A PROCEDURE FOR COUPLED THERMAL-HYDRAULIC SUBCHANNEL AND NEUTRONIC CODES USING COBRA-TF AND PARCS

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ABSTRACT

A code system based on coupling the 3D neutron diffusion code PARCS v2.7 and the thermal-hydraulic sub-channel code COBRA-TF (CTF) has been developed as a tool for detailed nuclear reactor core calculations.

The increased importance of detailed reactor core and fuel assembly description for light water reactors (LWRs) as well as the sub-channel safety analysis requires coupled neutronic/thermal-hydraulic codes. The methodology to simulate a 3D-neutronic problem coupled with 1D thermal-hydraulics based on system and/or channel based codes has already been developed in the past, and is applied regularly in the analysis of different kind of asymmetric transients.

The transport of neutrons depends on several parameters, like fuel temperature, moderator temperature and density, boron concentration and control rod insertion. These data are calculated by the CTF code with high local resolution and passed to the neutronic code for calculating a 3D nodal power distribution that will be returned to the thermal-hydraulic code.

Since two different nodalizations are used to discretize the same physical system, a physically consistent averaging and interpolating procedure is needed to realize an effective data exchange. This procedure has been developed based on General Interface routines offered by PARCSv2.7 and the data exchange by means of Parallel Virtual Machine (PVM) software package.

Results of a steady state calculation of a single PWR fuel assembly are presented. The results are compared with calculations performed with PARCS code alone with fixed thermal-hydraulics boundary conditions and with coupled RELAP5/PARCS in order to validate the results obtained with COBRA-TF/PARCS and to assess the consistency and correctness of the coupling and calculation procedure.

1. INTRODUCTION

Nowadays the use of coupled three-dimensional (3D) kinetics/thermal-hydraulics coupled codes for best-estimate nuclear safety calculations is a common practice. Nevertheless, as the software and computer technology advances, the evaluation on local pin level has been introduced to operate nearest of the real safety margins.

In this work, in a first step for develop a system code for local pin level calculations have been performed coupling the 3D neutron diffusion code PARCS v2.7 and the thermal-
hydraulic sub-channel code COBRA-TF (CTF) [1], [2]. The system code RELAP5 supply CTF with static boundary conditions.

The neutronic code needs on its calculations several parameters like fuel temperature, moderator temperature and density and boron concentration. These data are calculated by the CTF code with high local resolution and passed to the neutronic code for calculating a 3D nodal power distribution that will be returned to the thermal-hydraulic code.

Since two different nodalizations are used to discretize the same physical system, a physically consistent averaging and interpolating procedure have been developed to realize a consistent data exchange. This procedure has been implemented on CTF and lets an automatic mapping between the two different nodalizations used.

The coupling algorithm is developed based on General Interface routines offered by PARCSv2.7 and the data exchange by means of Parallel Virtual Machine (PVM) software package [3]. To assess the consistence of the coupling algorithm a steady state and an inlet mass flow transient have been simulated with PARCS coupled RELAP and CTF.

2. COUPLING ALGORITHM

This coupling scheme, between COBRA-TF and PARCSv2.7, is a first step into multi-level coupling algorithm that uses the parameters calculated by a thermal-hydraulic system code as boundary conditions of sub-channel code. The data exchange is done by means of PVM environment.

The coupling and auto-mapping routines implemented in CTF introduce a coupling environment that permits the use of coupled codes CTF/PARCS on several cases: in case of a group of fuel elements, whole core or single fuel assembly.

In the auto-mapping routine PARCS send to CTF its geometry data, number of axial radial and axial nodes (and core map in case of more than one fuel element), and the latter code sets its averaging scheme to the PARCS geometry. As a necessary condition for the auto-mapping the number of nodes of PARCS and CTF models must be proportional by an integer number.

The moderator density, moderator temperature and fuel temperature is calculated by CTF in the first time step, using a CTF stand-alone boundary conditions. When the solution of CTF is converged, this information pass thought PVM interface to PARCS which calculates the total and nodal flux and power. Finally, PARCS send the nodal and total power to CTF and this loop begins in another time step. The figure 1 shown the data flow between the codes.
3. DESCRIPTION OF THE MODELS AND CALCULATION PROCEDURE

In this work, a single fuel PWR fuel assembly CTF model was performed. The selected assembly is situated at central position of the core of a 3010 MW PWR reactor. The figure 2 shows the radial map of the core in which the selected assembly is highlighted in blue.

Figure 1. Scheme of RELAP5/PARCSv2.7 and COBRA-TF/PARCSv2.7 data flow.

Figure 2. Radial thermal-hydraulic/neutronic mapping for RELAP5/PARCS model.
First, a coupled steady state RELAP5/PARCS calculation of whole core has been done to obtain the power of the selected fuel assembly. The relative power per fuel element obtained in the calculation is shown in figure 3.

![Figure 3. Radial power distribution of whole core RELAP/PARCS calculation.](image)

As the mean power for fuel assembly is 17.005 MW, the selected assembly has 12.872 MW distributed in 236 fuel rods of 3.4 m of active length. The fuel assembly is 16x16 with 236 fuel rods and 20 guide tubes. The pin and fuel pitches are 1.43 cm and 22.88 cm respectively.

The COBRA-TF assembly model consists of 256 flow channels (in an array of 16x16), with 208 fuel rods and 16 water rods. Axially, the flow channels have been divided in 34 nodes. The figure 4 shows a radial scheme of CTF model with the situation of the fuel and water rods.

The fuel assembly has 6 space grids situated in axial levels 4, 9, 14, 19, 24 and 29 respectively. Also in the first and last axial level are localized the lower and the upper plenum of the fuel element.
First a calculation of coupling codes RELAP5/PARCSv2.7 provides the initial axial boundary conditions to CTF. A CTF stand-alone execution was performed before the coupling with PARCS. Next a coupled steady state calculation was done and finally the coupled transient was run.

4. VALIDATION OF COUPLING ALGORITHM

In order to validate the coupling between CTF and PARCS, a comparison with the RELAP/PARCS results of most important steady state variables like axial power profile, \( k_{\text{eff}} \) and axial pressure and temperature profiles have been compared. Also, two different inlet mass flow transients have been simulated to compare the results obtained with RELAP/PARCS and CTF/PARCS coupled codes.

4.1. Coupled steady state

A coupled steady state calculation for the assembly model with RELAP5/PARCS and CTF/PARCS has been done. In the figure 5 are compared the axial power profiles obtained with execution of PARCS alone, with fixed boundary conditions, coupled RELAP/PARCS and CTF/PARCS. The results show good agreement, especially between PARCS alone and CTF/PARCS.
In the table 1 appear the root mean square error of axial profiles, respect to the PARCS alone solution and the error obtained in the $k_{eff}$ calculations.

**Table 1. Axial profile and eigenvalue errors**

<table>
<thead>
<tr>
<th>Code</th>
<th>Axial profile error RMS (%)</th>
<th>$k_{eff}$</th>
<th>$k_{eff}$ error</th>
<th>$k_{eff}$ error (pcm)</th>
</tr>
</thead>
<tbody>
<tr>
<td>PARCS</td>
<td>---</td>
<td>0.823175</td>
<td>---</td>
<td>---</td>
</tr>
<tr>
<td>RELAP/PARCS</td>
<td>8.11</td>
<td>0.835017</td>
<td>0.011842</td>
<td>1184</td>
</tr>
<tr>
<td>CTF/PARCS</td>
<td>3.29</td>
<td>0.836241</td>
<td>0.013066</td>
<td>1306</td>
</tr>
</tbody>
</table>

The axial evolution of pressure and temperature for RELAP and CTF models are shown in figures 6 and 7. The figure 6 shown small differences between the pressure drops calculated with both codes, nevertheless better adjust in the loss coefficients should be necessary for RELAP5 model to obtain a more consistent model. The temperature evolution is practically coincident between the two calculations except for the upper axial nodes.
Figure 6. Axial pressure drop profile.

Figure 7. Axial temperature profile.
4.2. Coupled transient calculation.

Two inlet mass flow transients calculation with RELAP5/PARCS and CTF/PARCS has been done. In the figure 8 is represented the inlet mass flow for the two different transients. In the transient 1 a smooth sinusoidal perturbation is applied and transient 2 an abrupt sinusoidal perturbation.

![Figure 8. Mass flow time evolution for the two studied transients.](image)

The figures 9 and 10 show the power evolution calculated by PARCS coupled with RELAP5 and CTF for both transients.
Figure 9. Total power evolution for transient 1.

Figure 10. Total power evolution for transient 2.

The void fraction evolution for the abrupt perturbation (transient 2) is shown in the figure 11. For transient 1, the void fractions calculated by both coupled codes are 0.0 and not represented.
Figures 11 and 13 show the temperature evolution at the outlet of fuel assembly calculated by PARCS coupled with RELAP and CTF for both transients. As can be seen the agreement is quite good.

Figure 11. Void fraction evolution of transient 2.

Figure 12. Exit temperature evolution for transient 1.
The figures 14 and 15 show the temperature evolution at axial level 15 of fuel assembly calculated by PARCS coupled with RELAP and CTF for both transients.

Figure 13. Exit temperature evolution for transient 2.

Figure 14. Temperature evolution for transient 1 at axial level 15.
Figure 15. Temperature evolution for transient 2 at axial level 15.

The figures 16 and 17 show the pressure evolution at axial level 15 of fuel assembly calculated by PARCS coupled with RELAP and CTF for both transients.

Figure 16. Pressure evolution for transient 1 at axial level 15.
Figure 17. Pressure evolution for transient 2 at axial level 15.

The figures 18 and 19 show the pressure evolution at axial level 4 of fuel assembly calculated by PARCS coupled with RELAP and CTF for both transients.

Figure 18. Pressure evolution for transient 1 at axial level 4.
Finally, in the figure 20 the cross flows in transversal axial section of fuel assembly are represented.

Figure 20. Cross flow distribution in an axial transverse section.
5. CONCLUSIONS

A code system based on coupling the 3D neutron diffusion code PARCS v2.7 and the thermal-hydraulic sub-channel code COBRA-TF (CTF) has been implemented and validate as a tool for detailed nuclear fuel assembly calculations.

The coupled CTF/PARCS steady state calculation and validation, against PARCS and coupled RELAP5/PARCS, show good agreement in the calculation of axial power profile and thermal-hydraulic parameters.

The transient calculations showed the same tendencies in the calculation of main thermal-hydraulic variables with CTF/PARCS and RELAP5/PARCS coupled codes. There are some differences that could be explained by the different level of detail of the nodalization of CTF and RELAP5.

With the results obtained, could be affirm that the primordial objective of this work, assess the consistency and correctness of the coupling scheme and calculation procedure, has been accomplished.

REFERENCES

