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THE GENERALIZED SYNTHETIC DIFFUSION METHOD. IMPLEMENTATIONS IN THE SEANAP CODE SYSTEM AND VALIDATION FOR PWR CORE ANALYSIS

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INTRODUCTION

In nuclear reactor core analysis, as well as in other physics and engineering fields, the equations that result from particle or radiation transport problems should be solved for a large, multidimensional and heterogeneous configuration space, with significant changes in material densities and temperatures (thermal hydraulics), isotopics (fuel burnup and poison depletion), control rod insertions and fuel loading patterns. There is obviously a high incentive for reducing the order (i. e. finite-difference diffusion) and the number of mesh points and energy groups in the final discretized solutions for the whole system, not only to reduce the computer memory and time required for each solution sweep; but also, mainly, to avoid the slowing-down of convergence rates, and thus the increase of iterations, when the number of loosely coupled mesh-group elements increases.

This reduction to low-order solutions in coarse-mesh homogenized nodes and a few (or one) energy groups raises two key questions: how to average or homogenize the terms and constants of the reduced discretized equations while preserving the accuracy (in nodal integrated values) and simplicity (plus stability) of the low-order solution; and how to recover or reconstruct accurately the detailed intranode solution (i. e. the fuel pin power distribution).

For PWR core analysis, most previous computational development work has proceeded along two different lines. The first, continuing the earlier succesful experience in PWR design after the US naval reactor program, relies mainly on detailed finite-difference diffusion calculations of whole two-dimensional (XY) core plus reflector plane configurations. A fine-mesh (one point per fuel cell) two-group discretization is used, with point or block depletion and other feedback effects. Single fuel cell spectral and depletion calculations, with auxiliar one-dimensional transport calculations for absorbers and reflector components, directly provide the cell homogenized two-group cross section libraries. Core axial one-dimensional diffusion codes and three-dimensional nodal core simulators, used for extensive core analysis, are "adjusted" through several parameters (i. e. bucklings, nodal factors, albedos, removals or absorptions) to match, more or less accurately, the fine-mesh solutions at some "reference" cases.

The other line, initiated in the 70's, relies on more sophisticated fuel assembly codes and advanced nodal methods. Whole fuel assemblies, or color-sets made up of four quarters of neighboring assemblies, are calculated separately (zero currents at boundaries) by detailed multicell spectral and two-dimensional multigroup transport calculations along burnup and perturbed conditions, obtaining the average

assembly cross sections and heterogeneity factors. The three-dimensional two-group core simulators use high-order nodal methods and improved superposition procedures for intranode pin-power recovery. Good accuracies and efficiencies are thus achieved for fuel assemblies with heterogeneities (i. e. absorber cells) far from or regularly configured over the boundaries. But for assemblies with large and asymmetric burnup gradients or boundary spectral changes (i. e. those close to heavily poisoned assemblies or to reflectors), the independent assembly calculations are far from the actual boundary conditions and internal burnup distributions, so that increasingly different combinations of color-sets found in successive reload cycles.

In collaboration with the Spanish electric utility owners of PWR power plants, we have undertaken since 1980 the development and validation of a code system for PWR core analysis as an independent in-house support capability. The first version was developed in 1980-81, including mainly: PWR fuel assembly calculations using the cylindricalized multicell cluster models of WIMS-D; detailed XY two-group diffusion calculations using the CITATION and VENTURE codes with a pin-by-pin mesh in the whole core plus reflector planes; and a new synthetically corrected one-group nodal method for 3D PWR core simulation, with all correction factors precalculated explicitly from the detailed two-group XY (and Z) solutions. A summary of those developments and the validation results were published in 1982 (1). That early code system was improved, documented and made available through our participation in a Coordinated Research Program of the International Atomic Energy Agency (2).

Our synthetic diffusion method, which proved to be very effective for one-group nodal calculations in PWR cores, was later (3,4) generalized and implemented for synthetic fine/coarse-mesh fewgroup diffusion calculations of two-dimensional (XY) PWR core planes and for one-group synthetic diffusion acceleration of multigroup calculations. As a result, a very efficient and accurate code, named COBAYA, was developed (4,5) for detailed pin-by-pin PWR core calculations and explicit generation of the correction parameters for the one-group nodal simulator and introduced in our code system, named SEANAP (acronym for Sistema Español de Análisis de Núcleos de Agua a Presión). Validation results with measurements and tests for 11 cycles of 4 spanish PWR units were obtained (6) in collaboration with the utilities. One of our research associates has introduced (7) the relevant neutronic and thermal hydraulic feedbacks and detailed burnup at the cell level in a new extended version, called COBAYA-87, and is implementing the extensions to multigroups (several thermal) with advanced synthetic acceleration techniques and transport corrections at the cell level. Another of our colleague professors has successfully applied (8) our synthetic diffusion method to accelerate highly nonlinear radiation transport calculations.

In this review paper, we discuss first, our generalized synthetic diffusion method, that is based on a very simple, but stable and regular, correction of the first-order finite-difference diffusion discretizations. The fine structure variations in space (heterogeneities), energy and angle of the neutron flux is completely accounted for in the low-order solution (coarse-mesh and few-group diffusion), while preserving its simplicity, efficiency and stability, by means of the extra local correction coefficients introduced, the generalized interface flux discontinuity factors (IFDFs). The IFDFs are explicitly obtained in terms of the net interface currents and average fluxes, diffusion coefficients and widths of the adjacent nodes, obtained from previous (reference, still unconverged or reduced in space or dimensions) high-order diffusion or transport solutions in fine-mesh (heterogeneous) and multigroups. The general properties of the IFDFs are then translated to the performance of the synthetic method, concluding that it answers in a simple, but general and accurate way, the questions raised at the beginning.

Next, we summarize the implementations of the method, including: the one-group coarse-mesh synthetic nodal method; the iterative synthetic fine/coarse-mesh few-group two-dimensional (XY) diffusion method for PWR core planes; and the one-group synthetic acceleration of multigroup diffusion calculations.

In other Section, we describe the main characteristics and capabilities of the SEANAP code system for PWR analysis, and, specially, the role and utilization of our synthetic diffusion method. Next Section includes some selected SEANAP validation methodology and results compared with measurements in tests and operation of Spanish PWR cores. Finally, last Section is the summary and conclusions of the paper.

GENERALIZED SYNTHETICALLY CORRECTED FINITE DIFFERENCE FORMULATION

Instead of correcting the diffusion coefficient in Fick's law, we correct locally the finite difference approximation of the flux gradient at the mesh interfaces, so that the net currents are given in terms of the actual mesh average fluxes and corrected discontinuous interface fluxes. At the interface $I + 1/2$ between mesh nodes I and $I + 1$, the actual net current in group G is locally given from the left side by:

$$J^G_{I+1/2} = -\frac{D^G_I}{h_I}\left(f^G_{I,I+1/2}\,\Phi^G_{I+1/2} - \Phi^G_I\right), \qquad (1)$$

where

D^G_I = average diffusion coefficient of node I in group G.
h_I = half-width of node I in the direction transverse to the interface.
$\Phi^G_{I+1/2}$ = actual flux at the interface.
Φ^G_I = average flux in node I.
$f^G_{I,I+1/2}$ = interface flux correction factor.

The same net current from the right side is given by

$$J^G_{I+1/2} = -\frac{D^G_{I+1}}{h_{I+1/2}}\left(\Phi^G_{I+1} - f^G_{I+1,I+1/2}\,\Phi^G_{I+1/2}\right), \qquad (2)$$

where the constants and variables are now those of node $I + 1$. The interface flux correction factor $f^G_{I+1,\,I+1/2}$ from the right node is different to the $f^G_{I,I+1/2}$ from the left node. Both are to be synthetically obtained from a previous high-order calculation.

As in the standard finite difference diffusion formulation in terms of mesh average fluxes, the actual interface flux is eliminated from Eqs. (1) and (2), resulting in

$$\Phi^G_{I+1/2} = \qquad (3)$$

$$= \frac{h_{I+1}\,D^G_I\,\Phi^G_I + h_I\,D^G_{I+1}\,\Phi^G_{I+1}}{h_{I+1}\,D^G_I\,f^G_{I,I+1/2} + h_I\,D^G_{I+1}\,f^G_{I+1,I+1/2}}$$

and

$$J^G_{I+1/2} = \frac{D^G_I\,D^G_{I+1}}{h_I\,D^G_{I+1} + h_{I+1}\,D^G_I\,r^G_{I+1/2}} \cdot$$

$$\cdot\left(\Phi^G_I - r^G_{I+1/2}\,\Phi^G_{I+1}\right) \qquad (4)$$

where

$$r^G_{I+1/2} = \frac{f^G_{I,I+1/2}}{f^G_{I+1,I+1/2}} \qquad (5)$$

is the ratio between the left and right interface flux correction factors introduced in Eqs. (1) and (2). We call to this ratio the interface flux discontinuity factor (IFDF), because the general correction factors introduced in Eqs. (1) and (2) make the flux discontinous at the interface, so that their ratio is that of the corrected interface flux from the left divided by the corrected one from

the right. Then, the IFDF (r factor) in the last term in Eq. (4) restores that interface discontinuity jump. If the mesh nodes are heterogeneous, the correction also takes into account the spatial variation of the local diffusion coefficients, so that the use of node average diffusion coefficients results, in general, in differences between the actual flux gradients and the piecewise linear gradients at the interfaces.

The introduction of this IFDF-corrected finite difference diffusion formulation results in the following discretized linear equations for each group G and mesh node I, in the one-dimensional case:

$$-\alpha^G_{I-1/2}\,\Phi^G_{I-1} + (r^G_{I-1/2}\,\alpha^G_{I-1/2} + R^G_I + \alpha^G_{I+1/2})\;\Phi^G_I - r^G_{I+1/2}\,\alpha^G_{I+1/2}\,\Phi^G_{I+1} = S^G_I, \qquad (6)$$

where the coefficients are given by

$$\alpha^G_{I+1/2} = \frac{D^G_I D^G_{I+1}}{h_I D^G_{I+1} + h_{I+1} D^G_I r^G_{I+1/2}}\, A_{I+1/2} \qquad (7)$$

and

R_I^G = integrated removal (macroscopic cross section by mesh volume).

S_I^G = integrated source to group G in node I.

$A_{I+1/2}$ = area of the interface between nodes I and I + 1.

The formulation is directly extendible to two –and three– dimensional problems in any coordinates. In addition to one α constant per interface and group, as in the standard finite difference formulation, only one additional r factor per interface and group is needed. The fast convergence and stability of the iterative solution of the linear equations (6) are preserved as long as the diagonal dominance is not lost. This requires the α coefficients and the r factors to be nonnegative and bounded, a condition that can be guaranteed, as shown later.

The additional IFDFs (r factors) can be explicitly determined or synthetized by introducing Eqs. (1) and (2) in (5), to obtain:

$$r^G_{I+1/2} = \frac{\Phi^G_I - h_I J^G_{I+1/2} / D^G_I}{\Phi^G_{I+1} + h_{I+1} J^G_{I+1/2} / D^G_{I+1}} \qquad (8)$$

in terms of the average node fluxes and diffusion coefficients and the net currents at the interfaces in the coarse-mesh few-group low-order discretization, which are obtained by directly adding or averaging any high-order solution, i. e. in a fine-mesh (heterogeneous) and multigroups, obtained by diffusion or transport calculation on smaller domains or dimensions, thus with faster convergence rates.

Direct inspection of Eq. (8) gives the condition for stability discussed previously ($0 \leq r < \infty$), as a "flux-limited" condition on the net current at each interface, in the form:

$$J^G_{I+1/2} \leq D^G_I \frac{\Phi^G_I}{h_I}, \text{ if } J^G_{I+1/2} > 0 \qquad (9)$$

$$|J^G_{I+1/2}| \leq D^G_{I+1} \frac{\Phi^G_{I+1}}{h_{I+1}}, \text{ if } J^G_{I+1/2} < 0$$

When the equality is reached, either α or rα in (6) is set to zero and the other coefficient is set to the current-to-flux ratio using the leading (upstream) flux.

The effectiveness and stability of this formulation rely on those properties for linear discontinous piece-wise discretizations of first-order operators. In this IFDF corrected finite difference diffusion formulation, we introduce a balanced distribution of the two resulting correction terms (α and r) over the diagonal and nondiagonal terms of the discretized equations (6). Hence, its range of application is much larger than if only the diffusion coefficient is corrected. With explicitly limited net interface currents, this formulation offers a high potential for effective and stable synthetic diffusion solutions in coarse mesh and few groups, even in extremely difficult cases of neutron or radiation transport problems (steep flux variations and angular and spectral changes within the coarse-mesh nodes and broad groups). Since the IFDFs are ratios of current-to-flux ratios, first order errors in fluxes and currents reduce to second order (or less) in the IFDFs and, thus, in the low-order solutions. These properties, confirmed by all of our numerical experiences carried out this far, provide grounds to expect a higher effectiveness for this synthetic corrected diffusion formulation as compared with other linear and non-linear formulations for diffusion synthetic acceleration (DSA), where only the removal or source terms are synthetically corrected.

IMPLEMENTATIONS OF THE SYNTHETIC DIFFUSION METHOD

This synthetic IFDF finite difference diffusion formulation can be applied, in general, to several synthetic diffusion correction or acceleration methods. We will discuss here our implementations related to reactor analysis, that can probably also be extended to other computational fields, where diffusion or diffusion-like equations are solved.

ONE-GROUP COARSE-MESH NODAL METHOD

The three-dimensional one-group IFDF-corrected finite difference neutron diffusion equations in coarse-mesh nodes with fission multiplications are given by:

$$\sum_{m=1}^{6} \frac{2D_l D_m}{H^2} \frac{\Phi_l - r_{lm}\Phi_m}{D_m + r_{lm} D_l} + \sum_{al} \Phi_l = \frac{1}{K_{eff}}\, \nu \sum_{fl} \Phi_l, \qquad (10)$$

that reduce to the nodal neutron source equations:

$$\sum_{m=1}^{6} (W_{lm} S_l - W_{ml} S_m) + \frac{S_l}{k_{\infty l}} = \frac{1}{K_{eff}}\, S_l, \qquad (11)$$

where the transport kernels, between node ι and m, are obtained as:

$$W_{lm} = \frac{2}{1 + r_{lm} \dfrac{M_l^2}{M_m^2}} \quad \frac{M_l^2}{k_{\infty l} H^2} \qquad (12)$$

We have developed (1,2) a three-dimensional PWR nodal core simulator, named SIMULA-3, that solves iteratively the nodal source equations (11). The nodal constants (k_∞ and M^2) are calculated from input correlations in terms of the local nodal properties (moderator density, power density, xenon and boron concentrations, burnup, and control rod insertion) in actual operating core conditions.

The one-group nodal IFDFs (r_{lm}) and albedos, as well as the spectral correction factors of the k_∞ and M^2 obtained from infinite-lattice calculations, are determined explicity (1,2,5) by detailed (fine-mesh, two-group) solutions in the reduced core X-Y and Z representations. The COBAYA-2 code is used for these heterogeneous detailed X-Y reference core calculations within the SEANAP code system.

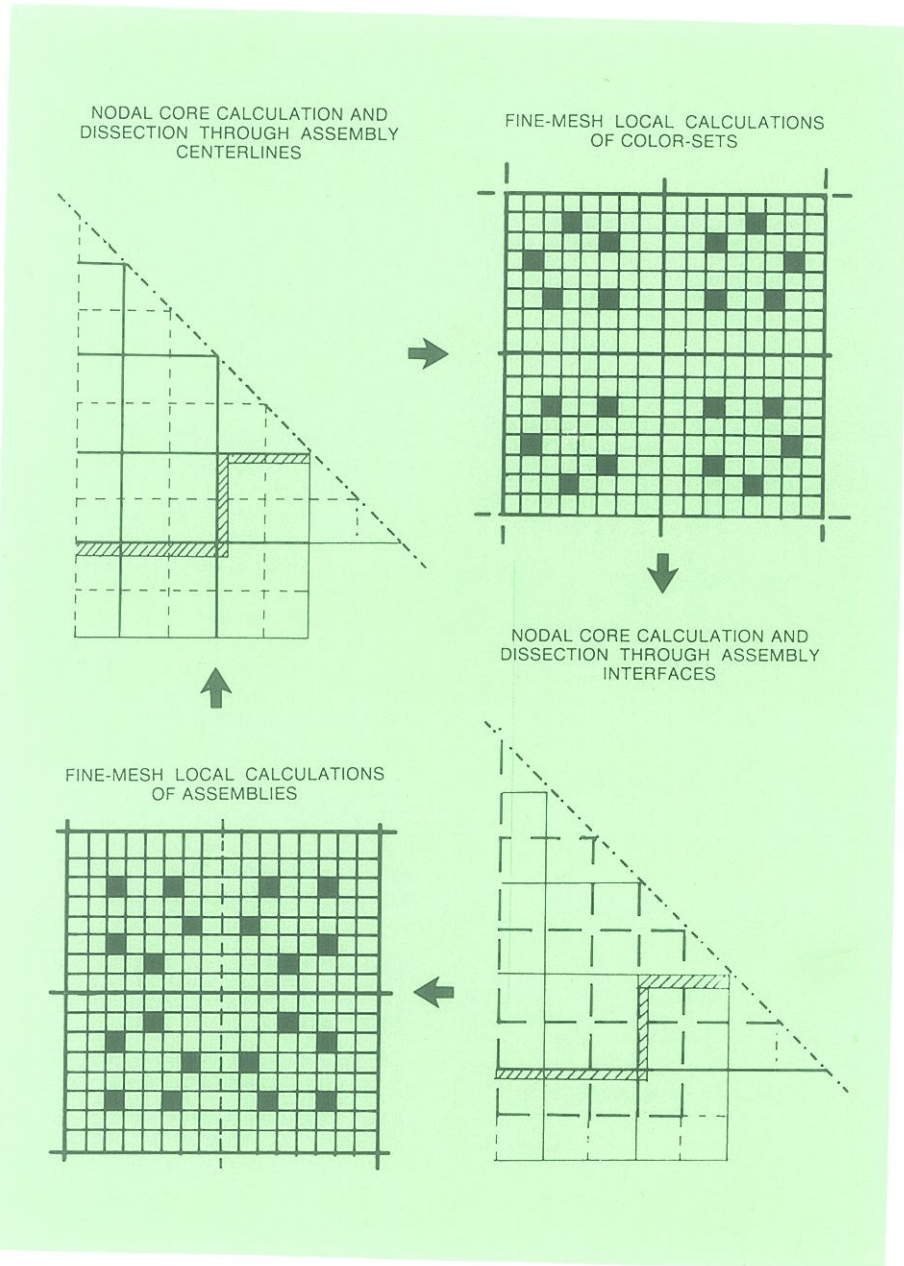
We have found that for PWR cores those one-group correction factors and albedos in quarter-assembly nodes are nearly the same for, or between, nodes of the same fuel type. This regularity allows us to use averaged values per fuel type, although additional relative factors for each node position and direction can also be used. The deviations of the uncorrected method (up to 12 %) are reduced (5) to ~1 % in maximum deviations with our corrected

method, and the nodal solution reproduces rather accurately the node-integrated powers of the detailed heterogeneous solution.
The SIMULA-3 is our three-dimensional nodal PWR simulator code, using the previous methods, implemented in the SEANAP system. It is used to determine: average fuel assembly power distributions, critical boron concentrations, control rod worths, reactivity coefficients, and other core design parameters.

ITERATIVE FINE/COARSE-MESH FEW-GROUP SYNTHETIC DIFFUSION METHOD

In this implementation, we attempt to provide an answer in a strict, yet effective, way to the two key questions for heterogeneous reactor analysis posed in the introduction, namely, those of homogenization and dehomogenization of fuel assemblies or nodes. Our synthetic IFDF corrected finite difference diffusion formulation answers accurately to the homogenization problem, provided that accurate enough local fine-mesh-heterogeneous solutions are known within each node for the synthetic determination of the averaged constants and IFDFs. The local solutions also provide the answer to the dehomogenization problem by reconstruction.
Thus, the only remaining key issue is related to the boundary conditions for the local (intranode) detailed solutions. In our corrected finite difference diffusion formulation for coarse and heterogenous nodes, the IFDFs account for all the coarse-mesh, heterogeneity, and spectral interaction effects, so that realistic node heterogeneous properties and boundary conditions have to be considered in obtaining the detailed local solutions.
The analysis (4) of fine-mesh two-group whole-core solutions for PWRs, obtained with standard fine-mesh calculations, has shown that different actual intranode solutions are required for practically every fuel assembly in operating cores to obtain accurate pin powers and burnups. Then, with little extra effort, those local intranode solutions can be iterated together with the synthetic IFDF-corrected nodal core solution in order to obtain an accurate converged solution.
In two-dimensional problems, the procedure for nodal/local coupling becomes elaborate due to the need for the detailed shape of the interface fluxes or currents (or their ratios) in each group and fine-mesh interval along the boundaries of the local problems. The global nodal IFDF-corrected solution provides only the actual interface fluxes and currents in each group averaged or integrated over the node interfaces. If a "single core dissection" is done along the fuel assembly interfaces or centerlines, resulting in local problems for each fuel assembly or color set (four quarters of neighboring assemblies), respectively, an interpolation is required to obtain the fine-mesh shape of the fluxes and currents along the dissection boundaries. For PWR cores, we have investigated (4) several core dissections (through assembly interfaces or centerlines), boundary conditions (fluxes, currents, or current-to-flux J/Φ ratios), interpolation schemes (linear or quadratic with continuity for the corner fluxes), and node sizes (assemblies or quarters of assemblies).
We found (4,5) that the last of those named options are required to reduce the maximum errors in pin powers to below 2 %.
To achieve global/local convergence to the reference (full-core) fine-mesh solution, we have developed (4) an original procedure of "alternate core dissections" through the assembly and color-set local problems, in alternate iteration steps. The global IFDF-corrected solution is always done in quarter-of-assembly nodes and in the same groups as the local solutions. After each local solution, aside from calculating the homogenized group constants and IFDFs for the four nodes and respective interfaces, the detailed fine-mesh J/Φ ratios

F I *Scheme of the global/local method of COBAYA-2*



subsystems or codes: the MARIA subsystem (2) for fuel assembly calculations, the COBAYA-2 code for detailed (pin-by-pin) core calculations and the LOLA subsystem for three dimensional one-group corrected-nodal core simulation. A scheme of the system is shown in Fig. II.

The MARIA subsystem is based on the WIMS-D4 Winfrith code, with some extensions for PWR fuel assembly calculations.

The COBAYA-2 code performs the detailed (pin-by-pin) two-group two-dimensional XY core diffusion calculations at some selected core conditions and configurations along each cycle, including several unrodded and rodded configurations at hot-zero-power (HZP) and hot-full-power (HFP), at beginning and end-of-cycle (BOC and EOC), and at several steps along the nominal burnup. The full core or a half, quarter of octant, according to the actual symmetries, is represented. The detailed core-shroud (steel plate) and reflector (borated water) is included in the fine-mesh grid.

The LOLA subsystem for PWR core simulation includes the synthetically corrected one-group three-dimensional nodal code SIMULA-3 and several auxiliary codes, of which the most relevant ones are the MELON and SIMPAT codes. The MELON code fits the polinomical correlations of reactivity and migration area used in the SIMULA-3 model, to the values given by the MARIA subsystem at different operational and physical conditions. The SIMPAT code generates the data sets to be input to the core simulator for defining the fuel type per node (quarters of assemblies), and the shuffling of fuel from previous cycles, also at the node level.

The main characteristic of the SEANAP system is its well-defined procedures and specifications of the detailed assembly and core calculations that explicitly produce all the data files required by the core simulator, without any adjustement or ad-hoc correction. It provides a reliable, systematic and efficient capability to perform the neutronic calculations required for the extense analysis of the design, tests and operation of PWR cores. For each cycle, an expert can complete the preparation of all the data files required by the core simulator (including the MARIA and COBAYA calculations for all fuel assembly types and reference core conditions) in one or two weeks with a medium computer (1-4 Mips, 100-200 Kwords). The calculations for the analysis of the startup tests and nominal design of one cycle can be completed in another week or two, because the SIMULA-3 core simulator is quite fast and simple. Then, the detailed follow-up of actual core operation throughout the cycle can be easily performed analyzing the deviations and updating the previsions, as well as the modifications during the fuel reload.

SEANAP VALIDATION METHODOLOGY AND RESULTS

The SEANAP system has been applied to the analysis of the core design, tests and operation for all the previous and present cycles of five Spanish PWR units (Almaraz I and II, Ascó I and II and Vandellós-II) of the same design (Westinghouse 900-MWe PWR's). All first cycles had similar fuel assemblies and loading patterns (157 assemblies of three enrichments with hollow burnable absorber clusters). The reload cycles, designed by the Spanish nuclear fuel company ENUSA (with Westinghouse license), had only minor differences in feed enrichments, lengths and loading patterns (out-in without burnable absorbers), with the exception of the third cycle of Almaraz-II and the fifth of Almaraz-I, with a larger fresh fuel batch of higher two-split enrichments and a low-leakage loading pattern using wet advanced burnable absorbers.

The validation of the SEANAP system has been performed (6,9) by a systematic comparison of its calculated results with the available core parameters from the measurements in the nuclear test at startup and the in-core monitoring throughout the actual operation of the cycles included in Table I. The results of SEANAP have also been compared with the design ones; including, in addition to the former parameters at testing and nominal conditions, the core reactivity coefficients and defects (at BOC and EOC); differential and integral worths and power peaking factors for successive and overlapped insertion of the control banks and for the worst ejected or sutck control rod; and the reactivity and power distribution evolution with xenon transients in the startup, shutdown and power changes at various power levels, burnups and control insertions.

We will focus in this article on the validation against measurements of the detailed pin-by-pin and the assembly average power distributions, as key validation of our detailed (coupled fine/coarse mesh) and corrected coarse-mesh nodal codes and procedures, described in the previous sections.

Throughout the actual operation of these PWR's, the boron concentration, power level and control bank position are measured and registered frequently (several times per day) and in-core flux maps are measured and computer processed regularly (several times per month). In our analysis of the actual core operation throughout each cycle, we follow very closely the power and control insertion history in the three-dimensional core simulation, with rather short burnup steps. Critical boron concentrations and detailed power distributions are calculated for every core condition and configuration of each in-core flux map. The parameters of our validation results selected here are the boron concentrations and peak assembly relative powers. The differences between the SEANAP calculated values and the measured ones for all the cycles analyzed are given in Fig. III (see Table I for symbols). The assembly relative power distributions averaged over the symmetric fraction of the core are also compared with the measured ones in Fig. IV, at the beginning, middle and end of three of these cycles.

T I Symbols used for each cycle and unit

Cycle	CN Almaraz		CN Ascó		CN Vandellós
	Unit I	Unit II	Unit I	Unit II	Unit II
1	□	□	◇	□	D
2	·	+	*	D	
3	⊠	⊗	⊗		
4	X	D			
5	⊠				

Note: Only the design analysis has been done for cycles marked with D

at the centerlines of each local box are also calculated. Since these centerlines will be the boundary lines for the next alternate core dissection, we obtain the detailed shape of the interface J/Φ ratios to be adjusted only to the node interface averages of the global nodal solution. A fast convergence is achieved because the J/Φ shapes at the centerlines of the local problems depend on the intranode properties, so that good approximations result even if the boundary conditions are far from convergence as in the first iteration. Figure I is a scheme of this global/local method with alternate core dissections in its two-dimensional implementation, where the main calculation steps are also summarized.

The method has been implemented (3-5) in a computer code, called COBAYA-2, for PWR core calculations in two dimensions and two groups (the extension to multigropus is now under development). The code has very small central memory requirements, only for cell fluxes and pin powers of each box at one time, and they can be stored in extended memory or disk. Another advantage is that the local calculations can be done in any order (i. e. by fuel type), which is useful for the efficient feedback of cell cross sections (7), depending on cell burnups and other point properties, with tables stored in extended memory or disk by fuel type.

The best computational property of this method, however, is that the local solutions are small, so that they converge very fast, and can be done in parallel on small computers or processors. The code requires only ~ 100 CPU seconds per total core calculation on a CYBER-835 (~0.5 MFLOPS).

The COBAYA-2 is the code for detailed core calculations in the present SEANAP code system and it is used to determine the pin-by-pin power distribution, and maximum local peaking factors, as well as the synthetic one-group nodal correction factors for the SIMULA-3 code.

ONE-GROUP SYNTHETIC ACCELERATION OF MULTIGROUP DIFFUSION CALCULATIONS

The multigroup, standard, or IFDF-corrected, finite difference diffusion solution provides the net currents and fluxes at the mesh interfaces in terms of the mesh average fluxes Φ_i^g by Eqs. (1) through (5) applied at each interface (i + 1/2) and group g. The integrated one-group fluxes and net currents are obtained by simply summing up groups, so that by introducing the one-group (G = 1) values of f, by Eqs. (1) and (2), in the same mesh grid, we obtain (4,5) the following synthetic definition for the one-group IFDF:

$$r^1_{i+1/2} = \frac{f^1_{i,i+1/2}}{f^1_{i+1,i+1/2}} = \frac{\bar{D}^1_{i+1}}{\bar{D}^1_i} \frac{\sum_g D^g_i f^g_{i,i+1/2} \Phi^g_{i+1/2}}{\sum_g D^g_{i+1} f^g_{i+1,i+1/2} \Phi^g_{i+1/2}} \quad (13)$$

where the one-group averaged diffusion coefficients in each mesh are obtained by the direct flux weighting:

$$\bar{D}^1_i = \frac{1}{\Phi^1_i} \sum_g D^g_i \Phi^g_i \; ; \; \Phi^1_i = \sum_g \Phi^g_i . \quad (14)$$

Equation (13) determines the one-group IFDFs by a simple ratio of the weighted diffusion coefficients of both adjacent meshes with the mesh average fluxes and the corrected interface fluxes with the respective correction factors f^g (unity in the standard multigroup finite difference diffusion solution). Since all terms involved are positive and bounded, the one-group IFDFs will always be positive and bounded, too, so that the one-group solution will be unconditionally stable.

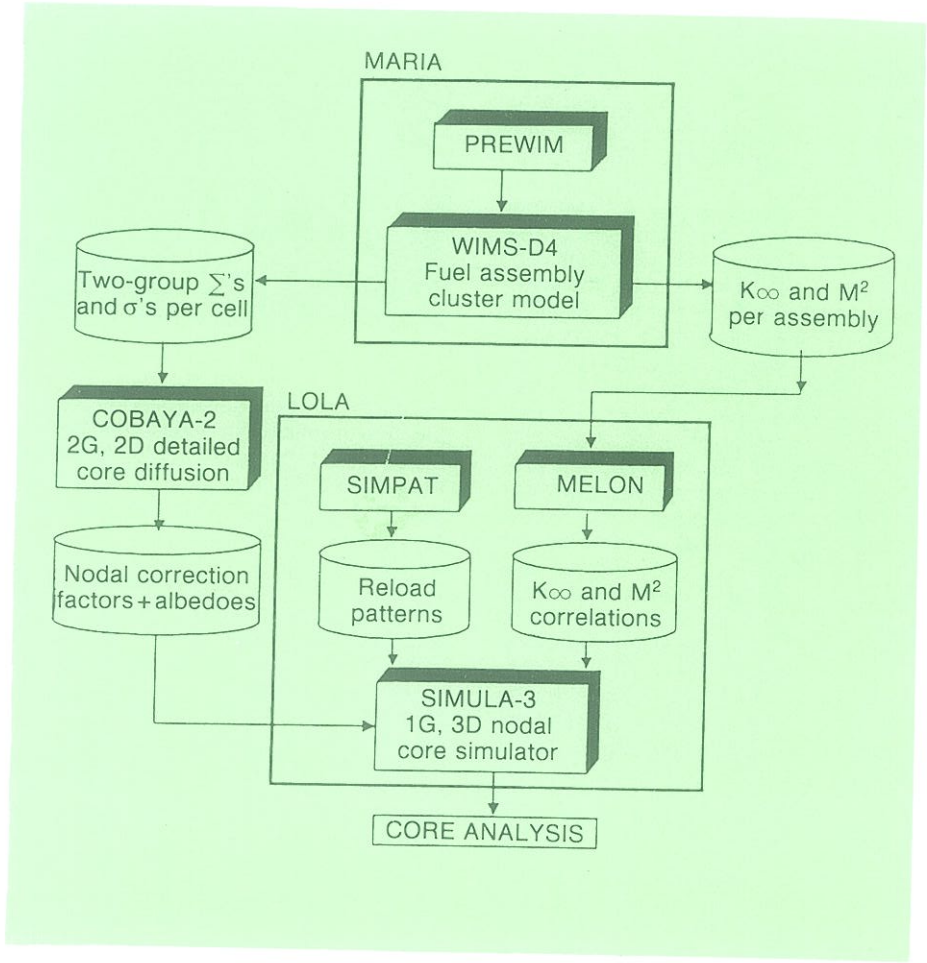
The one-group equation can be solved, after the sweep over all multigroups is completed, by the same numerical techniques as the inner equations for each group. In eigenvalue problems, the one-group solution is iterated up to convergence in the eigenvalue. The one-group fission source and the eigenvalue are then used for the next sweep over each multigroup.

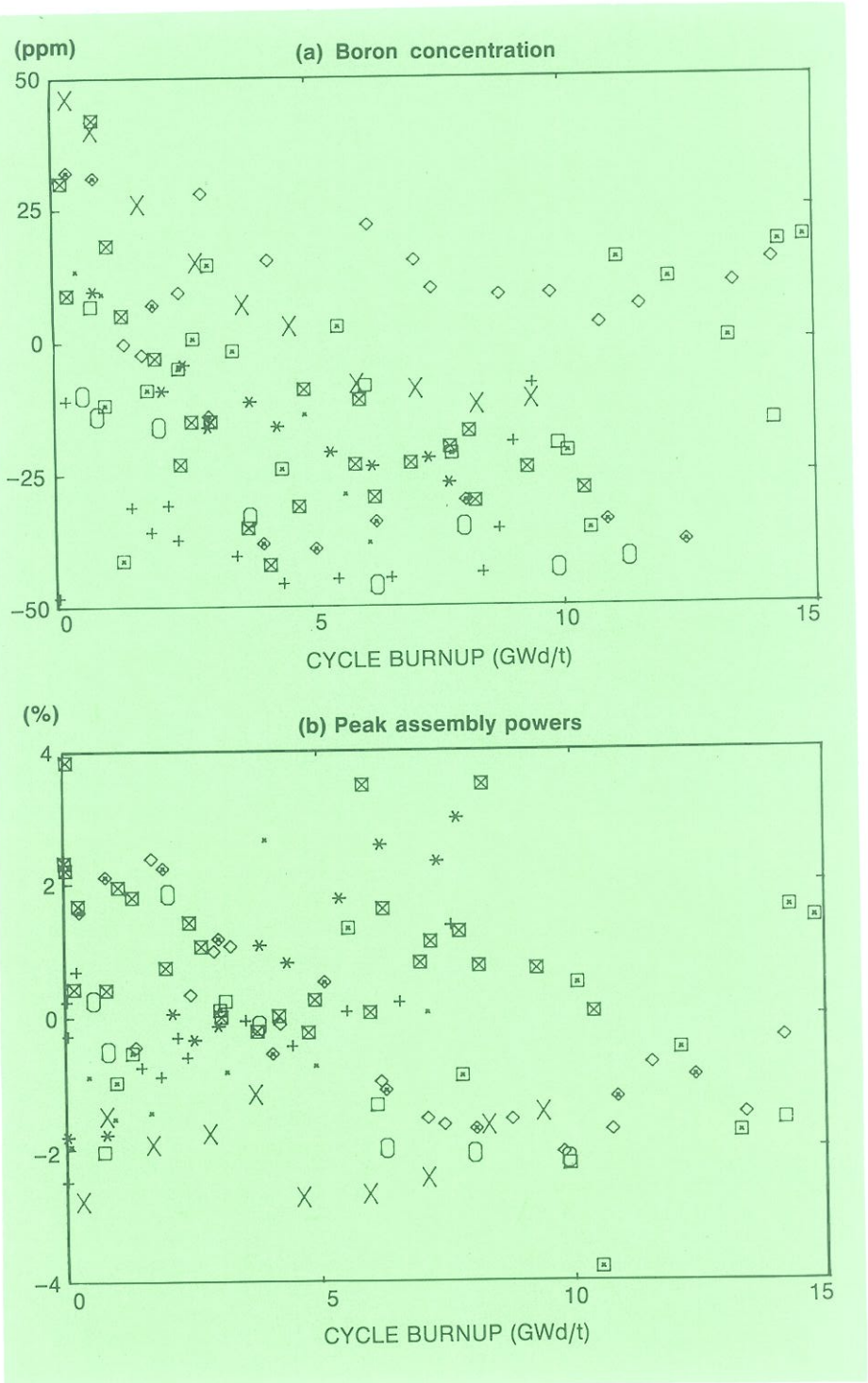
This method can be used very efficiently to accelerate the fission source (outer iterations) of the multigroup diffusion calculations. It has shown (4,5) outstanding performance (one order of magnitude reduction in the number of outer iterations as compared with standard source overrelaxation or extrapolation techniques) in multigroup neutron diffusion problems, as well as in other similar problems (8). It is being implemented also in the multigroup version of COBAYA-2.

THE SEANAP CODE SYSTEM FOR PWR CORE ANALYSIS

The SEANAP code system is integrated by three main interconnected

F II *Scheme of the PWR Core Analysis System SEANAP-86*





F III *Differences between calculated and measured boron concentrations (a) and peak assembly powers (b) versus burnup throughout actual cycle operation*

SUMMARY AND CONCLUSIONS

A new synthetic diffusion method has been developed in terms of IFDFs that are explicity obtained from detailed (fine-mesh and multigroup diffusion or transport) solutions. The formulation allows the accurate and stable reduction (synthesis) in directions (transport to diffusion), spatial mesh (fine to coarse mesh), and groups (multigroups to a few or one group). Since the average fluxes (thus reaction rates) and interface net currents and fluxes are preserved on the coarse mesh and few groups, the reduced synthetic solutions can be used to accelerate or approximate the detailed (heterogeneous) solutions.

The simplicity and stability of the formulation allow a wide scope of applications in reactor analysis and other computational fields. Several implementations have been discussed and presented for three of them; namely,

- one-group coarse-mesh synthetic diffusion three-dimensional core calculations
- iterative fine/coarse-mesh few-group synthetic diffusion calculations
- one-group synthetic diffusion acceleration of multigroup diffusion calculations

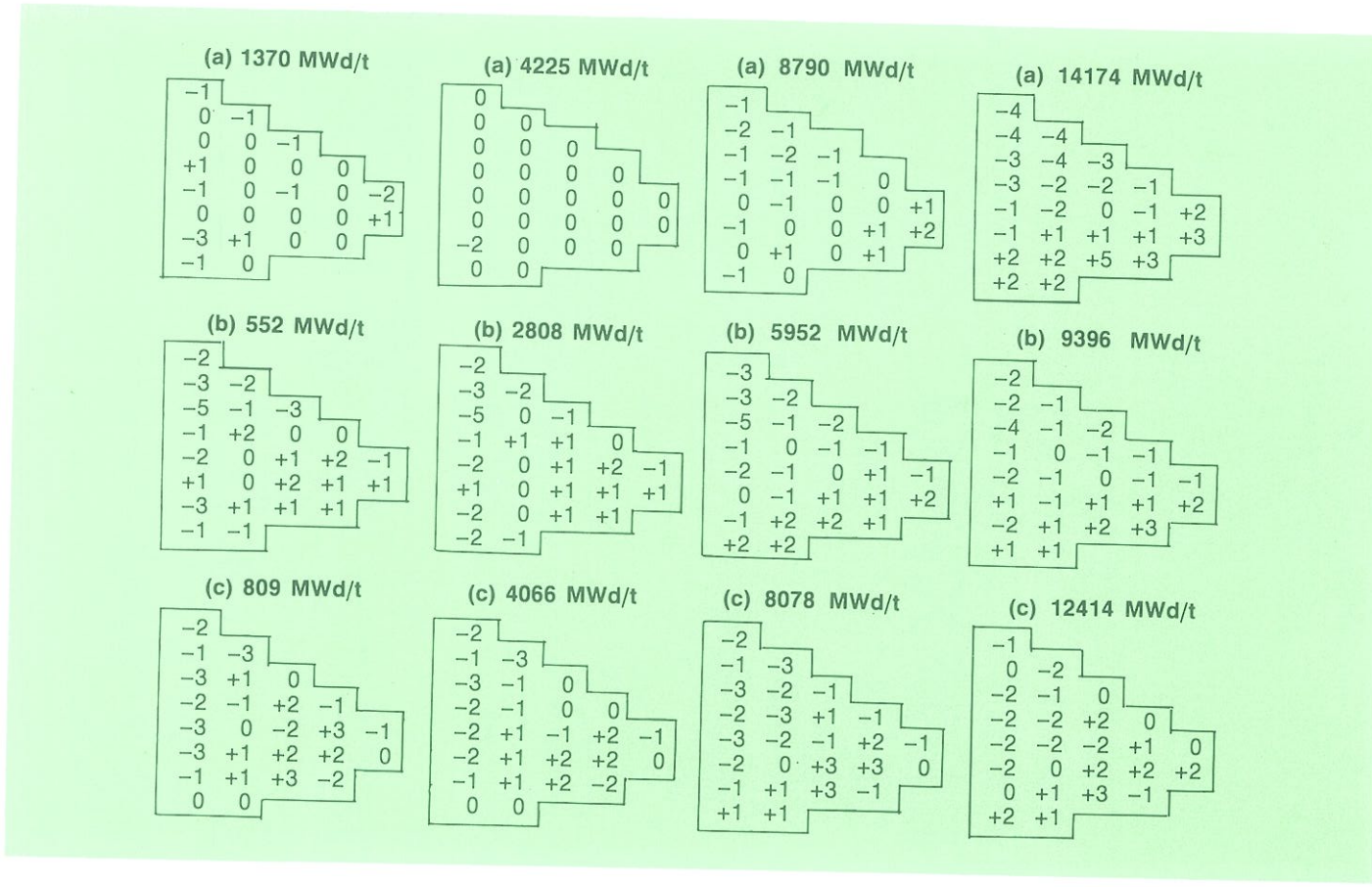
The results demonstrate, for all the applications considered, the following main properties of the new proposed formulation:

- regular behavior of the correction factors (IFDFs)
- simplicity and stability
- outstanding acceleration and convergence

In particular, from the above-mentioned applications and results, the following main conclusions can be drawn:

- The IFDF synthetic one-group coarse-mesh (nodal) three-dimensional core simulator achieves high accuracy with a relatively simple and explicit precalculation (by reduced X-Y and Z detailed solutions at reference conditions) of the correction factors, which show a regular behavior in operating PWR cores.
- The coupled and iterative global (IFDF-corrected coarse-mesh)/local (standard fine-mesh) calculations with alternate core dissections are very efficient for exact detailed diffusion solutions in heterogeneous reactor cores. Enhanced convergence is achieved both in the local solutions and in the successive global core solutions. The method is optimal for parallel computations of the local solutions.
- The one-group IFDF synthetic diffusion calculations are very simple and effective in accelerating the fission source and eigenvalue in multigroup diffusion calculations for multiplicative media.

From the validation analysis shown here and in previous articles (6,9), since the differences obtained between the calculated results and the measured results are well within the acceptance design criteria, we can conclude that the SEANAP system is well-suited to the analysis of most of the relevant parameters in the nuclear design, start-up tests and operation of PWR cores



F IV *Differences (%) between calculated and measured assembly relative powers at several burnups throughout actual operation of: (a) ASCO-I Cycle 1, (b) ALMARAZ-I Cycle 4, and (c) ALMARAZ-II Cycle 3*

with present fuel designs, loading patterns and base operation. Thus, the goal of providing the nuclear utilities with a validated, experimented, well-known, flexible and efficient capability of inhouse independent core analysis has been met, at least at the first level. The application of the SEANAP system to the extended core analysis of currently operating and future cycles of these and other Spanish PWR units is being continued, in close cooperation between the research and development group and the support core engineering groups of the nuclear power plants.

These results encourage us to continue development of the SEANAP system along two lines, to improve its efficiency and accuracy and to extend its scope to the nominal and transient analysis of the core power capabilities. The COBAYA-2 code has been extended (7) to include the feedbacks on the cross sections, at the cell or fuel-pin level, of the local burnup, xenon concentration, fuel temperature, water density and boron concentration. It is also being extended to multigroups. In the other line, the SIMULA-3 code is being extended to time-dependent corrected nodal calculations. This code, plus a reduced one and two dimensional versions, will be coupled to available core-subchannel and plant thermal-hydraulics transient codes for fast transient analysis.

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