FRENETIC, and in the absence of easily available thermal-hydraulic data for lead-cooled multi-FA structures, a reasonable alternative is to perform this validation on sodium-cooled geometries relevant to EBR-II [5], see Fig. 1, for which some experimental data are going to be available within the framework of a multi-party benchmarking exercise [6].

Scope of the work

This report will present a qualification of the TH module of the FRENETIC code in a multi-assembly geometry, by means of a benchmark with the RELAP5-3D© code [7].
The final goal is the simulation of two different transients, in a simplified EBR-II geometry, starting from the steady state experimental data of the Shutdown Heat Removal Test (SHRT) #17 [6]:

a) complete loss-of-flow,
b) locked rotor in one of the main coolant pumps.

The rationale in the definition of these scenarios is to reproduce transients with significant non-uniformities in the core cross section, in order to test the thermal coupling model between different assemblies. Indeed, the first transient is characterized by quasi symmetric flow at the core inlet and non-uniform power distribution in the core, while the second transient has asymmetric flow and power distribution in the core.

This report contains the results in steady state conditions to assess the inter-assembly thermal coupling strategy adopted in the FRENETIC model.

1 Model description

As the FRENETIC code can only deal with the fuel bundle region of the core (0.61 m axial extension in the EBR-II case), the RELAP5-3D© code [7] is used to provide detailed time-dependent boundary conditions (BC) at the inlet and outlet of each FA in that region. Moreover, the same axially uniform power distribution on the core cross section is provided to both codes. At steady state, the SHRT-17 test data are used, while in transient conditions the RELAP5-3D© code will be used.

For the time being, the radial extension of the FRENETIC model is limited to the 127 central FAs (first 7 rings) of the EBR-II core (region inside the red dashed circle in Fig. 1), i.e. those connected with the high pressure inlet plenum, with adiabatic BC on the side surface of the core. On the other hand, the RELAP-3D© model includes also the thermal coupling with the 42 FAs of the blanket in the next ring, which are connected with the low pressure flow line.
The inlet Na temperature is set to 700 K.

This section contains the description of the two models adopted in the two different codes. The numbering order of the fuel assemblies (FA) in the core is shown in Fig. 2 and Fig. 3.

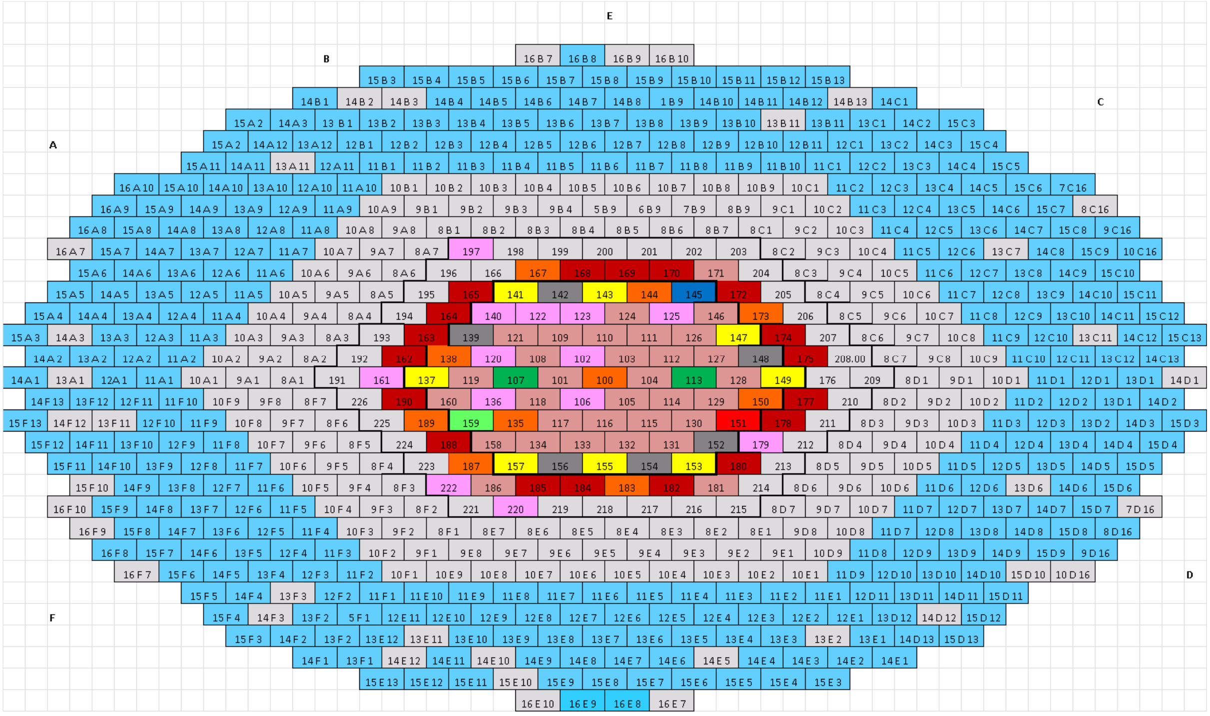

Fig. 2. Assemblies numbering order in the present work. The central FA (100 in the figure) will be referred to as number 1, and so on.

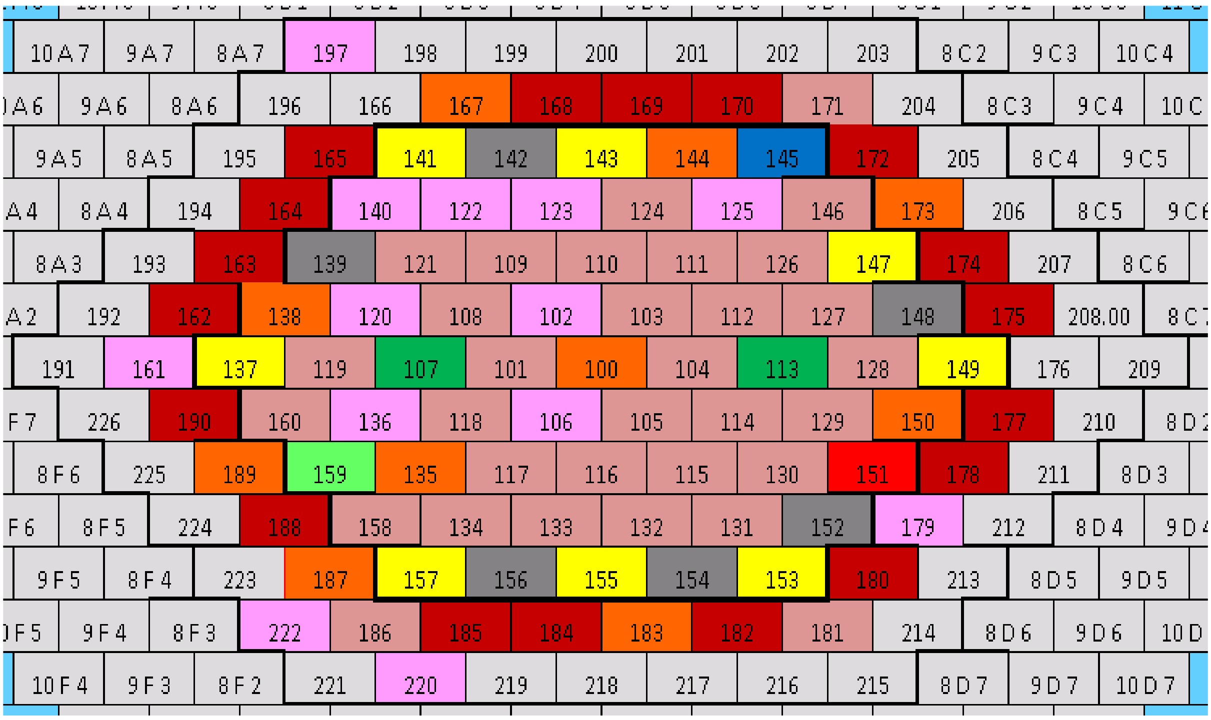

Fig. 3. Zoom of the assemblies numbering order in the present work on the zone of interest (first seven rings).

The core load, reported in Fig. 4, corresponds to the load of the Shutdown Heat Removal Test (SHRT) 17, i.e. the run 129C.

Table 1 reports the types of FA present in the first seven rings.

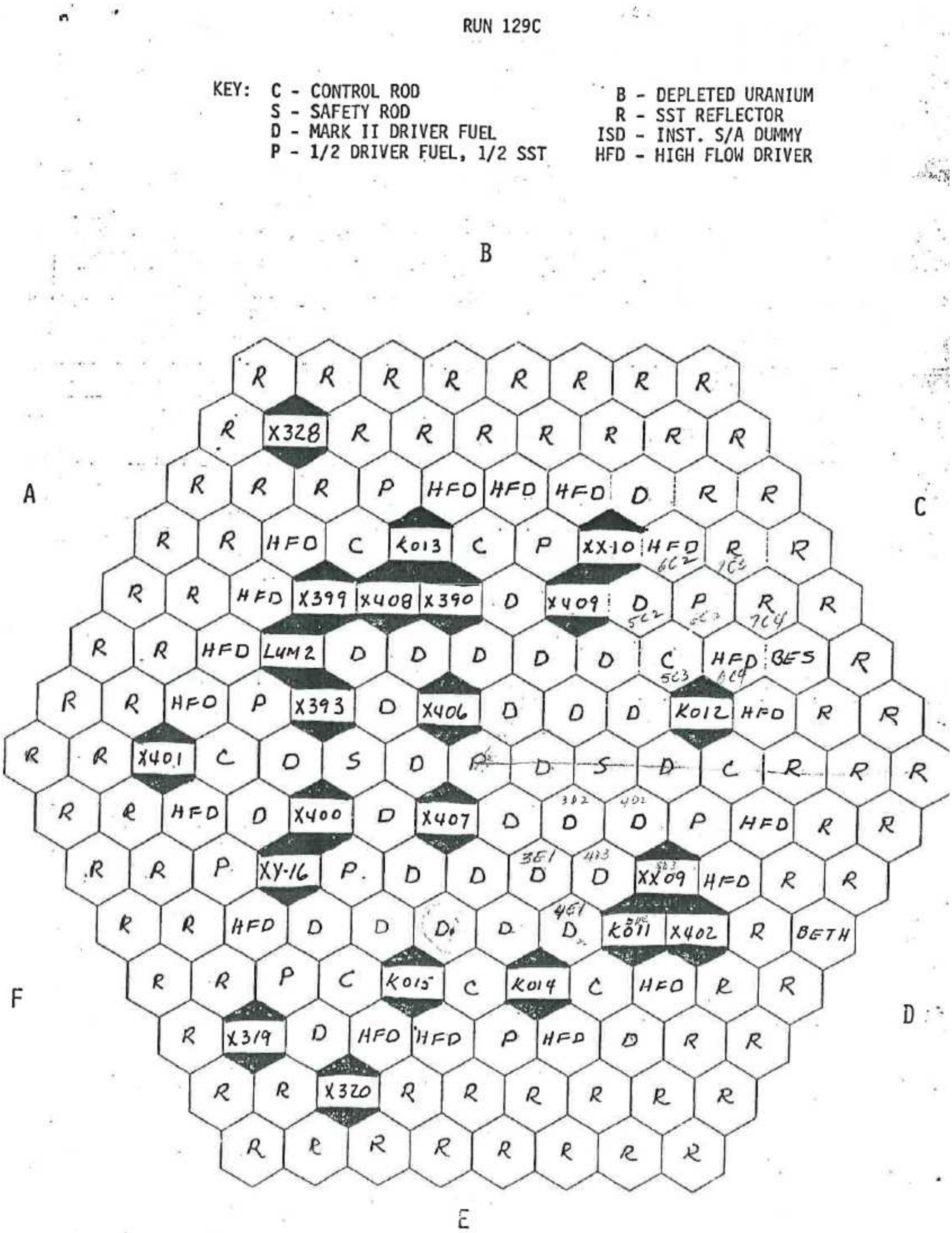

Fig. 4. SHRT-17 Core Loading Pattern (First 8 rings).

Table 1. Types of FA in each ring.

EBR-II - ring	Type	No FA
Ring 7 - Tot. FA 36	Reflector	33
	Exp	3
Ring 6 - Tot. FA 30	Reflector	2
	Exp	2
	Driver	3
	Partial D	5
	High flow	18
Ring 5 - Tot. FA 24	Exp	1
	Driver	3
	Partial D	3
	CR	8
	SS FA	6
	XX09	1
	XX10	1
	Dummy	1
Ring 4 - Tot. FA 18	Exp	5
	Driver	12
	Partial D	1
Ring 3 - Tot. FA 12	Driver	10
	Safety	2
Ring 2 - Tot. FA 6	Exp	2
	Driver	4
Ring 1 - Tot. FA	Partial D	1

1.1 RELAP5-3D model

The RELAP5-3D model of the EBR-II code is described below.

- Core model
  - Core modeled with 127 separate channels connected with the high pressure flow line (one by one)
  - Blanket modeled with 24 separate channels connected with the low pressure flow line (not relevant for the comparison)
- Pressure at Z-pipe imposed (1.953×10^5 Pa).
- Pressure at fuel assembly inlet is 6.29×10^5 Pa in steady state conditions, which is 1.0×10^5 Pa higher if compared with the estimation provided in the specifications.
  - The pressure drops calculated between the core inlet pressure (in the plena) and the fuel core at wire wrapped fuel bundle outlet is provided, for each channel.
- Power imposed (see Fig. 5 and Fig. 6):
  - Fuel assembly power based on SHRT-17 benchmark specification
  - Uniform axial power distribution.
- Mass flow rates @ different FA types.
- The coolant inlet temperature is set to 700K.

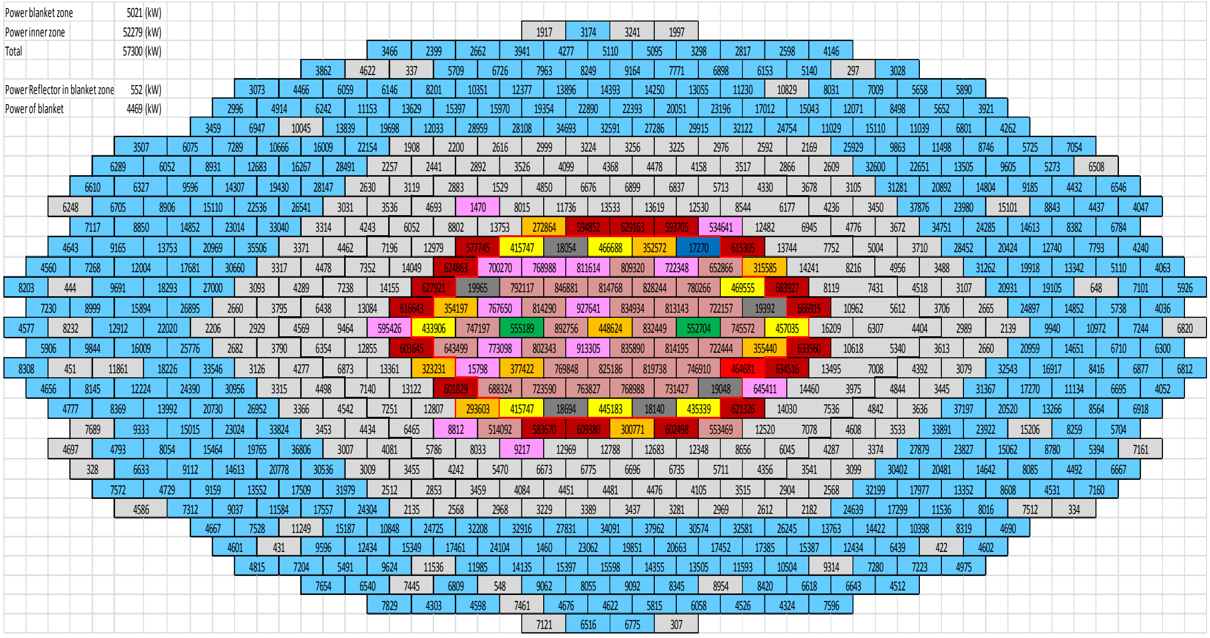

Fig. 5. Power distribution in the EBR-II assemblies.

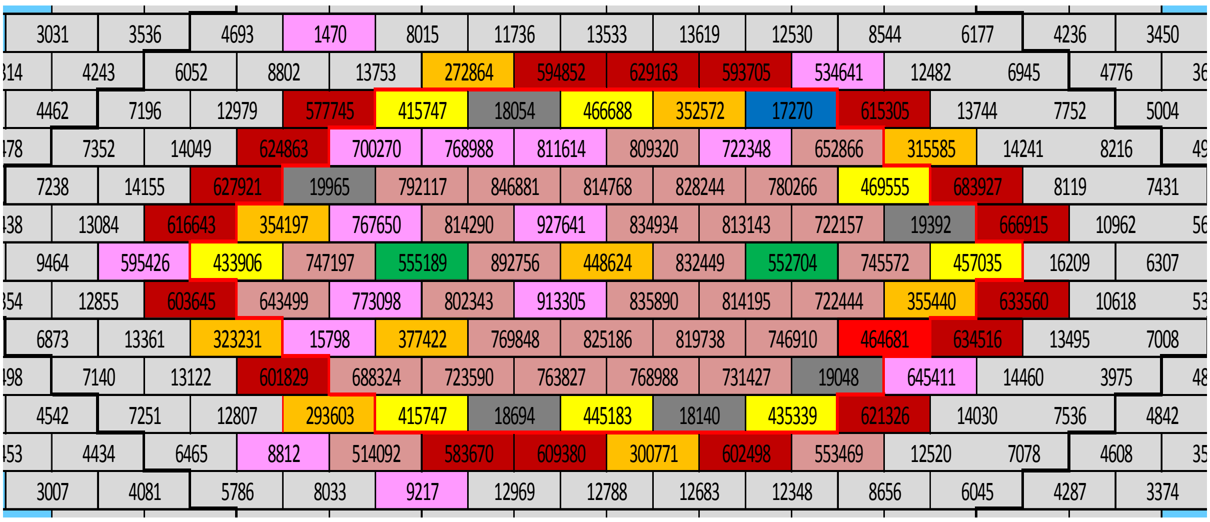

Fig. 6. Zoom of power distribution in the EBR-II assemblies on the zone of interest (first seven rings).

1.2 *FRENETIC model*

The FRENETIC model of the EBR-II code is described below.

- Core model
  - Core modeled with 127 separate channels
  - Blanket NOT modeled (adiabatic BC prescribed on the outer surface of the 7th ring)
- Power imposed (see Fig. 5 and Fig. 6):
  - Same power as Relap5-3D
  - Uniform axial power distribution in the active core (0.3429 m from FA inlet).

- BC (from Relap5-3D)
    - Mass flow rates each FA inlet.
    - Pressure at each FA outlet (pressure at the core bundle outlet).
    - Coolant inlet temperature (set to 700 K).
- Additional BC: all the pins are adiabatic at their ends.
- Friction factor from [8] Chapter 9, "Single-Phase Fluid Mechanics", VI - Pressure drop in rod bundles, for bare rod bundles.
- Heat transfer coefficient: Mikityuk correlation [9] (for FA without pins Dittus-Boelter [10] is used). Schad correlation [8] is also used.

Neighboring FAs are thermally coupled in FRENETIC through a nominal 1D thermal resistance – series of the two stainless steel hexagonal wrappers (each 1.016 mm thick) and of the Na, assumed stagnant, in the clearance (0.764 mm thick) between FAs (see Fig. 7).

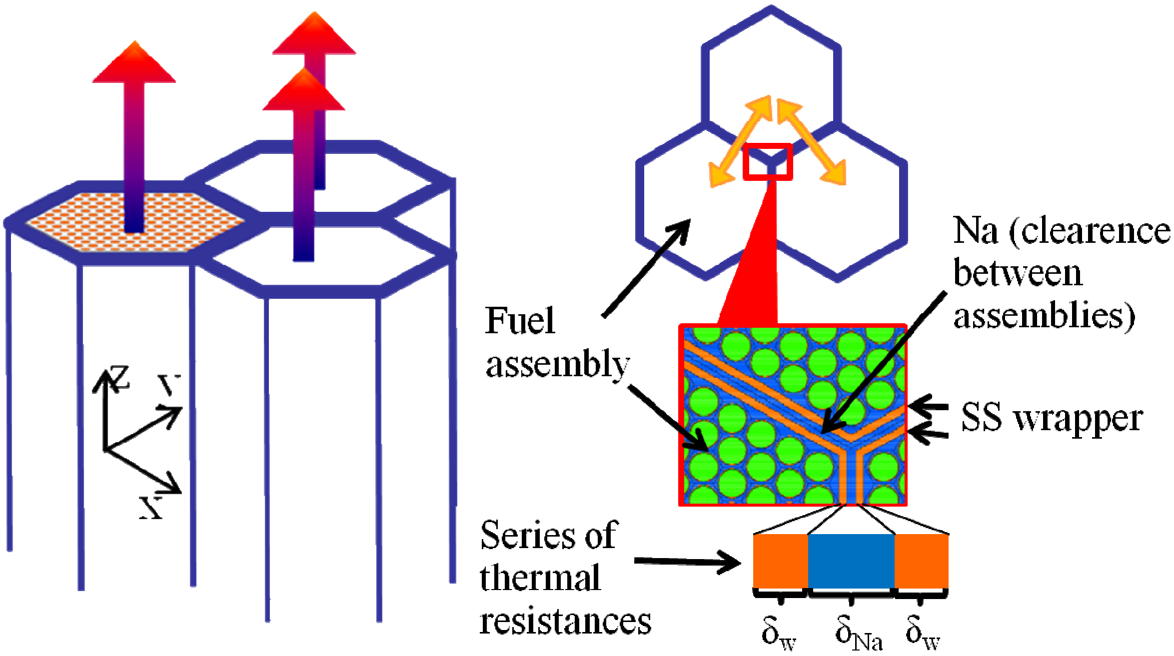

Fig. 7. 2D thermal coupling between the neighboring assemblies in the FRENETIC model.

2 Steady state results

The comparison between the FRENETIC and RELAP-3D© preliminary results at steady state is presented in this Section.
In a first test, the FAs are assumed to be adiabatic (decoupled). The results of the two codes, shown in Fig. 8, are qualitatively in good agreement, while the FRENETIC outlet temperature is almost everywhere ~10 K lower than RELAP5-3D© results. We discovered that this is due to a difference in the Na properties implemented in the two codes (taken from [11] in FRENETIC, from [12] in RELAP5-3D©), and in particular to a discrepancy of ~10% in the specific heat at constant pressure (see Fig. 9). In the future this discrepancy will be eliminated by adopting in both codes a consistent set of properties.

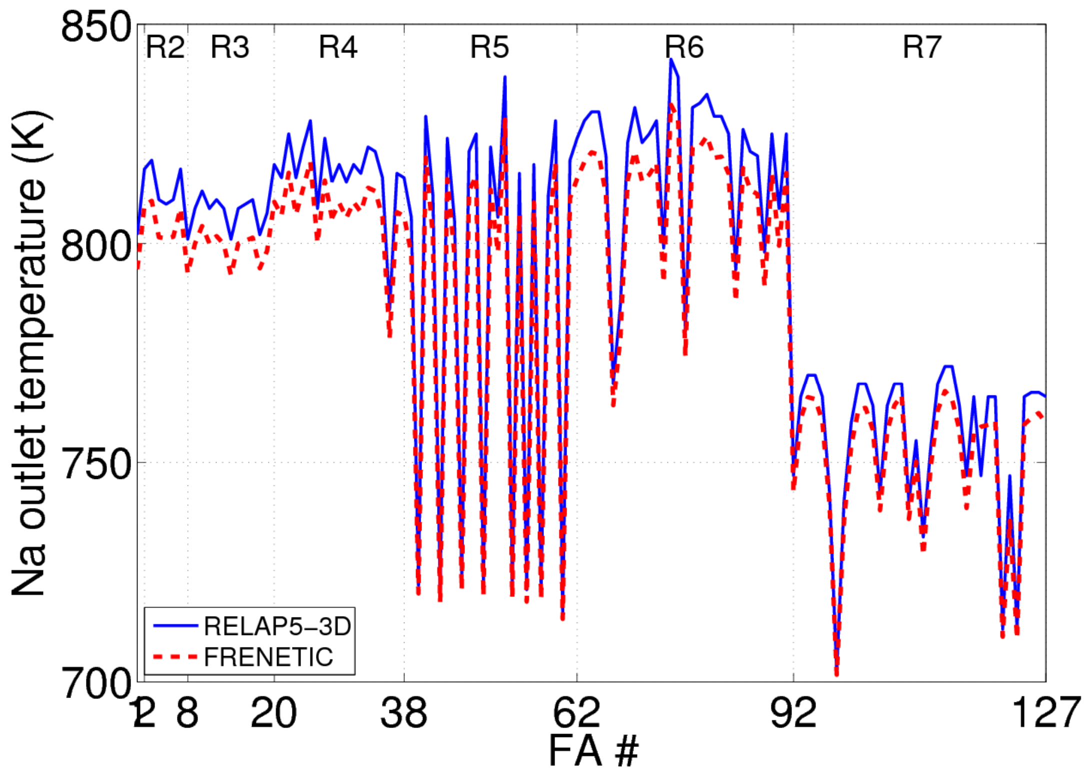

Fig. 8. Comparison between computed steady state Na temperatures at the outlet of the fuel bundle region, in each *adiabatic* FA. The number R* of the corresponding ring is reported at the top of the figure.

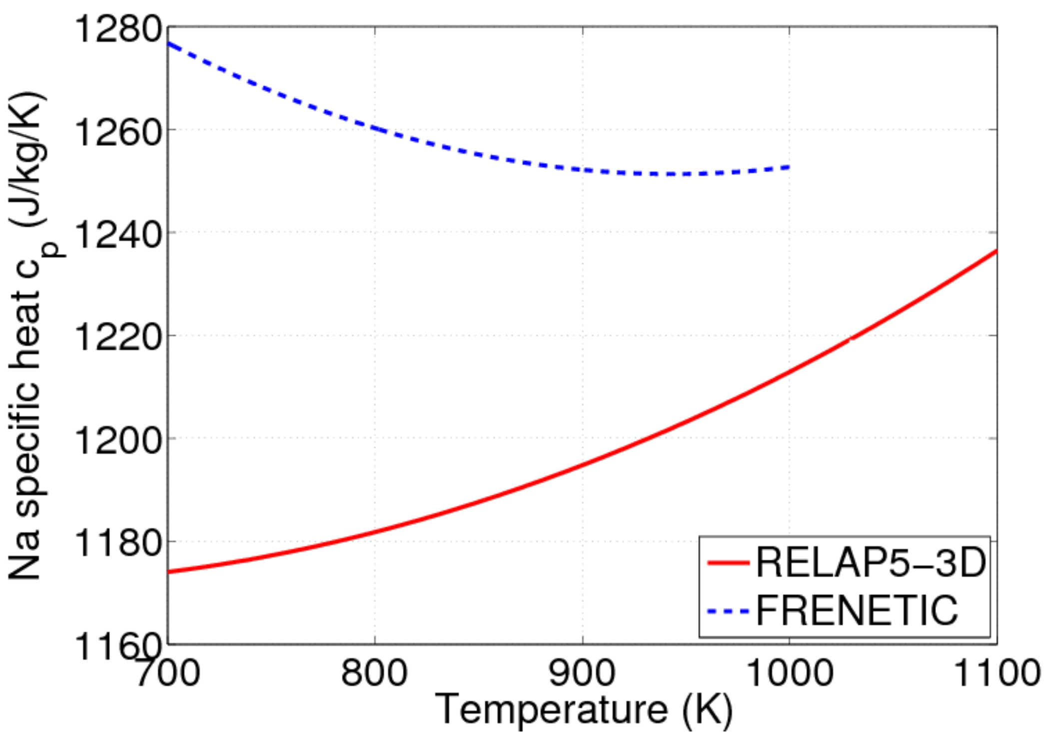

Fig. 9. Comparison between Na specific heat at constant pressure implemented in the two codes.

We also considered a more realistic and interesting second test (see Fig. 10), where the FAs are thermally coupled to each other. Of course, in such a comparison the outermost (7^{th}) ring is somehow anomalous, due to the above-mentioned different BC applied in the two codes (FRENETIC outlet temperatures are obviously going to be much higher than RELAP5-3D© ones, as the FAs in that ring are heated by the 6th ring FAs, but not cooled by the 8th ring ones, as in the RELAP5-3D© case).

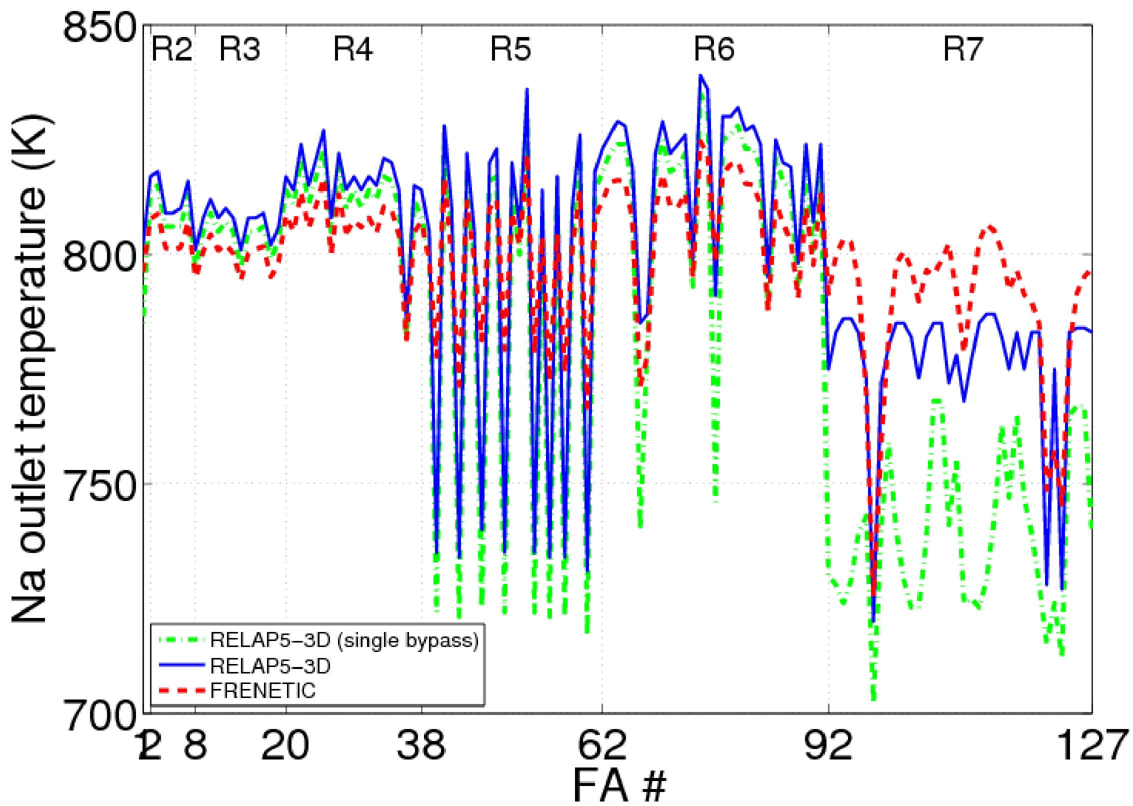

Fig. 10. Comparison between computed steady state Na temperatures at the outlet of the fuel bundle region, in each FA. The number R* of the corresponding ring is reported at the top of the figure. For the RELAP5-3D® results, both the cases with single (dash-dotted green) and double (solid blue) bypass model are reported.

However, it also turned out that our first model in RELAP5-3D© of the Na bypass flow in the clearances between the FAs was too rough: indeed, this flow was modeled so far in RELAP5-3D© with a single channel, collecting the flow from the clearances in the entire core, most of which is not heated. This results in a very low temperature increase of the bypass, so that in the in RELAP5-3D© model the coolant flowing in the clearances of the FAs in the central part of the core (the hottest, and the only one analyzed here) is *cooling* the FAs. On the contrary, the simplified thermal coupling model between FAs adopted by the FRENETIC code tends to smooth the temperature difference between neighboring FAs, leading to a temperature *increase* of the colder FAs.

A more detailed RELAP5-3D® model of the bypass flow has then been developed, where the bypass flow is divided into two bypass channels: one for the 127 central assemblies included in the FRENETIC model and another one for all the other assemblies. The results of the comparison at steady state are also reported in Fig. 10, showing that the coolant outlet temperature is indeed higher with respect to the one computed with a single bypass channel. The effect of the different BC applied at the interface between the 7th and the 8th rings is still evident, as the RELAP5-3D® temperature is again lower than the FRENETIC one due to the cooling effect of the reflector (cold) assemblies in the 8th ring.

3 Conclusions

An encouraging agreement between FRENETIC and RELAP-3D© results at steady state has been shown above [13]. This first test has also highlighted some issues, concerning the Na properties and the model of the Na in the clearances between FAs, which are being taken care of. After the successful completion of this benchmark, it should become meaningful to apply FRENETIC to the analysis of the actual SHRTs in EBR-II.

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5 Breve CV del gruppo di lavoro

Il gruppo di lavoro impegnato nell'attività opera presso il Dipartimento Energia del Politecnico di Torino ed è costituito da un professore ordinario (Roberto Zanino, impianti nucleari), un ricercatore confermato (Laura Savoldi, impianti nucleari), e da un dottorando (Roberto Bonifetto) iscritto al III anno di Dottorato in Energetica.

Il gruppo ha una lunga esperienza nella ricerca nel campo dell'ingegneria nucleare, sia nel settore della fusione, che, più recentemente, nel settore della fissione.
Nel settore della fissione l'attività riguarda lo sviluppo di strumenti per il calcolo accoppiato (neutronico e termo-idraulico) della dinamica dei reattori, in particolare per applicazioni ai sistemi nucleari avanzati (reattori innovativi di IV generazione). Nel settore della fusione il gruppo si è occupato dell'analisi termofluidodinamica di componenti di reattori a confinamento magnetico e in particolare dello sviluppo di codici per la modellazione del sistema dei magneti superconduttori e dell'applicazione di software CFD per l'analisi di blanket, first wall e vacuum vessel.
L'attività di Roberto Bonifetto comprende lo sviluppo e l'applicazione di codici per la modellazione di reattori nucleari a fissione (il codice presentato in questo lavoro) e a fusione (il codice 4C per l'analisi termofluidodinamica dei magneti superconduttori).
Nel lavoro presentato in questo rapporto sono state utilizzate le metodologie di simulazione termoidraulica messe a punto nel settore della fusione per lo sviluppo di un modulo di codice di multifisica per l'analisi dinamica di un reattore veloce.

Maggiori dettagli e l'elenco delle pubblicazioni più recenti dei membri del gruppo si possono trovare sul sito Web del Politecnico di Torino:
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