COMPARATIVE ANALYSIS OF BWR CORE TRANSIENT BENCHMARKS

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ABSTRACT

This paper describes the research performed jointly by the Pennsylvania State University Nuclear Engineering Department and Toden Software Incorporation to evaluate the effect of employed neutronics model on coupled three-dimensional (3D) transient predictions. The TRAC-BF1/NEM and TRAC-BF1/ENTRÉE codes use the same thermal hydraulics code applied to different neutronics codes. These codes were used to model the OECD/NEA BWR core transient benchmarks. A brief description of these benchmarks is given in this paper. Modeling schemes are described in detail, followed by comparisons of steady state and transient results obtained from both codes. Some results from other coupled codes are also presented for a more complete comparison. The performed comparative analysis suggests that the neutronics effects (in terms of nodalization schemes, cross-section modeling algorithm and used method in the framework of transverse integrated nodal procedure) are small in both steady-state and transient predictions. Once the optimal spatial overlay (mapping) scheme is chosen the major contributor in the observed deviations in the different coupled code predictions seems to follow from different thermal-hydraulics models and their degree of sophistication.

1. INTRODUCTION

With the fast development of computer technology in recent years, more attention has been focused on the coupling of thermal hydraulic (T-H) system code with three-dimensional (3-D) neutronics core simulator. A coupled code, named TRAC-BF1/NEM[1], was developed and tested at the Nuclear Engineering Program of the Pennsylvania State University (PSU). This code uses TRAC-BF1[2], a T-H system code that provides a consistent and unified analysis capability for a wide range of BWR transients. On the neutronics side, TRAC-BF1/NEM uses a 3-D code[3] with an implemented nodal expansion method based on the response matrix technique. The parallel virtual machine (PVM) technology is employed to exchange data between the T-H system code and the neutronics core simulator. TRAC-BF1/ENTRÉE[4] is another coupled code that couples TRAC-BF1 with ENTRÉE, a 3-D neutronics code developed by the Toden Software Inc from Japan. ENTRÉE is also based on the nodal expansion method, using response matrix method and nonlinear iterative method. Similar coupling scheme is used as the one in TRAC-BF1/NEM. The purpose of this research is to use the OECD/NEA BWR numerical benchmark problems[5] to verify and compare the performances of the coupled 3-D neutronics/T-H system code TRAC-BF1/NEM and TRAC-BF1/ENTRÉE. Since both codes utilize the same T-H model and coupling scheme the performed comparative analysis evaluates the effect of the neutronics model employed (in terms of numerical methods for solving space- and time-dependent few-group diffusion equations and cross-section modeling) on transient predictions. Other objectives are to establish basis for future optimization of spatial and temporal coupling depending on the nature of analyzed transient, as well as to have a well defined set of benchmark problems for future TRAC-BF1/NEM and TRAC-BF1/ENTREÉ modifications.

2. THE OECD/NEA BWR CORE TRANSIENT BENCHMARKS

The OECD/NEA BWR core transient benchmarks, which were defined in 1992, attracted wide participation.

The reactor core is composed of 185 fuel macro-elements and 64 reflector macro-cells. The definition of a fuel macro-element is a homogeneous average of four real BWR fuel elements with the associated control rod. Axially, the reactor is divided into 14 layers, which results in a total of 3486 neutronics spatial nodes. All nodes have the same size: 30.48x30.48x30.48cm.

Each node in the reactor core has a complete cross-section set associated with it. Each cross-section set has not only a complete set of two-group macroscopic cross sections at the reference point P_0, but also a complete set of derivatives of macroscopic cross sections at the reference point. Therefore, given the local water density and Doppler temperature, the actual macroscopic cross sections for any spatial mesh can be calculated during both steady state and transient calculations by using the polynomial fitting procedure.

Neutron modeling features from the benchmark specifications are as follows: two prompt neutron energy groups and six delayed neutron groups, no delayed energy release, and zero flux boundary conditions.

The total pressure drop in a channel is the sum of the inlet pressure drop and the frictional pressure drop. The inlet orifice diameter is reduced in the peripheral core macro-elements, which leads to a larger inlet pressure drop. The inlet mass flow through the core must be properly distributed to obtain the same total pressure drop across the whole core for steady state runs. Coolant flow through the core is considered as a constant during the transients.

Two classes of problems, core pressurization (case E1) and cold water injection (case D1), were defined for 3-D solutions. At the beginning of the transients, the reactor is in a steady state, the obtained steady state multiplication factor, K_{eff}, should be used to divide the number of neutrons produced per fission in order to obtain a critical steady state. The disturbance for case D1 is an exponential decrease of inlet coolant enthalpy of 46.52KJ/KG with a time constant of 2.5s. Originally case E1 defines a very fast increase of core pressure. Up to this moment, no reported results for case E1 in the open literature are available. Later, a slower, more realistic transient scenario named E2 was defined, which attracted some participation[6].

The problems are to some extent complementary, since the core pressurization case emphasizes instantaneous void collapsing and slower thermal feedback, while the cold-water injection case features simultaneous neutronics and T-H response.

3. TRAC-BF1/NEM AND TRAC-BF1/ENTREE MODELING

The same basic model was used for both TRAC-BF1/NEM and TRAC-BF1/ENTREE. Since boundary conditions are provided for both problems, only the reactor core is modeled, using a coupled 3-D neutronics/core T-H boundary conditions model.

In the thermal hydraulic model, the core is represented by 33 CHAN components, each with a FILL component at the bottom and a BREAK component at the top, resulting in a total of 99 components, which is only one component less than the maximum number of components allowed by TRAC-BF1. It is obvious that each CHAN component should contain only macro-elements of similar thermal hydraulic properties and no CHAN component should contain both types of macro-elements. After an analysis of the results obtained from a simpler model which uses only 25 CHAN components, it was determined that the mapping scheme shown in Figure. 1 is the best mapping scheme.

The inlet pressure drop is modeled by a form loss at the first cell of each CHAN component. Different form loss coefficients are applied depending on whether the inlet orifice is regular or reduced.

TRAC-BF1 has its own way to calculate frictional pressure drop, which is different from the way described in the benchmark specifications. Later it was found that the formula specified in the benchmark overestimates frictional pressure drop by a factor of 3. The surface roughness to hydraulic diameter ratio was set to 0.1 in order to match TRAC-BF1 calculated frictional pressure drop with the formula proposed in the benchmark specifications.

In order to obtain the same total pressure drop for each CHAN component, a mass flow control system, a diagram of which is shown in Figure 2, is implemented in the thermal hydraulic model during the steady-state run. Because mass flow rate is considered as a constant during the transients, this system is removed during transient runs.

Even though the benchmark specifications suggest that a polynomial fitting procedure be used to calculate macroscopic cross sections, the current version of TRAC-BF1/NEM uses a cross section table look-up. An additional program was written to convert the polynomial fitting procedure to the table look-up approach used in TRAC-BF1/NEM algorithm. The current version of TRAC-BF1/ENTRÉE can handle both the polynomial fitting procedure and table look-up approach. In order to determine the impact of converting the polynomial fitting procedure to the table look-up approach on the final results, two ENTRÉE decks were built. In one case the polynomial fitting procedure is used. The second case uses the table look-up approach. In the rest of the paper, “TRAC-BF1/ENTRÉE-1” corresponds to the table loop-up approach and “TRAC-BF1/ENTRÉE-2” corresponds to the polynomial fitting procedure.

4. RESULTS

4.1 STEADY STATE RESULTS

Steady state results produced by TRAC-BF1/NEM and TRAC-BF1/ENTRÉE agree with each other very well and they agree with the results produced by other coupled codes.

TRAC-BF1/NEM predicted a multiplication factor of 0.98322, which is very close to the TRAC-BF1/ENTRÉE-1 result of 0.98271 and the TRAC-BF1/ENTRÉE-2 result of 0.98275. All these predictions fall in the middle of the results predicted by other coupled codes[7] (Figure. 3). TRAC-BF1/NEM predicted a core averaged outlet density of 460.90Kg/m^3. TRAC-BF1/ENTRÉE-1 and TRAC-BF1/ENTRÉE-2 predicted 452.51Kg/m^3 and 452.53Kg/m^3, respectively. Again these results fall in the middle of the results predicted by other coupled codes (Figure 4).

The normalized axial power predicted by TRAC-BF1/NEM matches the result predicted by TRAC-BF1/ENTRÉE-1 or TRAC-BF1/ENTRÉE-2 perfectly, and they are in good agreement with the results predicted by QUABOX-CUBBOX[8] (Figure 5).

Figure 6 to Figure 8 show the radial power distribution at layer number 7 predicted by TRAC-BF1/NEM and TRAC-BF1/ENTREE. Figure 9 to Figure 11 show the difference between TRAC-BF1/NEM and TRAC-BF1/ENTRÉE results and QUABOX-CUBBOX results, which appeared to be trivial.

4.1.1 CASE D RESULTS

The TRAC-BF1/NEM prediction of total relative power response for case D agrees with the TRAC-BF1/ENTRÉE prediction very well, and they are in good agreement with the prediction by DYN3D (Figure.12), which is the only result available in form of electronic data. Because of the lack of electronic data from other participants, the predictions of TRAC-BF1/NEM and TRAC-BF1/ENTRÉE cannot be compared graphically with the results from other coupled codes. However, it should be mentioned that the transient “trend” and the magnitude and timing of total peak power, calculated by TRAC-BF1/NEM or TRAC-BF1/ENTREE, are within the range where most of the reported results are clustered (Table I).

4.1.2 CASE E2 RESULTS

While TRAC-BF1/NEM and TRAC-BF1/ENTRÉE predictions of total relative power response for case E2 agree well with each other, they don't seem to be in good agreement with the prediction of DYN-3D (Figure.13), the only result available for comparison. DYN-3D[9] uses simpler T-H model.

CONCLUSIONS

It could be seen from the comparisons presented that all results produced by TRAC-BF1/ENTRÉE-1 are in excellent agreement with results produced by TRAC-BF1/ENTRÉE-2. This shows that the converting of the polynomial fitting procedure to the table look-up approach has very little effect on the final results in this case. Also noticed is that TRAC-BF1/NEM predictions agree well with those of TRAC-BF1/ENTRÉE, which is evidence that the two neutronics codes employed tend to give similar predictions for reactor transients although they have different features. In general, TRAC-BF1/NEM or TRAC-BF1/ENTRÉE results are within the range where most reported results from other coupled codes are clustered except for case E2, for which not many results are reported.

REFERENCE

1. Shanlai Lu, "GE simplified Boiling Water Reactor Stability Analysis in Time Domain", Ph.D. Thesis, The Pennsylvania State University, 1997.

2. J.A.Borkowski, N.L.Wade, "TRAC-BF1/MOD1: An Advanced Best-Estimate Computer Program for BWR Accident Analysis, Model Description", Idaho National Engineering Laboratory

3. B. B. Bandini, "A Three-Dimensional Transient Neutronics Routine for the TRAC-PF1 Reactor Thermal Hydraulic Computer Code", PhD Thesis, The Pennsylvania State University, 1990.

4. Akitohsi Hotta, Hisashi Ninokata, Anthony Baratta, "Development of Three Dimensional Kinetic Code ENTRÉE Coupled with TRAC-BF1", Ninth International Topical Meeting on Nuclear Reactor Thermal Hydraulics, 1999

5. H. Finnemann and A.Galati, "NEACRP 3-D LWR Core Transient Benchmark Final Specifications", NEACRP-L-335, Revision 1.

6. K. Elja, et al. "Calculation of BWR Transients with the 3D Dynamics Code TRAB-3D", ICONE 5, 1998, USA

7. H. Finnemann, H. Bauer, A. Galati, and R. Martinelli, "Results of LWR Core Transient Benchmarks", NEA/NSC/DOC(93)25, 1993.

8. K. Velkov, GRS, Germany, private communication (1999).

9. U. Gunndmann, Research Center Rossendorf, Germany, private communication (1999)

Table I. Relative total power response of Case D by different coupled codes

Code	Peak power	Time of peak power (s)	Power at end of transient
ARROTTA	1.62	3.1	1.28
DYNAS	1.78	1.6	1.30
TNK-XC	1.43	2.3	1.29
KICOM	2.24	2.1	1.34
QUANDRY-EN	1.52	1.8	1.29
STAND	1.27	4.8	1.18
QUABOX-CUBBOX	1.65	1.6	1.30
TRAC-BF1/NEM	1.88	1.75	1.29
TRAC-BF1/ENTRÉE-1	1.83	1.71	1.28
TRAC-BF1/ENTRÉE-2	1.83	1.71	1.28

	A	B	C	D	E	F	G	H	I	J	K	L	M	N	O	P	Q
1					0	0	0	0	0	0	0	0	0				
2				0	0	94	97	97	91	97	97	94	0	0			
3			0	0	88	82	76	73	70	73	76	79	88	0	0		
4		0	0	88	64	58	55	52	49	52	55	58	61	88	0	0	
5	0	0	88	67	46	40	37	34	31	34	37	40	43	61	88	0	0
6	0	94	85	58	40	28	25	22	19	22	25	28	40	58	79	94	0
7	0	97	76	55	37	25	16	13	10	13	16	25	37	55	76	97	0
8	0	97	73	52	34	22	13	7	4	7	13	22	34	52	73	97	0
9	0	91	70	49	31	19	10	4	1	4	10	19	31	49	70	91	0
10	0	97	73	52	34	22	13	7	4	7	13	22	34	52	73	97	0
11	0	97	76	55	37	25	16	13	10	13	16	25	37	55	76	97	0
12	0	94	79	58	40	28	25	22	19	22	25	28	40	58	82	94	0
13	0	0	88	61	43	40	37	34	31	34	37	40	46	64	88	0	0
14		0	0	88	61	58	55	52	49	52	55	58	67	88	0	0	
15			0	0	88	79	76	73	70	73	76	85	88	0	0		
16				0	0	94	97	97	91	97	97	94	0	0			
17					0	0	0	0	0	0	0	0	0				

Figure.1 Thermal Hydraulic Channel Radial Map

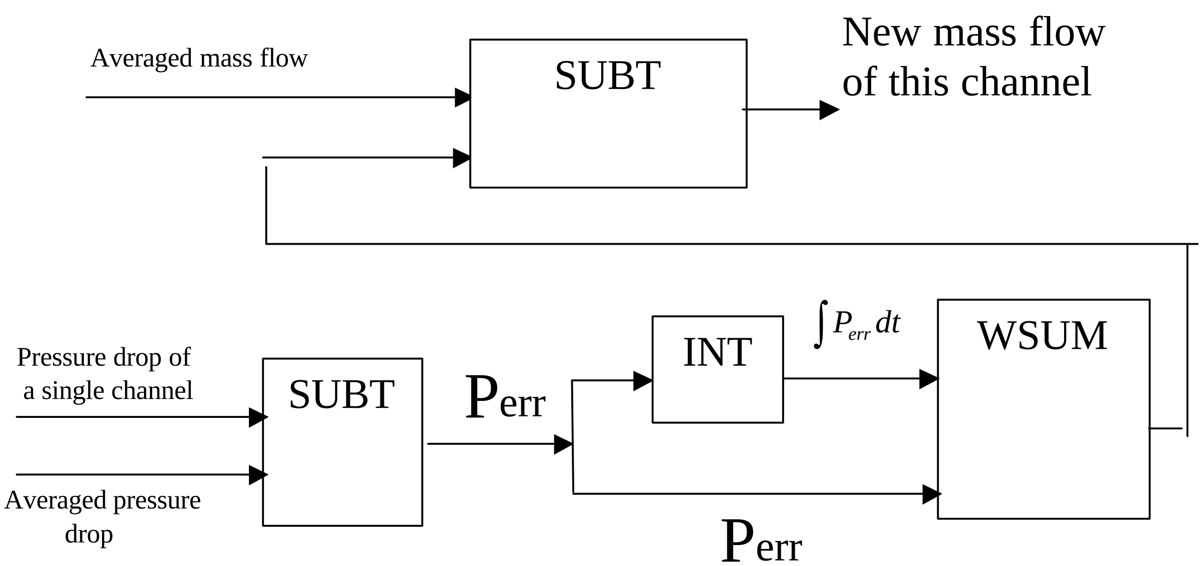

Figure.2 Inlet mass flow rate control system

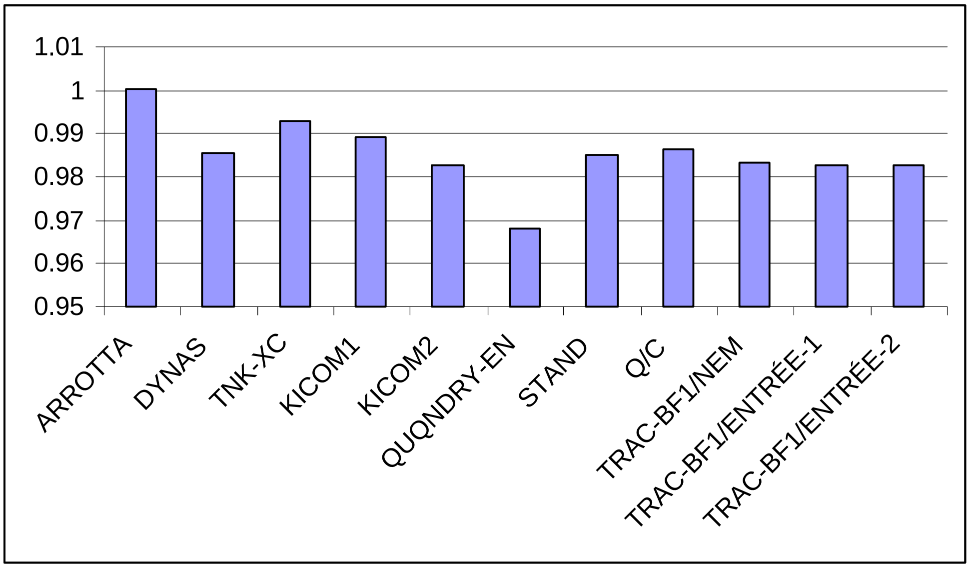

Figure.3 Calculated multiplication factor by different coupled codes

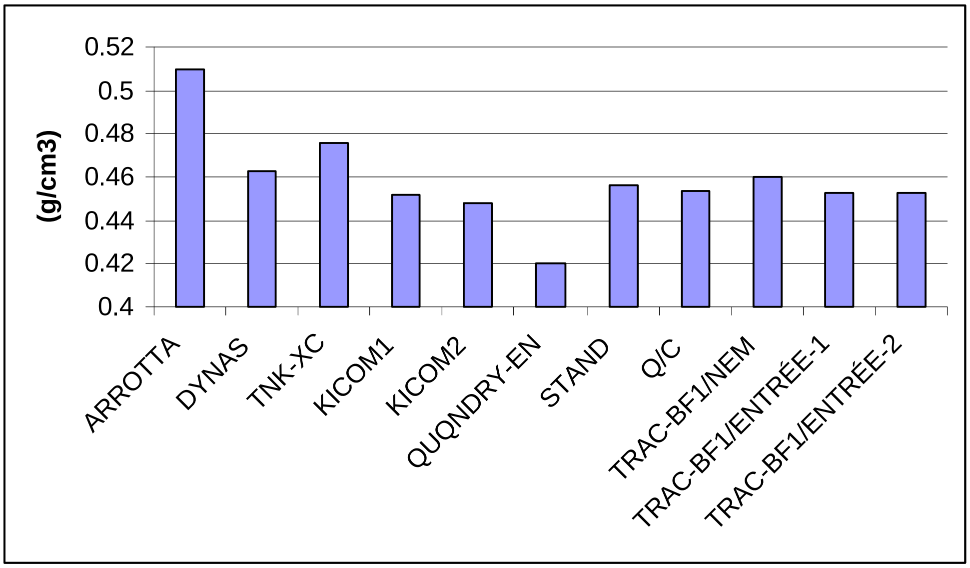

Figure.4 Calculated core averaged outlet density by different coupled codes

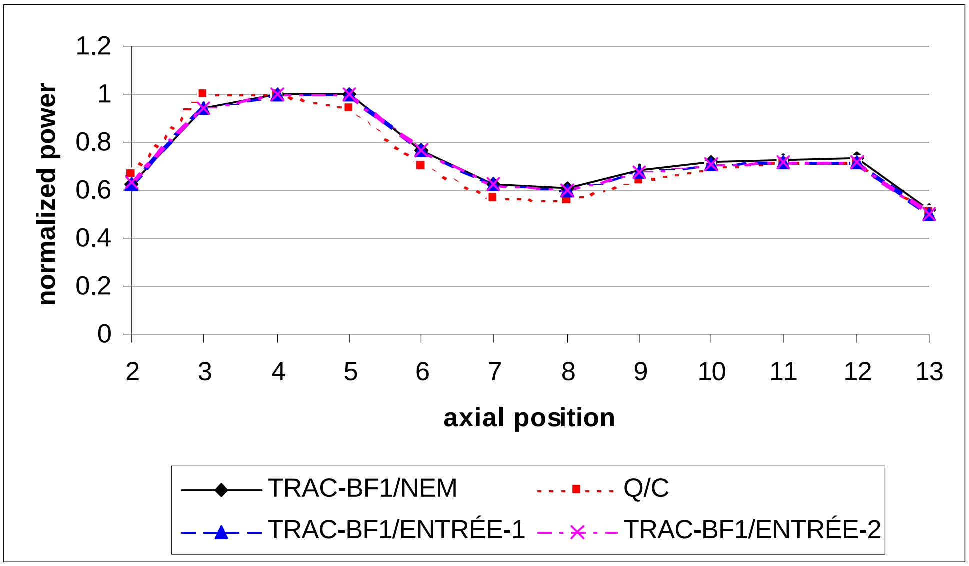

Figure.5 Normalized axial power distribution

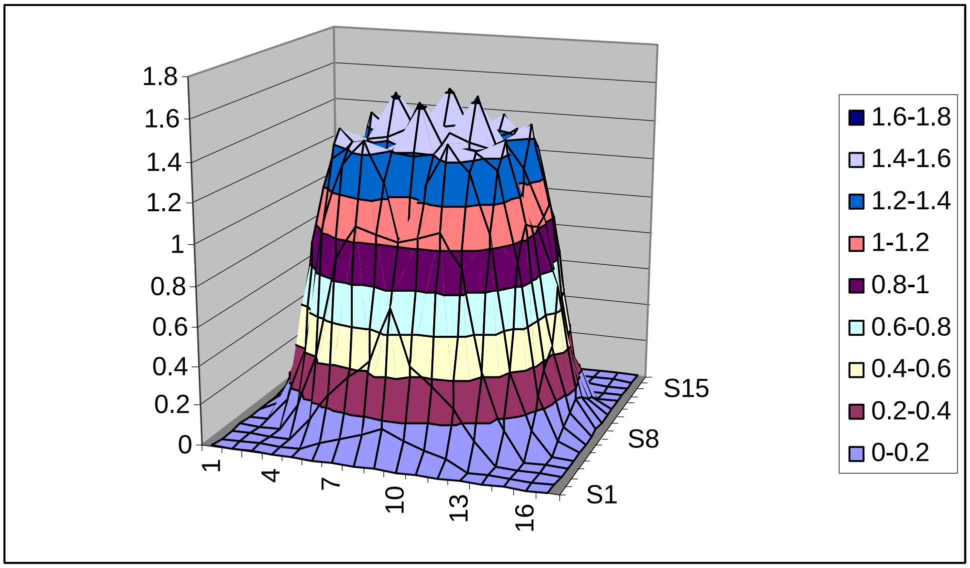

Figure.6 Normalized radial power distribution at layer #7 by TRAC-BF1/NEM

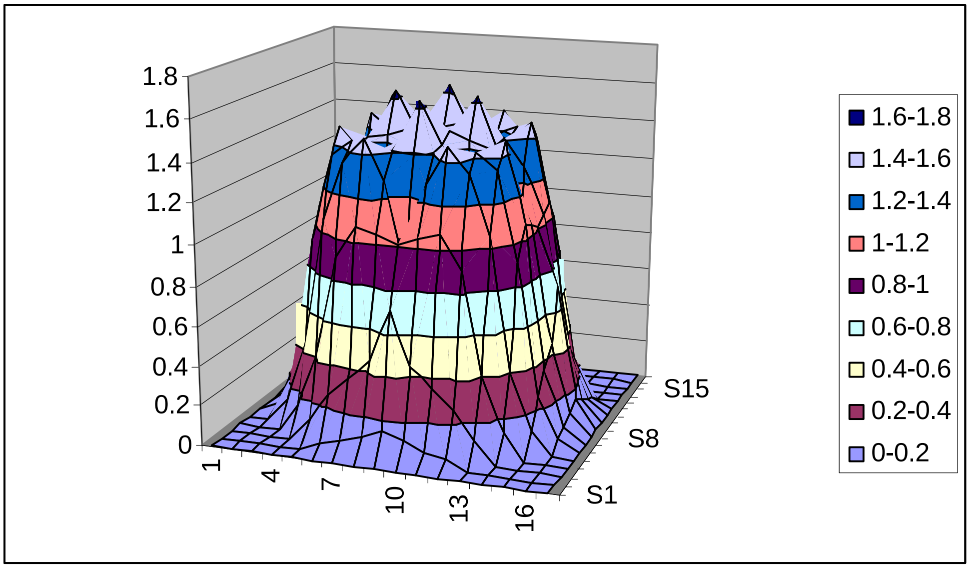

Figure.7 Normalized radial power distribution at layer #7 by TRAC-BF1/ENTRÉE-1

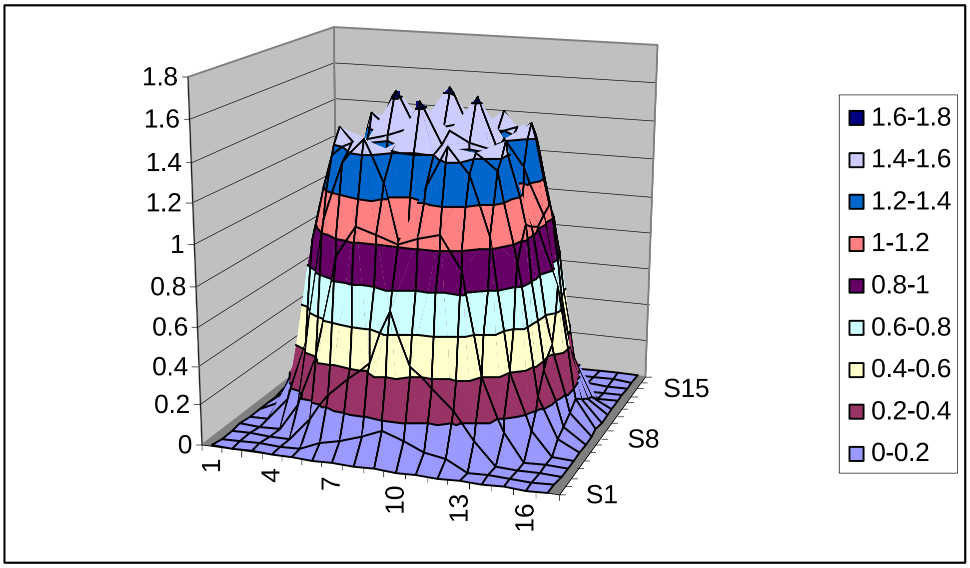

Figure.8 Normalized radial power distribution at layer #7 by TRAC-BF1/ENTRÉE-2

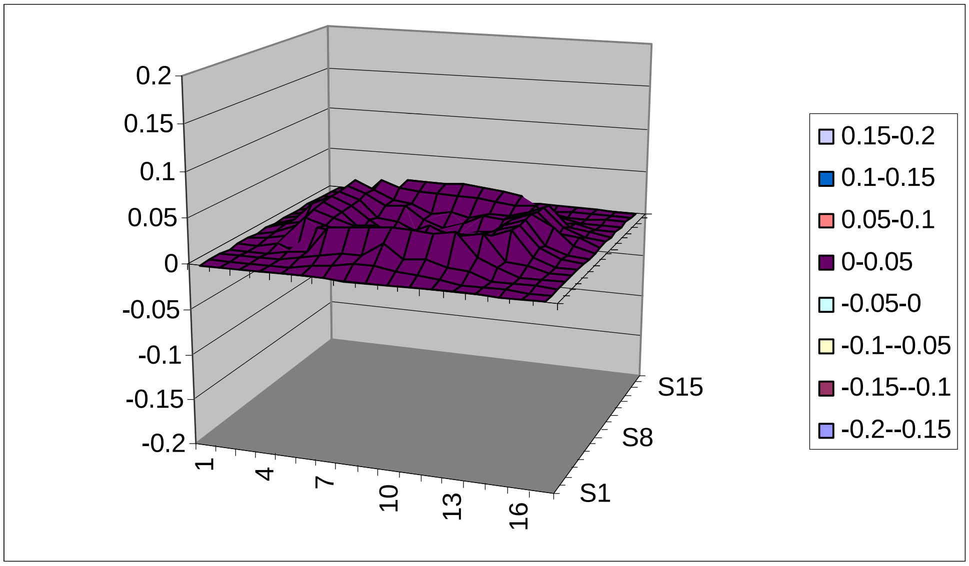

Figure.9 Difference of calculated normalized radial power distribution at layer #7 by TRAC-BF1/NEM and Q/C

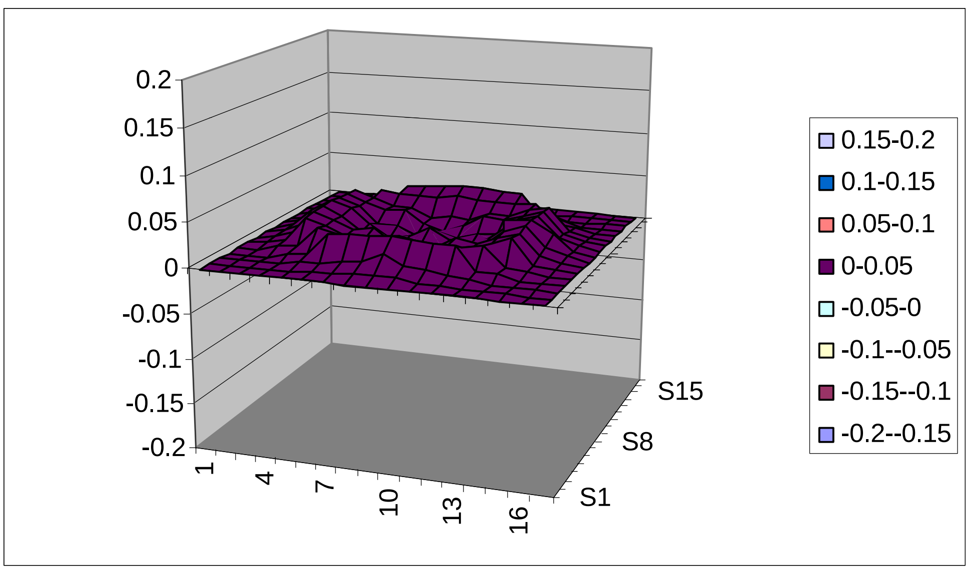

Figure.10 Difference of calculated normalized radial power distribution at layer #7 by TRAC-BF1/ENTRÉE-1 and Q/C

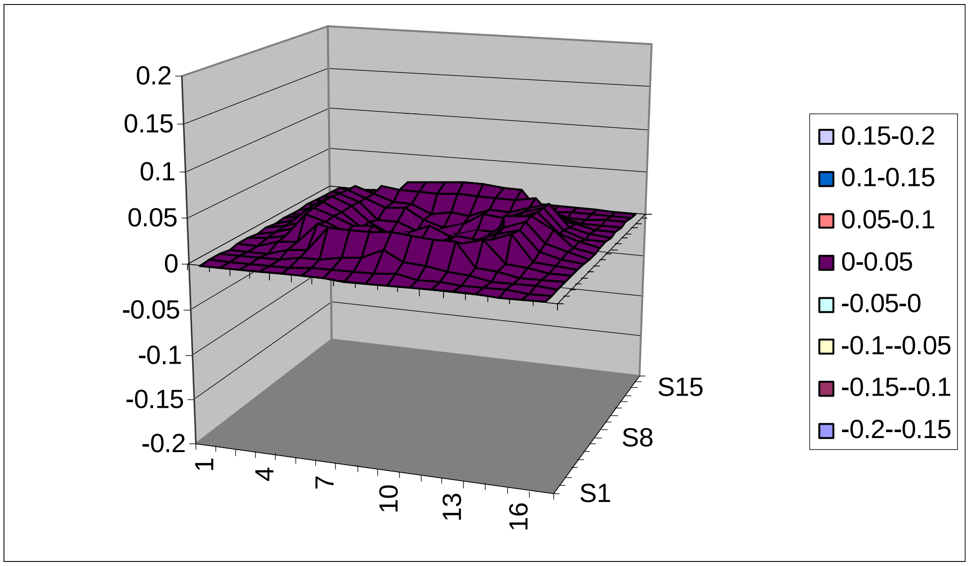

Figure.11 Difference of calculated normalized radial power distribution at layer #7 by TRAC-BF1/ENTRÉE-2 and Q/C

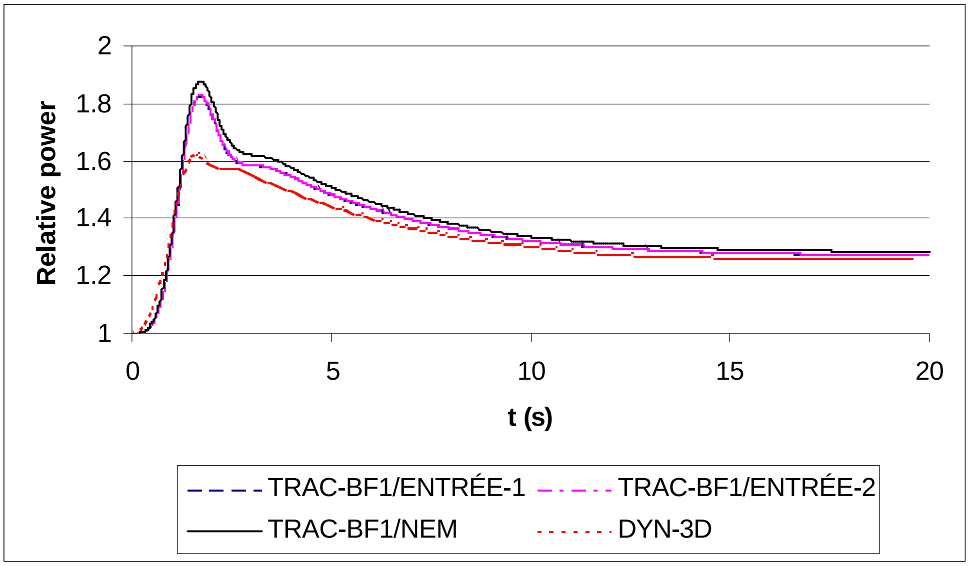

Figure.12 Total relative power response for Case D

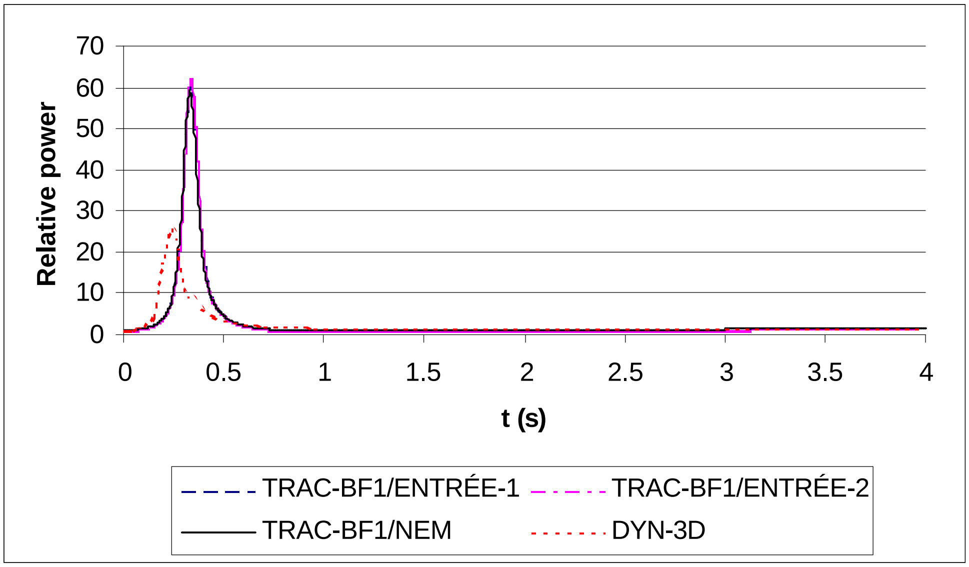

Figure.13 Total relative power response for Case E2