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THERMAL HYDRAULIC MODELS FOR NEUTRONIC AND THERMALY HYDRAULIC FEEDBACK IN CITVAP CODE

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Abstract. The improvements in the computational systems (increasing the memory and storage capacity and the enhancement of calculation power) allows the development of innovative methods for reactor calculations, including not only more accurate theories and numerical methods, but also adding more prediction capabilities and additional engineering information to perform the numerical analysis of the system. As an example, nowadays it is possible to integrate different tools with interdisciplinary or multi engineering information allowing an integrated approach to the problem to be solved. The INVAP calculation line is used in a wide range of applications (mainly MTR, CAREM and CNA-2), where some of them requires a thermal hydraulic model to evaluate the distribution of the fuel, coolant and moderator temperatures and the coolant and moderator densities. For those cases, such parameters are needed for a proper evaluation of neutronic behavior of the core.
Currently CITVAP has several thermal-hydraulic models depending on the geometry considered; 1D geometry, to calculate each channel of the reactor core, and a 3D geometry to use more complex thermal-hydraulic model of the core. The models are for MTR and NPP. The simulations involve natural convection and forced circulation. In the last upgrade of CITVAP, the possibility to use any external thermal hydraulic model was added.
All these models are used for the thermal hydraulic feedback of the neutronic calculation, but in some cases can be used for further key analysis in reactor design. Accordingly, there is the possibility to evaluate thermal margins to critical phenomena, perform the evaluation of the temperature distribution for the oxide layer growth calculation and perform the calculation of the power feedback coefficient.

1 INTRODUCTION

The variables needed in reactor physics analysis depend on the interaction of the neutrons with matter. The properties of matter are basically given by the nuclear cross sections (XS) and engineering data. The engineering data depends on operational conditions (for example then temperature, density, cooling conditions, etc) and the nuclear XS basically depends on energy and temperature, but also in the engineering data like crystal and molecular structure of the materials. Finally, the transport equation is used to describe the neutron distribution (through the neutron flux), which is usually solved with different methods and tools.

In a reactor core the power distribution can be obtained using the XS of the system in an average thermo-hydraulic condition. With this power distribution it is possible to calculate the fuel temperature distribution, the coolant temperature and density distribution, and (in some reactors) the moderator temperature and density distribution. These parameters are used then to modify the XS in order to perform a recalculation of the power distribution if it is required. As it can be seen, this process generates feedback loop that allows us to properly determine all the variables of the reactor core. Figure 1 conceptually shows this neutronic and thermal-hydraulic feedback loop.

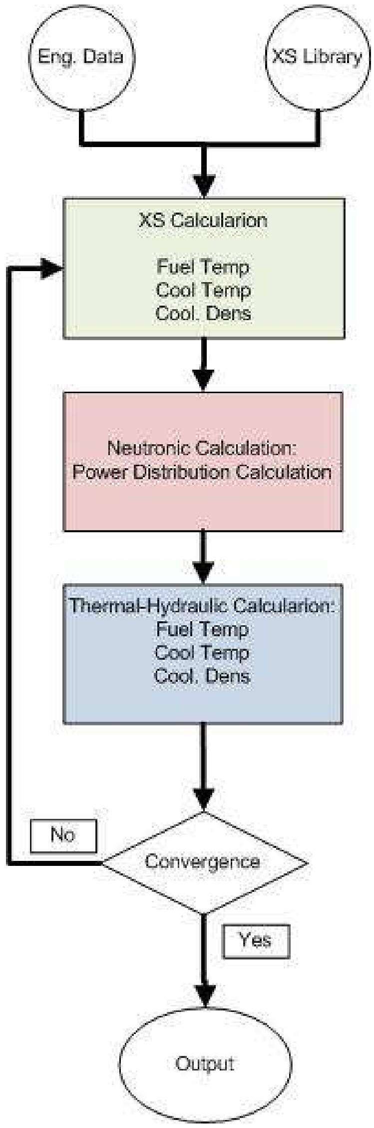
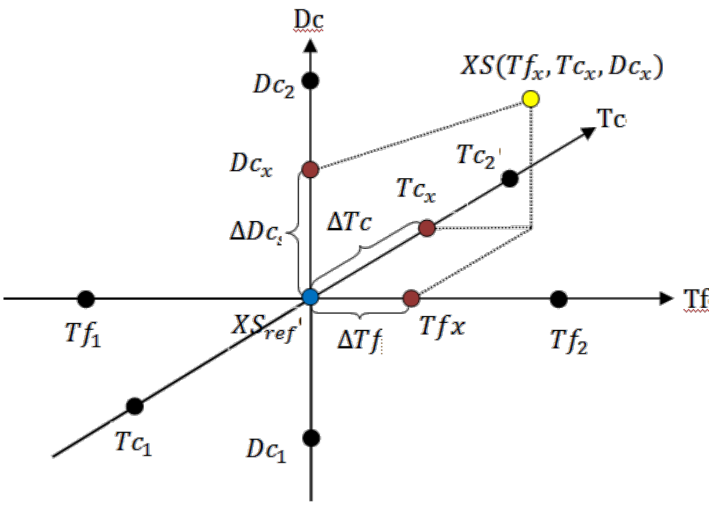


Figure 1: Scheme of the Neutronic and thermal-hydraulic feed loop

Core code CITVAP (Villarino E and Lecot C, 1993; Mochi 2011) performs the steps given in Figure 1 as follows:

- Engineering Data is given by the user.
- The macroscopic XS library is provided by HXS program (Mochi 2011). The most important characteristics of this library are that the burnup dependent XS also depends in the thermal-hydraulic parameters, such as: fuel temperature, coolant density and temperature, and (in some cases) moderator density and its temperature. This library provides also the initial (or guess) values for these parameters.
  The XS are calculated by CONDOR code (Villarino E, 2002).
- XS calculation is carried out by interpolation in the above mentioned parameters. The interpolation scheme used is based on partial derivates. Figure 2 shows the concept and the required parameters. Where linear interpolation is carried out in all variables except the fuel temperature, which is done using the square root interpolation.
- CITVAP calculates the power distribution using the Diffusion method.
- CITVAP performs thermal-hydraulic calculation based on different methods, which will be explained in the following sections.
- If there is no convergence, CITVAP repeats this loop up to reach convergence. Normally this process takes 2 or 3 iterations.



$$XS(Tf_x, Tc_x, Dc_x) = XS_{ref} + \Delta Tf + \Delta Tc + \Delta Dc$$

Figure 2: Scheme of the Neutronic and thermal-hydraulic feed loop

The application of this thermal-hydraulic loop is mandatory in NPP for accurate calculations, in the case of MTR reactors it is usually applied mainly for two purposes:

- The calculation of the Power Feedback coefficient.
- The calculation of the oxide layer growth in a high performance MTR reactor.

2 CITVAP THERMAL HYDRAULIC MODEL FOR NPP

For this type of reactors, CITVAP has two different models based on COSTHA developments for light (Soto A. 1989) and heavy water (Gambetta M, Mazufri C. and Mochi I., 2008).

COSTHA was the first modeling capability introduced in the CITVAP code. The main objective of this first model was the simulation of the CAREM NPP reactor, but unfortunately this reactor has a very strong cross flow that was not possible to model with a channel model.

In spite of this, it showed to be a powerful tool, and several years later it was modified to allow the calculation of CNA-2 NPP reactor with good results.

COSTHA is a steady state, 1D fuel element vertical channel model for a PWR or PHWR reactors. This is a two phase flow, forced circulation model for pressurized light or heavy water reactors. It can calculate the average fuel temperature, coolant temperature and coolant density and in case of the PHWR the moderator temperature and density, but no capabilities to calculate thermal margins are included.

One of the input parameters is the global inlet flow of the coolant and moderator in case of PHWR. The code distributes the coolant flow in the fuel channels according to the conditions of each of the channels.

This thermal-hydraulic module is applicable within the following limits:

- Reactor type: PWR or PHWR with vertical fuel channels.
- Steady State calculation.
- Fuel assembly: Rod cluster (square or triangular).
- No cross flow among fuel assemblies.
- Coolant and moderator: light or heavy water.
- Coolant Flow Regime: two phases.
- Moderator Flow Regime: single phase.
- Fuel: Incorporated UO2 and Zircalloy properties (or defined by the user).

This model was verified against RELAP and PUMA models for CNA-2 NPP reactor with a very good agreement (Gambetta M, Mazufri C. and Mochi I., 2008).

3 CITVAP THERMAL HYDRAULIC MODEL FOR MTR

For this kind of reactors, CITVAP includes two different thermal-hydraulic modeling capabilities, one for natural circulation and the other one for forced circulation (Pieck D., 2010)

Both models are only for research reactors with flat type fuel assemblies where only single phase light water for a non pressurized system in a steady state condition can be modeled.

3.1 TERMIC Model for RR reactors in forced circulation

TERMIC (Abbate P., 2003) is a steady-state thermal-hydraulic code developed to perform core thermal hydraulic design using plate type fuel assemblies. TERMIC is intended to calculate maximum allowable powers and heat fluxes using selectable limiting criteria of Onset of Nucleate Boiling (ONB), CHF and Flow Instability as a function of the coolant velocity. The coolant is light water, in upward or downward flow.

As TERMIC was programmed as an engineering code it includes a statistical and multiplicative treatment for uncertainty factors of the input data.

The main characteristics of usage are:

- Reactor type: pool type.
- Fuel Assembly: MTR.
- Coolant channel: Rectangular.
- Coolant: light water.
- Operating conditions: low pressure, low temperature.
- Flow type: Forced convection, single-phase conditions.
- Operating conditions: Steady State.

- Operating Regime: up and down flow. Laminar, transition, turbulent.
- Critical Heat Flux: Mirkshak correlation, Sudo correlation, and Bernarth correlation.
- Flow instability: Whittle & Forgan correlation, Saha & Zuber correlation.
- Power shape: CITVAP provides the power shape.
- Heat Transfer: Single-phase convection, subcooled boiling, 1D conductivity model.

This model can predict thermal margins to different critical phenomena, including uncertainties factors. It also calculates thermal hydraulic data to be used by FLUX post-processing tool for the calculation of the oxide layer growth in aluminum cladded fuel assemblies.

The model included in CITVAP was verified against its standalone version, giving the same results and validated against power feedback coefficients measured in the ETRR-2, OPAL and RA-6 reactors.

3.2 CONVEC Model for RR reactors in natural convection

During thermal-hydraulic design and calculation of nuclear reactors, the cooling performance of a core when the flow is driven by natural circulation forces needs to be calculated. This can be the case after a pump trip, or in a core whose normal operation mode is natural circulation.

The CONVEC program (Abbate P., 2002) was specially developed for thermal-hydraulic calculation in steady state regime of MTR reactors operating in Natural Convection mode.

Its unique features have been specially tailored to the geometry of the rectangular channels of MTR fuel elements.

Given the power to be removed from a fuel element, the program calculates: mass flow and velocity in the cooling channel, fluid temperature distribution, wall temperature distribution, mean wall and fuel temperatures, and the margins to the Onset of Nucleate Boiling and to Critical Phenomena.

The program can calculate using either light water in one phase flow.

The main characteristics of this version of the program are the following:

- Reactor type: pool type.
- Fuel elements: MTR.
- Coolant channel: rectangular.
- Coolant: light water.
- Operating conditions: low pressure, laminar regime.
- Flow type: single phase.
- Operating conditions: steady state.
- Operating regime: upward flow, driven by the natural convection mechanism.

The model included in CITVAP was verified against its standalone version, giving the same results, and was validated against power feedback coefficients in natural convection measured in the OPAL reactor.

4 GENERAL PURPOSE THERMAL HYDRAULIC MODEL

Currently CITVAP can use any thermal-hydraulic code and model for the thermal-hydraulic feedback. This was a significant improvement, developed mainly for the CAREM reactor due to the requirement of a 3D model for the proper calculation of its thermal hydraulic parameters.

The code used for this 3D calculation is THERMIT (Kelli J. & Kazimi M. 1979), which is a 3D thermal hydraulic code for LWR developed by Massachusetts Institute of Technology sponsored by EPRI (Electric Power Research Institute). This is a two phase flow for rectangular and polar geometries.

In early versions of CITVAP, this thermal-hydraulic feedback was done using both standalone codes integrated in an operating system script. This methodology was hard to use and certain plant simulations were not possible to be modeled.

The current version of CITVAP allows the preparation of a script to be run inside the CITVAP execution using any thermal-hydraulic code. The only requirement to do this integration is to use the following steps (see Figure 3):

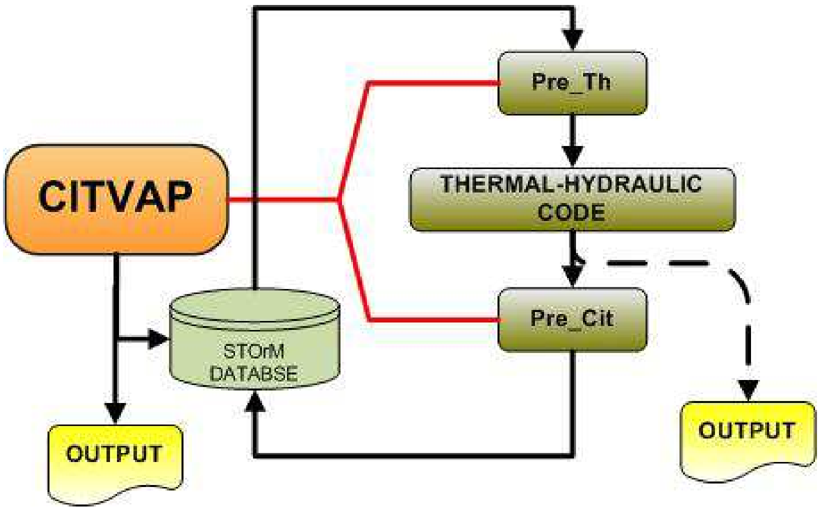


Figure 3: Methodology for the external thermal-hydraulic calculation.

- CITVAP generate a script to call at least 3 Different Programs.
- The First Program (in Figure 3 is named Pre_Th) is used to read the power distribution from the CITVAP database and convert it to the required format for the Thermal-hydraulic code.
- The second program is used for the proper thermal-hydraulic calculation. For example in CAREM NPP case this program is THERMIT code.
- The third program (in Figure 3 is named Pre_Cit) is needed to read the output of the thermal-hydraulic code, convert and save them in the CITVAP database. Normally the fuel and coolant temperature, and the coolant densities are needed. But in some cases another states variables can be modified (for example coolant boron concentration, or moderator temperature, density and boron concentration).

CITVAP properly update in his arrays the modified data, and checks the convergence of this loop.

5 CONCLUSIONS

CITVAP code has the capabilities to perform thermal-hydraulic feedback in the neutronic calculations for several Reactor Types and Cooling conditions. Nowadays four different models are imbedded in the program that allows the user to model: PWR, PHWR, MTR in natural convection and MTR in forced circulation.

All these methods were verified against other codes, and validated against the experimental measurements of the Power Feedback coefficients of the ETRR2, OPAL and RA-6.

The latest improvement mentioned allows using any thermal-hydraulic code to model this feedback, and this option was properly verified in the frame of the CAREM project.

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