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Review Article

INVAP's Nuclear Calculation System

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Since its origins in 1976, INVAP has been on continuous development of the calculation system used for design and optimization of nuclear reactors. The calculation codes have been polished and enhanced with new capabilities as they were needed or useful for the new challenges that the market imposed. The actual state of the code packages enables INVAP to design nuclear installations with complex geometries using a set of easy-to-use input files that minimize user errors due to confusion or misinterpretation. A set of intuitive graphic postprocessors have also been developed providing a fast and complete visualization tool for the parameters obtained in the calculations. The capabilities and general characteristics of this deterministic software package are presented throughout the paper including several examples of its recent application.

1. Introduction

INVAP utilizes a proprietary software package to design nuclear installations such as research reactors, radio-isotopes production reactors, and nuclear power plants. With every project, the codes became more flexible and user-friendly to the point that the two graphic postprocessors provide intuitive and self-explanatory 1D, 2D, and 3D (currently in development) representation of the variables involved in the calculation and the results obtained.

This software package includes the following.

(i) CONDOR, a *cell code* which performs neutronic calculation of fuel assemblies of nuclear reactor cores and produces sets of data, like the homogenized and condensed cross-sections to be used in the reactor (or core) code.

(ii) CITVAP, a *reactor code* which performs neutronic burnup-dependent calculation of nuclear reactor cores, calculating the nuclear parameters associated with several states of the reactor taking into account the feedback of many thermal-hydraulic parameters. INVAP has a nodal reactor code in development which uses automatic pinpower reconstruction.

(iii) ESINLM, a *nuclear data library manager* which performs the update and upgrades of the nuclear data required.

(iv) HXS, a *library manager* which provides the link library between the cell code and the reactor code allowing the user to create or modify the data library to be used in the reactor code.

(v) POSCON and FLUX, two *graphic postprocessors* which enable the user to view in a fast and easy way the calculated parameters of both the cell code and the reactor code and export information and graphics from the respective databases.

(vi) Several *utility programs* that serve specific functions to ease the interface with the user by graphically preprocessing the geometry input data, exporting data in MCNP format, and many other functions required (The program used to export data to MCNP is called NDDUMP and its usage is briefly described in [1].).

The nuclear calculation system (the most relevant programs are shown in Figure 1) includes the calculation codes, utility programs, processors and an external calculation code called MCNP (MCNP is a general purpose Monte Carlo N-Particle code that can be used for neutron, photon, electron, or coupled neutron/photon/electron transport. This widely spread code has been developed by Los Alamos National Laboratory (EEUU) and has been extensively tested worldwide with more than satisfactory results in many benchmarks).

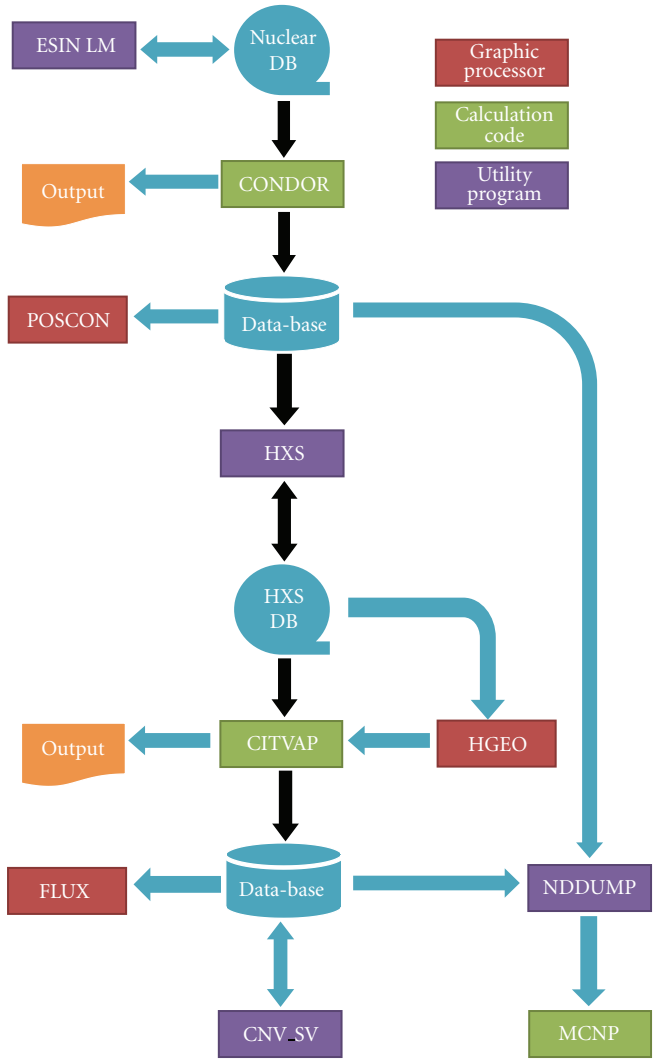

Figure 1: INVAP's nuclear calculation system.

This external code is used as an alternative method in the cases where the diffusion approximation that the reactor code uses is not accurate enough or cannot provide the desired parameters. Normally, the conditions that require the use of this stochastic particle transport method are vacuum media, the vicinity of strong absorbers, and the flux or power estimation on small targets. The most common scenarios of application for the Monte Carlo code are neutron beams, the crosschecking of control rod worths, shielding calculations, estimation of gamma-heating, and energy deposition (or radiation damage) on specific small targets.

This calculation system has been used by INVAP and several of its clients for the design, optimization, and followup of several reactors throughout the world obtaining optimum results:

(i) RA6 (research reactor in Argentina): operating since 1982,

(ii) NUR (research reactor in Algeria): operating since 1988,

(iii) RA-8 (critical facility in Argentina): reached Criticality 1997,

(iv) ETRR2 (research and radio-isotope production reactor in Egypt): operating since 1998,

(v) OPAL (research and radio-isotope production reactor in Australia): operating since 2006,

(vi) CAREM (small nuclear power plant in Argentina): currently under design [2],

(vii) CNA II (nuclear power plant in Argentina): currently in construction.

The results of the calculation system have been evaluated providing very good agreement with other calculation systems and experimental data in many benchmarks (see [3–14]), and it is currently being used for design in many state-of-the-art reactors to be built in the near future. It has also been utilized for the modifications to the original design that some of our clients desire in order to keep up with the new

requirements of modern day physics and material testing and the dynamic radio-isotope production market.

All the codes and utility programs were developed using FORTRAN programming language. This ensures the scalability of all the software in the calculation system, making it easy for INVAP to update, combine or improve any single piece of the package in a fairly independent manner. There are of course some standards that link some of the programs. A clear example of this is the database architecture. At the present state, CONDOR and CITVAP databases are generated in different architectures. This is being changed in the versions in development in order to homogenize the interaction of all the programs with a single proprietary database access library that will provide fast (indexed) access to the information as well as great potential for further development.

The graphic processors are also programmed in FORTRAN language and currently use a commercial graphic development library for generating and managing the GUIs which has proven to be both programmer and user-friendly.

All the software is compiled (and has been successfully tested) for use in Microsoft Windows and Linux systems. Furthermore, the database architectures of both CONDOR and CITVAP are fully portable between these two platforms.

The most relevant codes of the calculation system are introduced in the following sections with several examples of their recent application.

2. Cell Calculations

The two-dimensional cell code used to generate burnup-dependent cell parameters is called CONDOR. This code (developed by INVAP; see [15]) uses a two-dimensional heterogeneous response method (HRM) (This method is also known as the CCCP (current-coupling collision-probability) method and is used by HELIOS (see [16–22]) and PHOENIX-H as shown in [23]) with angular dependent coupling currents to calculate flux distribution along the components of the fuel assemblies.

In research reactors, fuel assembly optimization is a recurrent challenge, and this method provides a fast and accurate alternative to the most common (also available in CONDOR) collision probability method (PC) for large, complex fuel assemblies (Figure 2).

In addition, this method uses the subgroup theory for treating resonances in fuel zones. This provides an accurate method for calculating resonances in 2D MTR (MTR: material testing reactor. In this case it refers to fuel assemblies that are not a cluster of pin cells like the PWRs.) fuel assemblies, taking into account the heterogeneous character of the fuel assembly. This is a key development for the correct calculation of complex plate fuel assemblies that contain very thin burnable poisons in the frames like the one depicted in (Figure 3).

The preprocessing capabilities of CONDOR enable the use of regular expressions to create the necessary geometries in a simple way. Using HRM, the final model can be constructed repeating or combining several different cell-blocks. These blocks are then individually calculated using

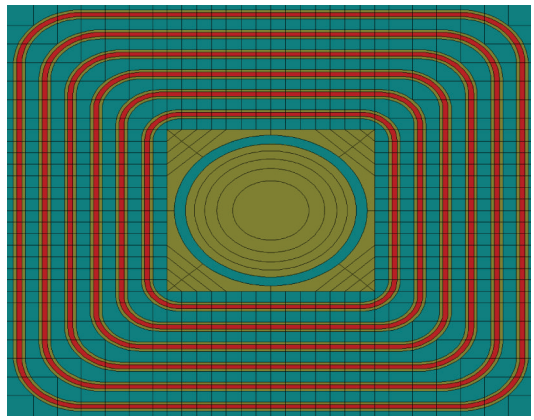

Figure 2: WWR fuel assembly.

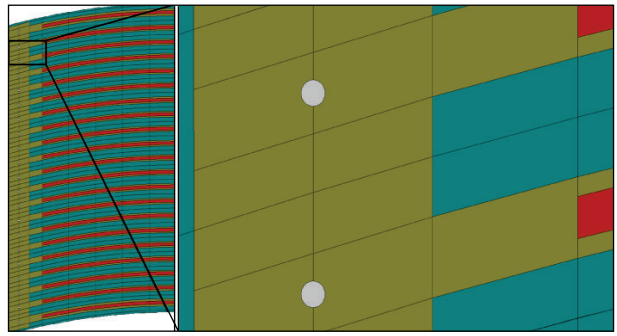

Figure 3: Half MTR fuel assembly (left) and a close-up of the cadmium wire section (right).

the collision probability method and coupled by the inward and outward angular dependent currents of the connecting boundaries. This methodology is suitable for most calculations and provides a more compact geometric definition of the problem and a shorter calculation time.

CONDOR has the capability to calculate fuel assemblies (with fuel rods or plates) with full geometrical detail (without homogenization). This has been used in many designs, especially for research reactors, in which the geometry is another optimization variable exploited in order to obtain the best performance possible. However these are not the only cases. For example, in CNA II the tube-shaped control rods are located between the fuel cluster assemblies. In this case, the control rods are not just a pin in the fuel assemblies. They affect all nearby channels, locally depressing the neutronic flux. Because of this (and some other factors), the model used for calculating cross-sections for these control rods should consist of more than one fuel cluster assembly and has been modeled as an arrangement of seven fuel assemblies as can be seen in Figure 4. This represents one of the applications of the flexibility of CONDOR for nuclear power plants.

CONDOR also possesses a postprocessor called POSCON that provides an easy way of plotting and analyzing the results of the cases executed by the user (see Figures 2, 3, and 4). The graphic interface is versatile and

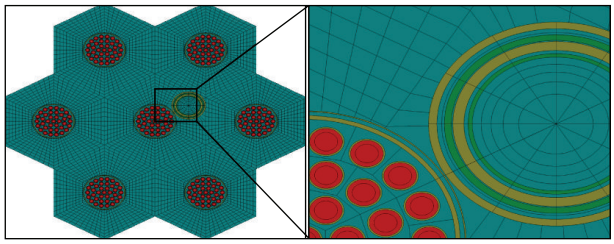

Figure 4: CNA II control rod model (left) and a close-up of the control rod section (right).

powerful as to mathematically combine one-dimensional curves using simple arithmetic expressions. This feature facilitates the analysis of diverse parametric studies required for many innovative designs.

The calculation speed and extensive geometric capabilities of CONDOR allow 2D full core, burnup-dependent calculations to be made on MTR reactors to evaluate certain specific parameters, giving the user the ability to perform parametric studies with the actual gradient of the flux in a geometrically accurate model, also taking advantage of the thin energy discretization. This has been recently used in with optimum results for the design of critical facilities, in which the core consists of a heterogeneous array of pin cells and in reactors like OPAL, in which the fuel assemblies possess thin burnable poisons.

Another specific application of the full core model is the precise placement of the correct temperatures of each pin in all the fuel assemblies. This provides a very accurate model that can be used for certain applications like the estimation of power feedback coefficients for fresh cores. However, such precision is normally not required for calculation feedback coefficients. These values are usually calculated using a full 3D reactor model with a core code (CITVAP in our case).

3. Core Calculations

CITVAP (see [24]) is a code derived from the well-known diffusion code CITATION II (CITATION II is a three-dimensional reactor code originally released in 1971 by Oak Ridge National Laboratory (EEUU) to solve problems using the finite difference representation of neutron diffusion theory (see [25])). CITVAP greatly enhances the capabilities for design and fuel management provided by CITATION II, giving the user an easy-to-use set of free-format keywords that provide a more intuitive input file.

This three-dimensional diffusion code is in constant development, enhancing (as were required) mostly its administrative capabilities. The fuel management, search for equilibrium cores, and followup characteristics of the code have been greatly improved in the last couple of years. This was motivated primarily by the development of new, different designs that require certain special features. The CNA II reactor, for example, contains oblique-entering control rods. This particular unique design needed the modification of the code in order to let the user model, position and operate these control rods in a simple and efficient manner. This lead

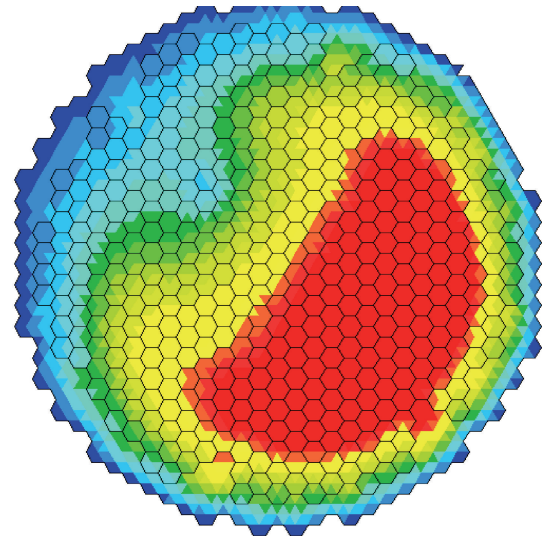

Figure 5: TZ model of CNA II.

to a new set of related keywords that assured a simple and self-contained input file that satisfied the client's needs.

CITVAP can handle rectangular (XYZ), cylindrical (θRZ), and triangular (TZ) geometry, with being the latter, particularly useful for modeling hexagonal fuel reactors like CNA II (see Figure 5).

However, as the most common geometry used for modeling complex designs has proven to be the rectangular representation, an XYZ preprocessor has been designed that allows the user to input the general rules for discretization and the shapes of the actual components and converts them to approximate XYZ representations maintaining the volumes of each component. This graphic preprocessor is called HGEO (see Figure 6) and allows the user to change the free-format input file and view the changes almost instantly using a very fast GUI (graphic user's interface). HGEO is a program that has been greatly expanded in the past few years, taking advantages from the input processing capabilities originally developed for CONDOR. Some of these features allow the user to define generic templates which have several input parameters that can be used to construct arrays of complex similar objects without making the input file too hard to interpret and/or check.

The operation followup characteristics of CITVAP are being permanently used by our clients for cross-checking purposes and to verify new experiments and irradiations in their reactors. Some small utility tools are also designed and implemented for specific use certain clients like filters for operational data (mainly power and control rod positioning) produced by the reactor monitoring system or postprocessors that extract large portions of specific data from CITVAP database to be input to another program.

As model and cycle detail was raised by increasingly complex new designs, CITVAP was adapted to take advantage of multicore modern-day computers. This was achieved using the available OPENMP directives. These directives proved to utilize the full extent of the multiple cores in new processors in a very efficient way.

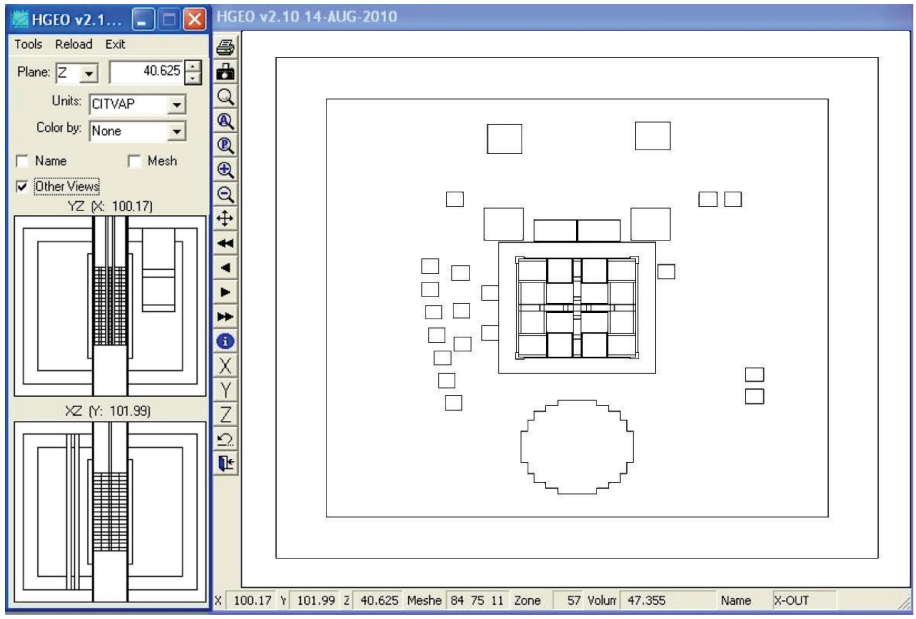

Figure 6: Example visualization of a CITVAP input model using HGEO.

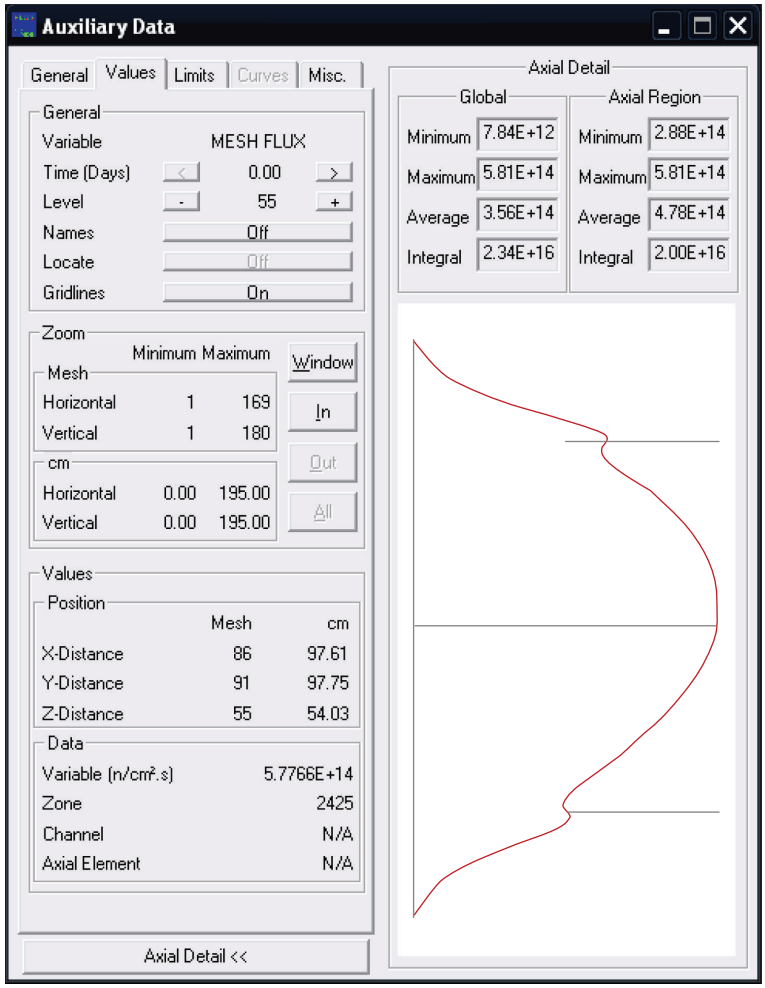

Figure 7: FLUX's auxiliary data child window.

As required by the design of nuclear power plant cores, CITVAP contains several codes that provide thermal-hydraulic feedback in stationary conditions. These codes use different theories to cover the fundamental behavior of MTR and NPP designs. The code can handle natural and forced convection primary cooling systems in one or two-phase flow conditions. This feedback capability is mandatory for the correct analysis of NPPs primarily because of the widely different temperature and density conditions throughout the reactor core.

In the same way as POSCON was created to visualize the parameters calculated by CONDOR, FLUX graphic postprocessor has been developed in order to maximize the design orientation of CITVAP (see Figures 5 and 7). This GUI lets the user plot a wide set of variables involved in the calculation: temperature, density, fuel burnup, control rod insertion level, selected isotopes concentration (Boron, Xenon, Samarium, Promethium, and Iodine), neutronic flux, neutronic fluence, power density, multiplication factor, reactor power, power peaking factor, fuel movements during refueling, and many more. All these variables can be represented as a function of position (if applicable) and time. It also allows the user to plot these parameters following a particular fuel assembly that might be relocated several times by the refueling schedule.

The program was designed to present most numeric data in a floating auxiliary child window (as shown in Figure 7) that pops out when plotting a two-dimensional graph. This window can be relocated or minimized according to the required use and shows specific data regarding a particular point as the user hovers the 2Dplot with the pointer. It also shows a one-dimensional plot of the variable in the direction that is not being shown in the 2Dplot, giving the necessary information for a complete evaluation of the three-dimensional characteristics of the selected variable.

The recent resurgence of nuclear industry has triggered the worldwide design of many NPPs. This type of reactors focuses on very efficient use of the nuclear fuels, requiring the most accurate calculations available to date in order to make all the assemblies be burnt to the same level (i.e., keep the power peaking factor to the lowest value possible).

To achieve these and other specific design requirements in the most precise and efficient way, INVAP is developing a new, state-of-the-art reactor calculation code using nodal method and with similar calculation capabilities as CITVAP, but which is much faster and includes more detailed characteristics as pin power reconstruction and pin-dependent isotopic concentration, among others. This code is currently in the final development phase and will be added to the nuclear calculation system in the near future.

4. Conclusion

The calculation system used by INVAP consists of a wide variety of calculation codes, utility programs, and graphic processors which, coupled with Monte Carlo particle transport code MCNP, covers the design necessities for facing the near future challenges. This deterministic calculation package provides a very good agreement with both measurements in INVAP reactors and other methods used worldwide, including MCNP (see [3–14]), which make it suitable for the design of new reactors and the characterization of existing ones.

This software package is in constant improvement, ensuring that the users receive the full extent of the currently available technology to date in order to enhance the evaluation and optimization of new, state-of-the-art nuclear installations.

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