APPLICATION OF THE RELAP5/Pb-Bi CODE TO SAFETY STUDIES FOR THE ADS MYRRHA FACILITY

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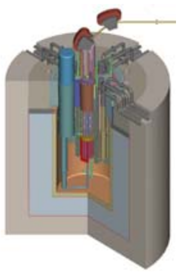
Abstract

RELAP (Reactor Excursion and Leak Analysis Program) is a code developed by Idaho National Engineering Laboratory (INEL) for best estimate transient simulations of LWR coolant systems in a broad range of normal or accidental situations such as LOCAs (**L**oss **O**f **C**oolant **A**ccidents).

The code is widely used for high pressurised reactors PWR (150 bars, 300°C) and BWR (70 bars, 280°C) safety studies for which it has been validated. A version bis of the code with the Pb-Bi as coolant was developed by Ansaldo Nucleare (Genova, Italy) and we applied it to simulate the behaviour of the ADS MYRRHA core in the framework of safety studies.

The paper describes this model and provides first calculation results in steady state conditions.



Introduction

This paper concerns the modelling of the core of the MYRRHA facility in its pre-design phase.

The model was built to allow thermohydraulic calculations with a β version of RELAP5.3.2 for the coolant eutectic Pb-Bi. The version 3.2 of RELAP5 which normally deals with water as coolant, was modified by the ANSALDO Nucleare (Genova, Italy).

The description of a partial model is showed here as well as first calculation results corresponding to the nominal core operation.

To perform safety studies, the model has to include pumps, heat exchangers and control elements. This work is still in progress.

Model description

In a first step this partial model is used to perform the simulation in steady state conditions and to analyse the influence of some design parameters on the temperature and flow distributions in the core. More particularly it addresses:

- the thermohydraulic behaviour in a hot, a medium and a cold channels (see the definition below);
- the heat exchange between the core and the vessel;
- the physical parameters at the core outlet;
- the flow circulation in the medium plenum;
- the total power and power distribution effects on the thermohydraulic behaviour inside and outside the core.

At the present stage, the model does not include the pumps, the heat exchangers, the vessel and the vessel structures. The spallation loop is partially modelled, only the part in the core is represented (the pump and heat exchanger of the spallation loop are not included).

The left part of Figure 1 given below represents the general sketch of the MYRRHA primary circuit and the right figure corresponds to the equivalent RELAP model of the core.

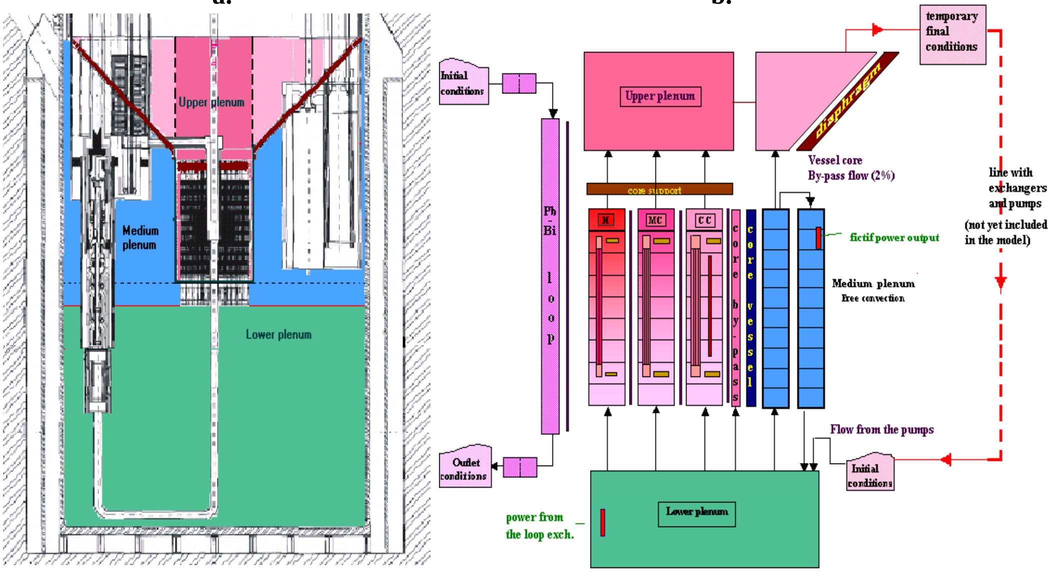
The RELAP building blocks showed on the right Figure can be divided into four fundamental groups: thermal-hydraulic, heat structures, trips, and control variables.

The thermal-hydraulic group is composed of elements designed to simulate fluid passages and fluid-handling equipment.

- The volumes (rectangular boxes) represent the parts of the flow in which temperature, pressure and phase concentrations are considered to be uniform. Temperature and pressure are the average values in a volume taking into account an average flux and the heat transfer between the structures and the flow. The type of flow in the volume is chosen in function of some criteria (velocity….) calculated on the basis of these average parameters.

Figure 1. **Comparison between a sketch of the MYRRHA primary circuit (a) and the equivalent RELAP model of the core (b)**

a. **b.**



- Junctions (arrows) connect two adjacent volumes and represent the pressure drop and the mass flow rate.
- Heat structures are designed to simulate solid structures and the interactions between the wall structures and the fluid flowing along it. They simulate the behaviour not only of the core fuel rods in a reactor system, but also the various plant structures
- The heat structures are represented by slabs, which may be finely nodalised to provide a rather detailed temperature distribution in one dimension. Plane, cylindrical or spherical structures are allowed for any slab. The code assumes that energy flow to and from the heat structures is in a direction normal to the stream-tube flow direction. Consequently, the heat structure nodes are aligned in the direction normal to the fluid flow. The radial and axial distributions of the flux are introduced with each structure if necessary.
- Special components such as pumps, accumulators, tanks or valves are modelled separately.
- Trips are designed to simulate the signals that initiate equipment actions of various sorts (e.g., turning on a pump at a desired time or causing a valve to open at one pressure but to close at another pressure).
- Control systems are designed to give the code modelling added capability by allowing equipment control systems (e.g., proportional- integral-differential controllers and lead-lag controllers) and "lumped-node" systems to be simulated.

These three element types are not yet included in the model. They are necessary to simulate transients behaviour for safety studies.

Core model

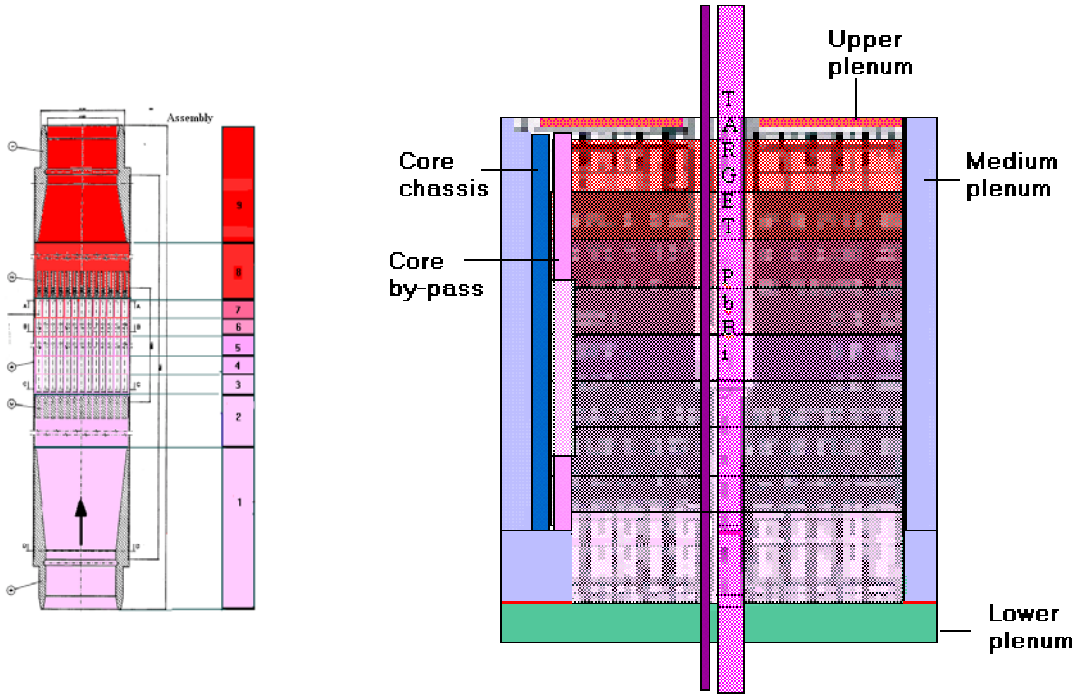
Description of an assembly

Each assembly is vertically subdivided in 9 sub-volumes:

- 2 vertical sub-volumes for the assembly inlet;
- 5 vertical sub-volumes for the active fuel length;
- 2 vertical sub-volumes for the assembly outlet.

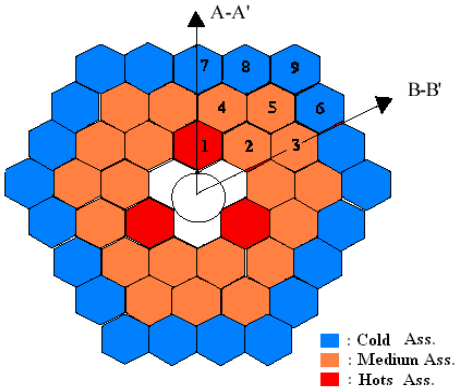
The hydraulic parameters are calculated in each sub-volume. Figure 3 shows the correspondence between the axial zones in an assembly and the axial zones in the core. The inert parts of the fuel are included in the top and bottom structures of the assembly. The active structure (fuel length) is modelled such as in a PWR core model, i.e. in a half cylindrical geometry with conservation of the area exchanging heat with the flow in the adjacent hydrodynamic volume. The centre of the fuel is an axis of symmetry and the power is distributed in the 5 fuel sub-volumes via axial and vertical power factors (see Figure 2 below).

Figure 2. **Correspondence between the axial zones in an assembly (a), the length of volumes in the equivalent pipe (b) and the axial zones in the core (c)**



a b c

Figure 3. **Core fuel distribution**



Description of a group of assemblies

The sub-critical core of MYRRHA with 45 assemblies is subdivided into 3 zones (or groups): hot (3 assemblies), medium (21 ass.) and cold channel (21 ass.), as shown on Figure 4. Each group is modelled by one representative channel supposed to reproduce the average behaviour of the corresponding group.

A group of n assemblies is simulated by only one pipe, (group of adjacent volumes) and 3 structures.

The pipe corresponds to a channel having a coolant flow equivalent to the flow of the n assemblies in which the flow velocity and the pressure drop are kept. (left Figure 2).

The inert metallic parts of the assemblies in a group are simulated by 3 structures:

- the wall of the assemblies, in contact with the core by-pass flow at the outer side and with the assemblies cooling flow at the inner side;
- the top grids including inert parts of the fuel elements in contact with it;
- the bottom grids including inert parts of the fuel elements in contact with it.

The third structure represents the total active fuel parts of the n assemblies in which the power is distributed according to the explanation given in chapter 4. The core by-pass is the volume between the assemblies and the annular free volume around the core assemblies. It is represented by a dummy pipe-volume.

Power distribution

The model is frozen with the configuration described in Figure 3. The total power is radially and axially shared by means of distribution factors defined for each radial and axial nodes.

We have therefore 3 groups with 5 sub-structures for each one, i.e. in total 15 sub-structures characterised every one by its group number g and its axial position i. The "partial" power P_{gi} delivered in the sub-structure gi is equal to:

$P_{gi} = a_g \; a_{gi} \; P_{tot}$ where P_{tot} is the total power of the core (38.8 MW), a_g is a radial distribution factor a_{gi} is an axial distribution factor. The values of the a_g and a_{gi} are given in Table 1. In this table the a_i factors are "over-all" axial distribution factors defined by:

$$a_i = \sum_{g=1}^{3} a_g \, a_{gi} \quad \text{as} \quad \sum_{g=1}^{3} a_g = \sum_{i=1}^{5} a_{gi} = 1,$$

$$\text{we may also write: } \sum_{g=1}^{3}\sum_{i=1}^{5} P_{gi} = P_{tot}$$

Table 1. **Radial and axial power distribution factors**

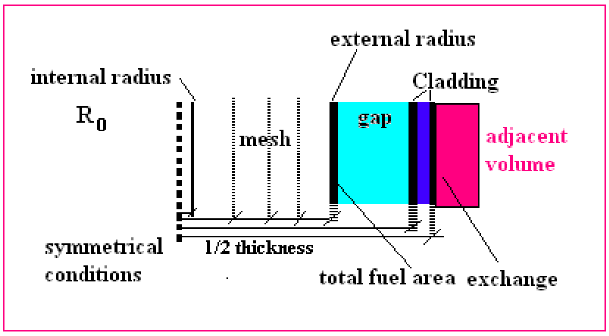
	AXIAL	Hot Ass A (3)	Med.Ass BCDE (21)	Cold Ass FGHI (21)	
			a_{gi}		
	0.162	0.155	0.160	0.167	
	0.218	0.220	0.219	0.216	
	0.241	0.251	0.242	0.234	
	0.218	0.219	0.219	0.216	
	0.161	0.155	0.159	0.167	
total	1.000	1.000	1.000	1.000	
RADIAL	a_g	0.093	0.565	0.342	1.000
					total
P_{tot}	38.8 (MW)				

Finally the fuel region of each sub-structure is cut in 4 radial intervals with a uniform power generation in it. It means that 25% of the sub-structure power is delivered in each interval. Two radial intervals are added to represent the gap and the cladding thickness respectively.

These radial node intervals are showed on Figure 4.

One should remark that all the assemblies within a group (hot, medium or cool) have the same core-radial coefficient distribution.

Figure 4. **Radial nodes in the fuel**



Modelling of the plena

The lower plenum is represented by the volume in the vessel under the core. It receives the flow from the pumps and it supplies the core in Pb-Bi. Some calculations with the FLOW 3D CFD code are in progress and will give information on the flow circulation in this volume. Later the lower plenum will be remodelled by taking account this information.

The volume of fluid around the core chassis between the low assemblies level and the medium level of the diaphragm is **the medium plenum.** In this volume free convection is induced by the heat exchange with the core chassis. To simulate the fluid motion, this plenum must be modelled by 2 parallel channels interconnected via cross-flows. Until the modelling of the primary loop is completed, a small fictitious cooling structure is introduced to remove the heat and to allow the free convection.

The upper plenum is represented by the hot volumes above the core; one part is included in the diaphragm and the other one is outside the chassis above the diaphragm walls. The 2 volumes are connected by a junction simulating the flow through the openings in the wall chassis above the core.

The spallation loop

At the present stage, the spallation loop is modelled by a vertical pipe with inlet and outlet horizontal pipes. The vertical pipe represents the part of the spallation loop through the core. The pressure, temperature and coolant flow are given at the initial state. A small power could be introduced to represent the thermal power produced in the loop. The wall of the loop, submitted to the thermal exchange between the spallation flow and the by-pass core volume representing the flow between the assemblies, is modelled with a structure.

First results of the calculations in nominal conditions.

A classical calculation with a nominal power of 38.8 MW and the power distribution described in section 3.3 was done.

After stabilisation of the flow in the channels (between 0 and 300 s.), at zero power, a power ramp is applied between 300 and 400 s. (Figure 5)

Figure 5. **Power transient**

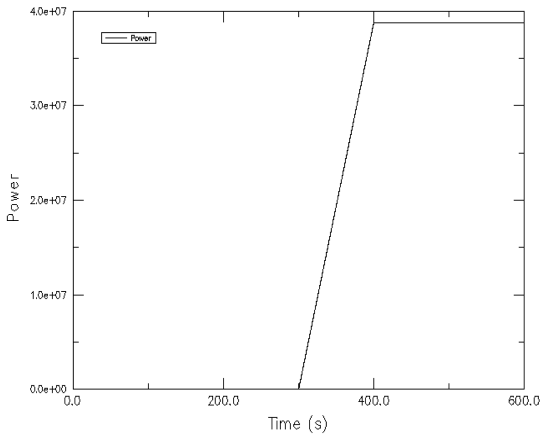
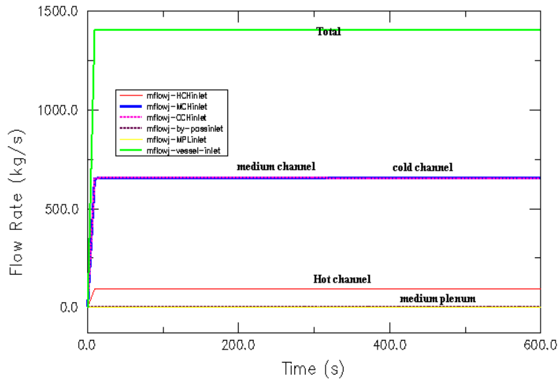


Figure 6. **Distribution of the flow in the hot, medium and cold channels**



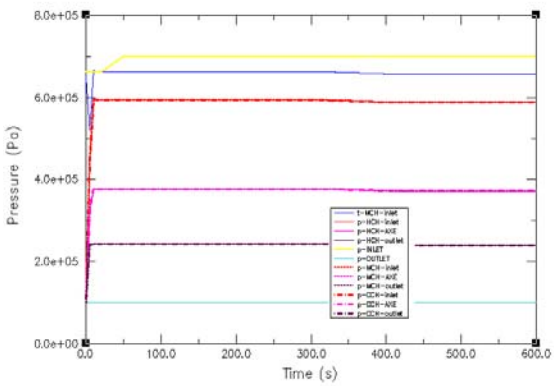
The total coolant flow is 1 400 kg/sec and distributed as follows:

- 94 kg/s in the hot channel;
- 653 kg/s in the medium and cold channels;
- 653 kg/s in the cold channel considering that the dummy assemblies are plugged (the fluid flows in 21 assemblies only with a small power 54 assemblies are plugged).

The velocities of the coolant are identical in the tree channels.

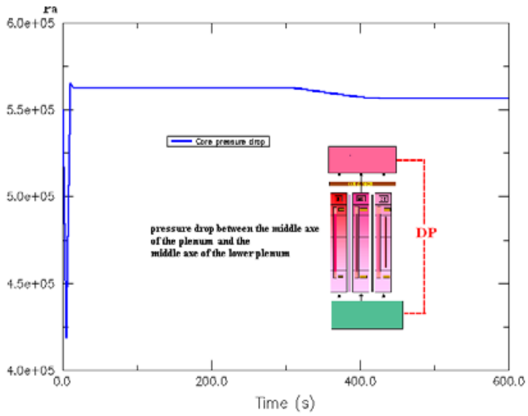
The repartition of the pressures in the vessel and core is shown in Figure 7 below. The initial pressure (pressure at the pump outlet) was 0.7 MPa, and the pressure drop between the middle axes of the upper and the lower plena was 0.56 Mpa.

Figure 7. **Pressure distribution in the different channels**



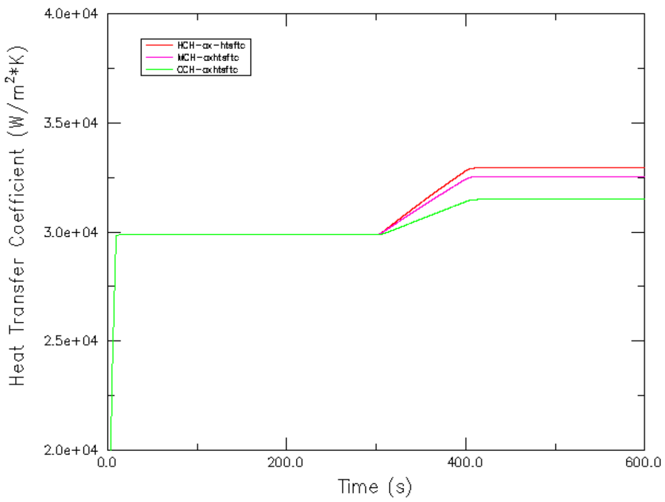
The stabilisation at 38.8 MW with the power distribution given in Table 1, shows a maximum temperature of the coolant at the channel outlets equal to 730, 697, 611°K respectively in the hot, medium and cold channels.

Figure 8. **Pressure drop between the upper and the lower plena**



The average temperatures of the fluid in the core axial level are: 634, 611, 556 K respectively in the hot, medium and cold channels. The heat transfer coefficient and the heat flux, at the outer cladding surface of the fuel elements, at the mid core level, are shown on Figures 9 and 10 for each group of assemblies. The heat transfer coefficient is about 3.25e+04 W/m²K.

Figure 9. **Fuel surface heat transfer coefficient in the hot, medium and cold channels** (mid core level)



In the hot channel, the maximum fuel temperatures are 1 500°K and 1 035°K at the centre and at the pellet surface respectively. The cladding temperature at the inner and outer surfaces, are 691 and 658 K (mid core level). The maximum temperature of the core vessel chassis walls is 623°K in the upper level.

Figures 11,12,13 and 14 show the temperature profiles at various levels and in the different channels.

Figure 10. **Fuel surface heat flux in the hot, medium and cold channels** (mid core level)

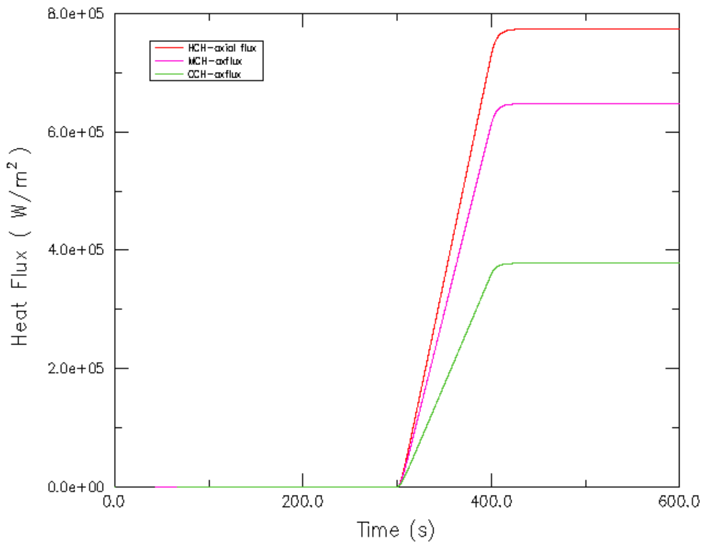


Figure 11. **Temperature profile in the hot channel fuel** (mid core level)

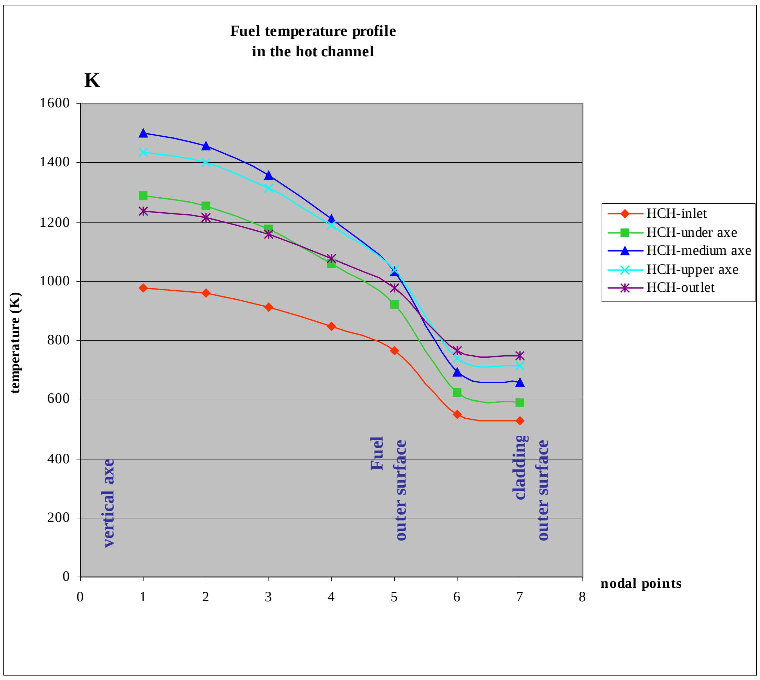


Figure 12. **Temperature profile in the medium channel fuel**

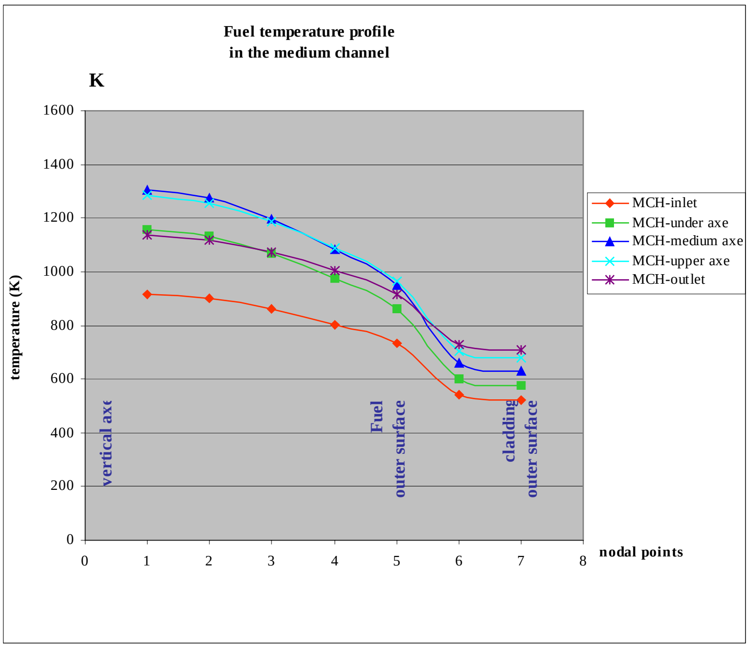


Figure 13. **Temperature profile in the cold channel fuel**

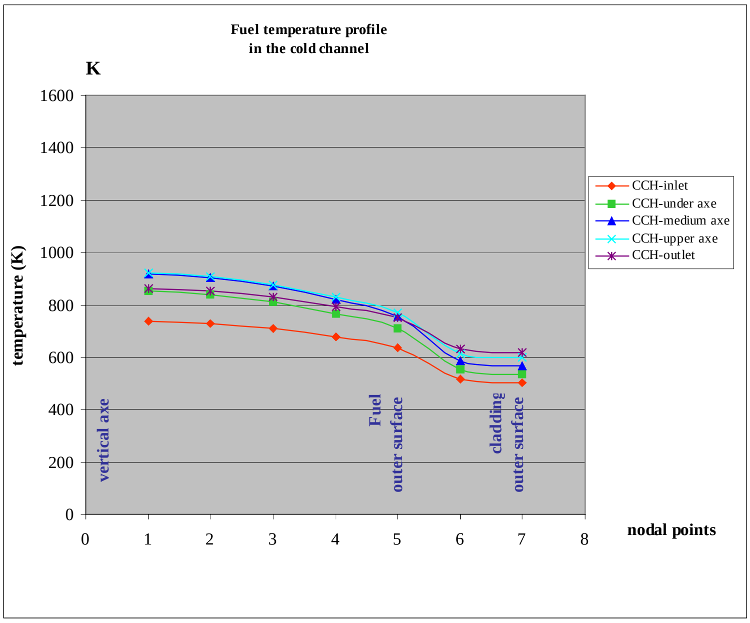


Figure 14. **Temperature profile in the dummy fuel of the cold channel**

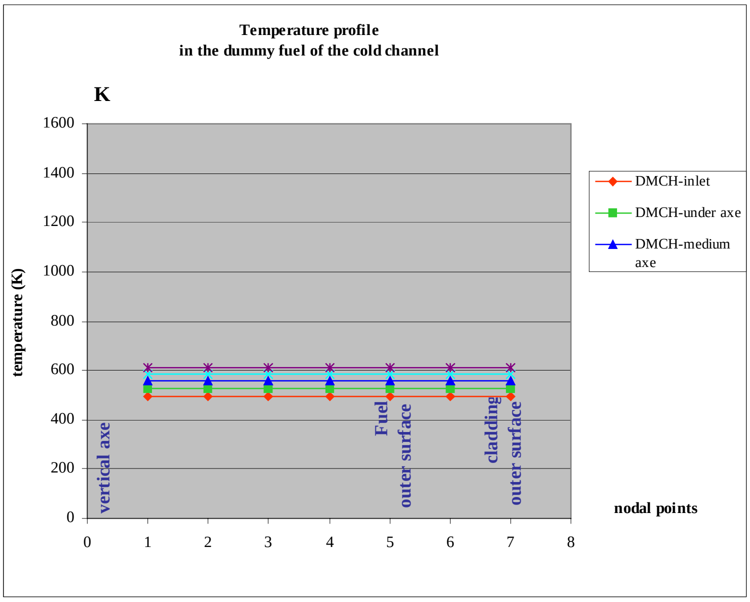
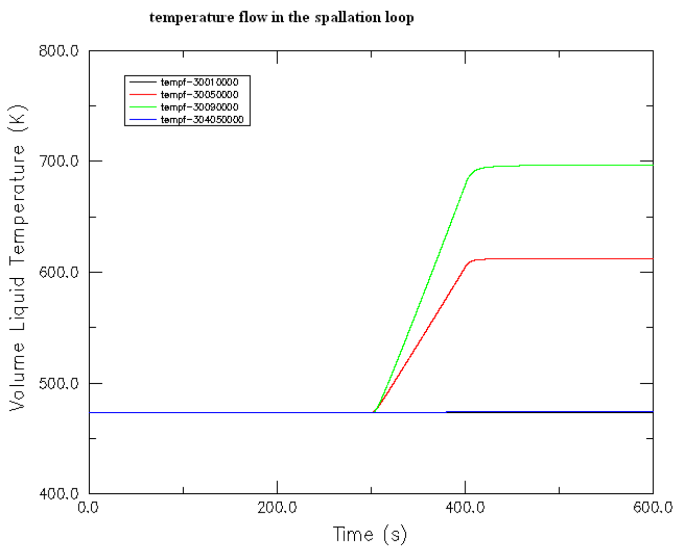


Figure 15. **Pb-Bi temperature at various position in the spallation loop**



In these first calculations, no power was injected in the spallation loop. The temperature is only function of the thermal exchange between the core and the loop.

At the outlet of the spallation pipe the temperature reaches 700°K (see Figure 15).

Conclusions

This first calculations in nominal conditions are in agreement with the preliminary calculation of the pre-design (Ref 1 and 2). The pressure drops were calculated approximately, especially in the sections of the grid and support plate. Only the restrictions and expansions of the cross -sections were taken into account. These pressure drops have to be calculated more accurately.

The by-pass flows through narrow paths in the diaphragm were not taken into account neither. The free convection was not included, and some sensitive pre-calculations must to be done. The medium plenum is considered as a stagnant volume.

A lot of parameters could be tested and evaluated. The pumps, the heat exchangers and the control system must be included in the model to perform safety studies.

Acknowledgement

We are grateful to Ansaldo Nucleare, in particular Doctor Alessandro Alemberti, for their contributions and for the version Pb-Bi of RELAP/mod 3.2.2 to be tested by the MYRRHA team.

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