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AZTLAN: Mexican Platform for Analysis and Design of Nuclear Reactors

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Abstract – *The AZTLAN Platform project is presented in this paper. This project aims to modernize, improve and incorporate the neutronics, thermo-hydraulics and thermo-mechanical codes developed in the Mexican institutions of higher education as well as in the Mexican nuclear research institute, in an integrated platform, established and maintained for the benefit of the Mexican Nuclear knowledge.*

I. INTRODUCTION

The Aztlan Platform project is a Mexican national initiative led by the National Institute for Nuclear Research, which brings together the nuclear institutions of higher education in Mexico: the National Polytechnic Institute, the National Autonomous University of Mexico and the Autonomous Metropolitan University, in an effort to take a significant step towards positioning Mexico, in the medium term, in a competitive international level on nuclear reactors analysis and modeling software. The project is funded for the next five years by the Sectorial Fund for Energy Sustainability CONACYT-SENER and one of its main goals is to build up as well as strengthen the national development of specialized nuclear knowledge and human resources. Particularly, the Aztlan Platform will reinforce substantially the research institutes along with the universities, in the area nuclear reactors design and analysis. This project aims to modernize, improve and incorporate the neutronics, thermo-hydraulics and thermo-mechanical codes developed in the Mexican institutions of higher education, in an integrated platform, established and maintained for the benefit of the Mexican Nuclear knowledge.

One of the innovative aspects of the project includes the establishment of a user's group; consisting of the member institutions of the project and by the Nuclear Power Plant - Laguna Verde (CNLV), the Regulatory Body - National Commission for Nuclear Safety and Safeguards (CNSNS), the Ministry of Energy (SENER) and the Karlsruhe Institute of Technology, Germany. This user's group will be in charge of utilizing the developed tools with the objective of providing feedback to the development team in order to meet the needs of the NPP and the regulatory body. This project represents an opportunity to enrich not only the science but also academics. Furthermore, the purpose of these developments is to provide validated simulation tools that can be easily coupled to the end user according to their specific needs. The Aztlan Platform project will positively impact Governmental Organizations, Research Institutes, Academic Institutions, and the NPP.

For the development of the Platform, the project is organized in four Working Groups: 1) Neutronics, 2) Thermo-hydraulics, 3) Coupling and Uncertainty and Sensitivity Analysis, and 4) the User's Group. Besides the neutronics, thermo-hydraulics and thermo-mechanical areas, the Platform will generate its Cross Sections in order to properly incorporate the thermal-hydraulic feedback in core calculations. The Aztlan Platform will follow stringent verification and validation (V&V) processes in order to meet international standards. The V&V will support and provide reliability to the platform. The V&V processes are diverse and will be applied to each developmental stage; considering the international Benchmarks, Monte Carlo reference solutions and even the use of full-scale plant data, experimental data, and comparisons against well validated computer codes. Sensitivity and uncertainty analysis will be applied as part of the development methodology of the Aztlan Platform.

With the aim of bridging the gap between similar developments worldwide, the Aztlan Platform will make use of the latest technologies of supercomputing to improve efficiency and speed up the calculation times. This work presents the Aztlan Platform, describing its proposed methodology as well as the goals and objectives to be pursued during the development.

I.A Nuclear Institutions

The National Polytechnic Institute (IPN), through the participation of students, teachers and researchers from the School of Physics and Mathematics (ESFM), has developed computational tools in order to solve numerically, using classical finite element and nodal methods, various models arising from the physics and engineering of nuclear reactors. These main neutronics models apply different approaches: the S_N discrete ordinate and the P_L approximation of spherical harmonics. These models have been studied and solved numerically with highly satisfactory results.

The Faculty of Engineering of the National Autonomous University of Mexico (UNAM) has developed and validated methodologies in addition to calculation schemes for the analysis and design of the core of nuclear reactors as well as for in-core nuclear fuel management, particularly related to fuel optimization. Its group has extensive experience on codes of the state of art as MCNPX, TRIPOLI and SERPENT, for the analysis and design of any type of nuclear reactor.

The Autonomous Metropolitan University (UAM) has especially focused on the development of mathematical and numerical models to simulate processes of heat transfer by conduction in fuel rods and the heat transport phenomena in two-phase flow.

At the National Institute for Nuclear Research (ININ), it is being carried out the development of a calculation tool with the purpose of evaluating the thermo-mechanical behavior of the fuel rods which are introduced in the CLV reactor, regardless of the operating conditions of the plant; considering the stationary state and in the case that a transient event occurs.

II. AZTLAN PLATFORM

The platform will consist of several modules coupled and integrated into a common platform (like SALOME platform, the base of the NURESIM European platform), which will provide the final user a friendly graphical user's interface with powerful pre- and post-processing tools.

The neutronics modules under development are the following:

- AZTRAN: A 3D transport code which solves numerically the multi-group time independent Discrete Ordinates neutron transport equation.
- AZKIND: a 3D diffusion module that solves numerically the time dependent neutron diffusion equations in Cartesian geometry.
- AZNHEX: a 3D diffusion module that solves numerically the time dependent neutron diffusion equations in Hexagonal-*Z* geometry.

The thermal-hydraulics module is called AZTHECA. This module is based in lumped and distributed parameters approximations, which includes the reactor vessel, the recirculation loops, the fuel pin temperature distribution, the core, lower and upper plenums and the pressure and level controls.

In the next sections a brief description of each module is presented.

III. NEUTRONICS MODULES DESCRIPTION

III.A The AZTRAN Neutronics Model

AZTRAN is a code resulting of an extension to 3D of its antecessor the 2D code TNXY [1, 2]. It solves numerically the multi-group time independent Discrete Ordinates neutron transport equations in XYZ geometry given by [3]

$$\begin{aligned}\Omega_n \cdot \nabla \psi_{ng} + \Sigma_{tg}\psi_{ng} &= \sum_{m=1}^{M}\omega_m \sum_{g'=1}^{G}\Sigma_{sg'\to g}(\vec{r},\Omega_n\cdot\Omega_m)\psi_{mg'} \\ &+\frac{1}{\lambda}\chi_g \sum_{m=1}^{N}\omega_m \sum_{g'=1}^{G}\nu\Sigma_{fg'}(\vec{r})\psi_{mg'}(\vec{r}), n=1,..,M, g=1,..,G\end{aligned} \tag{1}$$

where the parameters involved have the usual meaning. The method to approximate the space dependence of the neutron flux ψ_{ng} for the energy-group g and the n-th angular direction is based on the Raviart-Thomas-Nédélec-0 (RTN-0) nodal approximation [4]. In its present status, inner iterations are accelerated using coarse mesh rebalance [5], this is planned to be changed by a DSA technique [6]. Up to now, the AZTRAN code has been tested using some in-house exercises focused on the calculation of the k_∞ multiplication factor as well as the radial and axial power distribution in a given fuel assembly (FA). Figs. 1 and 2 show the 3D and 2D views of a typical nuclear core respectively. Nonetheless, the main objective is to use the AZTRAN code in full core calculations with some details inside a FA. It is also planned to extend the

AZTRAN static version to a time dependent version including the ones corresponding to the delayed precursors.

A neutron cross sections XS library will be built using the SERPENT code [7] with several burn-ups along with fuel temperatures, void fractions, control rod densities and their historic behavior. Finally, feedback effects will be taken into account by solving the thermal-hydraulic part using the AZTHECA code which is described in Section IV.

III.B AZKIND and AZNHEX Neutronics Model

The AZTLAN Platform includes also several codes, AZKIND and AZNHEX, which solve numerically the time dependent neutron diffusion equations in *XYZ* and Hexagonal-*Z* geometries, respectively. The equations are given by the following set of *G+I* partial differential equations. The first ones correspond to the neutron density balance for each one of the *G* energy groups and they are as follows:

$$\frac{1}{v_g}\frac{\partial \phi_g}{\partial t} - \nabla \cdot D_g \nabla \phi_g + \Sigma_{Rg}\phi_g = \sum_{g' \neq g}^{G} \Sigma_{sg' \to g}\phi_{g'} + \chi_g(1-\beta)\sum_{g'=1}^{G} \nu\Sigma_{fg'}\phi_{g'} + \sum_{i=1}^{I}\chi_{gi}\lambda_i C_i, g = 1,..,G \quad (2a)$$

and for the *I* groups of delayed precursor concentrations:

$$\frac{\partial C_i}{\partial t} = \beta_i \sum_{g'=1}^{G} \nu\Sigma_{fg'}\phi_{g'} - \lambda_i C_i, i = 1,..., I \quad (2b)$$

AZKIND is based on a Galerkin Finite Element Method using the RTN-0 polinomial nodal approximation to estimate both the scalar neutron flux and the neutron precursor concentrations. For the time dependence a θ-method was implemented. The algebraic system resulting is solved on each time step using the BiCGSTAB method. AZKIND was developed from the code NRKin3D [8].

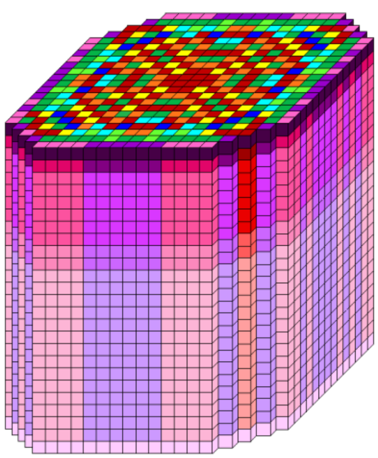


Fig. 1. A typical XYZ array for AZKIND

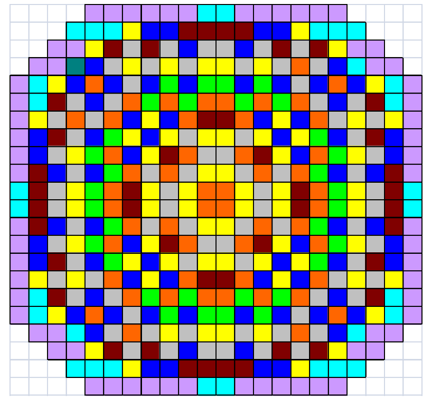


Fig. 2. A cross section view of the core for AZKIND

Regarding AZNHEX, it is an extension of the code developed in [9, 10]. It solves numerically Eq. (2) for domains like the one shown in Fig. 3, which 2D view is provided in Fig. 4. In order to accomplish this, one first applies a Gordon-Hall transfinite interpolation [11] to each one of the four quadrants shown in Fig. 5a to transform it in a cube as it is shown in Fig. 5b. Once that this is done, the remaining procedure to discretize the resulting equations is analogous to the one followed for AZKIND including the numerical algorithm for the time dependence and to solve the resulting algebraic system.

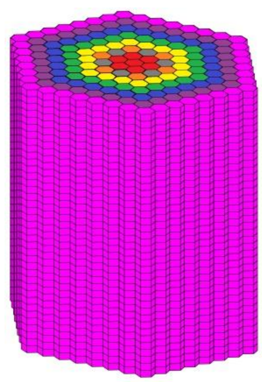


Fig. 3. A typical hexagonal-z array for AZNHEX

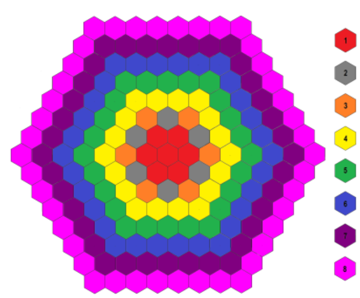
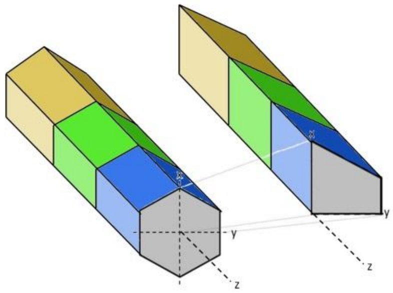
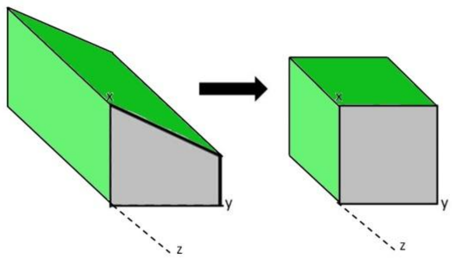


Fig. 4. A cross section view of the core for AZNHEX



a)



b)

Fig. 5. Gordon-Hall transformation for an hexagonal prism. a) Hexagonal prism is divided in four quadrants; b) One quadrant is transformed in a cube.

IV AZTHECA DESCRIPTION

The AZTHECA model is based in lumped and distributed parameters approximations, which includes the reactor vessel, the recirculation loops, the fuel pin temperature distribution, the core, lower and upper plenums and the pressure and level controls [12, 13].

The thermal-hydraulic model (THM Model) that describes the dynamic behavior of the lower and upper plenums and the reactor core, as well as the fuel temperature model, is based on the distributed parameters approximation. The vessel dome, downcomers and recirculation loops are based on the lumped parameters approximation.

The thermal-hydraulic model consists of a five equations model, which are based on liquid and gas phases mass balances, mixture momentum, mixture energy and liquid phase energy, together with a drift flux approach [14], for the analysis of phase separation. The non-equilibrium two-phase flows for the volumetric vapour generation rate in subcooled boiling is considered using Saha and Zuber's approximation [15]. The hydraulic model is based on a hypothetical averaging channel for the core, where the coolant is rising and is uniformly heated in each cell by the fuel (FUELHT Model). In this channel, coolant can be liquid, and in the two-phase flow and the convective heat transfer coefficient are both evaluated at these conditions.

The vessel dome is modeled as a two-region volume, one region being liquid and the other vapor. The two regions are assumed to be at the same pressure but not necessarily at the same temperature.

The recirculation model includes the pressure drops and flows from the downcomer, recirculation pumps, nozzles, jet pumps throat and diffuser, lower and upper plenum, reactor core and steam separators (Fig. 6) in order to obtain the momentum balances. The Feedwater System (FW) and Main Steam Line (MSL) models are considered as dummy or auxiliary models. The formulation of these models is based on resistive node approach. The reactor model is completed by including control models. In addition, this model uses a set of empirical correlations valid for the normal range of BWR operating conditions.

The reactor vessel was divided into five zones. Two of these zones, the vessel dome and the downcomer, have a variable volume according to the vessel water level. The three fixed volume zones are the lower plenum, which includes the jet pump volume; the upper plenum and steam separators, and the reactor core. Due to its importance on model performance, the latter can be subdivided into a significant number of cells.

IV.A THM Model

The THM model is the thermo-hydraulics model for single phase and two-phase core description, which is part of AZTHECA code. The THM model is based on liquid (subscript l) and gas (subscript g) phases, mass and energy balances, and mixture balances (subscript m).

$$\frac{\partial}{\partial t}(\rho_g \alpha_g) + \frac{\partial}{\partial z}(j_g \rho_g) = \Gamma \tag{3}$$

$$\frac{\partial}{\partial t}[\rho_l \alpha_l)] + \frac{\partial}{\partial z}(j_l \rho_l) = -\Gamma \tag{4}$$

$$\frac{\partial}{\partial t}(\rho_l h_l \alpha_l) - \alpha_l \frac{\partial p}{\partial t} + \frac{\partial}{\partial z}(\rho_l h_l j_l) = \frac{q_l'' P_H}{A_{x-s}} + q_l''' \alpha_l - \Gamma h_f \tag{5}$$

$$\frac{\partial}{\partial t}(\rho_m h_m) - \frac{\partial p}{\partial t} + G_m \frac{\partial h_m}{\partial z} = \frac{q'' P_H}{A_{x-s}} + q''' \tag{6}$$

$$\frac{\partial G_m}{\partial t} + \frac{\partial}{\partial z}\left(\frac{G_m^2}{\rho_m}\right) = -\frac{\partial p}{\partial z} - \frac{f\, G_m^2}{2\rho_m D_H} - g\rho_m \tag{7}$$

where

$$\rho_m = \rho_g \alpha_g + \rho_l \alpha_l \tag{8}$$

$$\rho_m h_m = \rho_g h_g \alpha_g + \rho_l h_l \alpha_l \tag{9}$$

$$G_m = \rho_g j_g + \rho_l j_l \tag{10}$$

$$\alpha_l = 1 - \alpha_g \tag{11}$$

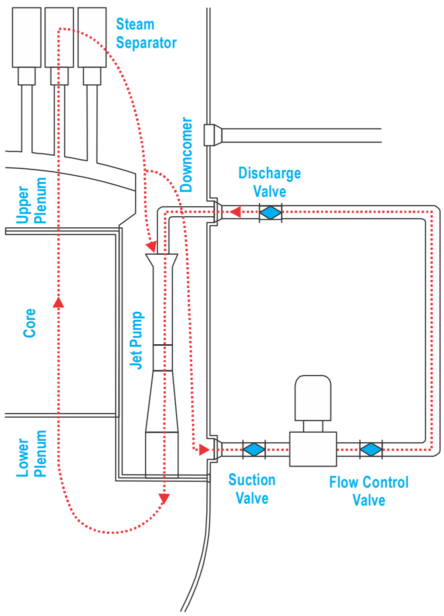


Fig. 6. Recirculation system flow path.

In these equations ρ is density, α_g is the void fraction, j is superficial velocity, Γ is the volumetric vapour generation rate, h is the enthalpy, p is the pressure, q'' is the heat flux, P_H is the heated perimeter, A_{x-s} is the cross-sectional area, f is the friction factor, D_H is the hydraulic diameter, and q''' is the volumetric heat generation by gamma radiation. For unheated sections (e.g., downcomer, lower and upper plenums), $q''=0$. The state equations for liquid phase are $\rho_l=\rho_l(p,h_l)$ and $T_l=T_l(p,h_l)$. For gas phase, $\rho_g=\rho_g(p)$ and $h_g=h_g(p)$.

For saturated boiling ($h_f<h_m<h_g$), the volumetric vapour generation rate Γ, is given by:

$$\Gamma=\frac{1}{h_{fg}}\left[\frac{q''P_H}{A_{x-s}}+q'''+\left(1-\rho_g\alpha_g\frac{\partial h_g}{\partial p}-\rho_l\alpha_l\frac{\partial h_f}{\partial p}\right)\frac{\partial p}{\partial t}\right] \quad (12)$$

For subcooled boiling ($h_m<h_f$ and $h_l<h_f$), the volumetric vapour generation is given by Lahey [16]:

$$\Gamma=\frac{1}{h_{fg}}\left[\frac{q''P_H}{A_{x-s}}\left(1-\frac{h_f-h_l}{h_f-h_{ld}}\right)+q''' -H_o\frac{h_{fg}}{v_{fg}}\varepsilon_g(T_s-T_l)+\varepsilon_g\left(1-\rho_g\frac{\partial h_g}{\partial p}\right)\frac{\partial p}{\partial t}\right] \quad (13)$$

where H_o is the condensation parameter. The subcooling liquid (h_f-h_{ld}) at the initiation of the subcooled boiling is given in [15].

The mass flow rate of each phase is given by:

$$W_g=j_gA_{x-s}\rho_g \quad (14)$$

$$W_l=j_lA_{x-s}\rho_l \quad (15)$$

where $j_l=j-j_g$ and $j_g=(C_oj+v_{gj})\alpha_g$, which is the drift flux model [14]. The distribution parameter C_o and average drift velocity v_{gj} are represented by a correlation for each of the two-phase flow regimes, bubbly and slug in the present case, in vertical pipes. The superficial velocity is calculated by:

$$j=\left[\left(\frac{1}{\rho_g}-\frac{1}{\rho_l}\right)\Gamma-\left(\frac{\alpha_g}{\rho_g}\frac{\partial\rho_g}{\partial p}+\frac{\alpha_l}{\rho_l}\frac{\partial\rho_l}{\partial p}\right)\frac{\partial p}{\partial t}-\xi_{sub}\right]\times\Delta z+\left(\frac{W_{gin}}{\rho_g}+\frac{W_{lin}}{\rho_l}\right)\frac{1}{A_{x-s}} \quad (16)$$

where ξ_{sub} is defined in [13], Δz is the cell length in the axial direction and W_{gin} and W_{lin} are the inlet mass flow rate of the gas and liquid phases, respectively.

In the BWR core the single phase at the bottom where $\alpha_g\approx0$, is predominant, after gives rise two-phase flow as subcooled boiling and saturate boiling, as shown in Fig 7.

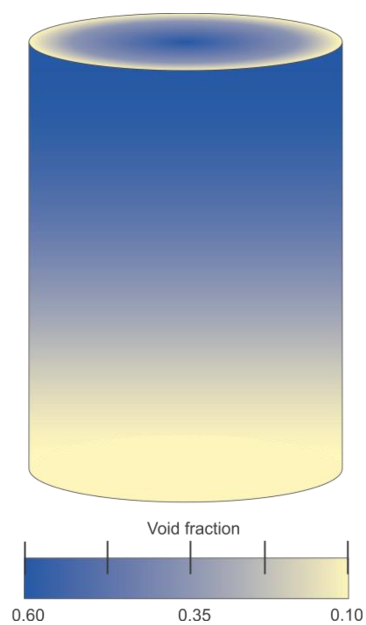


Fig. 7 Results of AZTHECA Code. Void fraction in the BWR core.

IV.B FUELHT Model [17]

The fuel heat transfer formulation is based on the following fundamental assumptions: (i) Axis-symmetric radial heat transfer, ii) the heat conduction in the axial direction is negligible with respect to the heat conduction in the radial direction, iii) the volumetric heat rate generation in the fuel is uniform in each radial node, and iv) storage of heat in the fuel cladding and gap is negligible. Under these assumptions, the transient temperature distribution in the fuel pin, initial and boundary conditions leads to:

$$\rho Cp \frac{\partial T}{\partial t} = \nabla \cdot (\mathbf{K} \cdot \nabla T) + q'''(t)\,, \quad \text{at} \quad r \le r \le r_f \qquad (17)$$

$$\text{I.C.} \quad T(\mathbf{r},0) = T_0(\mathbf{r})\,, \qquad \text{at } t = 0 \qquad (18)$$

$$\text{B.C.1} \quad -K_{rr}\frac{\partial T}{\partial r} = H_\infty (T|_{r=r_{cl}} - T_m)\,, \quad \text{at } r = r_{cl} \qquad (19)$$

$$\text{B.C.2} \quad \mathbf{e}_r \cdot \nabla T = 0\,, \qquad \text{at } r = r_0 \qquad (20)$$

where $q'''(t) = 0$, for $r_f \le r_{cl}$, i.e., for gap and clad regions. In these equations, r is the cylindrical radial coordinate, r_0, r_f and r_{cl} are the centroid, fuel and clad radius, respectively, $q(t) = P(t)/V_f$ at each axial node, where P is the neutronic power, T_m is the moderator temperature, and H_∞ is the convective heat transfer coefficient.

The differential equations described previously are transformed into discrete equations using the control volume formulation technique in an implicit form [18].

Application of the control volume formulation enables the equations for fuel, gap and cladding to be written as a single set of algebraic equations for the sweep in the radial direction:

$$a_k T_k^{t+\Delta t} = b_k T_{k+1}^{t+\Delta t} + c_k T_{k-1,}^{t+\Delta t} + d_k \qquad (21)$$

where $T_{k-1}^{t+\Delta t}$, $T_k^{t+\Delta t}$, and $T_{k+1}^{t+\Delta t}$ are unknowns, the coefficients a, b, c and d, are coefficients, which are computed at the time *t*.

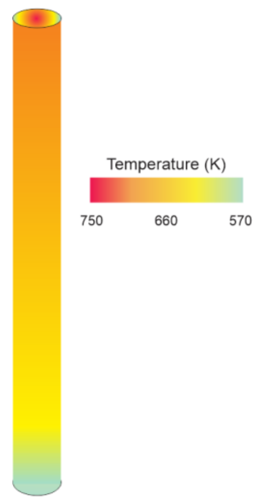


Fig. 8a. shows some fuel temperatures in the BWR core calculated with AZTHECA.

V. COUPLING METHODOLOGY

The coupling of multiphysics phenomena is a very complex subject with several possible combinations. An extensive description of this is given in [19], and some details and key points are found in [20, 21 and 22].

In the past, the thermal-hydraulic analysis used simplified neutronics models, like point kinetics, including when needed the balance of plant. The result of such simulations provided the necessary boundary conditions for the core, such as mass flow and temperature distribution of the coolant at the core inlet together with the time-functions for pressure, which could be analyzed with detailed 3D neutronics models in order to get more information. However, in reality, these boundary conditions are functions of the power generation in the reactor core. The application of these models is, therefore, limited by the consideration of proper core thermal-hydraulic interface conditions and it may lead to very unrealistic accident conditions if all uncertainties are taken into account by demanding conservative boundary conditions [23, 24].

The coupled code calculation approach constitutes the normal evolution of these methods. This is especially true in cases where strong feedback between the core neutronics behavior and the plant thermal-hydraulics is present, as well as in situations in which neutron flux distortions and excursions are important and its spatial distribution changes during the transient.

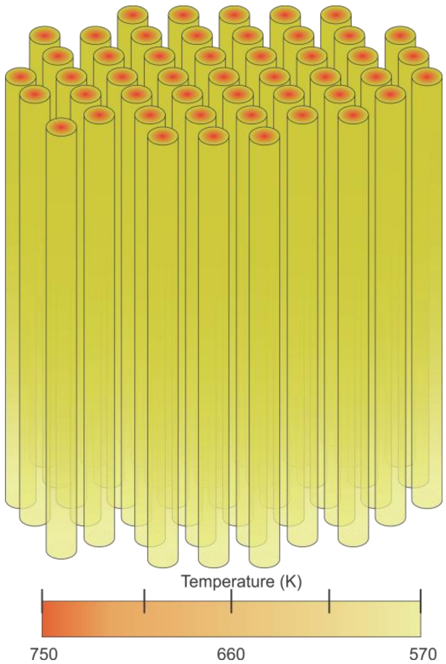


Fig. 8b. Fuel temperatures in the BWR core calculated with AZTHECA.

In the case of system codes coupled with 3D neutron kinetics models, six basic components of the coupling methodologies have been identified in order to be able to couple two codes [20 and 21]. Thus, the way of coupling (internal or external); the coupling approaches (serial integration or parallel processing coupling), the spatial mesh overlays (fixed or flexible), the coupled time steps algorithms (synchronization of the time steps), the numeric coupling (explicit, semi-implicit and implicit) and the coupled convergence schemes must be considered and implemented.

In a coupling scheme System code – 3D Neutronics, an integration algorithm usually considers the treatment of the neutronics code as a subroutine of the system code [20, 22 and 23], whereas in a 3D Neutronics – Subchannel code coupling, the subchannel code is implemented as a subroutine in the 3D Neutronics solver [25, 26 and 27].

The development of coupled codes in the past years has been fostered by safety analysis requirements. Although the most important part of a nuclear reactor is the core, several accidents are originated in the primary or secondary loops or even in some other components like the turbine. The coupling can be achieved in three different ways: internal, external and combined [24].

In the case of the AZTLAN Platform, an internal coupling with an explicit scheme for the numeric coupling, like the ones shown in the Figs. 9 and 10 are to be implemented. The 3-D nodal neutron kinetics model (AZKIND or AZHNEX) is integrated into the core section of the thermal-hydraulics module AZTHECA, Fig. 11. Each neutronics node is to be coupled directly to a core thermal-hydraulic computational volume in the system code. The mesh sizes could be different, but in such a case an interpolation procedure for the parameter transfer would be necessary. Even though this method requires the exchange of a significant amount of information between the two codes (power and thermal-hydraulic feedback "TH–FB"), it also allows for detailed and direct system calculations. One major disadvantage of this method is that it involves significant modifications in both codes. The modifications, however, can be done in such a way that if new versions of the codes are released, or if it is desired the coupling with some other code, no changes or minimal changes of the new coupling routines are necessary to generate the coupled code.

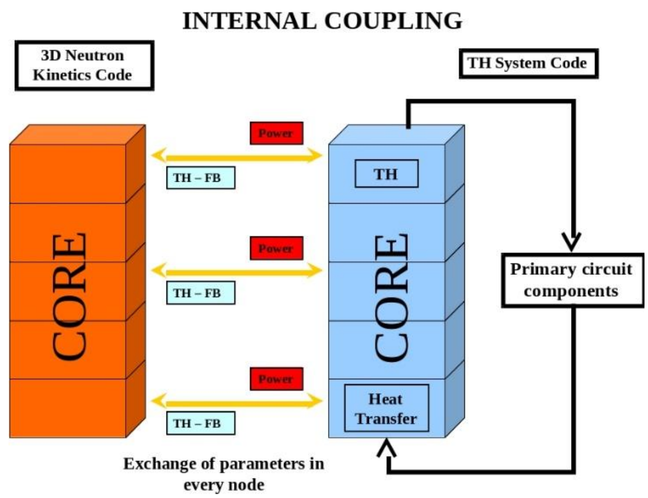


Fig. 9. Internal coupling of neutronics and thermal-hydraulics codes.

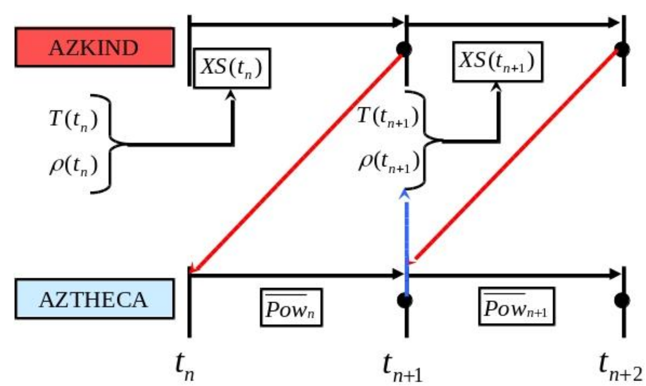


Fig. 10. Example of an explicit numeric coupling scheme between the AZTLAN modules AZKIND and AZTHECA.

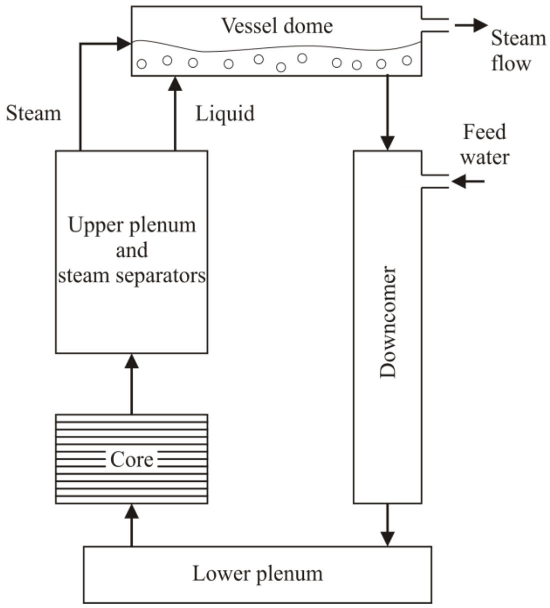


Fig. 11. Simplified structure of the AZTHECA thermal-hydraulics model.

In order to close the coupling, the Monte Carlo code SERPENT [7] is to be used for the cross sections (XS) data generation. Together with some tools, under development, the generation and extraction of XS from SERPENT to be used in the neutronics models of AZTLAN is foreseen. A flow diagram of such strategy is shown in Fig. 12.

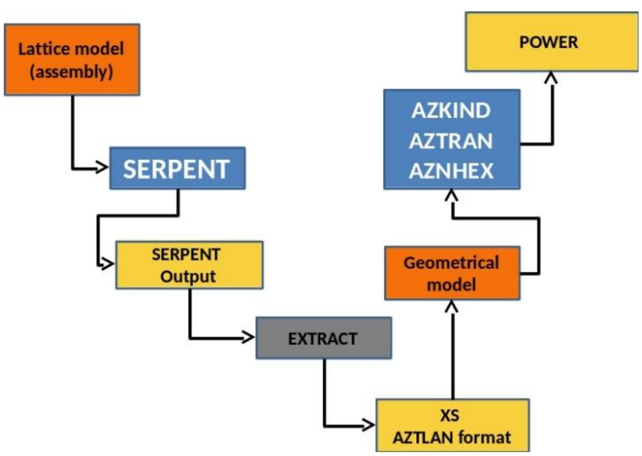


Fig. 12. Flow diagram of XS generation and utilization in the neutronics modules of AZTLAN.

VI. SENSITIVITY AND UNCERTAINTY ANALYSIS

The AZTLAN Platform intends to implement Sensitivity and Uncertainty (S&U) analysis along the calculation chain. The aim is to use the uncertainties' propagation methodology and the goal is to use the SUSA code (Software System for Uncertainty and Sensitivity Analyses) developed by GRS, Germany [28]. The SUSA code is based on the Wilks formula and on random sampling in Monte Carlo on a set of probability density functions (pdf's).

The steps to be considered in the S&U analysis with SUSA are: 1) Identification of potential uncertainties. 2) Uncertainties range definition, i.e. selecting maximum and minimum values for the input options. 3) Pdf's specification on these ranges. 4) Identification and quantification of dependencies between parameters. 5) Generate a random sample of size N for creating input files using a Monte Carlo type method. 6) Perform N runs using the code under analysis; from each execution, a response is obtained that can be for example the peak power. 7) Quantitative parametric sensitivity calculation to identify uncertain parameters that contribute the most to the uncertainty of the answer. 8) Interpretation of results.

VII. CONCLUSIONS

This paper introduces the Mexican platform for analysis and design of nuclear reactors, which will be developed under the Aztlan Platform project. This Mexican national initiative is a collaboration project between the National Institute for Nuclear Research, National Polytechnic Institute, the National Autonomous University of Mexico and the Metropolitan Autonomous University. The objective of this project is to have an internationally competitive tool for calculation, analysis and design of nuclear reactors. The methodology of this project involves mathematical models, as well as fully developed and implemented numerical models, by the Mexican institutions.

This project represents an opportunity to enrich science and academics. Furthermore, the purpose of these developments is to provide validated simulation tools that can be easily coupled by end users according to their specific needs, for example: research institutions (ININ/IIE), academic institutions (IPN, UNAM, UAM), industry, the regulatory body (CNSNS), and the nuclear power plant (CLV/CFE).

In general terms, the methodology proposed is comprised by the following major areas: neutronics, thermal-hydraulics, thermo-mechanics, coupling, S&U, V&V, and the user's group. The project is funded for the next five years by the Sectorial Fund for Energy Sustainability CONACYT-SENER. The authors humbly believe, and strongly hope, that this will be the beginning of the first stage of the platform and one of its main goals is to build up as well as strengthen the national development of specialized nuclear knowledge and human resources.

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NOMENCLATURE

AZKIND: a 3D diffusion module that solves numerically the time dependent neutron diffusion equations in Cartesian geometry.
AZNHEX: a 3D diffusion module that solves numerically the time dependent neutron diffusion equations in Hexagonal-*Z* geometry.
AZTRAN: A 3D transport code which solves numerically the multi-group time independent Discrete Ordinates neutron transport equation.
CNLV: Power Plant - Laguna Verde.
CNSNS: the Regulatory Body - National Commission for Nuclear Safety and Safeguards.
ININ: National Institute of Nuclear Research.
IPN: the National Polytechnic Institute.
KIT: the Karlsruhe Institute of Technology.
UAM: the Autonomous Metropolitan University.
UNAM: the National Autonomous University of Mexico.
SENER: the Mexican Ministry of Energy.
S&U: sensitivity and uncertainty analysis.
V&V: verification and validation.

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