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Neutronic investigations of an equilibrium core for a tight-lattice light water reactor

Dedicated to Professor Werner Oldekop on his 65th birthday

Calculation procedures and first results concerning the neutronic design of an equilibrium core of an advanced pressurized water reactor with mixed oxide fuel in a compact light water moderated triangular lattice are presented. Principle and qualification of the cell burnup calculations with the KARBUS program are briefly discussed. The fuel assembly design with single control rod positions filled with control rod material or coolant water requires special transport theory calculations, which are performed with a one-dimensional supercell model. The macroscopic fuel assembly cross section data is collected in a special library to be used in a new calculational procedure, ARCOSI, for multi-cycle reactor core simulations. Its first application for a reference design resulted in an equilibrium configuration with moderator density reactivity coefficients which are satisfactory as regards safety.

Neutronenphysikalische Untersuchungen des Gleichgewichtskerns für einen fortgeschrittenen Druckwasserreaktor. *Rechenverfahren und erste Ergebnisse der neutronenphysikalischen Auslegung des Gleichgewichtskerns eines fortgeschrittenen Druckwasserreaktors mit kompaktem leichtwassermoderiertem Gitter mit Mischoxid-Brennstoff werden vorgestellt. Grundlagen und Qualifikation der Zellabbrand-Rechnungen mit dem Programm KARBUS werden kurz erläutert. Für die Brennelemente mit Einzelstabpositionen für die Regelung werden eindimensionale Super-Zellen-Rechnungen durchgeführt. Die makroskopischen Brennelementdaten werden in einer speziellen Bibliothek gespeichert und in dem neuen Reaktorkern-Simulationsprogramm ARCOSI weiterverwendet. Die ersten Berechnungen eines Gleichgewichtskerns im Rahmen eines Referenzentwurfs haben gezeigt, daß eine Konfiguration mit sicherheitstechnisch zufriedenstellenden Moderatordichte-Reaktivitätskoeffizienten gefunden werden kann.*

1 Introduction

The characteristics of a pressurized water reactor with mixed oxide (MOX) fuel in a closely packed light water moderated triangular lattice have been investigated in the Nuclear Research Center Karlsruhe (KfK) since 1978 [1]. This reactor concept was proposed in 1975 by *Edlund* [2]. In cooperation with other partners, e.g., the reactor manufacturer Siemens KWU, the Technical University of Braunschweig (TUBS) and the Paul Scherrer Institute, all aspects of the replacement of the core of a modern pressurized water reactor designed by KWU have been considered. In the literature this reactor type is referred to as *advanced pressurized water reactor* (APWR).

This paper describes the neutron physics investigations concerning the consistent design of an equilibrium core for the APWR. Some parts of the work were performed in close cooperation with the Technical University of Braunschweig, e.g., the validation of the basic cell calculations [3] and the first multi-cycle investigations [4]. A more extensive documentation of the investigations at KfK, related to the neutron physics design of APWR cores, is in progress [5].

2 The calculational procedures

2.1 General

The spectral properties of tight light water moderated lattices deviate significantly from those of established reactor systems as fast breeder reactors (FBR) or pressurized water reactors (PWR). The differences are demonstrated by means of the cumulative reaction rates for fission and absorption of neutrons in Fig. 1. The curves give the accumulated number of reaction rates occurring below a certain energy. They start with zero at the lowest and increase to unity at the highest energy. The steepest slope of the curves indicates the energy region with maximum reaction rates. It can be seen that the APWR curves have the steepest slope in the energy region between approximately 5 eV and 1 keV. The neutron cross section resonances play a dominant role in this energy region. As a consequence, calculation of the effective neutron cross sections in the resonance region has to be performed with high accuracy for an APWR system. A further characteristic property of the APWR lattice is the high Pu content of the MOX fuel, see also [6]. The high capture resonances of ^{240}Pu at 1 eV and ^{242}Pu at 2.67 eV, together with the resonance of ^{238}U at 6.68 eV, may lead to a positive reactivity effect with decreasing moderator density. This effect would have an unacceptable impact on the

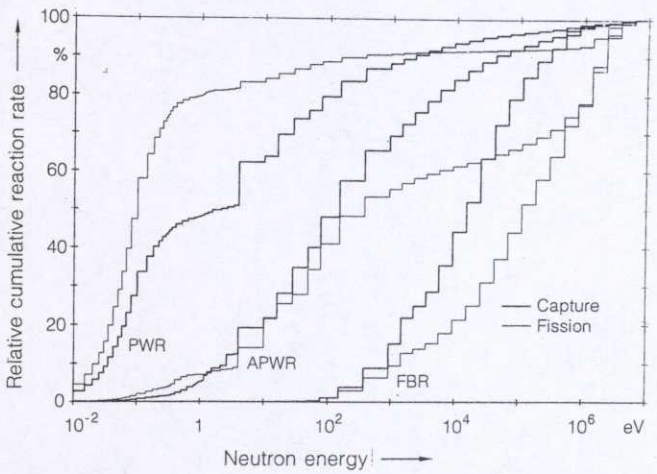

Fig. 1. Relative cumulative reaction rates for the reactor types PWR, APWR and FBR

actual licensing procedures and must be avoided. Unfortunately, first exploratory tight lattice calculations showed a tendency to a positive reactivity effect for a change from normal to zero moderator density (void effect). In consequence, very accurate calculations are required to determine the reactivity effects for all moderator densities from normal to zero. In reactor systems with hard neutron spectrum (FBR and voided APWR systems) the neutron interactions at high energies are dominant and high accuracy for the calculation of these processes is necessary. In order to satisfy these requirements, new calculational procedures had to be developed and validated. The developments will be discussed briefly in the following Sections. More details may be found in Ref. [5].

2.2 *The calculation of infinite lattices*

2.2.1 Calculational procedures

The standard calculational procedures for the investigation of infinite lattices of fast and thermal reactors are usually based on different principles.

Procedures for *thermal reactor calculations* are based on one-dimensional Wigner-Seitz cell calculations with a transport theoretical solution. The neutron processes at energies below approx. 1 keV are described in detail. For the high energy processes, less accurate approximations are applied. As an example, the worldwide used English code WIMS [7] enables a detailed description of the neutron processes in the thermal and resonance energy regions, but cannot process isotope dependent spectra of the fission neutrons. The distribution of the neutrons after elastic or nonelastic scattering processes and after (n,2n) reactions are all processed with the same transfer matrix between energy groups.

Standard procedures for *fast reactor calculations* are usually based on homogenized reactor zones, sometimes with simple approximations for the description of heterogeneity effects in the resonance energy region. In the standard 26-group calculations, the treatment of neutron processes at thermal energies is restricted to one energy group. On the other hand, the important neutron reactions at higher energies may be handled separately. Cell codes are available for more detailed FBR investigations, e.g., the KAPER4 code [8] in the KfK FBR code system KAPROS [9].

For *APWR investigations,* a new procedure was established, combining the advantages of the calculation schemes of thermal and fast reactors. The 69-energy-group structure and the first flight collision probability method for the flux calculation in the Wigner-Seitz cell were adapted from the thermal reactor code WIMS. The principles of the library with nuclear group constants and the code for the calculation of zone-dependent macroscopic cross sections were taken from the KfK FBR methods.

2.2.2 Calculation of resonance self-shielding

In standard calculations, resonance self-shielding is treated with precalculated tabulations of self-shielding factors depending on material dilution and temperatures (f-factor table look-up concept). The determination of the dilution parameter σ_0 for a resonance material is straightforward only when all other materials in the mixture have no resonances in the energy region of interest. In the case of a mixture of isotopes with resonances in the same energy region, approximations for the determination of the dilution parameter are required:

- In the case of high, well separated, widely spaced resonances, use of the potential cross section without resonance contributions, σ_{pot}, may be the best solution in most practical cases. This method is applied to the important reactor material ^{238}U.
- For other materials, the groupwise unshielded total cross section, $\sigma_{t,\infty}$ is used in standard calculations.
- The shielded total cross section may be applied optionally. This solution needs an iterative calculation of the effective shielded total cross section, $\sigma_{t,eff}$.

With the help of equivalence considerations it can be shown that heterogeneities may be taken into account in the f-factor concept by increasing the dilution parameter. In KfK APWR calculations the following formula is applied [5]:

$$\sigma_{0,het}^{k} = \sigma_{0,hom}^{k} + \frac{1}{N^k} \cdot \frac{1 - C}{l_F} \cdot \frac{a}{1 + (a - 1)\,C} \qquad (1)$$

with

$\sigma_{0,het}^{k}$ dilution parameter for material k, heterogeneous case,
$\sigma_{0,hom}^{k}$ dilution parameter for material k, homogeneous case,
N^k number density of atoms of material k,
l_F mean chord length in the fuel lump,
C energy-dependant Dancoff factor,
a Bell-Levine correction factor.

For the Dancoff factor the formula of *Sauer* [10] is used, including cladding effects as proposed by *Williams* and *Gilai* [11]. Most of the KfK APWR investigations were performed with a Bell-Levine factor $a = 1$. In Ref. [5] the following relation between the Bell-Levine factor and the resonance escape probability P from a lumped absorber was derived:

$$a = \frac{P}{1 - P} \cdot \Sigma_t^F l_F \qquad (2)$$

with

Σ_t^F total cross section in the fuel.

In actual calculational procedures for the neutron physics APWR calculations, Eq. (2) is applied using a polynomial expression for P as proposed by *Raghav* [12]. In order to validate the f-factor table look-up method, improved approximations for the calculation of effective resonance group cross sections have been made available. In these procedures fine energy mesh neutron fluxes for the cell are calculated starting from the neutron slowing down equations. The fine energy fluxes obtained are used for direct calculation of the flux-weighted cell-averaged group cross sections. The comparison of results of cell calculations shows that the agreement between the f-factor formalism and the fine-flux method is improved if the new Bell-Levine factor from Eq. (2) is used instead of $a = 1$.

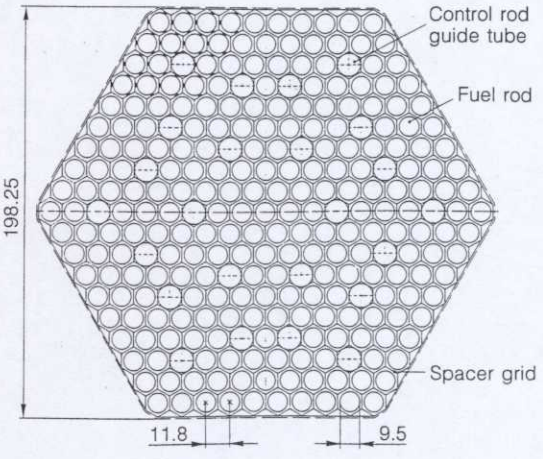

Fig. 2. Cross section of a typical APWR fuel assembly, from [25]. Dimensions in mm

2.2.3 Group constant libraries

Generation of a 69-group library containing data for about 140 reactor materials is a rather laborious task. This work started in 1981, using the standard FBR group constant generation code MIGROS-3 [13] and the Karlsruhe nuclear data file KEDAK-4 [14]. Thermal scattering data were taken from WIMS, fission product data were provided by the research center Petten [15]. The results with this early version of the library were published in 1985 [6]. Validation investigations indicated an underestimation of the reactivity of APWR systems [6]. This library is referred as KARBUS-85. Extensive validation investigations, using both results from the PROTEUS HCLWR experiments [16] and from a NEACRP HCLWR burnup benchmark [17], lead to the detection of some data errors and inconsistencies. The main error was neglection of the self-shielding of the capture resonance at 2.67 eV of ^{242}Pu; the same error occurred in the original WIMS code. The latest library version is referred to as KARBUS-91. Compared to KARBUS-85, this library has the following improvements:

- improved self-shielding table processing over the whole energy region for all materials;
- consistent data for first-order treatment of the anisotropy with transport calculations;
- a number of additional control materials;
- new consistent group constants for the fission product data.

The new materials were calculated with the Karlsruhe version of the NJOY [18] group constant processing code, mainly using nuclear data from the JEF-1.1 [19] library. The weighting spectrum for most of the group constant calculations was the collision density spectrum from a typical APWR system. The consequences of the choice of this weighting spectrum are briefly discussed in the next Section.

2.2.4 Influence of the weighting spectrum

A weighting function is required during generation of multigroup constants for reactor calculations. The choice of this weighting function determines the main application area, e.g., thermal or fast reactors. The influence of the weighting spectrum increases with broader energy groups. Although the scalar group constants, e.g., for capture or fission, also depend on the spectrum, the largest influence of the weighting spectrum on reactor calculations comes from the effects on elastic scattering. These effects were earlier observed for fast reactor calculations. Therefore special improvements for determination of the removal cross sections and fine energy group libraries were introduced. These effects are investigated in some detail by means of a fine group library with 334 energy groups and new approximations for the removal cross sections in Ref. [5]. Starting from a coarse group library with 69 groups, established with the weighting spectrum from a typical APWR system, the effect of the improved void reactivity calculation may be in the order of $\delta\Delta k_{void} = -0.015$, a rather large correction. The results of calculations with improved methods lead to smaller values of the void reactivity; that means a change into the favorable direction. These investigations are in agreement with earlier studies with separate libraries with weighting functions from normal and voided APWR lattices [20].

2.3 Burnup calculations

A new module, the program BURNUP [21], was developed for burnup calculations in a reactor zone. It has the following characteristics:

- The depletion equations are solved with the formalisms of the codes ORIGEN [22] and KORIGEN [23].
- The nuclear data for the depletion is stored on libraries which are part of the code. Cross section and decay data for about 1100 nuclides are available.
- For all nuclides on multigroup libraries, improved one-group cross sections for the depletion calculations may be provided for all important neutron reactions.
- The program BURNUP is an integral part of the FBR code system KAPROS.

The improved one-group cross sections may be obtained by weighting the basic multigroup constants with the best estimate for the actual neutron fluxes, e.g., from cell calculations or from whole-core calculations.

Several procedures are available for automatized application of the module BURNUP. The program KARBUS may perform the following tasks:

- determination of multigroup macroscopic group constants using cell calculations or modified FBR methods for homogenized zones;
- whole-core calculations with one- to three-dimensional diffusion or transport theory (only for selected geometries); these calculations are optional and may be omitted, e.g., for cell burnup investigations;
- preparation of one-group cross sections for burnup calculations;
- burnup calculations with the module BURNUP;
- automatic storage of the results on standard KAPROS-archives.

The procedure KARBUS has been extensively used for LWR and APWR cell burnup calculations.

2.4 Validation of the basic programs

One of the main tasks of the neutron physics APWR investigations at KfK was validation of the calculational procedures. This started with a common benchmark project together with the Technical University of Braunschweig and Siemens [3]. The conclusions from this work initiated a number of further activities, e.g., a NEACRP HCLWR benchmark program [17] and experimental programs in KfK and in Würenlingen. During development of the 69-group procedures the participation in NEACRP benchmarks and the analysis of the PROTEUS experimental data resulted in a number of improvements of data and methods, see also Section 2.2.3. The actual state and the validation of the calculational procedures for infinite lat-

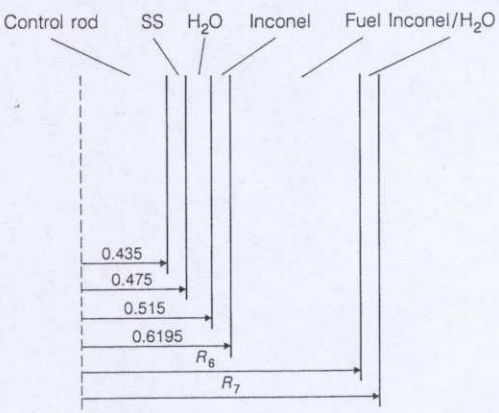

Fig. 3. One-dimensional 6-zone supercell model of an APWR fuel assembly with control rods. 18 rods with R_6 = 2.4038 cm, R_7 = 2.4533 cm; 24 rods with R_6 = 2.0817 cm, R_7 = 2.1246 cm; 30 rods with R_6 = 1.8619 cm, R_7 = 1.9002 cm. The radii (in cm) are determined from the volume partition of the material zones in the fuel assembly

tices and for cell burnup is described in more detail in Ref. [5]. It can be concluded that the available procedures are well suited for exploratory investigations of APWR systems.

2.5 *Fuel assembly calculations*

In the first APWR whole core calculations the influence of the fuel assembly (FA) structures was taken into account only by rather crude approximations. Several methods of incorporating the FA structures in the Wigner-Seitz cell were investigated [4]. In the early KfK APWR whole core calculations these effects were treated by changes of the clad material density and the coolant density in the Wigner-Seitz cell [24]. For more recent whole core calculations a more refined treatment has been applied. The main reason for the refinement is the complicated design of the reactor control system. A cross section of the actual KfK design for an APWR core is shown in Fig. 2. The design is based on two control mechanisms, viz., boric acid in the coolant and single control rods in the FA. The control rod material is boron carbide with a ^{10}B enrichment to approximately 60 %. In the case of withdrawn control rods, the rod positions are filled with coolant water. Such a strongly heterogeneous FA needs a more detailed treatment. A good solution would be a two-dimensional transport theory calculation with explicitly treated control rod positions. Starting from a 69-group constant library, no computer programs at KfK enable this solution at present. Therefore one-dimensional supercell transport calculations have been applied. The six-zone supercell model with equivalent volume arrangement of control rod tube guide, fuel and structural materials around one control rod position filled with control rod material or water is shown in Fig. 3. Application of this one-dimensional model is justified by Japanese validation investigations for this type of FA approximation [26].

2.6 *Whole core calculations*

Only the design investigations for the so-called "homogeneous design" with the same lattice in all FAs will be considered in this paper. The first exploratory whole core calculations were performed with strongly simplified core models. In most cases, one- and two-dimensional *R,Z*-geometries were applied. The burnup calculations were restricted to first core investigations with FAs with different fissile enrichments, needed to obtain a flat radial power distribution in the core. The first detailed multi-cycle burnup calculations, using the KfK calculational procedures, were aimed at an equilibrium core for a homogeneous APWR design [4]. In multi-cycle investigations, burnup calculations are more complicated compared to first core analysis, because optimized fuel reloading must be considered. Especially, the search for an acceptable fuel shuffling scheme can be a laborious task. It may be advantageous to use precalculated burnup data, instead of repeating calculations of similar depletion behaviour. The investigations described in Ref. [4] were performed with precalculated macroscopic data from cell burnup calculations. Each FA in the hexagonal reactor model was specified by a unique material specification. The three-dimensional diffusion code D3E [27] and the corresponding evaluation program AUDI3 [28] were used for the reactor calculations. The accumulated mean burnup in each FA was based on the AUDI3 results. Using the most reasonable tabulated burnup values from the precalculated data for the next cycle, a systematic approach to an equilibrium core, including fuel element shuffling, could be simulated.

The use of precalculated macroscopic burnup data enables improvements of the cell burnup results, e.g., with transport

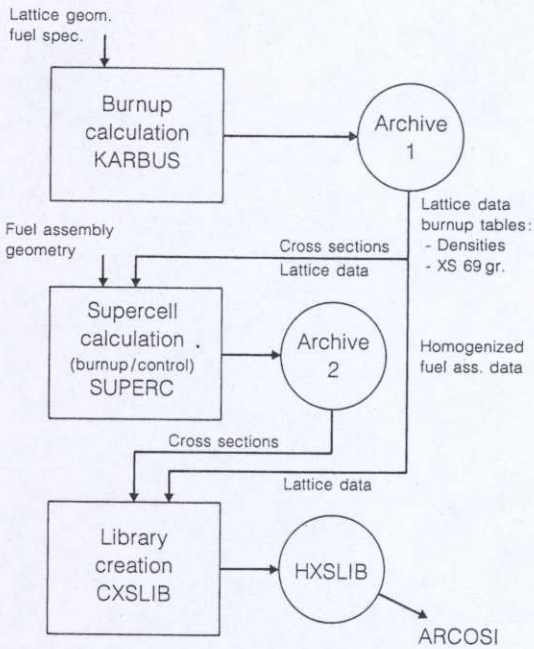

Fig. 4. Flowchart of the generation of a fuel assembly library

theory approximations for the description of FA control rods and structures, as described in Section 2.5.

The approach for burnup investigations with precalculated *macroscopic* data is different from the KWU methods used at present for PWR design [29] and recently developed for APWR studies [30]. These codes are based on precalculated *microscopic* cross section tables with a variety of reactor parameters like temperatures, boron concentrations, burnup, etc. In the KWU procedure, up to 10 energy groups may be used in the tabulations. During burnup calculations, the number densities of the atoms of selected fuel and fission product nuclides are determined at every time step and in all reactor zones in order to calculate the required macroscopic cross sections. This method is more flexible for taking into account history effects, like neutron spectrum changes or reactor shutdown times. On the other hand, the method using macroscopic data tabulations is faster in producing the required macroscopic data for reactor calculations. Moreover, the treatment of the resonance self-shielding with the full number of energy groups from the basic library and with all important nuclides could be more accurate.

Based on the experience from the early investigations [4], a new program for the simulation of a typical APWR core, based on macroscopic cross section tabulations, has been developed. It is described in the next Section.

2.7 *The program ARCOSI*

The program ARCOSI (Advanced Reactor Core Simulator) has been developed with the following objectives:

- performance of automatized multi-cycle calculations for advanced reactor cores with hexagonal fuel assemblies;
- capability of performing search to criticality by changes of boric acid concentration in the coolant;
- capability of calculating reactivity coefficients, like Doppler and moderator density coefficients, for all core situations;
- extensive evaluation aids to judge the results;
- acceptable demands on computer resources.

These requirements could be fulfilled with a new calculational procedure within the KAPROS code system with the following characteristics:

- implementation of the three-dimensional nodal flux calculation code in hexagonal geometries HEXNOD [30], the same code as in the KWU APWR investigations;
- introduction of a new data library with macroscopic multigroup FA cross section data and of appropriate storage and interpolation schemes.

The first version of this library contains four group data for the FAs of the equilibrium core of the KfK APWR reference design in [25]. The characteristics of this so-called HXSLIB library are described in the next Section.

3 Generation of a library with fuel assembly data

3.1 System variables

The data library for the program ARCOSI may be considered as a pool of data for irradiated FAs with precalculated data for a number of system variables. At present the system variables are the following:

- Geometry of the basic lattice. Until now data for one lattice is available.
- Enrichment of the fuel. Two enrichments are available: 8 and 8.5 % of fissile Pu.
- Temperatures in the lattice zones. Besides the temperatures at nominal conditions, the fuel temperatures 300 K and 2100 K are available for Doppler coefficient studies.
- Coolant density. For investigation of moderator density reactivity effects, 9 ratios to the nominal value are available: 0.01, 0.1, 0.3, 0.5, 0.7, 0.8, 0.9, 1.0 and 1.1.
- Content of ^{10}B in the coolant. Five concentrations are available: 0, 500, 1000, 2000 and 4000 ppm.
- Number and state of the control rod positions within the FA, see also Section 2.5. Data is available for FAs with 24 control rod positions. These positions are filled with B_4C (60 % ^{10}B) or with the coolant.

A flowchart of the calculational scheme for the generation of this library is presented in Fig. 4. The main components are described in the next Sections.

3.2 Cell burnup calculations

The first step towards an equilibrium core is the determination of the specification of the fuel to be used for the reload FAs. It is planned to use reprocessed PWR plutonium with about 10 years ex-core time and an estimated isotopic composition of 1.94 % ^{238}Pu, 56.63 % ^{239}Pu, 25.24 % ^{240}Pu, 9.87 % ^{241}Pu, and 6.32 % ^{242}Pu. Compared with other APWR investigations, this plutonium contains a relative small amount of the very favorable ^{241}Pu, with a high ratio of fission to absorption cross section (η) but with a half-life of about 14 years. Exploratory investigations resulted into a reload enrichment of 8 % of fissile Pu, i. e., the same value as in the KWU studies. The influence of boron in the coolant on the cell burnup was investigated in some detail. It turned out that moderate boron concentrations have no significant impact on the burnup. For this reason the cell burnup calculations were performed without boron in the coolant. The boron concentration was taken into account in all supercell calculations.

3.3 Supercell calculations

As pointed out in Section 2.5, a simplified one-dimensional geometry model is applied for the supercell FA calculations. These calculations were performed with the KAPROS version of the one-dimensional S_N transport theory program ONETRAN [31, 32], using a special-purpose KAPROS procedure SUPERC for the systematic parameter variations to be prepared for the library. Group collapsing to the final HXSLIB coarse group number is done after the 69-group supercell calculations. All intermediate results are stored on standard KAPROS archives.

3.4 Generation of the HXSLIB library

For the storage of FA data for multi-cycle reactor calculations, a new library was established. It comprises the required group constants for application in the three-dimensional flux calculation codes D3E and HEXNOD. It has a simple sequential structure with systematic variations of the parameters specified in Section 3.1. Starting from the results of the FA or cell burnup calculations stored on the standard KAPROS archive files, the program CXSLIB may be used to generate the HXSLIB library.

4 Determination of an equilibrium core

4.1 Design conditions

The present APWR core design investigations are aimed at an acceptable equilibrium core configuration. At the end of a reactor cycle, after a given number of full power days (FPDs), a certain amount of FAs has to be replaced by fresh ones with the same reload enrichment. The available FAs at that time have to be arranged in a way that the same reactor behaviour can be obtained in succeeding reactor cycles. If such a configuration can be found, it will be possible to design the pre-equilibrium cores, possibly with refined axial and radial enrichment differences in the FAs.

The approach to an equilibrium core is strongly influenced by a number of conditions coming from various origins. The present KfK design for an equilibrium APWR core was developed with the following conditions:

- Triangular lattice with moderate tightness. The fuel pin diameter is 9.5 mm, and the lattice pitch, 11.8 mm (p/d = 1.242). This lattice was chosen in order to avoid problems with the moderator density reactivity coefficient. For the same reason the fuel enrichment should not be too high.

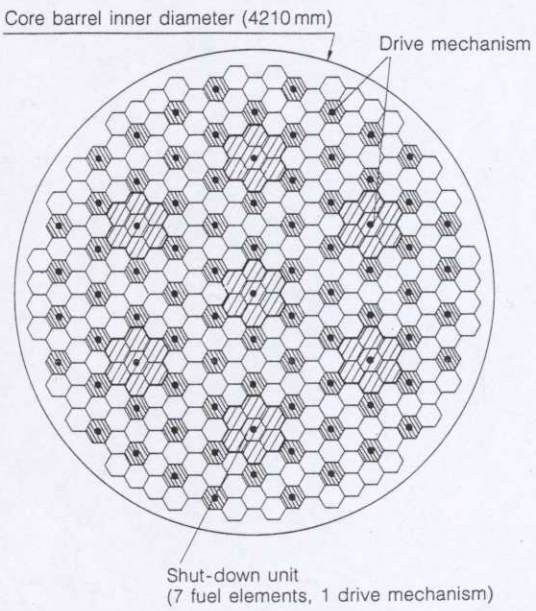

Fig. 5. Core cross section of KfK APWR reference design

- The power rating should be approximately 125 W/cm, being considerably lower compared to the ratings in existing PWRs and in previous APWR designs as, e.g., in Ref. [24].
- The discharge burnup should be equal to, or higher than, the claimed extended burnup in existing PWRs, i.e., 50000 MWd/t.
- The period between fuel reloadings (cycle time) should be approximately one year, as recommended in Germany for practical reasons [33]. This means a cycle time of 300 to 320 FPDs.

A number of exploratory investigations resulted in a useable solution for an equilibrium core, matching the conditions mentioned before.

4.2 Basic core layout

The basic core layout was determined in a close cooperation between KfK and KWU. The core cross section from the KfK design [25] is reproduced in Fig. 5. It is compatible with the KWU design for an APWR core, including the control and shut-down system.

4.3 Fuel assembly shuffling scheme

A satisfactory shuffling scheme could be obtained by laborious two-dimensional calculations. The principles of this scheme are the following.

- The core loading is subdivided into six batches. After each reactor cycle the "oldest" of these batches is removed from the reactor and replaced by fresh FAs with 8 % of fissile Pu. The batches are numbered from 1 to 6, starting with 1 for the fresh assemblies and going up to 6 for the "oldest" ones.
- Whereas the three freshest batches are arranged at the boundary of the core in an out-in shuffling scheme, the three "oldest" ones are placed in an in-out order from the core center. In this way the order of the batches from the core center to the boundary is: 4-5-6-3-2-1. The central FA is taken from batch 5.
- A fixed order of the FA arrangement with increasing burnup of the available assemblies is used within every batch. This order is determined by means of a trial-and-error procedure until a satisfactory power distribution over the whole reactor cycle is obtained.

4.4 Approach to equilibrium core

The final investigations for an APWR equilibrium core were performed with the program ARCOSI with a three-dimensional core model. The latter is based on a 30° section of the core with 16 axial planes. 38 FAs are explicitly represented in this section. In all 608 FA zones of the model, detailed burnup accumulation is determined on the basis of the power distribution from whole-core calculations with the KAPROS version of HEXNOD (HEXNODK) with 608 zones and 4 energy groups. The burnup-dependent cross sections of these zones are obtained by fast interpolations from the data on the HXSLIB library. In the reactor calculations, the number of full power days per cycle is specified by input. The cycle may be subdivided into an arbitrary number of micro-time-steps. At each micro-time-step, a search to criticality by means of boron concentration changes may be performed. An arbitrary number of cycles may be calculated in one computer run. Between two cycles, FA shuffling may be done, using the same shuffling scheme, specified by input. The calculations start with an estimation of the three-dimensional burnup distribution in the core, based on the radial distribution of the two-dimensional pre-calculations and a cosine shape for the axial distribution.

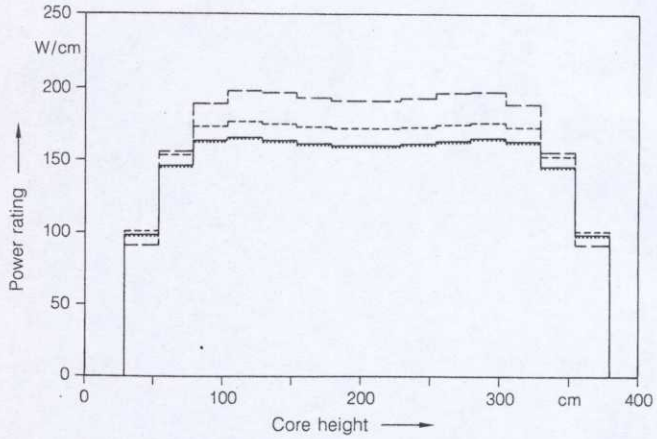

Fig. 6. Axial power rating of a selected fuel assembly near the core centre for different iteration cycles to equilibrium. 320 full power days. --- cycle 1, k_{eff} = 0.9991; ----- cycle 4, k_{eff} = 0.9996; ——— cycle 6, k_{eff} = 1.0006; ····· cycle 7, k_{eff} = 1.0003

4.5 Characteristics of the equilibrium core

The single control rods in selected FAs are only used for shut-down purposes. During normal reactor operation, all control rods are withdrawn and reactor control is performed by the boric acid in the coolant. After some exploratory three-dimensional calculations determining the possible number of full power days, the equilibrium core could be found after seven cycle simulations in a final computer run. Three micro-time-steps with search to critical boron concentration were applied for the 320 FPD period. The changes between cycle 6 and 7 in reactivity, power distribution and burnup distribution were small. The iterative convergence is demonstrated in Fig. 6 for the power rating in a near the core centre. Figs. 7 and 8 show results from the plot-file from the 7th cycle, a good approximation for the equilibrium core. In Fig. 7 the axial burnup distribution is plotted for selected FAs from each of the six batches. The maximum burnup at discharge time is close to 60000 MWd/t. Fig. 8 shows how the axial power shape is flattened for a fresh FA in its first cycle. The axial form-factor decreases from 1.29 to 1.15.

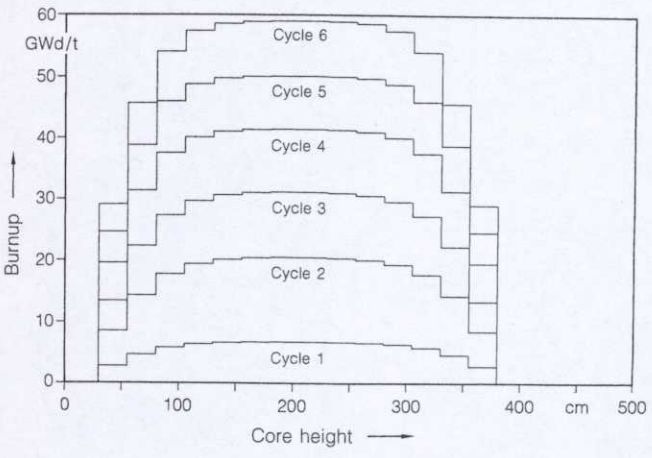

Fig. 7. Example of burnup build-up in the six fuel charges

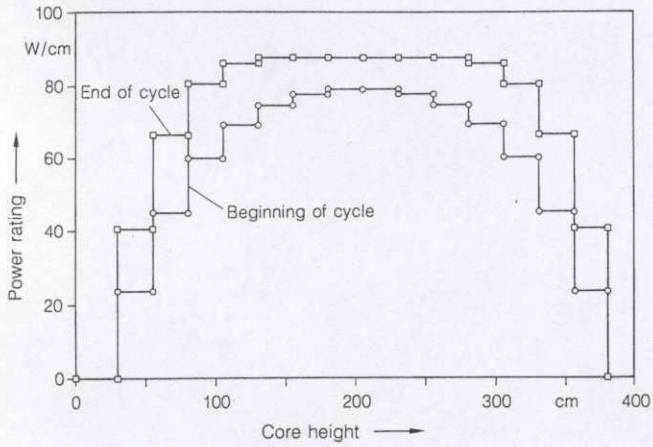

Fig. 8. Example of axial power flattening for fresh fuel assemblies

4.6 Reactivity coefficients

With the help of restart options of the procedure ARCOSI, a three-dimensional whole-core calculation can be performed for any arbitrary burnup distribution in the core and for any boron content in the coolant. With the help of changes of the coolant density, including the corresponding change of the boron content, and of changes of the fuel temperatures, reactivity coefficients can be calculated. More detailed results are given in Ref. [25]. The very important moderator density reactivity curves are reproduced in Fig. 9. For four values of the number of FPDs from cycle beginning the critical ^{10}B concentrations were searched, decreasing from 1799 ppm at beginning of cycle to 18 ppm after 320 FPDs. k_{eff} values for the different coolant densities were determined. It can be concluded that no severe problems are to be expected in the case of moderator density changes if the core of a PWR is replaced by the proposed APWR design.

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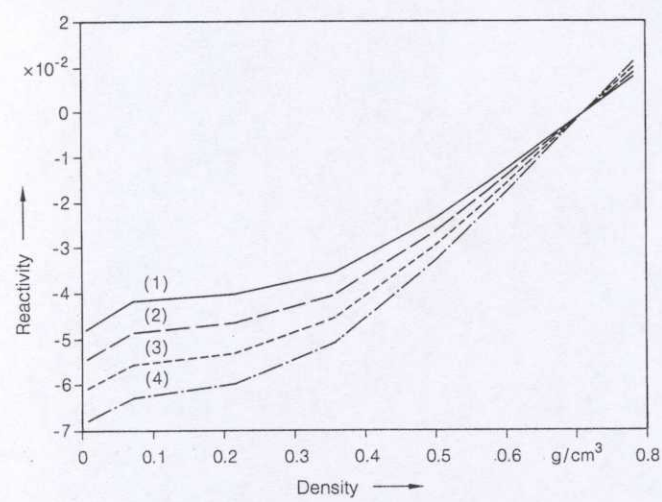

Fig. 9. Reactivity change to normal condition as a function of the coolant density. Equilibrium cycle, pitch-to-diameter ratio 1.24. (1) 0 FPD, 1799 ppm ^{10}B; (2) 106.5 FPDs, 1082 ppm ^{10}B; (3) 213 FPDs, 508 ppm ^{10}B; (4) 320 FPDs, 18 ppm ^{10}B

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Books · Bücher

Chernobyl: A Policy Response Study. Herausgegeben von *B. Segerstahl,* Springer Verlag, Berlin 1991. 180 Seiten, gebunden, DM 78,–.

Im Vorwort umreißt der Heruasgeber kurz die Vorgeschichte dieses Buches. Es entstand aus den Diskussionen während mehrmaliger Treffen einer Gruppe von Wissenschaftlern, die sich vor allem mit den nichttechnischen Auswirkungen des Tschernobyl-Unfalles befaßten. Spezielle Punkte waren u.a.:

- Wie reagieren Behörden?
- Was ist die Rolle und wie ist das Verhalten der Medien?
- Wie sind die Entscheidungsstrukturen?
- Auf welche Art und Weise arbeiten internationale Koordinationssysteme?

Das Buch gliedert sich in die folgenden Kapitel.

1. Introduction (*B. Segerstahl,* Oulu). Die Einleitung enthält nach der Aufzählung einer Reihe anderer Unfälle (Windscale, Three Mile Island u.a.) die ausführliche Beschreibung des Unfallvorganges im Kernkraftwerk Tschernobyl.

2. Monitoring and Assessment (*F. Schönhofer,* Wien). Es wird die Ermittlung der Radioaktivität und deren Bewertung nach dem Unfall beschrieben; der Schwerpunkt des Berichtes liegt hierbei auf den Meßprogrammen der Länder Finnland, Schweden, Ungarn, Schweiz und Österreich.

3. Health Effects (*L. Sztanyik,* Budapest). Nach der Beschreibung der durch erhöhte Strahlenexposition möglicherweise auftretenden gesundheitlichen Wirkungen werden die potentiellen gesundheitlichen Langzeiteffekte abgeschätzt; dies geschieht auf der Basis der für die europäische Bevölkerung ermittelten Kollektivdosis.

4. Agriculture and Trade (*P.S. Gray,* Brüssel). Es wird das Verhalten der Regierungen Europas in Hinblick auf die Limitierung der Kontamination von Lebensmitteln beschrieben sowie der Einfluß dieser Entscheidungen auf Landwirtschaft und Handel dargestellt. Die Kosten, die durch die Begrenzung verursacht wurden, werden abgeschätzt.

5. The International Response (*J. Linnerooth-Bayer,* Laxenburg). Es wird festgestellt, daß dieser Unfall bewußt machte, daß kerntechnische Sicherheit ein internationales Anliegen ist und deshalb auch internationale Antworten verlangt. Die Reaktionen der Internationalen Organisationen IAEA, WHO, NEA und EG werden beschrieben.

6. Perception of a Secondhand Reality (*B. de March* and *W. Tessarin,* Gorizia). Der Unfall wird als 'secondhand reality' beschrieben, eine Realität also, die im wesentlichen von den Massenmedien ausgeformt wurde. Die Hauptpunkte der öffentlichen Beunruhigung, die während des Unfalls ermittelt wurden, bezogen sich auf die Entscheidungsprozeduren, die Kompetenz und Verläßlichkeit der Entscheidungsträger und schließlich die Glaubwürdigkeit der demokratischen Institutionen.

7. The Media and Crisis Management (*H. Otway,* Ispra). Es wird untersucht, wie in sieben Ländern Europas (Dänemark, Deutschland, Frankreich, Griechenland, Großbritannien, Italien, Österreich) die Information von den Medien weitervermittelt wurde, und zwar sowohl die kerntechnische Information als auch die Information über potentielle gesundheitliche Auswirkungen. Es wird eine Liste von Empfehlungen herausgearbeitet, die eine Verbesserung der Kommunikation und des Krisen-Managements bewirken könnten.

8. The Credibility Crisis (*M. Poumadère,* Cachan). Der Unfall wird unter dem Aspekt, Ursache einer Glaubwürdigkeitskrise zu sein, untersucht. Es werden die Parameter, die das Ausmaß derartiger Krisen beeinflussen, identifiziert und am Beispiel des Unfallverlaufs ausführlich dargestellt.

Trotz der Heterogenität der Darstellung, die nun einmal einem Werk mit mehreren Autoren innewohnt, stellt das Buch eine sehr gute und übersichtliche Zusammenstellung der Beobachtungen, Überlegungen und Analysen, die nach dem Unfall im Kernkraftwerk Tschernobyl – aber auch schon teilweise nach dem Unfall im Kernkraftwerk Three Mile Island – angestellt wurden, dar. Die jeweils etwa zwanzigseitigen Einzeldarstellungen geben eine kompakte Einführung in die Problematik. Das Buch gibt dem Leser, der fachlich meist nur mit einigen wenigen Aspekten vertraut ist, einen guten Überblick über das gesamte Spektrum der nichttechnischen Auswirkungen des Unfalls im Kernkraftwerk Tschernobyl.

A. Bayer, Neuherberg