Ongoing activities at SCK•CEN with RELAP5-3D

D. Castelliti[1], A. Toti[1], B. Arien[1], F. Belloni[1], G. Scheveneels[1], G. Van Den Eynde[1]

[1]Studiecentrum voor kernenergie – Centre d'étude de l'énergie nucléaire (SCK•CEN)
Boeretang 200, Mol, Belgium

Different activities involving the use of RELAP5-3D code currently ongoing at SCK•CEN are presented.

Introduction

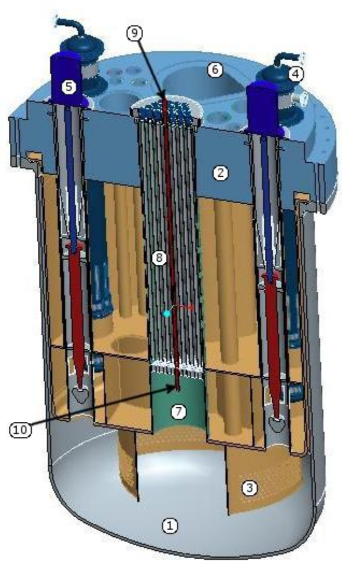
MYRRHA (**M**ulti-purpose h**Y**brid **R**esearch **R**eactor for **H**igh-tech **A**pplications) is a pool-type Accelerator Driven System (ADS) with the ability to operate also as a critical reactor. It is cooled by liquid lead-bismuth eutectic (LBE). Its main objectives can be summarized as:

- Flexible irradiation facility
- Minor Actinides (MAs) transmutation demonstration in support of R&D on a "closed fuel cycle"
- ADS demonstrator at a reasonable power level
- Heavy Liquid Metal (HLM) Fast Reactor demonstrator

Because of these objectives the MYRRHA project has been recognized as a high priority infrastructure for the nuclear research in Europe.

MYRRHA current configuration description

Figure 1, presents the main MYRRHA primary system components:



1. Reactor vessel
2. Reactor cover
3. Diaphragm
4. Primary heat exchanger
5. Pump
6. In-Vessel Fuel Handling Machine
7. Core barrel
8. Above Core Structure
9. Core plug
10. Spallation window

Figure 1 - Overview of the MYRRHA reactor (in ADS mode)

The primary system is completely enclosed in the primary vessel (pool-type system). The average upper plenum temperature is 325 °C. The LBE in the upper plenum flows through the four Primary Heat eXchanger (PHXs) units into the two Primary Pumps (PPs) (one PP serving two PHXs). From the PPs the LBE is recirculated into the lower plenum.

The primary system is linked to four independent Secondary Cooling Systems (SCSs) through the four PHX units. Each secondary system is operated with forced flow two-phase water mixture at 16 bar (~200 °C). The moisture is then separated in a steam drum, from where the steam is directed towards an air condenser (one per secondary loop) and the water is recirculated to the PHX. In normal operation the secondary water temperature is kept constant by the control system.

The steam dissipates the heat to the external environment through the tertiary system's air condenser.

All three systems are designed to operate in forced circulation (active mode) during normal operation. Nevertheless, the plant must also be able to remove the decay heat in accidental conditions in full natural circulation (passive mode).

Ongoing activities involving RELAP5-3D

SCK•CEN acquired the RELAP5-3D code in anticipation of the licensing activities for the MYRRHA reactor. The main purpose is to use the code as a validated tool to perform transient analysis for different operational and accidental conditions.

However, the application range of the RELAP5-3D code has been extended to several different activities related to the safety analysis, to the design and to the operation of the reactor.

Activities in support of MYRRHA licensing

A RELAP5-3D model of the most recent version of the MYRRHA reactor design has been realized by SCK•CEN. It includes all reactor systems, from the core to the tertiary air heat sink, and a preliminary simulation of the main control systems and devices.

At first, the reactor *steady state* has been simulated, in order to make sure that the nominal steady state conditions could be properly represented by the code. A number of *transients* have then been proposed for the preliminary safety analysis, which can be divided in three main categories:

- Protected transients to study the response and the capabilities of the DHR systems;
- Protected transients to analyze the risk connected to the primary LBE overcooling;
- Unprotected transients to understand the system physical limits and determine desired Instrumentation & Control (I&C) response times.

A series of enveloping transients has been discussed and set up, one event for each category:

- Loss Of Offsite Power (LOOP);
- Overcooling + Loss Of Flow Accident (LOFA);
- Unprotected Loss Of Flow (ULOF).

Uncertainty + Sensitivity analysis

A first test of the Uncertainty + Sensitivity (U + S) methodology was performed making use of the Software for Uncertainty and Sensitivity Analysis (SUSA) [1] coupled with RELAP5-3D. The procedure, tested in the framework of European FP7-MAXSIMA project [2], was applied to the Unprotected LOFA transient case.

The purpose of the U + S analysis consists in the quantitative evaluation, through multiple system code runs, of the variation of certain safety-relevant output parameters caused by certain input variations.

Every input parameter considered for the U + S analysis has been selected and identified through its reference value and a suitable variation range. A *Gaussian distribution* (± 3σ), centered on the reference value, has been considered for each parameter.

A list of relevant output parameters has been defined. *17 relevant output parameters* have been selected. In order to achieve a 95%/95% statistical accuracy, 100 RELAP5-3D input decks have been run.

The variations induced by the selected input parameters on the Unprotected LOFA transient evolution have proven to be not wide. The *Peak Clad Temperature* (PCT) shows maximum deviations within ± 20 °C from the best estimate value. In general, the Unprotected LOFA transient shows limited sensitivity to the considered input parameters.

The U + S methodology applied to Unprotected LOFA transient gives interesting first insights in a statistical approach for the MYRRHA safety analysis, but it requires some improvements in the definition of the parameters considered as most important for the reactor safety limits.

Applications on component design and verification

Design validation is part of the design process of a specific component. *Every component designed for the MYRRHA reactor* must be *verified* through a *suitable calculation tool* to prove the solution efficiency in all working conditions (especially considering the transient behavior).

SCK•CEN applies *RELAP5-3D* code *to prove* the concept of *many components* involved in the MYRRHA reactor design. Specific models, extrapolated from the main RELAP5-3D MYRRHA model, have been set up to test the components under various conditions.

The *Primary Heat Exchanger* thermal-hydraulic design has been *completed* at SCK•CEN under steady state conditions, granting the required performances. A *RELAP5-3D model* dedicated to the PHX has been built and *used to test* the nominal *steady state conditions* and the validity of the concept towards transients. In particular, a detailed analysis of the tube bundle instabilities has been performed [3].

The *RELAP5-3D* code has also been used as *support* calculation tool for the SCK•CEN *R&D program*. A dedicated model of the *European SCAled Pool Experiment* (ESCAPE) facility [4] has been set up and used to *predict* the *steady state* and the *planned transients* behavior of the plant. In the near future the facility will become operational and the evaluations will be compared to the experimental results, providing a valuable validation source for the MYRRHA licensing activities.

MYRRHA control logic studies

As a preliminary reactor control strategy developed for the MYRRHA plant, two main control systems are foreseen [5]:

- Reactivity control through the control rod system
- Secondary pressure control through the air fan velocity control

Reactivity control system

In the current MYRRHA plant *RELAP5-3D* model, *core power* is calculated using the *reactor point NK* model. The *Control Rods* (CRs) *model* is *based* on a *series of Control Variables* which elaborate the error between the requested power set-point and the actual power generating a suitable signal acting on the CRs velocity. The CRs reactivity worth is known as a function of its position. The model considers the 6 CRs foreseen by design collapsed in only one CR with the resulting total worth.

The control system input developed to simulate the control rod reactivity leads to determine the current CRs position and the reactivity associated with it as a function of the power level desired. The reactivity inserted is determined by an evaluation of the rod position.

The control variable logic must be able to determine the CRs position at each time step. The new position is calculated by integrating the CRs velocity over time, thus obtaining a position difference to be added at the current CRs position.

The shim speed is determined by the power error, which is the difference between the current core power and the desired set-point core power. The power error is then related to the current shim rod speed.

Secondary Cooling System pressure control

The reactor control strategy foresees a constant Secondary Cooling System pressure of 16 bar under all operating conditions. This requirement is assured through a pressure control system using the regulation of the tertiary air fan mass flow rate as controlling parameter. Such logic is implemented through a *Proportional-Integral* (PI) *controller*.

The *pressure error* has been calculated, through a *dedicated Control Variable*, as the difference between the set point pressure (16 bar) and the actual pressure value. This error enters the PI controller.

The PI controller is encoded in the MYRRHA RELAP5-3D input deck through the dedicated Control Variable ("prop-int") [6]. The K_I and K_P gain constants for the PI controller have been evaluated through the MATLAB optimization toolbox [7] basing on the response of the system.

The outcome of the PI Control Variable acts as input for the SCS pressure controller, giving as output an air mass flow rate maintaining the pressure constant.

RELAP5-3D coupling with Ansys FLUENT

SCK•CEN has recently undertaken the development and *validation of a code coupling* method between *RELAP5-3D* and the CFD code, *Ansys FLUENT*. The main purpose is to develop a computational approach able to *capture 3D fluid flow phenomena in system-scale thermal hydraulics simulations* (limited to certain specific zones where 3D flow relevance is greater).

Some issues related to *numerical instabilities* and conservation laws preservation *at the interfaces* arose from this first test. This is a *consequence of the sequential explicit scheme*, characterized by the fact that one of the two codes uses for the time step *n* only BCs calculated by the other code at the previous step *n-1* [8].

An *implicit coupling scheme* has been then *developed* to overcome the issues related to the explicit coupling. In the implicit algorithm, *inner code-to-code sub-iterations* are performed for each coupling step, until a specific convergence criteria at the interfaces is satisfied. The results are reported in Figure 3. As can been observed, the implicit scheme is *not affected by numerical instabilities* and leads to *improved accuracy* of results. As a drawback, the implicit scheme is *significantly computationally more expensive*, due to the sub-iterations performed at the each coupling step.

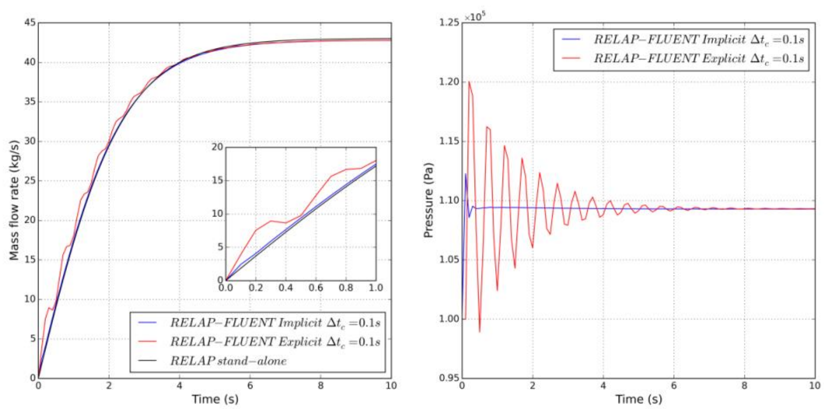


Figure 3 - Mass flow rate (left) and interface pressure (right) for the explicit and implicit coupling schemes

3-D Neutron Kinetics model with NESTLE code

A preliminary 3-D Neutron Kinetics (NK) model of the complete MYRRHA core has been set up with NESTLE code and coupled with RELAP5-3D reactor core, in order to simulate local transients involving reactivity variations, for which a point NK approach does not provide enough details.

RELAP5-3D core thermal-hydraulics model

The Thermal-Hydraulic (TH) MYRRHA core model has been refined in comparison to the model used in the overall plant model. It includes 12 channels (each channel divided into 22 sub-volumes):

- 5 channels to simulate 5 different FA burn-up levels
- 5 channels to simulate the core positions not filled with fuel
- 1 channel to simulate the core bypass

- 1 channel to simulate the radial reflector (not connected to the rest of the system, only used for NK simulation)

All the hydraulic channels are connected to a Heat Structure where the actual power generation occurs.

NESTLE model

The first step towards the 3-D model of the MYRRHA core is represented by the *macroscopic cross sections* generation to be inserted in the NESTLE model of the core itself. The cross sections have been evaluated through the Monte Carlo *Serpent code* [9].

Since the NESTLE code is limited to four neutron energy groups only, a cross section condensation has been adopted to reduce the available data from Serpent into a format compatible with NESTLE: being MYRRHA a fast reactor, higher importance has been given to the fast and high-epithermal spectrum.

The selection of the boundaries of the energy groups has proven to be very important in order to get a correct value for the K_{eff}, matching the Serpent value. Various sensitivity analyses have been performed before finding the right configuration providing similar steady state results.

The NESTLE core model, coupled with the TH core model developed in RELAP5-3D, consists in the representation of all 169 core channels. The model has been made according to the following assumptions:

- Model extended to the FA detail and coupled to the TH section according to the core channel modelization
- Core modeled using only fresh FAs (core at Beginning of Life state)
- FA divided in 7 axial nodes (5 for the active zone, 2 for the plena)
- Every axial volume of every FA characterized by its own cross sections set

Coupled calculations preliminary results

As preliminary calculation, a steady state run has been performed in the coupled core model. In Figure 4 it is possible to see the peak factor distribution in the complete core.

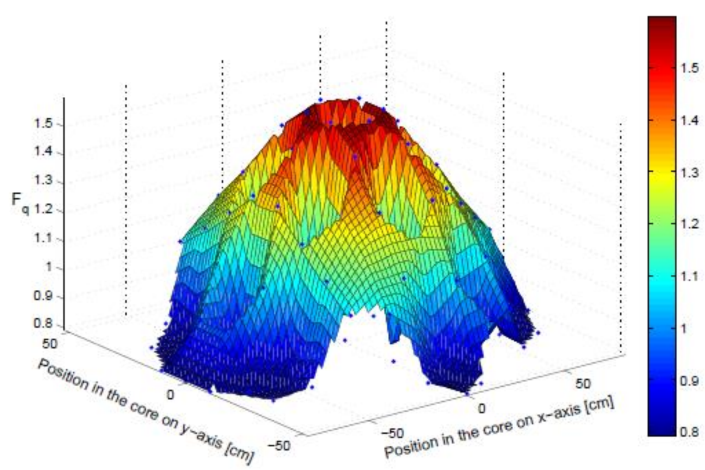


Figure 4 – RELAP5-3D/NESTLE Steady State core power peak factor distribution

Control Rod Ejection transient

An important safety-related transient event for the MYRRHA reactor is represented by the Control Rod Ejection (CRE). In order to evaluate the difference with a point NK approach, a 3D NK simulation of the CRE has been run using the RELAP5-3D/NESTLE core model.

References

[1] http://www.grs.de/en/simulation-codes/susa

[2] FP7 MAXSIMA project Grant Agreement, SCK•CEN, August 2012

[3] D. Castelliti, "Stability analysis of MYRRHA Primary Heat Exchanger two-phase tube bundle", IRUG 2014 meeting, Idaho Falls, 2014.

[4] M. Greco, F. Mirelli, S. Keijers, K. Van Tichelen, "Pre-test computational fluid dynamics and system thermal-hydraulics calculations of the E-SCAPE scaled LBE pool facility", NURETH-16, Chicago, 2015.

[5] G. Morresi, "Preliminary study of MYRRHA control system analysis", Master Thesis, Pisa, February 2015

[6] The RELAP5-3D Code Development Team, "RELAP5-3D ver. 4.2 Code Manual", Idaho Natianal Laboratory, Idaho Falls, June 2014

[7] http://www.mathworks.com

[8] W. L. Weaver, E. T. Tomlinson and D. L. Aumiller, A Generic Semi-Implicit Coupling Methodology For Use In RELAP5-3D, 2000 RELAP5 Users Seminar, Jackson Hole, Wyoming September 12-14, 2000.

[9] http://montecarlo.vtt.fi