METHODOLOGY FOR OPTIMAL GROUPING OF THERMAL HYDRAULIC CHANNELS IN 3D KINETICS

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ABSTRACT

The proposed methodology will manage to the choice of the thermal-hydraulic channels that group fuel assemblies with the same reactivity response, not only in the steady state, but also in the transient. This choice have into account the supposition that the thermal-hydraulic behaviour is similar in all these assemblies during the transient (mass flow, temperatures distribution, voids distribution, losses coefficients, etc). In this way, the purpose is to get a thermal-hydraulic core nodalization with the smallest number possible of thermal-hydraulic channels but sufficient for characterising properly the neutronic and/or thermal-hydraulic response by using a preset criterion.

One of the effects to be proved quantitatively in this paper is that if only one thermal-hydraulic channel is used for representing different fuel assemblies (different reactivities), the global effect on the final response in the thermal-hydraulic channel is averaged, masking the biggest reactivities; and therefore attenuating the feedback effect.

1. INTRODUCTION

By means of the study of some exercises proposed in the Peach Bottom Turbine Trip benchmark [3], we have tested the developed thermal-hydraulic and neutronics coupling (TRAC/BF1-NOKIN3D). In this paper we have studied the core-plant interaction and present a description of the numerical methods applied in the neutronics module. Secondly, a brief description of the coupling has been made. In this point and in order to test the coupling, exercises one and three of “Peach Bottom benchmark” have been result. With the purpose of simplifying the thermal-hydraulics core model, a methodology has been developed to choice the number of channels that define the core in an acceptable precision. To test the proposed methodology, we have built different thermal-hydraulic core models (different number of channels and different spatial distribution). In exercise one, we have obtained steady state results like axial void fraction and axial pressure profile. On the other hand, we present the lower plenum and the upper plenum pressure results to characterise the transient behaviour.

2. BOILING WATER REACTOR TURBINE TRIP (TT) BENCHMARK DESCRIPTION

Recently, appropriate Light Water Reactor transient benchmarks have been developed on a best-estimate level in order to verify the capability of coupled codes to analyse complex transients having into account coupled core-plant interaction and testing the thermal-hydraulic couplings.
In this way, the Nuclear Energy Agency (NEA-NRC) has developed a Boiling Water Reactor Turbine Trip (TT) Benchmark. This benchmark consists of a transient, which begins with a sudden turbine stop valve closure. The pressure oscillation generated in the main steam piping propagates with little attenuation into the reactor core. The induced core pressure oscillation results in dramatic changes of the core void distribution and fluid flow. At high power level production, the valve stop closure provokes the control rod insertion (scram).

This benchmark consists of three different exercises, but in this paper we only have into account exercises one and three due to interaction core-plant.
Exercise one: Power versus time plant system simulation with fixed axial power profile table (obtained from experimental data).
Exercise three: Best-estimate coupled 3D core/thermal-hydraulic system modelling.

3. COUPLED CODE SYSTEM TRAC/BF1-NOKIN3D

3.1 NOKIN3D module

The starting point is the standard time-dependent two groups neutron diffusion equation,

$$\left[v^{-1}\right]\frac{\partial\phi}{\partial t} + L\phi = (1-\beta)M\phi + \sum_{k=1}^{K}\lambda_k C_k \chi \,, \tag{1}$$

$$\frac{\partial C_k}{\partial t} = \beta_k\left[\nu\Sigma_{f1}\ \nu\Sigma_{f2}\right]\phi - \lambda_k C_k \,, \quad k = 1,\ldots,K, \tag{2}$$

where, K is the number of delayed neutron precursors considered,

$$L = \begin{bmatrix} -\nabla\cdot\left(D_1\nabla\right)+\Sigma_{a1}+\Sigma_{12} & 0 \\ -\Sigma_{12} & -\nabla\cdot\left(D_2\nabla\right)+\Sigma_{a2} \end{bmatrix}, \quad \left[v^{-1}\right] = \begin{bmatrix} \frac{1}{v_1} & 0 \\ 0 & \frac{1}{v_2} \end{bmatrix}, \tag{3}$$

and

$$M = \begin{bmatrix} \nu\Sigma_{f1} & \nu\Sigma_{f2} \\ 0 & 0 \end{bmatrix}, \quad \phi = \begin{bmatrix} \phi_1 \\ \phi_2 \end{bmatrix}, \quad \chi = \begin{bmatrix} 1 \\ 0 \end{bmatrix}, \tag{4}$$

To discretize the spatial part of these equations, the reactor core is divided in N neutronic nodes and a nodal collocation method [4], [5] is applied. This method makes use of an expansion of the neutron flux in each node in terms of Legendre polynomials and it allows to approximate equations (1) and (2) by the following set of ordinary differential equations

$$\left[v^{-1}\right]\frac{\partial\psi}{\partial t} + L\psi = (1-\beta)M\psi + \sum_{k=1}^{K}\lambda_k X C_k \,, \tag{5}$$

$$X\frac{\partial C_k}{\partial t} = \beta_k M\psi - \lambda_k X C_k \,, \tag{6}$$

where now ψ and C_k are vectors whose components are the Legendre coefficients of the expansions of ϕ and C_k in each node, at each time step. L and M have the following block structure

$$L = \begin{bmatrix} L_{11} & 0 \\ -L_{21} & L_{22} \end{bmatrix}, \quad M = \begin{bmatrix} M_{11} & M_{12} \\ 0 & 0 \end{bmatrix}, \; X = \begin{bmatrix} I \\ 0 \end{bmatrix}. \tag{7}$$

The dominant Lambda modes for a given configuration of a nuclear power reactor are obtained as a solution of a partial eigenvalue problem of the form

$$AX = X \,, \; A = L_{11}^{-1}\left(M_{11} + M_{12}L_{22}^{-1}L_{21}\right) \tag{8}$$

and X is a matrix, which has as columns the dominant eigenvectors associated to the dominant eigenvalues of matrix A.
L_{11} and L_{22} are not symmetrical, due to the fact that the assembly discontinuity factors (ADF) effect. This is an important point because it increases the problem dimension.

Recently, Sorensen et al. (Sorensen, 1992), (Lehoucq *et al.*, 1995) have developed an Arnoldi Method [8] with implicit shifted QR iteration (IRA). This a very efficient implementation of Arnoldi Method. We have adapted this method to our problem, including in the algorithm initial starting guess and restart options. The eigenvalues calculation algorithm has been implemented in the ARPACK software package (Lehoucq *et al.*, 1995), and the calculation of the dominant Lambda modes has been done with a program written in FORTRAN, called LAMBDA, which calls this library.

For the time dependent part [9], the set of ordinary differential equations (5) and (6) is integrated over a series of time steps $[t_n, t_{n+1}]$.
To simplify the calculations, Eq. (6) is integrated under the assumption that the term $[M_{11}\ M_{12}]\psi$ varies linearly from t_n to t_{n+1}, obtaining the solution C_k at t_{n+1}

$$C_k^{n+1} = C_k^n e^{-\lambda_k h} + \beta_k\left(a_k\begin{bmatrix}M_{11} & M_{12}\end{bmatrix}^n\psi^n + b_k\begin{bmatrix}M_{11} & M_{12}\end{bmatrix}^{n+1}\psi^{n+1}\right) \tag{9}$$

Where $h = t_{n+1} - t_n$, and the coefficients a_k and b_k are given by

$$a_k = \frac{\left(1+\lambda_k h\right)\left(1-e^{-\lambda_k h}\right)}{\lambda_k^2 h} - \frac{1}{\lambda_k} \,, \; b_k = \frac{\lambda_k h - 1 + e^{-\lambda_k h}}{\lambda_k^2 h} \tag{10}$$

To integrate Eq. (5), we must take into account that it constitutes a system of stiff differential equations, mainly due to the elements of the diagonal matrix [v^{-1}]. This requires the use of an implicit backward difference formula (BDF) for its integration, resulting in a system of linear algebraic equations to solve at each time step.

$$\frac{\left[v^{-1}\right]}{h}\left(\psi^{n+1}-\psi^{n}\right)+L^{n+1}\psi^{n+1}=(1-\beta)M^{n+1}\psi^{n+1}+\sum_{k=1}^{K_c}\lambda_k X C_k^{n+1} \qquad (11)$$

Taking into account Eq. (9) and the structure of matrices L and M, we rewrite (11) as the system of linear equations

$$\begin{bmatrix} T_{11} & T_{12} \\ T_{21} & T_{22} \end{bmatrix}\begin{bmatrix} \psi_1^{n+1} \\ \psi_2^{n+1} \end{bmatrix}=\begin{bmatrix} R_{11} & R_{12} \\ 0 & R_{22} \end{bmatrix}\begin{bmatrix} \psi_1^{n} \\ \psi_2^{n} \end{bmatrix}+\sum_{k=1}^{K_c}\lambda_k e^{-\lambda_k h}\begin{bmatrix} C_k^{n} \\ 0 \end{bmatrix}, \qquad (12)$$

where

$$T_{11}=\frac{I}{h}v_1^{-1}+L_{11}^{n+1}-(1-\beta)M_{11}^{n+1}-\sum_{k=1}^{K_c}\lambda_k\beta_k b_k M_{11}^{n+1},\ \ T_{21}=-L_{21}^{n+1}$$

$$T_{12}=-(1-\beta)M_{12}^{n+1}-\sum_{k=1}^{K_c}\lambda_k\beta_k b_k M_{12}^{n+1},\ \ T_{22}=\frac{I}{h}v_2^{-1}+L_{22}^{n+1} \qquad (13)$$

$$R_{11}=\frac{I}{h}v_1^{-1}+\sum_{k=1}^{K_c}\lambda_k\beta_k a_k M_{11}^{n+1},\ \ R_{12}=\sum_{k=1}^{K_c}\lambda_k\beta_k a_k M_{12}^{n},\ R_{22}=\frac{I}{h}v_2^{-1}$$

3.2 TRAC/BF1-NOKIN3D coupling

In this coupling process, we have used TRAC/BF1 code as thermal hydraulic module [1]. Thermal hydraulics processes are simulated solving the mass, energy and momentum balance equations for liquid and steam phases. The spatial part of the equations has been discretized with a first order finite difference method with staggered mesh and the time discretization with a semi-implicit two step method.

In a first step, to obtain the power distribution with NOKIN3D module, nuclear cross sections are needed. By means of the fuel temperature, moderator density and control rod insertion pattern, a nuclear cross sections set associated to each neutronic node is obtained interpolating in tables.

Secondly, this power distribution is used as an input for TRAC/BF1 to obtain the new set of thermal hydraulic variables.

4. THERMAL-HYDRAULIC CHANNELS GROUPING

In order to simplify the thermal-hydraulic model, different fuel assemblies could be grouped under the same channel. However, and due to the fact that every fuel assembly has its own reactivity and thermal-hydraulics properties, after the group process, an averaged value for each variable and for the reactivity is obtained in every channel. The global effect on the final response in the thermal-hydraulic channel is averaged, masking the biggest reactivities; and therefore attenuating the feedback effect.

TRAC/BF1 has a main restriction in the definition of thermal-hydraulic components (only 100 component definition is allowed). This limitation affects to the total number in channels that can be included in the model. In this way, the main goal of this work is to obtain the number of channels, which reproduces in an acceptable level precision neutronics and thermal-hydraulics core behaviour.

The proposed methodology will manage to the optimum choice of the thermal-hydraulic channels that group these fuel assemblies with the same reactivity response, not only in the steady state but also in the transient.
Furthermore, this choice is made under the supposition that the thermal-hydraulic behaviour is quite similar in all these assemblies during the transient (mass flow, temperature distribution, void distribution, loss coefficients, etc). In this way, the purpose is to get a thermal-hydraulic core nodalization with the smallest number possible of channels but sufficient for characterising properly neutronics and/or thermal-hydraulics response by using a preset criterion.

To apply this methodology, a local study of reactivities is required for the neutronic regions before the same thermal-hydraulic perturbations. It is also convenient the study of the variation of the thermal-hydraulic conditions (mass flow, temperatures distribution, void distribution, loss coefficients, etc) according their position in the core.

The core-simplified cross sections are obtained from the data obtained by the CASMO/SIMULATE physics code, provided by the benchmark team. This first phase tries to check that the 3D neutronic map in the core, adopted in the "Peach Bottom" benchmark characterises sufficiently the core neutronic behaviour. Each cross section set has been obtained from a data table by means of a linear interpolation process (fuel temperature vs. moderator density).

The process of validation of this methodology consists in the comparison of the steady and transient states. In the steady state case, the thermal-hydraulic part and the neutronic part are checked separately; always taking as reference the experimental data provided in the “Turbine Trip (TT) Benchmark”. In order to test this methodology, different thermal-hydraulic core models have been built as an input of TRAC/BF1-NOKIN3D coupled code. These models are described in the next chapter of the paper.

A series of perturbations are carried out over each neutronic composition, observing the feedback responses obtained. Moreover, other phenomena have been studied like xenon concentration effect (variation in absorption cross sections for each neutronic composition), moderator density variation produced by the existence of bypass mass flow and assemblies discontinuity factors (ADFs).

From the steady state obtained for each one of the different configurations, the axial power profile and the axial void fraction profile (and also other parameters like fuel temperature, etc…), of each one of the thermal-hydraulic channels (that groups several fuel assemblies) are compared. The same profile is also obtained for the whole core. For the different configurations (different number of channels and different distributions), the power transferred to each neutronic composition is calculated for each channel, obtaining then the core total power.

Thermal hydraulic models

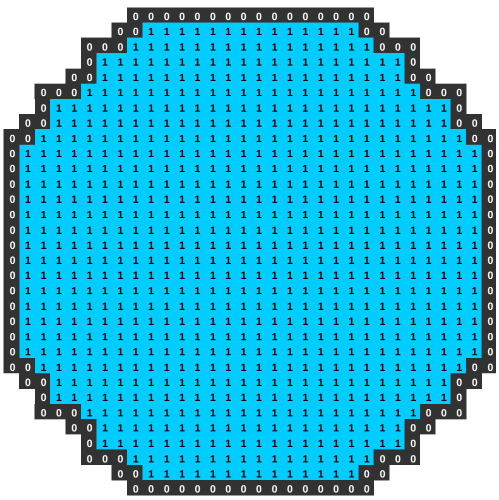
In order to observe the global effect when the number of thermal-hydraulic channels vary, four different core models have been built, thus, the feedback between neutronics and thermal-hydraulics can be characterized by choosing mapping schemes in the radial and axial core planes. In all models, homogeneous power distribution has been had into account and has been obtained by means of exercise two.

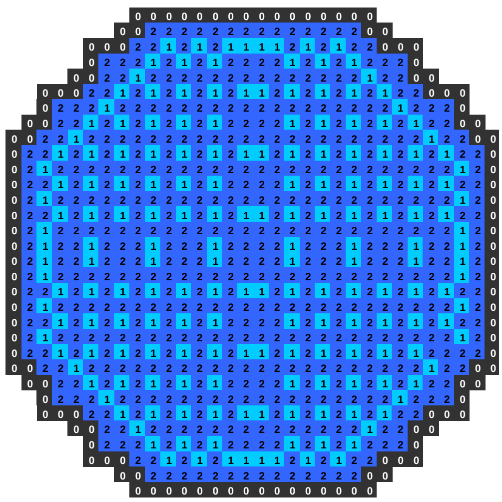
Model 1: It consists of two thermal hydraulic channels: one of them represents all fuel assemblies (764 in Peach Bottom) and the other one models a bypass channel.

Model 2: This model has into account the fuel assembly type (e.g., 7x7 or 8x8 fuel rod arrays). In this way, we distinguish three T-H channels: bypass, channel related to 7x7 (576) and channel related to 8x8 (188) fuel assemblies. Depending on the fuel rod array type, loss coefficients will be different.

Model 3: It considers two different channels related to 7x7 fuel rod arrays, according to its radial position. Then, this model consists of four channels: bypass, 8x8, 7x7 peripheral and 7x7 incore.

Model 4: 33 thermal-hydraulic channels and a bypass channel. This distribution has been supplied by the benchmark specifications.





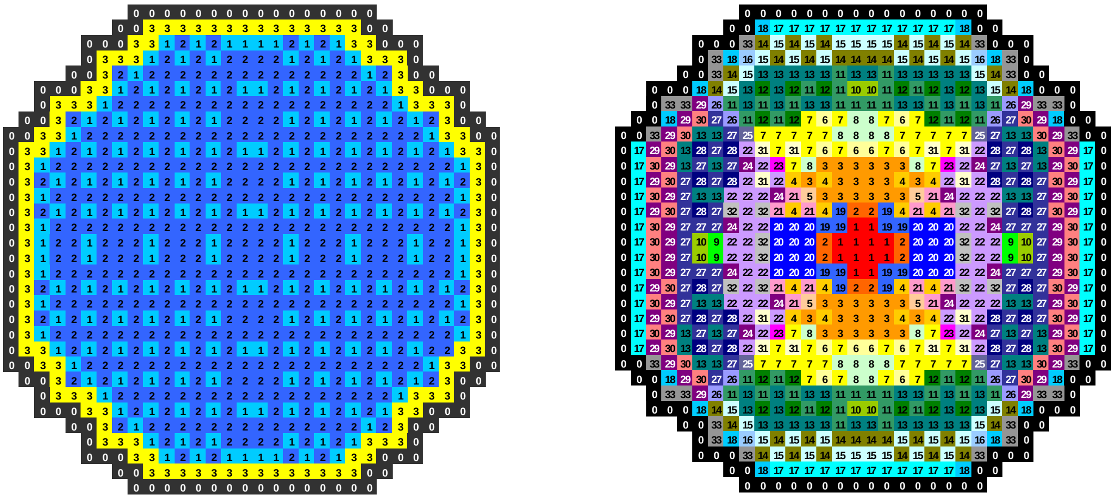


Figure 1: Thermal-hydraulic core models.

5. SENSITIVITY ANALYSIS

In this section we have made a sensitivity analysis in order to obtain a relationship between the upper plenum pressure peak and the geometrical features of the bypass valve and the turbine stop valve. We have had into account the length and the flow area of these valves. Moreover, the influence of the turbine break pressure over the initial pressure has been analysed. Figure 2 shows snap shots corresponding to initial time and time 0.85 seconds.

We can observe that the upper plenum pressure increases when the bypass valve length grows up. However the initial pressure reaches a plateau. Secondly, both the upper plenum pressure and the initial pressure are decreasing as flow area turbine stop valve increases. Finally, when the turbine stop valve length increases, the upper plenum pressure and the initial pressure grow up.

Upper and dome pressure depend on the turbine break pressure (PIN). When PIN grows up, pressure difference between upper and dome decreases. Moreover, upper plenum pressure depends on loss coefficients values. In this way, when we increase loss coefficients, upper plenum pressure grows up.

In figure 3, the influence of the bypass valve length over the upper plenum pressure transient has been represented. The 71 meters curve offers the best agreement with the average reference. Moreover, we can observe than the upper plenum pressure grows up when the bypass valve length value increases.

By means of this sensitivity analysis we have characterised the geometrical feature of the bypass valve and the turbine stop valve. In this way, the length and the flow area values obtained are included in the input TRAC/BF1 file for the following comparison work.

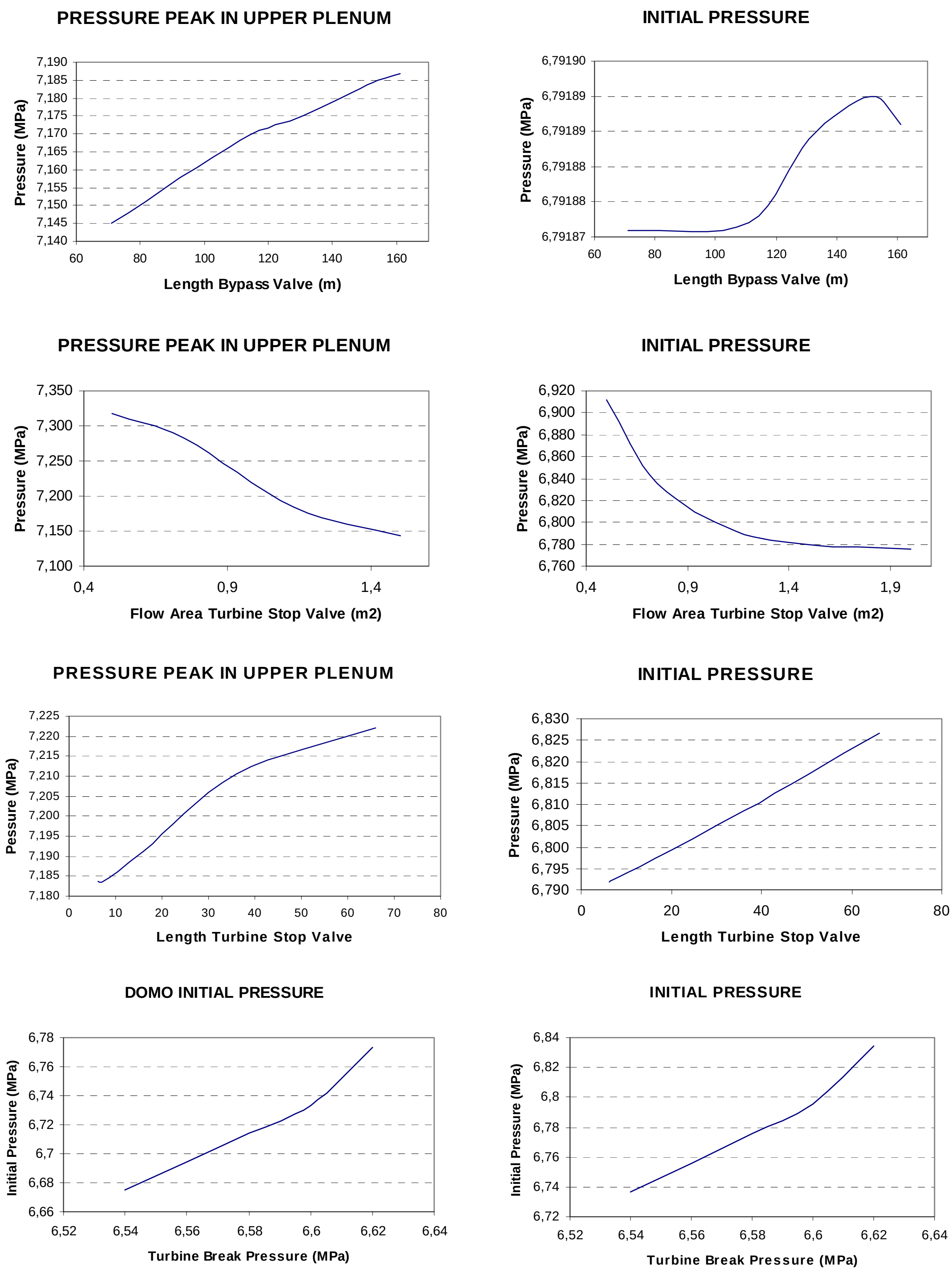


Figure 2. Sensitivity analysis results

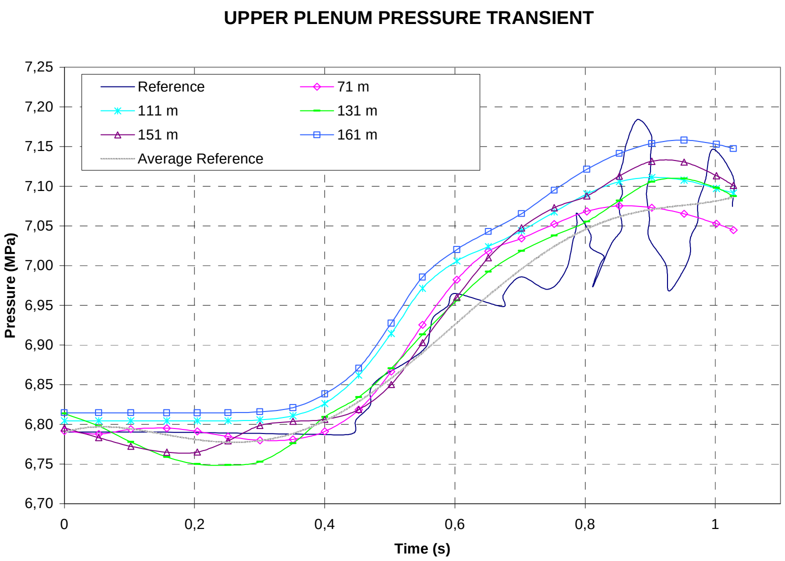


Figure 3. Sensitivity analysis results. Upper plenum pressure transient.

6. STEADY STATE RESULTS

In boiling water reactor, some of the most important variables in order to characterise neutronics and thermal-hydraulic core behaviour are the pressure distribution and the axial void fraction distribution. In this section we have studied the developed T-H models by means of these variables. Comparison results are showed in the next figures. In all models, the behaviour inside the core is quite similar.

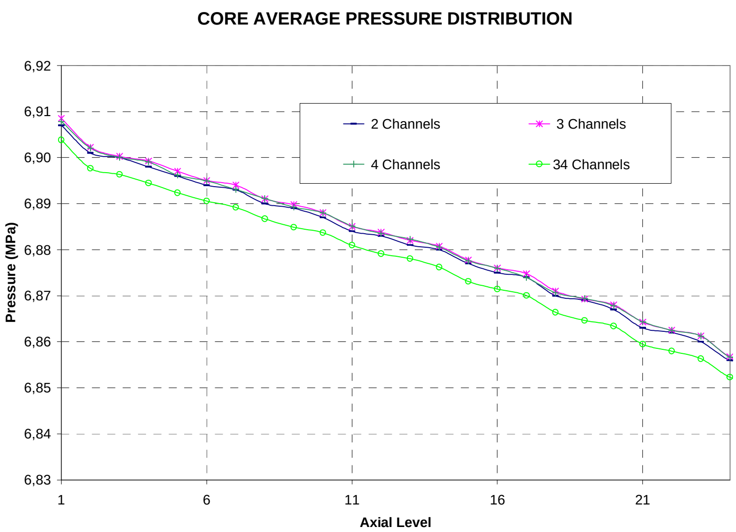


Figure 4. Steady state results. Core average pressure distribution

Table 1: PB2 TT2 initial conditions from process computer

	REF.	2 CHAN	3 CHAN	4 CHAN	34 CHAN
Feedwater Flow (kg/s)	980.26	986.9	986.9	987.2	985.8
Total Core Flow (kg/s)	10445	10500	10460	10190	10420
Jet Pump Driving Flow (kg/s)	2871.24	2876	2874	2863	2882
Bypass Flow (kg/s)	854	875.4	877.8	912.3	889.7
Dome Pressure (MPa)	6.798	6.800	6.801	6.801	6.801
Core Average Void Fraction	0.304	0.323	0.302	0.300	0.318

In the above table the initial conditions of some operations values are showed. These values are referred to the feed water flow, the total core flow, jet pump driving flow, bypass flow, dome pressure and core average void fraction. The reference and the calculated data are represented.

Figure 5 shows the axial void fraction distribution results. We can observe that models with 3 and 4 channels offer quite similar distributions. This result indicates us that there is no great differences if we have into account two 7x7 channels or only one 7x7 channel (according to its radial position inside the core). The main difference between model 1 (2 T-H channels) and model 2 (3 T-H channels) is the fact that we have distinguished fuel assemblies according to its fuel rod array (7x7 or 8x8). This modification does not represent an important difficult to build the input TRAC/BF1 file. However, taking as a reference the model 4 curve, we can observe that model 1 results are better than model 2 and 3. Also, we can observe that the most important differences between these curves are located in upper levels.

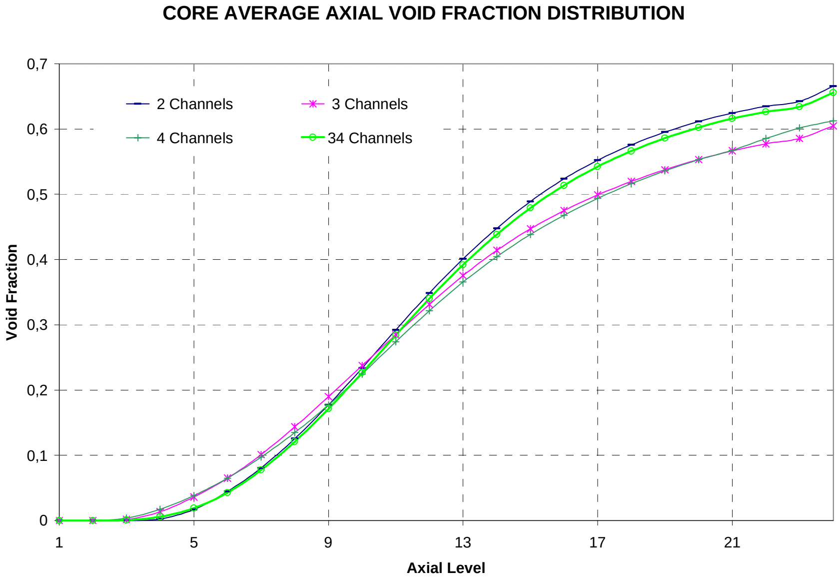


Figure 5. Steady state results. Core average axial void fraction distribution

7. TRANSIENT RESULTS

In the section, upper plenum and dome pressure have been studied, in order to show the effect when varying the thermal-hydraulic model considered. In the same way than the previous section, we have considered models with 2, 3, 4 and 34 thermal-hydraulic channels. In figures 6 and 7 the pressure difference transient in upper plenum and dome is represented. The represented value has been obtained by means of subtracting to the transient pressure results, the initial pressure.

As we can see from figure 6, it can be said that all models (1, 2, 3 and 4) present a very similar behaviour. The reference curve is characterised by important oscillation in pressure that makes it very difficult to approximate the value obtained by each model to the real one.

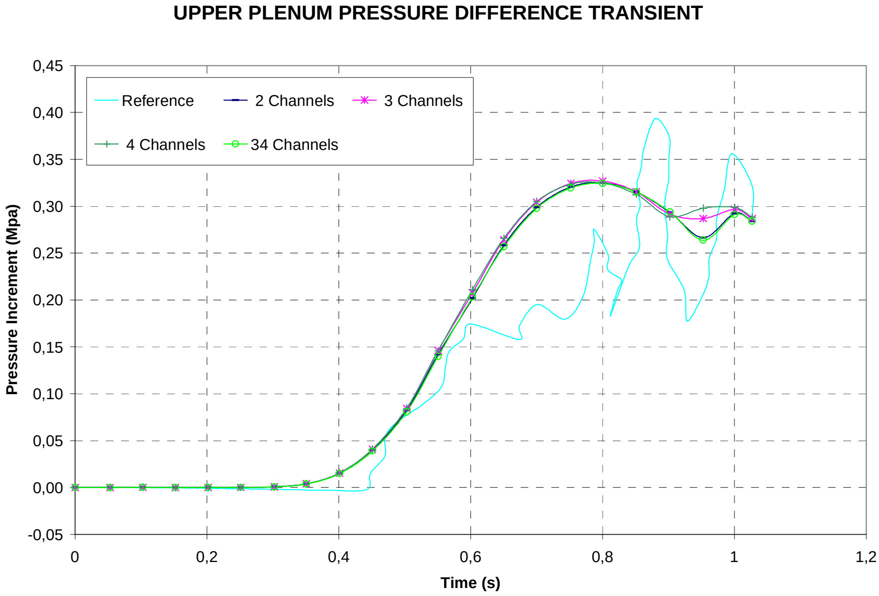


Figure 6. Transient results. Upper plenum pressure

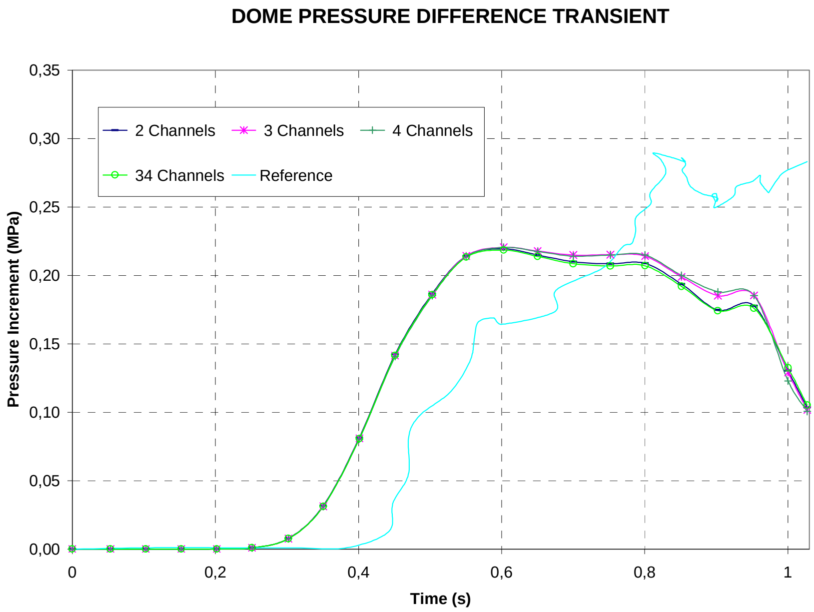


Figure 7. Transient results. Dome pressure

8. CONCLUSIONS

In this paper, we have presented the coupled code for BWR reactor analysis TRAC/BF1-NOKIN3D code. Comparison of thermal-hydraulic models has been realised studying the "boiling water reactor turbine trip benchmark". The TRAC/BF1 input file (TRACIN) elaboration has been a difficult task from the point of view of determining the geometrical features of some valves (bypass and turbine stop). These optimal values have been obtained by means of a sensitivity analysis. A restriction has been taken into account in characterising valves geometrically. In this way, the volume of every valve has been maintained invariable when varying either the length or the flow area.

The model that offers the best results according to the similarity to the reference pressure transient is the corresponding to 34 channels. However, the difference between this model and the other models, is not very important.

We can conclude that for this transient the coupling between the thermal-hydraulic module and the 3D kinetics neutronics module is very weak; therefore it could be unnecessary to use a full 3D kinetics. Also, a detailed nodalization of the core is not very important, confirming that the optimal grouping of thermal-hydraulic channels depends strongly of the transient analysed. We believe that a full 3D kinetics and a detailed nodalization will play an important role in the extreme scenario of the exercise 3.

9. REFERENCES

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