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Reactor Analysis System for PWRs
Volume 1 - Methodology Description
Volume 2 - Benchmarking Results

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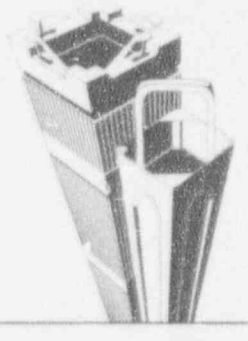

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Reactor Analysis System for PWR's
Volume 1
Methodology Description

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1.0 INTRODUCTION

This report presents a core physics code system for pressurized water reactors which will be used for Siemens Power Corporation (SPC) and Siemens KWU/B to perform neutronics design analyses. The code system, named SAV95 (SAV is an acronym for Standard Design Process, German : Standardauslegungsverfahren), will replace the SAV90 code system currently used by KWU/B and the XTGPWR/PDQ/CASMO – 2E codes used by SPC.

The SAV95 code system will use CASMO – 3G to generate cross sections and a code named PRISM to determine core reactivity and power distributions. PRISM is a reactor simulator code with significant improvements over SPC's previous reactor simulator. These improvements include:

- Number densities and depletion of key isotopes are evaluated on a nodal basis using microscopic cross sections.
- The nodal expansion method is utilized in solution of the two group diffusion theory representation of the reactor core.
- The code will generate pin power distributions using a pin power reconstruction technique.

Improvements in the SAV95 code system reduces or eliminates the use of biases and results in increased accuracy in startup and operations data.

The SAV95 system reduces the number of reactor simulator codes and auxiliary codes used by SPC and KWU, thereby reducing engineering effort and enhancing quality and efficiency.

The code system also incorporates simplified data input formats and utilizes automated calculation sequences to further enhance ease of use.

2.0 SUMMARY

This document describes the Siemens SAV95 code system. SAV95 replaces the PWR neutronics methodology both for Siemens Power Corporation (SPC) and for Siemens KWU/B.

The currently approved PWR neutronics methodology for SPC is described in References 1, 2, 3, 4, 5 and 6. This methodology is used to generate acceptable fuel cycle designs, predictions of the fuel behavior for comparisons with the startup test results, and as input for the safety and mechanical design analyses. To perform these calculations, the methodology uses MICBURN-2/CASMO-2E to calculate the microscopic cross sections as a function of exposure for the different assembly lattice configurations (e.g., gadolinia loadings and distributions, enrichment distributions, etc.). These cross sections are used as input to calculate the power distributions, the kinetics parameters, the burnup for the assembly, and the core cycle design. The codes used by SPC for this purpose are PDQ-7/HARMONY and XTGPWR. The PDQ-7/HARMONY code package is used for calculating reactivity and radial (x-y) power distributions. The XTGPWR[7] code extends the core simulation including three-dimensional analyses. CASMO – 2E is also used in conjunction with the PDQ calculations to determine the transport theory correction factor for quarter-core PDQ calculated activation rates in instrumented assemblies.

Siemens KWU/B currently uses the code system SAV90 for pressurized water reactor analysis.[8, 9] The experience with SAV90, together with its predecessor code package SAV79, includes about 300 cycles for KWU, Westinghouse and Framatome PWRs with power levels ranging from 365 to 1420 MWe. The SAV90 system can be divided into a spectral code system which generates reactor cross section libraries and heterogeneous form functions, a nodal reactor depletion program and a dehomogenization program which is also used for the post-processing of nodal reactor calculations.

The SAV90 spectral code system consists of FASER (one-dimensional, 85 energy groups, based on MUFT/THERMOS)[10,11,12] and MULTIMEDIUM (two-dimensional, 10 energy groups, Nodal Discrete Ordinates Methods NDOM).[13,14] The core simulator MEDIUM is based on the two main concepts of a modern nodal code, the microscopic cross section representation and isotopic depletion,[15] and the Nodal Expansion Method NEM.[16, 17, 18, 19, 20, 21] The reconstruction of heterogeneous power, burnup and flux distributions is performed by the post-processing code PINPOW. This code is based on the

burnup-corrected spectral interpolation method[22, 23, 24] and combines a general three-dimensional methodology with two-dimensional elements.

The SAV95 code system is defined by an extension of the spectral code package used at SPC (MICBURN-3/CASMO-3G instead of MICBURN-2/CASMO-2E) and an extension of the core simulator used at Siemens KWU/B (PRISM instead of MEDIUM/PINPOW).

MICBURN-3/CASMO-3G [25, 26, 27, 28, 29, 30, 31, 32] is used to generate the microscopic cross sections. These codes are updates to the previously accepted codes. MICBURN-3 is a multigroup, one-dimensional transmission probability code which calculates the microscopic burnup in an absorber rod containing initially homogeneously distributed poison. These cross sections as functions of absorber number density are input to CASMO – 3G. CASMO-3G is a multi-group, two-dimensional transport theory code for burnup calculations on PWR and BWR assemblies or simple pins. The microscopic depletion is calculated in each fuel rod and burnable absorber rod. The output consists of cross sections, discontinuity factors, and heterogeneous form functions for the core simulator code, PRISM. MICBURN-3/CASMO-3G is currently approved for use by SPC for BWR neutronic calculations.

The PRISM code replaces the PDQ-7/HARMONY and XTGPWR codes in the SPC neutronic methodology. PRISM will calculate the core-wide power distributions in three dimensions. It also uses pin power reconstruction to establish the individual rod histories and reactivities. The main task of this document is the description of this newly developed reactor code, PRISM.

The evaluation and validation of the SAV95 code system is presented in EMF – 96 – 029(P),"Reactor Analysis System for PWR's", Volume 2, "Benchmarking Results". The results in the report EMF – 96 – 029(P) demonstrate the adequacy of the SAV95 code system for neutronic design analyses.

2.1 Development Goal

The goal for the development of SAV95 was to provide Siemens with a modern reactor code system which is applicable for all types of PWRs (Westinghouse, Combustion Engineering, Framatome, KWU, etc.) on the basis of the same international standard methodology. The

spectral code CASMO-3G and the reactor code PRISM will be the key components of the new Siemens PWR Standard Design Procedure SAV95.

2.2 Development Basis

The three-dimensional reactor program MEDIUM (the nodal simulator of SAV79 and SAV90 during the last two decades) was chosen as the development basis for PRISM. PRISM inherits the key components of a modern nodal code: a) microscopic cross section representation and isotopic depletion, b) Nodal Expansion Method, and c) pin power reconstruction.

2.3 New or Improved Models

- Nuclide chain

  The nuclide chain of SAV79 and SAV90,

  , was extended to

  the nuclide chain of SAV95 by

- Cross sections

  A continuous cross section representation was developed covering all combinations of thermal-hydraulic parameters possible in stationary reactor states. This was achieved by:

- Flux solution

  The flux solution module of MEDIUM was completely replaced by the fast running flux solution module for stationary reactor states, FLUXS, which is part of the KWU transient code system PANBOX. FLUXS is a state-of-the-art module which takes advantage of the tremendous hardware improvements during the last two decades, combining the efficiencies gained by the application of multi-level techniques and the use of vectorization capabilities.[33, 34, 35, 36]

- Introduction of Discontinuity Factors

  The concept of heterogeneity factors,[23, 37] used in SAV79 and SAV90, is fully equivalent to the concept of discontinuity factors[38] in the case of radially symmetric fuel assemblies.

- Integrated Pin Reconstruction Methods

  Within the SAV79 and SAV90 code system, the determination of pinwise flux, power, and burnup values was performed by the post-processor PINPOW. This code combines a general three-dimensional methodology with two-dimensional elements. A full three-dimensional pin interpolation scheme was directly integrated into the reactor code for SAV95.

2.4 Major Code Characteristics

The reactor core can be modeled in two- or three-dimensional cartesian (X-Y-Z) geometry. The size of the nodes in X and Y direction is the same for all nodes; whereas, the node size in the Z direction is variable.

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For fuel management calculations and PWR operation analysis, PRISM offers the following capabilities:

- Core calculations in 1/8, 1/4 (mirror symmetry and 90 degrees rotational symmetry) and full core geometry.
- Optional calculation of reactor eigenvalue at specified boron concentration or critical boron at a specified eigenvalue.
- Optional calculation of operational parameter values such as reactor power and control bank position.
- Calculation of control rod dependent parameters (control bank worth, differential rod worth, estimated critical position).
- Calculation of reactivity coefficients: fuel temperature (doppler) coefficient, moderator temperature coefficient (MTC and ITC), boron coefficient.
- Calculation of nuclide worth for single nuclides or combinations of user-selected nuclides.
- Calculation of equilibrium or time dependent xenon and samarium.
- Calculation of flux, power, and burnup per fuel assembly, node, and pin.
- Calculation of F_Q and $F_{\Delta H}$ for the core.

- Flexible edit control; a large variety of edits is available.
- Prediction of the in-core detector signals for fission, rhodium and aeroball detectors.
- Prediction of excore detector signals.
- File output for restart capability.
- File output for postprocessing; e.g., DNB calculation, calculation of corrosion and automated report generation.
- Generation and maintenance of plant and fuel assembly history data.
- Fuel shuffling.
- Startup physics test calculations (boron dilution or rod swap method for control rod worth).
- Simulation of PWR core operational transients including deep or shallow control bank movement and xenon and/or samarium evolution.

The connection of the main elements of the SAV95 code system is presented in Figure 2.1.

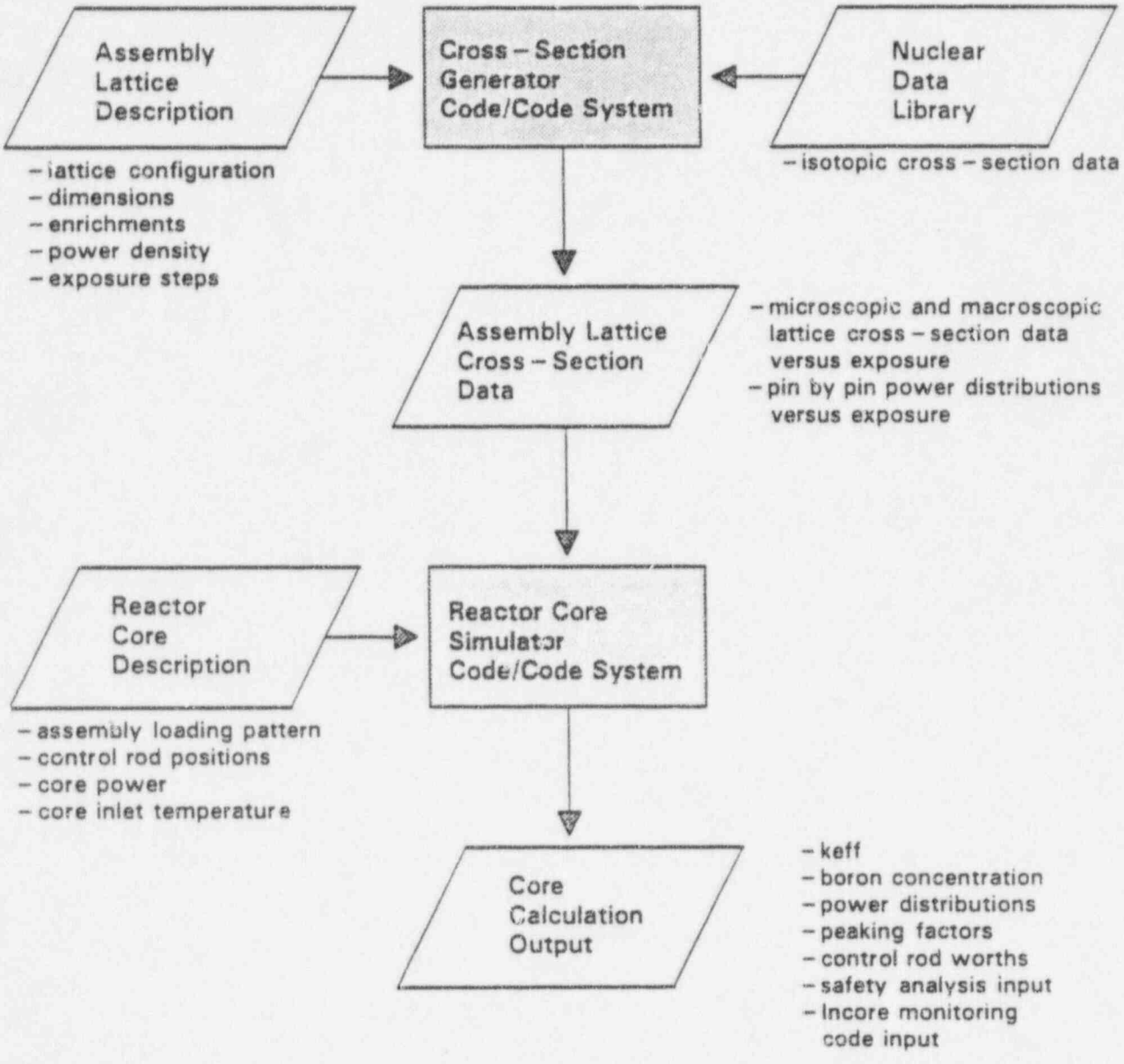

Figure 2.1 PWR Neutronics Design Methodology Flowchart

3.0 NODAL CROSS SECTIONS AND DISCONTINUITY FACTORS

3.1 Cross Section Generation with MICBURN/CASMO

The cross sections versus exposure in an absorber rod containing an initially homogeneous distribution of burnable absorber are calculated in MICBURN-3. It generates effective cross sections as a function of the absorber number density to be used in CASMO-3G.

. The input required for MICBURN-3 are geometry and material composition, and instructions for the choice of options in the calculations. Nuclear data are read from the CASMO-3G library.

The radial distribution of the gadolinium in the burnable absorber pin varies strongly with time. Initially, the pin is black for thermal neutrons and the burnup of the gadolinium isotopes is concentrated on the pin surface. The initial number densities of ^{155}Gd and ^{157}Gd in naturally occurring gadolinium are approximately the same, but the cross section of ^{157}Gd is about four times that of ^{155}Gd. This means that the ^{157}Gd is depleted faster than ^{155}Gd, and the effective cross section for each nuclide averaged over the pin varies considerably with burnup. The effective cross section for ^{155}Gd and ^{157}Gd, taken together, varies more slowly with burnup. It is, therefore, advantageous to calculate the total absorption cross section of the burnable absorber for the use in the lattice depletion program. At high burnup, when the number densities of ^{155}Gd and ^{157}Gd become small, the remaining Gd-isotopes are of importance in evaluating the residual gadolinium reactivity effect. The contribution to the total absorption from all naturally occurring Gd-isotopes is,therefore, included. MICBURN – 3 edits micro group cross sections by flux/volume weighting over the burnable absorber region. The cross section for each Gd – isotope is then weighted by its isotopic number density and these products are summed over all the Gd – isotopes. This sum is then divided by the sum of the various isotopic number densities with weighting factors

. (The standard Studsvik weighting factors are 0, 1, 0, 1, 0). CASMO – 3G then uses these flux/volume/isotope weighted cross sections for the heterogenous bundle calculation.

A realistic spectrum for the burnup calculation is generated by surrounding the burnable absorber pin with a homogenized pin cell lattice (buffer zone). The total absorption cross

sections are found to be fairly insensitive to the size of the buffer zone in MICBURN – 3. However, when self shielding of the Gd – isotopes in the resonance region is accounted for, the composition of the buffer region should reflect the fuel to water ratio of the fuel assembly. An annular buffer region is constructed by preserving the various volumes of material within the assembly: non – Gd fuel, gap, cladding, water rods, coolant, etc.

CASMO-3G is a multigroup, two-dimensional transport theory code for burnup calculations on PWR and BWR assemblies or simple pin cells. The code handles a geometry consisting of cylindrical fuel rods of varying composition in a square pitch array with allowance for fuel rods loaded with a burnable absorber, burnable absorber rods, cluster control rods, in-core instrument channels, water gaps, boron steel curtains and cruciform control rods in the regions separating fuel assemblies.

Additional features of CASMO-3G are the capability to handle four bundle BWR and PWR colorset cases and a model for the generation of baffle/reflector data. The CASMO-3G version contains a gamma transport module to calculate gamma detector responses for BWRs. Typical fuel storage rack geometries can also be handled.

Some important characteristics of CASMO-3G are:

- Nuclear data are collected in libraries containing microscopic cross sections in up to 70 energy groups. Neutron energies cover the range 0 to 10 MeV. A library containing data in 40 energy groups is typically used in production calculations.

- Effective flux/volume/isotope weighted cross sections generated by the MICBURN – 3 code are used for gadolinium loaded fuel.

  (This library is only used in MICBURN – 3; CASMO – 3G uses the effective cross sections generated from MICBURN – 3 for the Gd – isotopes.) The self shielding treatment of the heavy metals in MICBURN – 3 has been extended to include the Gd – isotopes.

- CASMO-3G can accommodate non-symmetric fuel bundles containing up to 19x19 rods. However, half, quadrant, or octant symmetry (mirror symmetry) can be utilized in calculations, where applicable.

- Absorber rods or water holes covering 1x1, 2x2, or 3x3 pin cell positions are allowed in the assembly.
- Effective resonance cross sections are calculated individually for each fuel pin.
- The calculational sequence starts in a simplified geometry. Energy groups are then collapsed as spatial detail is increased.
- Up to 12 energy groups are allowed in the two-dimensional transport theory calculation.
- A fundamental mode calculation is performed to account for leakage effects.
- A spatial collision probability treatment is used to take into account the heterogeneous structure of cruciform control rods.
- The microscopic depletion is calculated in each fuel pin and burnable absorber pin.
- The output is flexible and gives few-group cross sections and reaction rates for any region of the assembly for use in overall reactor calculations.
- Flux discontinuity factors are calculated at the boundary between bundles or quadrants and for reflector regions.

3.2 Nuclide Chain

The default nuclide chain of PRISM is shown in Figure 3.1.

The less important remaining heavy metal nuclides of CASMO – 3G are lumped together in a macroscopic background cross section block.

3.3 Representation of Nodal Cross Sections and Discontinuity Factors

The cross sections, discontinuity factors, and heterogeneous form functions for PRISM are generated by evaluating a file produced by CASMO-3G. The evaluation is performed by a code named TRAFO.

The nodal cross sections used in PRISM are

represented as polynomials

with respect to nodal state parameters. The polynomial coefficients are derived by TRAFO during the evaluation process of various branch calculations of CASMO – 3G.

The reference state parameter values of the basic depletion calculation of CASMO – 3G are:

Each nuclide of the nuclide chain is represented according to the description above. The polynomials are further discussed in section 4.1.

The following three special cross section blocks used are:

- MAC-Block

  The MAC-block containing the macroscopic cross sections characterizing the whole fuel assembly is derived for PRISM using the same procedure described above for the microscopic data. Additionally, a fast fluence ($E \geq 1.855$ eV) dependency of the macroscopic cross sections is derived

- DELTA-Σ-Block

  The above listed CASMO – 3G reference and branch calculations are performed in exactly the same way for the uncontrolled and the controlled fuel assembly.

- DET-Block

  The microscopic cross section representing the detector material is modified so that the product of the undisturbed flux at the detector location and the modified XS equals the reaction rate of a CASMO – 3G calculation with the detector explicitly modelled. The DET-block is only used for the determination of detector signals within PRISM.

Finally, the discontinuity factors for are stored together with the cross section data for use in PRISM.

4.0 STRUCTURE AND MAIN COMPONENTS OF PRISM

The structure of the code PRISM is shown in Figure 4.1. The main components are:

- Keyword based input
- Shuffle module
- Flux solver module
- Thermal hydraulics
- Cross section characterization
- Depletion
- Pin power determination
- Output edits

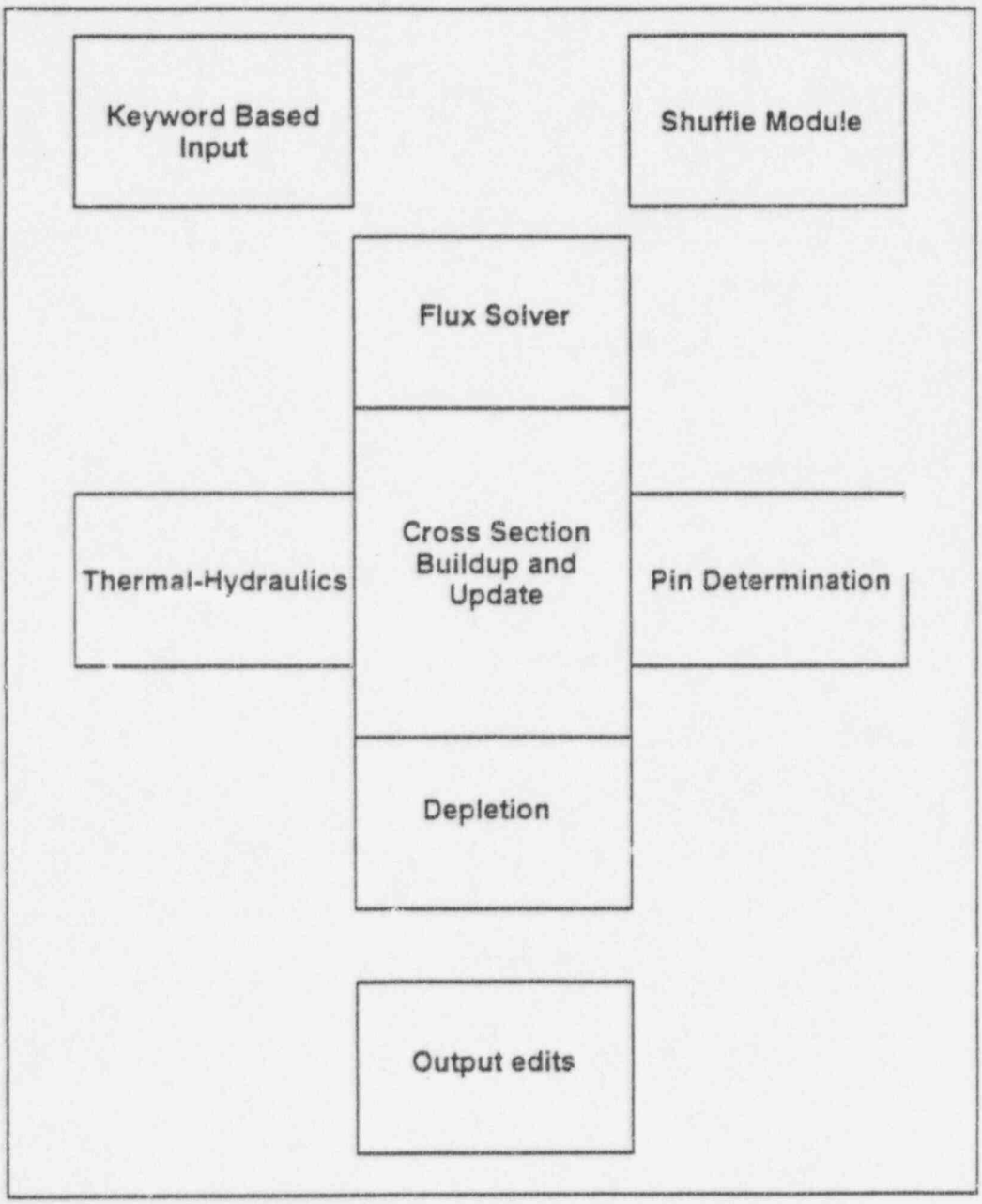

Figure 4.1 Structure and Main Components of PRISM

4.1 Usage of Nodal Cross Sections and Discontinuity Factors

PRISM is based on microscopic nodal cross sections. Their usage defines essentially the overall structure of the code and has a significant influence on its performance.

The cross sections in PRISM are represented as polynomials

Suppressing nuclide index, cross section type, and neutron energy group index, the cross sections are calculated as follows:

Reference cross sections and derivatives are determined

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The edge and corner discontinuity factors used in PRISM are and are determined

The insertion of control rods is modelled in PRISM by:

- Modification of Nodal Cross Sections

- Usage of Discontinuity Factors for the Controlled State

In the case of partially controlled nodes the usage of straight volume homogenized cross sections would result in the so called Cusping effect, i.e. the calculated integral and differential control rod worth would show nonphysical cusps as a function of the insertion depth. This effect is eliminated by using the heterogeneous control rod model which is characterized by:

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The intra-nodal burnup shape is defined by the six nodal edge values and the node average value

, therefore, results in:

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4.2 Neutronic Flux Solver – FLUXS

The nodal neutronics solution is calculated by means of the Nodal Expansion Method (NEM) or the Nodal Integration Method (NIM). Also the combined method NEM in the core and NIM in the reflector can be calculated (default option within PRISM). The methods can be applied in different approximations for the flux and the transverse leakage shapes.

Method	Flux Shape	Transverse Leakage Shape	Order

The starting point for the derivation of this method is the set of multigroup neutron diffusion equations in P_1 form for steady state conditions

$$\nabla \vec{j}_g(\vec{r}) + \left[\Sigma_{ag}(\vec{r}) + \sum_{g'=1}^{G} \Sigma_{g'g}(\vec{r})\right] \varphi_g(\vec{r}) = \sum_{g'=1}^{G} \left(\Sigma_{gg'}(\vec{r}) + \frac{1}{\lambda} \chi_g \nu \Sigma_{fg'}(\vec{r})\right) \varphi_{g'}(\vec{r}) \qquad 4\text{-}1$$

$$\vec{j}_g(\vec{r}) + D_g(\vec{r}) \nabla \varphi_g(\vec{r}) = 0 . \qquad 4\text{-}2$$

The notation is fairly standard and is presented below: φ_g and $\vec{j}_g$ are the flux and current, respectively, in each energy group g.

χ is the fission spectrum

$\nu\Sigma_{fg}$ is ν times the macroscopic fission cross section in group g

Σ_{ag} is the macroscopic absorption cross section in group g

$\Sigma_{g'g}$ is the scattering cross section for neutrons of initial energy g

D_g is the diffusion constant in energy group g

The eigenvalue λ is unity for a physically critical reactor. Equations (4-1) and (4-2) are to be solved subject to the boundary condition that $\varphi(\vec{r})$ be continuous everywhere and that the components perpendicular to internal surfaces of the current vector $\vec{j}_g$ be continuous across these surfaces. Along the exterior boundary of the reactor the incoming current vanishes. A particular class of nodal methods is characterized by the fact that spatial coupling is expressed in terms of interface currents. Formally exact nodal equations of this type can be obtained by integrating (4-1) over the volume

$$V^m = a_x^m \; a_y^m \; a_z^m$$

and (4-2) over the six surfaces of box m. The result is

$$\sum_{u=x,y,z} \frac{1}{a_u^m} \left[\left(j_{gul}^{-m} + j_{gur}^{+m} \right) - \left(j_{gul}^{+m} + j_{gur}^{-m} \right) \right] + \left(\Sigma_{ag}^m + \sum_{g'=1}^{G} \Sigma_{g'g}^m \right) \phi_g^m = \sum_{g'=1}^{G} \left(\Sigma_{gg'}^m + \frac{1}{\lambda} \chi_g \; \nu\Sigma_{fg'}^m \right) \phi_{g'}^m \qquad 4\text{-}3$$

$$D_g^m \frac{\partial \psi_{gu}^m}{\partial u} / s + \left(j_{gus}^{+m} - j_{gus}^{-m} \right) = 0 \quad (s = l, r; \; u = x, y, z) \qquad 4\text{-}4$$

where

$$\phi_g^m = \frac{1}{V^m} \int \varphi_g(\vec{r}) \, dV \qquad 4\text{-}5$$

$$\Sigma_g^m = \frac{1}{V^m \phi_g^m} \int_{V^m} \Sigma_g(\vec{r})\, \varphi_g(\vec{r})\, dV \qquad 4\text{-}6$$

$$\frac{1}{V^m} \int_{A_u^m} \vec{j}_{gur}\, \vec{n} dA = \frac{1}{a_u^m} \left(j_{gur}^{+m} - j_{gur}^{-m} \right) \qquad 4\text{-}7$$

$$\frac{1}{V^m} \int_{A_u^m} \vec{j}_{gul}\, \vec{n} dA = \frac{1}{a_u^m} \left(j_{gul}^{-m} - j_{gul}^{+m} \right) \qquad (u = x, y, z)$$

A_u^m denotes the surface of box m, the normal of which points into the positive or negative u-direction ($u = x, y, z$); j_{gus}^{+m} and j_{gus}^{-m} represent average partial currents on the right ($s = r$) or left ($s = l$) surface of box m.

ψ_{gus}^m ($s = l, r$) are average surface fluxes defined by

$$\psi_{gus}^m = \frac{1}{A_u^m} \int_0^{a_v^m} \int_0^{a_w^m} \varphi_g(\vec{r})\, dv dw \,. \qquad 4\text{-}8$$

The 1-D neutron flux is now expanded into a fourth order polynomial

$$\begin{aligned} \psi_{gu}^m(u) &= \phi_g^m + \frac{\psi_{gur}^m - \psi_{gul}^m}{2} h_1(u) \\ &+ \left(\phi_g^m - \frac{\psi_{gur}^m + \psi_{gul}^m}{2} \right) h_2(u) \\ &+ a_{3gu}^m h_3(u) + a_{4gu}^m h_4(u) \\ &= \sum_{i=0}^{4} a_{igu}^m h_i(u) \end{aligned} \qquad 4\text{-}9$$

4-10

The first three coefficients of expansion (4-9) are determined by the nodal balance equations and continuity conditions. The higher-order coefficients can be determined by requiring that (4-9) solves the equivalent 1-D diffusion equation

$$- \frac{\partial}{\partial u} D_g^m \frac{\partial}{\partial u} \psi_{gu}^m + \left(\Sigma_{agu}^m + \sum_{g'=1}^{G} \Sigma_{g'gu}^m \right) \psi_{gu}^m$$

$$= \sum_{g'=1}^{G} \Sigma_{gg'u}^m \psi_{g'u}^m + \frac{1}{\lambda} \sum_{g'=1}^{G} \chi_g \nu \Sigma_{fg'u}^m \psi_{g'u}^m - D_g^m L_{gu}^m \qquad 4\text{-}11$$

in a weighted residual sense (see below) .

Equation (4-11) is again obtained by a formal integration process. The index u at the cross sections indicates a possible spatial dependency

$$\Sigma_{gu}^m \psi_{gu}^m = \frac{1}{A_u^m} \int_0^{a_v^m} \int_0^{a_w^m} \Sigma_g^m (\vec{r}) \varphi_g (\vec{r}) \, dv \, dw . \qquad 4\text{-}12$$

If Σ_{gu}^m is constant inside the box, no assumption about the flux shape has to be made. In case of heterogeneous boxes, a polynomial approximation of the flux shape can be used in (4-12).

The term $D_g^m L_{gu}^m$ denotes the transverse leakage

$$D_g^m L_{gu}^m = - \frac{1}{A_u^m} \int_0^{a_v^m} \int_0^{a_w^m} \left(\frac{\partial}{\partial v} D_g \frac{\partial}{\partial v} + \frac{\partial}{\partial w} D_g \frac{\partial}{\partial w} \right) \varphi_g (\vec{r}) \, dv \, dw . \qquad 4\text{-}13$$

$$L^m_{gu}$$

$$L^m_{gu}$$

Third- and fourth-order coefficients of the equation (4-9) are now obtained by solving the weighted residual scheme

$$\int_0^{a^m_u} \frac{du}{a^m_u}\, w_i \left[\left(\left(\Sigma^m_{agu} + \sum_{g'=1}^{G} \Sigma^m_{g'gu} \right) \psi^m_{gu} \right.\right.$$

$$- \frac{\partial}{\partial u} D^m_g \frac{\partial}{\partial u} \psi^m_{gu} + D^m_g L^m_{gu} \qquad \text{4-14}$$

$$\left.\left. - \sum_{g'=1}^{G} \left(\Sigma^m_{gg'u} + \frac{1}{\lambda} \chi_g \nu \Sigma^m_{fg'u} \right) \psi^m_{g'u} \right] = 0 \right.$$

As weight functions, the following set of functions has been chosen for reasons of computational accuracy and efficiency:

4-15

Using the diffusion theory expression $\psi_{gus} = 2 \left(j^{+m}_{gus} + j^{-m}_{gus} \right)$ and inserting the expansion (4-9) into Fick's law (4-4), the final form of the nodal balance equations is given by

$$\left[\Sigma^m_{ag} + \sum_{g'=1}^{G} \Sigma^m_{g'g} + \sum_{u=x,y,z} \frac{2C^m_{1gu}}{a^m_u} \right] \phi^m_g$$

$$= \sum_{g'=1}^{G} \Sigma^m_{gg'} \phi^m_{g'} + \frac{1}{\lambda} \sum_{g'=1}^{G} \chi_g \nu \Sigma^m_{fg'} \phi^m_{g'} \qquad \text{4-16}$$

$$+ \sum_{u=x,y,z} \frac{1}{a^m_u} \left[\left(1 - C^m_{2gu} - C^m_{3gu} \right) \left(j^{+m}_{gul} + j^{-m}_{gur} \right) - 2C^m_{1gu} a^m_{4gu} \right]$$

and

$$j_{gul}^{-m} = C_{1gu}^{m}\left(\phi_{g}^{m} + a_{4gu}\right) + C_{2gu}^{m} j_{gul}^{+m} + C_{3gu}^{m} j_{gur}^{-m} - C_{4gu}^{m} a_{3gu}^{m}$$

$$j_{gur}^{+m} = C_{1gu}^{m}\left(\phi_{g}^{m} + a_{4gu}\right) + C_{3gu}^{m} j_{gul}^{+m} + C_{2gu}^{m} j_{gur}^{-m} + C_{4gu}^{m} a_{3gu}^{m} \qquad 4\text{-}17$$

where

4-18

The set of equations (4-14), (4-16) and (4-17) are solved for each box as an inhomogeneous boundary value problem (summarized in Figure 4.2 for two dimensions).

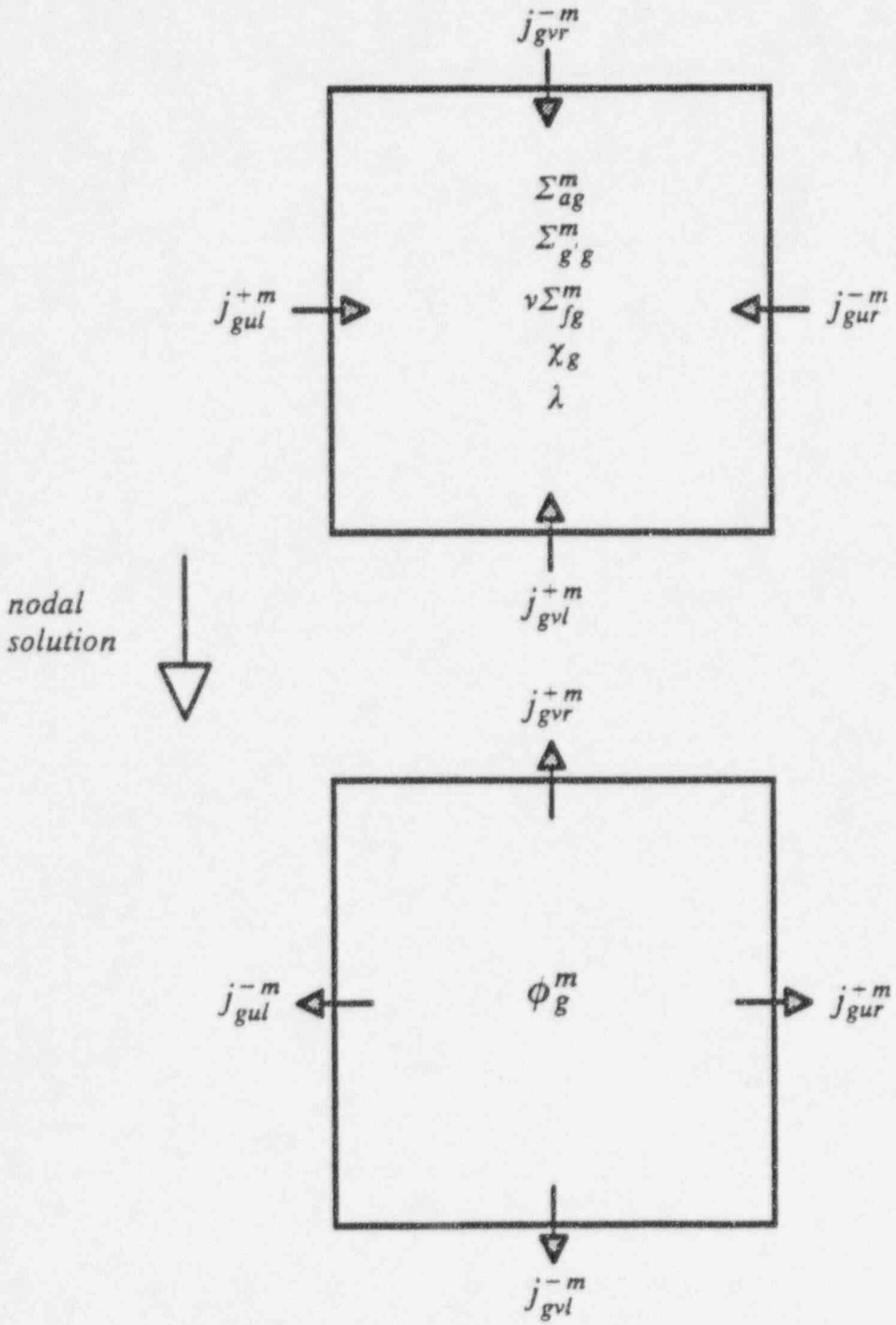

Figure 4.2 Nodal Inhomogeneous Boundary Value Problem

The iterative process to obtain the neutronics solution is accelerated both by the well-known coarse mesh rebalancing (CMR) procedure and by asymptotic extrapolation.

-
-
-
-

4.3 Thermal-Hydraulic Feedback

PRISM models a unique hydraulic channel for each radial node.

From the point of view of the steady state neutronics calculation, these assumptions are generally valid.

It is noted that PRISM is intended to be used for the analysis of normal PWR operations and minor operational transients which do not exhibit significant thermal hydraulic transients.

4.4 Determination of Pin Fluxes, Powers and Burnups: PFF

One basic requirement for the development of the new standard design procedure SAV95 was the implementation of a full three-dimensional pin determination module in the nodal reactor code PRISM.

The determination of local values is a complex task and its accuracy depends very much on a consistent methodology combining (see Figure 4.3) :

- the generation of homogenized cross sections, discontinuity factors and heterogeneous form functions,

- the feedback corrected nodal method,
- the interpolation technique(s) to determine the so called homogenized pin distributions, and finally
- the usage of the heterogeneous form functions to determine the heterogeneous (i.e., physical) pin distributions.

The SAV95 approach to determine local values is defined by the following principles:

a) Spectral code:

- Generation of homogenized cross sections, discontinuity factors and form functions
- Introduction of corner discontinuity factors

b) Nodal solution:

- Usage of edge discontinuity factors.
-
-
-

Therefore, values for flux, power, burnup and cross sections are provided as basis for the radial pin interpolation:

c) Pin interpolation (determination of homogeneous pin distributions):

-

4-20

$\Sigma_{1\text{-}2}$ = macroscopic scattering cross section

Σ_{a_2} = macroscopic thermal absorption cross section

4-21

4-22

4-23

4-24

D_2: thermal diffusion coefficient

k: mean free path length of thermal neutrons]

4-25

4-26

4-27

$\varkappa \Sigma_{f_1}$ $\varkappa$

$\varkappa \Sigma_{f_2}$ $\varkappa$

4-28

•

$$i = 1,2 \qquad \text{4-29}$$

•

4-30

d) Pin modulation (determination of heterogeneous pin distributions):

•

4-31

4-32

•

4-33

•

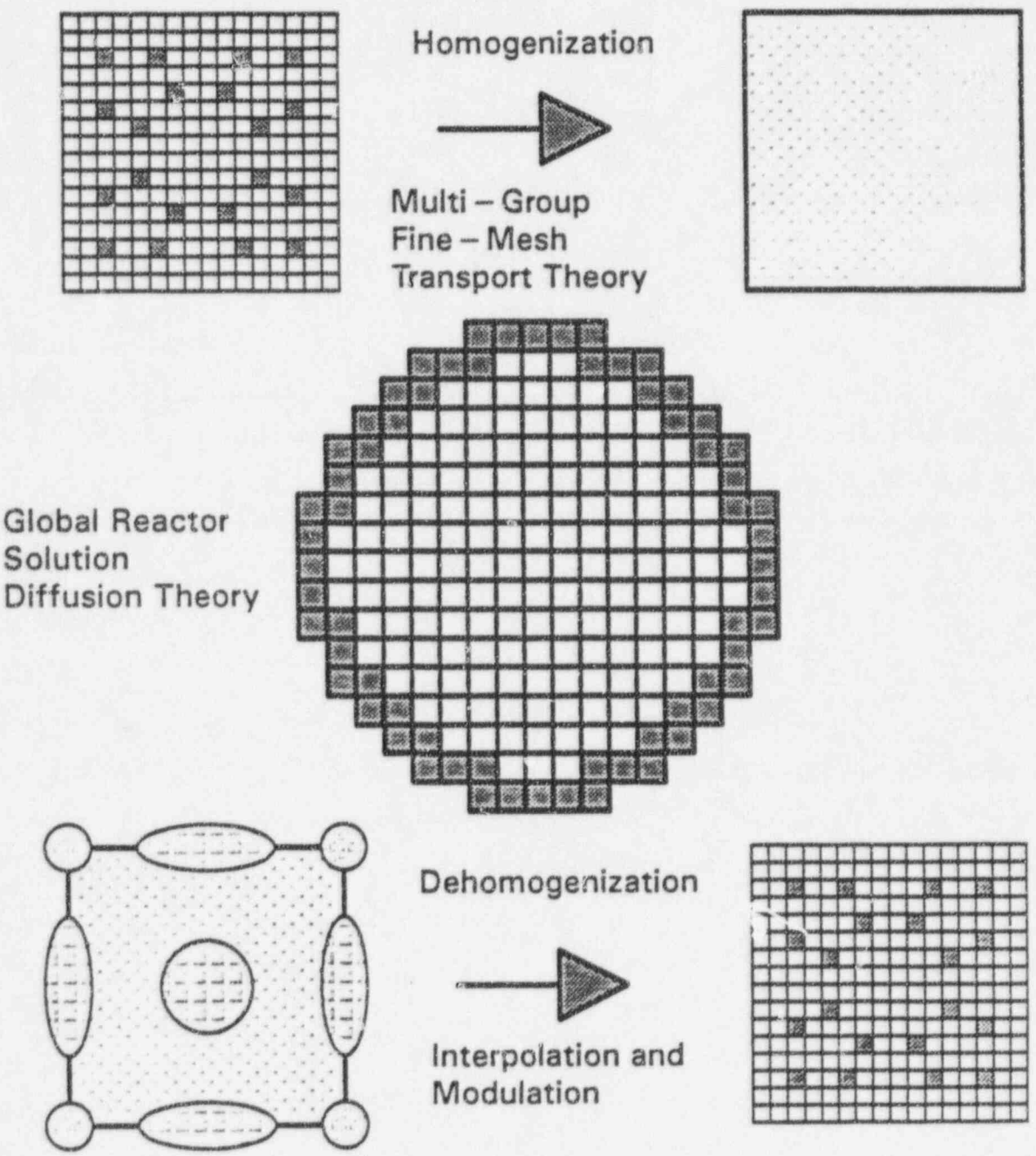

Figure 4.3 Homogenization and Dehomogenization

4.5 Depletion

The nuclear burnup equations to be solved are assumed to have the following form ($d_{ii} < 0$)

$$\frac{dx_i}{dt} = d_{ii}\, x_i + \sum_{j=1}^{i-1} d_{ij}\, x_j, \qquad i = 1, \ldots\ldots, N_x \tag{4-34}$$

or in compact matrix notation

$$\frac{dx}{dt} = Dx\ , \tag{4-35}$$

where x is a vector whose components comprise the nuclear densities of all burnable nuclides in the problem. The coefficients d_{ij} of Equation (4-34) are the elements of the lower triangular matrix D and represent the reaction rates for the transmutation of nuclide j into nuclide i. The diagonal elements d_{ii} correspond to the total removal rate of nuclide i from the system.

D

, the burnup equations for all fuel chains are solved first to obtain the fuel concentrations $\bar{x}_i$, $i = 1, \ldots, N_{fuel}$, averaged over the time step Δt. For each fission product $j = N_{fuel} + 1, \ldots, N_x$; a constant average production rate $\bar{S}_j$ is then calculated

$$\begin{aligned} \bar{S}_j &= \sum_{i=1}^{N_{fuel}} d_{ji}\, \bar{x}_i \\ &= \sum_{i=1}^{N_{fuel}} \Gamma_{ji}\, \bar{x}_i \sum_{g=1}^{G} \sigma_{fi}^{g}\, \phi_g \end{aligned} \tag{4-36}$$

to replace the actual time dependent source $S_j(t)$ for the time step $t_{..}$ Γ_{ji} is the yield of fission product j from fissions of nuclide i.

$a = a_t$

i

$i' < i$ $i'' > i$

i

During the last two decades, only two major improvements were added to the original method:

- Flux renormalization,

- Feedback in the depletion chains

After a depletion step and the corresponding update of all nodal nuclide particle number densities, the code determines

Both distributions are extensively used in the Spectral Interpolation Method (see Section 4.4).

4.6 Shuffle Module

Some basic capabilities of the fuel shuffle module are:

- Renaming of fuel assemblies
- Rotation of fuel assemblies
- Marking fuel assemblies as defective
- Marking fuel assemblies as repaired
- Marking fuel assemblies which have reached their target burnup
- Marking fuel assemblies as being removed from the plant site

The complete history of all fuel assemblies which are or have ever been at the plant site, including the reactor core, pool and dry storage is maintained. Together with the accumulated history information, the actual fuel assembly data (e.g., nodal nuclide particle number densities and burnup distributions) are read, processed and updated by the reload operator.

Furthermore, these data can be transferred to a postprocessor program which generates various tabular or graphic output and to a program which serves for the transition between different core geometries (e.g., change of axial node heights) or for the setup of reactor depletion calculations in higher cycles (jump start).

4.7 Keyword Based Input: KBI

A Keyword Based Input processor (KBI) was developed for PRISM.

One special feature of KBI is defined by its hierarchical data structure. KBI provides three levels of hierarchy for the input data. The primary layer has a collection of keywords called primary keywords. A primary keyword can be defined to open a secondary layer called a group. The group has its own collection of keywords called group keywords. A group keyword can be defined to open a third layer called a subgroup. The subgroup also has its own collection of keywords called subgroup keywords.

The special keyword $INCLUDE offers the capability to construct the input for a complex job by the usage of predefined standard input tasks. The keyword $INCLUDE causes an external file to be embedded within the current file, replacing the special keyword record. Nested include files are supported to a depth of ten.

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Reactor Analysis System for PWRs
Volume 2 Benchmarking Results

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REACTOR ANALYSIS SYSTEM FOR PWRS
VOLUME 2 BENCHMARKING RESULTS

1.0 INTRODUCTION

The evaluation and validation process for the Siemens SAV95 code system is presented in this document. (SAV is an acronym for Standard Design Procedure.)

The Siemens SAV95 code system is described in Reference 1. The evaluation and validation results consist of comparisons of SAV95 calculational results with measured data and with the calculational results of the currently approved code system. The purpose of these comparisons is to demonstrate the adequacy of the SAV95 code system to perform neutronic design analyses for PWRs.

2.0 SUMMARY

The SPC PWR neutronics design methodology consists of a cross-section generator computer code system and a reactor core simulator computer code system. A flowchart which summarizes these code systems is shown in Figure 2.1.

The cross-section generator is used to calculate basic nuclear parameters that are required by the reactor core simulator. Input to the cross-section generator includes a nuclear data library (e.g. ENDF/B files), as well as the user input describing the assembly lattice.

The reactor core simulator is used to model the reactor core and perform the basic core calculations required for fuel cycle design, safety analyses, and core operation follow. This includes input required for an incore monitoring code such as pin-by-pin power distributions. Input to the reactor core simulator includes the basic nuclear parameters calculated by the cross-section generator, as well as the user input describing the core.

The demonstration of the adequacy of the design methodology consists of verification that the methodology performs satisfactorily. The neutronics design methodology must produce accurate predictions, as exemplified by comparison with measured results. Additionally, if the methodology is used to produce input for an incore monitoring system, then the associated power distribution measurement uncertainties must be within the previously approved measurement uncertainties for that incore monitoring system.

These considerations are used to establish requirements which, when satisfied, demonstrate that the SPC PWR neutronic design methodology is acceptable for use in reload design, licensing, and calculation of startup and core follow data. These requirements include methodology requirements, validation requirements, and verification of previously approved measurement uncertainties for incore monitoring.

Results of the benchmark calculations demonstrate the adequacy of the SAV95 code system to perform neutronic design analyses.

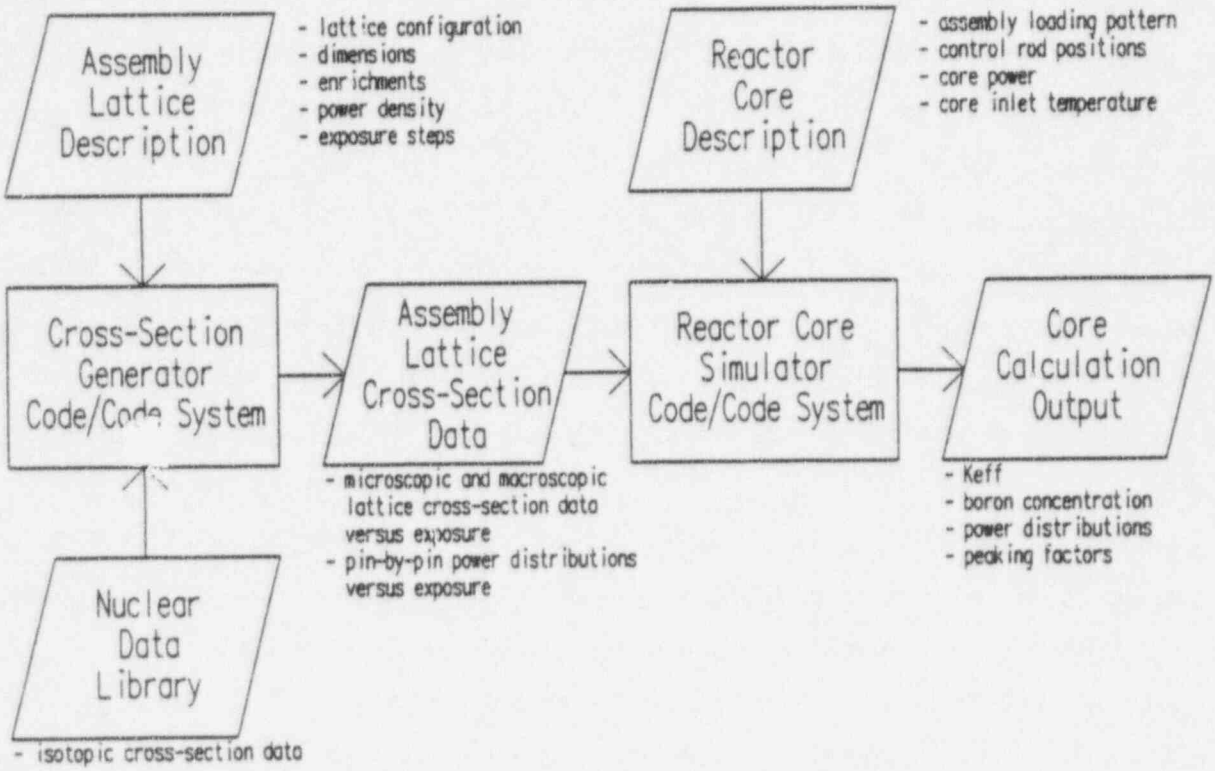

Figure 2.1

SPC PWR Neutronics Design Methodology Flowchart

3.0 METHODOLOGY DESCRIPTION

The SAV95 code system uses CASMO-3 to generate cross sections and PRISM to determine core reactivity and power distributions. PRISM evaluates number densities and burnup of key isotopes on a nodal basis using microscopic cross sections. The nodal expansion method is utilized in solution of the two group diffusion theory representation of the reactor core. Pin power distributions are generated by PRISM using a pin power reconstruction technique.

4.0 VALIDATION

The SPC neutronics design package, SAV95, consists of a cross-section generator code system and a reactor core simulator computer code system. Data calculated with SAV95 are compared to measured data in order to demonstrate the accuracy and capability of the code system. Following are comparisons of data calculated with the SAV95 code system to measured data obtained from critical experiments, startup physics tests, and core follow data obtained from commercial reactors. Additionally, comparisons to data calculated with previously approved analytical models are presented to show that the SAV95 code system provides similar information. The validation criteria chosen for the SAV95 model is based on ANSI/ANS-19.6.1 standard Reference 2.

4.1 Critical Experiment Reactivity Measurements

Comparisons of the SAV95 code system to critical experiments are performed. These calculated results are from the cross section generation portion of the SAV95 code system.

Examples of critical experiments for which data are available are the Strawbridge-Barry room temperature critical experiments (Reference 3), the KRITZ increased temperature critical experiments (Reference 4), and the Babcock and Wilcox (B&W) room temperature criticals (Reference 5). The measured data from these experiments includes reactivity for a variety of lattice configurations, enrichments, burnable absorber loadings, boron concentrations, and temperatures. Pin-by-pin fission rate distribution data are also available for various lattice configurations, enrichments, and burnable absorber loadings.

The critical experiments describe conditions at which criticality is achieved, i.e. a measured k_{eff} of approximately 1.0. The results of the cross-section generator calculation do not indicate any significant trends with respect to the various conditions considered.

4.2 Validation With Commercial Reactor Measurements

4.2.1 Description of Reactor Cores

The validation calculations involve comparisons with measurements obtained at commercial reactors. Each sample plant has an associated plant type specification and fuel type specification.

The 'plant type' specification refers to the NSSS vendor and the total number of fuel assemblies in the core, such as:

- Westinghouse reactor containing 121 assemblies
- Westinghouse reactor containing 157 assemblies
- Westinghouse reactor containing 193 assemblies
- Combustion Engineering reactor containing 133 assemblies
- Combustion Engineering reactor containing 217 assemblies
- Combustion Engineering reactor containing 241 assemblies

The 'fuel type' specification refers to the size of the rod array and the guide tube configuration in an assembly, such as:

- 14x14 array with Westinghouse guide tube configuration
- 15x15 array with Westinghouse guide tube configuration
- 17x17 array with Westinghouse guide tube configuration
- 14x14 array with Combustion Engineering guide tube configuration
- 15x15 array with Combustion Engineering guide bar configuration
- 16x16 array with Combustion Engineering guide tube configuration

A unique combination of a plant type and a fuel type will be referred to as a 'plant/fuel type'.

For each sample plant, measured data from one or more cycles is used in comparisons with predictions. A description of each plant and cycle used in the validation calculations is provided in order to illustrate the range of parameters used in the benchmarking. This description includes:

- Description of the fresh and burnt fuel assemblies, including:
  - Initial enrichments
  - Burnable poison loading
- Cycle length
- Any unique features of this plant/cycle

A number of sample plants, each with a unique plant/fuel type combination, are used in the validation against commercial reactor measurements.

4.2.2 Startup Physics Test Measurements

Comparisons include a number of different plant/fuel types, including at least one Combustion Engineering or Westinghouse plant type. Additionally, many cycles of operation are evaluated, with more than one cycle of operation per plant type. Startup physics test predictions are compared to the available measured data at beginning-of-cycle for each cycle of each plant/fuel type. The parameters include:

1) All rods out critical boron concentration
2) Individual control bank worths
3) Total control bank worth
4) All rods out isothermal temperature coefficient

The above list of parameters is representative of the startup physics test data typically available from commercial reactor measurements at zero power conditions. The criteria for the parameters are based on the recommended test criteria provided in Reference 2. All results are within the specified criteria.

4.2.3 Core Follow Measurements

Comparisons are to include different plant/fuel types, including at least one Westinghouse plant type and at least one Combustion Engineering plant type. Additionally, a minimum number of cycles of operation should be considered. Core follow predictions are compared to the available data measured at or near hot full power for each cycle of each plant/fuel type. The parameters compared are:

1) Critical boron concentrations
2) Assembly average power distributions
3) Core average axial power distributions

Measured boron concentration and power distribution data are typically available from commercial reactor measurements at full power conditions as a function of cycle burnup. The criteria for these parameters are based on the recommended test criteria provided in Reference 2.

The results are within the specified criteria.

4.3 Previously Approved Methodology Calculations

Comparisons of parameters calculated with SAV95 to values calculated using the previously approved CASMO-2/XTGPWR code system (Reference 6) are also made. These comparisons along with the comparisons to measurements are representative of the parameters which are significant to the safety analyses. These results demonstrate that use of the SAV95 code

system to calculate parameters for safety analyses will not have a significant impact on the analyses.

5.0 VERIFICATION OF POWER DISTRIBUTION MEASUREMENT UNCERTAINTY

This section describes the verification of the continued applicability of previously approved power distribution measurement uncertainties. This verification is necessary if either the cross-section generator methodology or the reactor core simulator methodology is replaced.

The measurement uncertainty for the power distribution peaking factors has been verified for two specific incore monitoring code systems, movable incore detectors (Westinghouse design) and fixed incore detectors (CE design). The power distribution measurement codes used in conjunction with the specific incore monitoring systems are described in References 7 and 8. Extension of the revised neutronics design methodology to other incore monitoring systems requires additional validation/verification.

Data reduction and statistical treatment techniques described in Reference 7 and Reference 8 have been used to verify the one-sided 95/95 relative uncertainties.

5.1 Nodal and Assembly Power Distribution Uncertainties

The standard deviations of the relative uncertainty in the assembly, planar, or nodal power distributions have been determined for the movable detector incore monitoring system using measured data from independent reactors. The standard deviations of the relative uncertainty in the assembly and nodal power distributions have also been determined for the fixed detector incore monitoring system using measured operating data.

The available data measured at or near hot full power has been used in the comparisons. These results are statistically combined to yield the standard deviations of the relative uncertainty in the assembly, planar, or nodal power distributions for the specified incore monitoring system.

5.2 Local Peaking Factor Uncertainty

The standard deviations of the relative uncertainty in the local peaking factor were determined by comparisons of calculated pin-by-pin fission rate distributions with critical experiment measurements. These comparisons include a variety of lattice configurations, e.g. both Westinghouse and Combustion Engineering guide tube configurations, various enrichments, and the inclusion of burnable absorbers.

From comparisons of the calculated to measured pin-by-pin fission rate distributions, values for the combined uncertainties in calculation and measurement were determined.

5.3 Power Distribution Measurement Uncertainty

The standard deviations of the relative uncertainties associated with the assembly, planar, or nodal power distribution and the local peaking factor determined in sections 5.1 and 5.2 were statistically combined and expressed in terms of relative standard deviations. A one-sided 95/95 tolerance limit factor was applied to the corresponding relative standard deviation, thereby resulting in the final relative uncertainty factor for each peaking factor.

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Attachment

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May 8, 1996
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Attn: Chief, Planning, Program and Management Support Branch

Reference: EMF-96-029(P), Volumes 1 and 2, and EMF-96-029(NP), Volumes 1 and 2, "Reactor Analysis System for PWR's," Siemens Power Corporation, May 1996.

Enclosed are twenty-five copies of the proprietary referenced topical report (both volumes) and fifteen copies of the nonproprietary referenced topical report (both volumes). Siemens Power Corporation is submitting this topical report to the NRC for review. The topical report describes a revised PWR neutronic code system, SAV-95. This code system consists of CASMO as the cross-section generator and PRISM as the core simulator. These codes replace the current approved CASMO version used as the cross-section generator and XTGPWR as the core simulator. A code description is provided in Volume 1 and Volume 2 provides the benchmarking.

Siemens Power Corporation considers some of the information contained in the (P) version of this report to be proprietary. In accordance with the requirement of 10 CFR 2.790(b), an affidavit is enclosed to support the withholding of this topical report from public disclosure.

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Very truly yours,

R. A. Copeland, Manager
Product Licensing

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Enclosures

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Mr. R. C. Jones (USNRC) w/ reports
Mr. E. Weiss (USNRC) w/ reports

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SPC Restrictions on SAV95

Reference: EMF-96-029(P), Volumes 1 and 2, and EMF-96-029(NP), Volumes 1 and 2, "Reactor Analysis System for PWR's," Siemens Power Corporation, May 1996.

As discussed with Mr. L. Kopp of your staff, Siemens Power Corporation will impose the following restrictions on our use of the SAV95 neutronics methodology described in the referenced topical reports. The specific restrictions are:

1. SAV95 application will be supported by additional code validation to insure that the methodology and uncertainties are applicable:

   a) For designs differing from the Westinghouse reactors with 157 fuel assemblies with either 15x15 or 17x17 fuel rod arrays, and Combustion Engineering reactors with 217 fuel assemblies with 14x14 fuel rod arrays,

   b) When using incore monitoring systems differing from the INPAX-W and INPAX-2 systems contained in the evaluation when SPC provides input from SAV95.

2. Modifications to the code and methodology must be validated using the criteria approved in the referenced SAV95 topical report.

3. The validation will be maintained by SPC and be available for NRC audit.

If you have any questions, or if I can be of additional assistance, please call me at (509) 375-8290.

Very truly yours,

R. A. Copeland
Product Licensing

cc: Mr. H. D. Curet (SPC)
Mr. L. Kopp (USNRC)
Mr. E. Wang (USNRC)

bc: L. J. Federico
R. G. Grummer
J. S. Holm
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EMF-96-029(NP)(A)
Volume 1
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Attachment

Reactor Analysis System for PWRs
Volume 1 - Methodology Description
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USNRC/R. A. Copeland (15)

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Siemens Power Corporation - Nuclear Division

EMF-96-029(NP)(A)
Issue Date: 1/8/97

Reactor Analysis System for PWRs

Volume 1 - Methodology Description

Volume 2 - Benchmarking Results

Attachment - Correspondence

/smg

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

ENCLOSURE

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATING TO TOPICAL REPORT EMF-96-029(P), "REACTOR ANALYSIS SYSTEM FOR PWR'S" SIEMENS POWER CORPORATION

1. INTRODUCTION

In a letter of May 8, 1996 (Ref. 1), Siemens Power Corporation (SPC) submitted Volumes 1 and 2 of the topical report EMF-96-029(P), "Reactor Analysis System for PWR's" (Ref. 2) for U.S. Nuclear Regulatory Commission (NRC) review. This topical report presents a core physics computer code system for pressurized water reactors (PWRs) which will be used by SPC to perform neutronics design analyses. The code system, named SAV95, will replace the XTGPWR/PDQ/CASMO-2E codes currently used by SPC.

SAV95 consists of a neutron cross section generator computer code system (MICBURN-3/CASMO-3G) and a reactor core simulator computer code system (PRISM). The cross section generator is used to calculate basic nuclear parameters that are required by the reactor core simulator. Input to the cross section generator includes a nuclear data library as well as user input describing the assembly lattice. The reactor core simulator is used to model the reactor core and perform the basic core calculations required for fuel cycle design and safety analyses, which include the required input for an incore monitoring code.

2. SUMMARY OF TOPICAL REPORT

Section 1 of Volume 1 of the topical report presents an introduction describing the type of neutronic calculations to be performed by the SAV95 code system and the improvements of SAV95 over SPC's previous neutronics design codes. Section 2 presents the development goals and basis, the new or improved models, and the major characteristics of the SAV95 code system. The neutron cross section generation scheme, including the nuclide chain used in MICBURN-3 and the representation of nodal cross sections for PRISM, are described in Section 3. Section 4 presents the structure and main components of the PRISM code, and references are given in Section 5.

Volume 2 of the report presents comparisons of data calculated with the SAV95 code system to measured data obtained from critical experiments, startup physics tests, and core follow data obtained from commercial power reactors. Additionally, comparisons to data calculated with previously approved analytical models are presented.

3. TECHNICAL EVALUATION OF REPORT

The MICBURN-3 (Ref. 3) and CASMO-3G (Ref. 4) codes are used to generate the microscopic and macroscopic lattice cross sections versus burnup (exposure) and fuel rod by rod power distributions versus burnup. The NRC has approved

UNITED STATES
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October 29, 1996

Mr. H. D. Curet
Manager, Product Licensing
Siemens Power Corporation
2101 Horn Rapids Road
P.O. Box 130
Richland, Washington 99352-0130

Dear Mr. Curet:

SUBJECT: ACCEPTANCE FOR REFERENCING OF TOPICAL REPORT EMF-96-029(P), VOLS. 1 AND 2, "REACTOR ANALYSIS SYSTEM FOR PWR'S" (TAC NO. M95745)

We have reviewed the topical report submitted by Siemens Power Corporation (SPC) by letter dated May 8, 1996. We find the report to be acceptable for referencing in license applications to the extent specified and under the limitations stated in the enclosed safety evaluation. The evaluation defines the basis for accepting the report.

We will not repeat our review of the matters described in the report and found acceptable when the report appears as a reference in license applications, except to ensure that the material presented applies to the specific plant involved. Our acceptance applies only to the matters described in the report. In accordance with procedures established in NUREG-0390, "Topical Report Review Status," the NRC requests that SPC publish an accepted version of the report within three (3) months of receiving this letter. The accepted version shall incorporate this letter and the enclosed evaluation between the title page and the abstract and include an "-A" (designating accepted) after the report identification symbol.

If the NRC's criteria or regulations change such that its conclusions as to the acceptability of the report are invalidated, SPC and the applicants referencing the topical report should revise and resubmit their respective documentation, or submit justification for the continued applicability of the topical report without revising their respective documentation.

Sincerely,

Timothy E Collins

Timothy E. Collins, Acting Chief
Reactor Systems Branch
Division of Systems Safety and Analysis

Enclosure:
EMF-96-029(P) Safety Evaluation

these codes for use by SPC for boiling water reactor (BWR) neutronics calculations (Ref. 5). MICBURN-3 is a multigroup, one-dimensional transmission probability code that calculates neutron cross sections as a function of burnup in an absorber rod containing an initially homogeneous distribution of burnable absorber. These cross sections as functions of absorber number density are input to CASMO-3G. CASMO-3G is a multigroup, two-dimensional transport theory code used for burnup calculations on PWR and BWR fuel rods or assemblies. The code handles a geometry consisting of cylindrical fuel rods of varying composition in a square pitch array with allowance for fuel rods loaded with a burnable absorber, burnable absorber rods, cluster control rods, incore instrument channels, water gaps, boron steel curtains, and cruciform control rods in the regions separating fuel assemblies. The assembly lattice cross section data and reactor core description (e.g., assembly loading pattern, control rod position, core power, core inlet temperature) are input to the reactor core simulator code system, PRISM, which calculates k_{eff}, boron concentration, power distributions, peaking factors, control rod worths, and input for safety analyses and incore monitoring. PRISM uses the nodal expansion method (NEM) to solve the two-group diffusion theory representation of the reactor core.

A number of new models have been incorporated into SAV95. These include an extension of the available nuclide chain for isotopic depletion, a continuous neutron cross section representation covering all possible combinations of thermal-hydraulic parameters, a faster and more efficient flux solution module for stationary reactor states, the introduction of discontinuity factors at the boundary between assemblies or quadrants and for reflector regions, and a full three-dimensional fuel rod interpolation scheme in the determination of rodwise flux, power, and burnup values. These changes are acceptable because they offer improvements which result in both increased accuracy and efficiency and which enhance ease of use.

In order to qualify the SAV95 code system, SPC has compared data calculated with SAV95 to measured data from critical experiments as well as to startup physics test data and core follow data obtained from commercial power reactors. The validation criteria used for the SAV95 model are based on those suggested in ANSI/ANS-19.6.1-1985, "American National Standard Reload Startup Tests for Pressurized Water Reactors" (Ref. 6).

3.1 Critical Experiment Reactivity Measurements

Calculated results from MICBURN-3/CASMO-3 were compared to measured data from the Strawbridge-Barry (Ref. 7), KRITZ (Ref. 8), and the Babcock and Wilcox (Ref. 9) critical experiments and the calculated mean value of k_{eff} was 1.00039 ± 0.00107. The calculations were performed by Studsvik, not by SPC. However, the very good agreement between the calculations and the measurements validates the ability of the cross section generation portion of the SAV95 code system to accurately predict reactivity.

3.2 Power Reactor Measurements

Two Westinghouse (W) plants containing 157 fuel assemblies (one 15x15 fuel rod array and one 17x17 fuel rod array) and one Combustion Engineering (CE)

reactor containing 217 fuel assemblies with a 14x14 fuel rod array were used in the validation against commercial power plant measurements. A total of 14 cycles of operation were evaluated with at least three cycles evaluated for each plant. SPC compared the SAV95 model predictions to startup physics test data as well as to core follow measurements from each plant.

Startup physics test measurements are typically performed at hot zero power (HZP) conditions, and include critical boron concentrations, control bank worths, and isothermal temperature coefficients. The results of SAV95 comparisons with startup physics test measurements show a maximum absolute difference of 46 ppm in the all rods out (ARO) HZP critical boron concentration compared to the recommended validation test criterion of ±50 ppm (Ref. 6). The maximum absolute difference between calculated and measured individual HZP control bank worths was 14.3% and 93 pcm (where 1 pcm = $1x10^{-5}$ $\Delta k/k$) compared to the Reference 6 criteria of ±15% or ±100 pcm, whichever is larger. The maximum absolute difference between predicted and measured total HZP control bank worths was 7.3% compared to the recommended validation criterion of ±10%. The maximum absolute difference between the predicted and measured ARO HZP isothermal temperature coefficient was 1.01 pcm/°F compared to the validation criterion of ±2 pcm/°F. Therefore, SAV95 predictions of startup physics measurements met all applicable criteria.

Core follow measurements obtained during operation typically are measured at hot full power (HFP), and include critical boron concentrations and power distributions as a function of burnup. The results of SAV95 comparisons with core follow measurements show a maximum absolute difference of less than 50 ppm between the measured and calculated HFP critical boron concentration as a function of core burnup, thus meeting the validation criterion of ≤50 ppm recommended in Reference 6. The root mean square (RMS) difference of 0.018 between predicted and measured assembly average power distributions at beginning of cycle (BOC), middle of cycle (MOC), and end of cycle (EOC) is well within the recommended criterion (Ref. 6) of <0.05. Comparisons of measured and predicted core average axial power distributions at BOC, MOC, and EOC show a maximum RMS difference of 0.049, which is within the recommended criterion of <0.05. Therefore, SAV95 predictions of core follow measurements met all applicable criteria.

3.3 Previously Approved Methodology Calculations

In addition to comparisons involving measured data, selected safety analysis parameters (Doppler coefficient, differential boron worth, and delayed neutron fraction) were compared to values calculated with the previously approved neutronics design methodology, CASMO-2/XTGPWR (Ref. 10). The good agreement in these comparisons demonstrate that the proposed new SAV95 methodology is compatible with the safety and licensing analyses.

3.4 Verification of Power Distribution Measurement Uncertainty

Since a replacement is proposed for the previously approved neutronics methodology, verification of the continued applicability of previously approved power distribution measurement uncertainties was performed. The measurement uncertainty for the power distribution peaking factors was

5. REFERENCES

(1) Letter from R. A. Copeland (SPC) to Document Control Desk (NRC), transmittal of EMF-96-029(P), Volumes 1 and 2, and EMF-96-029(NP), Volumes 1 and 2, "Reactor Analysis System for PWR's", RAC:96:043, May 8, 1996.

(2) S. K. Merk, et al, "Reactor Analysis System for PWR's," Volume 1, "Methodology Description," and Volume 2, "Benchmarking Results," EMF-96-029(P), May 1996.

(3) M. Edenius, et al, "MICBURN-3, Microscopic Burnup in Burnable Absorber Rods: Methodology," STUDSVIK/NFA-86/28, 1986.

(4) M. Edenius, et al, "CASMO-3, A Fuel Assembly Burnup Program: Methodology," STUDSVIK/NFA-86/8, 1986.

(5) Letter from A. C. Thadani (NRC) to R. A. Copeland (ANF), "Acceptance for Referencing of Topical Report XN-NF-80-19(P), Volume 1, Supplement 3, Advanced Nuclear Fuels Methodology for Boiling Water Reactors; Benchmark Results for the CASMO-3G/MICROBURN-B Calculation Methodology,'" August 13, 1990.

(6) American National Standard Reload Startup Tests for Pressurized Water Reactors, ANSI/ANS-19.6.1-1985, American Nuclear Society, 1985.

(7) L. E. Strawbridge and R. F. Barry, "Criticality Calculations for Uniform Water-Moderated Lattices," Nuclear Science and Engineering, Vol. 23, pp. 58-73, 1965.

(8) R. Persson, et al, "High-Temperature Critical Experiments with H_2O-Moderated Fuel Assemblies in KRITZ," Technical Meeting No. 2/11, NUCLEX 72, 1972.

(9) L. W. Newman, "Urania-Gadolinia: Nuclear Model Development and Critical Experiment Benchmark," BAW-1810, Babcock and Wilcox Company, April 1984.

(10) "Exxon Nuclear Neutronic Design Methods for Pressurized Water Reactors," XN-75-27(A), and Supplements 1, 2, 3, 4, and 5, Exxon Nuclear Company.

(11) "Power Distribution Measurement Uncertainty for INPAX-W in Westinghouse Plants," EMF-93-164(P)(A), Siemens Power Corporation, February 1995.

(12) "Exxon Nuclear Analysis of Power Distribution Measurement Uncertainty for St. Lucie Unit 1," XN-NF-83-01(P), Exxon Nuclear Company, January 1983.

(13) Letter from R. A. Copeland (SPC) to Document Control Desk (NRC), SPC Restrictions on SAV95, RAC:96:066, October 11, 1996.

verified for two specific incore monitoring code systems, the Westinghouse design using movable incore detectors (Ref. 11), and the CE design using fixed incore detectors (Ref. 12).

The standard deviations of the relative uncertainties associated with the assembly, planar, or nodal power distribution were determined for the movable detector incore monitoring system (Westinghouse), using measured data from 11 cycles of operation of two different reactors, and for the fixed detector system (CE), using measured data from three cycles of a single reactor. The standard deviations of the relative uncertainty in the local peaking factor were determined by comparisons of calculated rod-by-rod fission rate distributions with critical experiment measurements. The standard deviations were statistically combined and expressed in terms of relative standard deviations. The data reduction and statistical techniques described in References 11 and 12 were used to verify that the one-sided 95/95 relative uncertainties are less than the measurement uncertainties previously approved by the NRC for the specified incore monitoring system. Therefore, the continued use of the previously approved power distribution measurement uncertainties for the Westinghouse and the CE detector systems with the proposed new methodology is acceptable. An extension of the methodology to other incore monitoring systems will require additional validation and verification of the acceptable uncertainties.

4. SUMMARY AND CONCLUSIONS

The NRC staff has reviewed the proposed SAV95 methodology as well as comparisons of the SAV95 code system with measured data from critical experiments, operating reactors, and previously approved methodology calculations. On the basis of this review, the staff finds the use of SAV95 acceptable for use by SPC in PWR reload core design, safety analysis parameter calculations, and startup and operations calculations. As stated in a letter of October 11, 1996, from SPC to the NRC (Ref. 13), SPC will impose the following restrictions on their use of the SAV95 neutronics methodology described in EMF-96-029(P), Volumes 1 and 2. The specific restrictions are:

1) SAV95 application will be supported by additional code validation to insure that the methodology and uncertainties are applicable:

   a) For designs differing from the Westinghouse reactors with 157 fuel assemblies with either 15x15 or 17x17 fuel rod arrays, and CE reactors with 217 fuel assemblies with 14x14 fuel rod arrays,

   b) When using incore monitoring systems differing from the INPAX-W and INPAX-2 systems contained in this safety evaluation when SPC provides input from SAV95.

2) Modifications to the code and methodology will be validated using the criteria approved in EMF-96-029(P).

3) The validation will be maintained by SPC and be available for NRC audit.