EXPERIMENTAL STUDIES OF MACROSCOPIC AND CELL HETEROGENEITIES IN LWHCR'S

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ABSTRACT

In a light water high conversion reactor (LWHCR) lattice, we investigated the influence of macroscopic and cell heterogeneities on reaction rate distributions and reaction rates, respectively. The experimental results are compared with calculations.
Reaction rate distributions were measured in the vicinity of a control rod and a water hole, corresponding to an inserted and withdrawn control rod. Self-shielding effects of various reaction rates were measured across the axial core/blanket interface. To assess shielding effects of fuel constituents in the fuel rods, important reaction rates were experimentally determined separately in the moderator and in the fuel.

INTRODUCTION

The basic idea of the light water high conversion reactor (LWHCR) is to increase the conversion ratio by hardening the neutron spectrum with an undermoderated tight-pitch lattice. In this intermediate neutron spectrum the main absorption reactions occur in the resonance energy region. The microscopic cross sections in this energy range are not so well known as those in the thermal range, resulting in much less accurate calculations of integral neutronic parameters. In the PROTEUS zero-power test facility at Würenlingen, Switzerland, integral neutron physics experiments in LWHCR test lattices with U/Pu mixed-oxide (MOX) fuel of 7.5% fissile plutonium have been carried out since 1981. The current hexagonal test lattice has a pitch of 10.7 mm and a fuel-to-moderator volumetric ratio of 2.07. The central test zone of 50 cm diameter is driven critical by thermal driver regions. The test zone region containing the MOX fuel is 84 cm high. Axial blankets of depleted UO_2 are located above and below the MOX fuel zone. The regular lattice geometry of the LWHCR MOX fuel zone is carried through into the 28 cm thick axial blankets. The plutonium isotopic composition is approximately 1% ^{238}Pu, 64% ^{239}Pu, 23% ^{240}Pu, 8% ^{241}Pu, and 4% ^{242}Pu. Details of the reactor and the test zone configuration are given in Ref. 1.

Reactivity control in an LWHCR will be based on the use of control rod clusters for which the physics design has considerable uncertainties. A comparison of calculational and experimental results for reaction rate distributions in the vicinity of such heterogeneities gives valuable information on the calculational methods and nuclear data tested. In order to assess the degree and propagation distance of flux depression and peaking caused by inserted and withdrawn control rods, we have measured radial reaction rate distributions near a control

rod and a water hole introduced in the center of the PROTEUS test zone. The calculations of reaction rate distributions near the fuel/blanket interface are difficult because of the transitional nature of the neutron spectrum in this region. Although the LWHCR-experiment in PROTEUS was not specially designed for blanket studies, we tried nevertheless to get some information by measuring axial reaction rate traverses across the upper MOX/blanket interface.

The measurements of core-center reaction rates provide a check on individual components of the neutron balance in the test lattice and its influence on the void coefficient of reactivity. For the measurement and interpretation of these reaction rates, self-shielding effects play an important role. The determination of moderator-to-fuel reaction rate ratios allows one to assess these effects. The spatial variation of self-shielding effects is of great importance for all kinds of lattice heterogeneities. Especially if control rods consist of resonance materials, the modelling of the self-shielding has to be done very carefully. Insufficient treatment of the interaction between the resonances in the fuel and in the control rods cause great uncertainties in the calculation of the rod worths.

EXPERIMENTAL AND CALCULATIONAL METHODS

The microscopic reaction rates which will be mentioned in the following discussions are fissions in ^{235}U (F5), ^{238}U (F8), ^{239}Pu (F9), and captures in ^{238}U (C8) and ^{242}Pu (C2). They are measured by foil activation and standard gamma ray counting methods. These include the counting of 278-keV and 84-keV gamma rays from activated ^{238}U and ^{242}Pu, respectively, and the counting of fission product gamma rays having energies above 600 keV for the fission rates.[2] The counting laboratory is equipped with two (HP)Ge detectors and two NaI(Tl) detectors in face-to-face arrangements, attached to automatic sample changers and controlled by a system of minicomputers. High quality measurements require the foils or samples to be activated between fuel pellets. The misalignments of the foil-pellet arrangements have to be kept as small as possible but cannot be suppressed completely. Any discontinuance causes a perturbation of the foil activation. The common method to investigate foil perturbations is to extrapolate from measurements with stacks of two or three foils to zero foil thickness. This procedure, however, can be misleading if resonance reactions with high peak cross sections are involved. Therefore, an alternative way to correct experimental foil results is presented in the last section.

To make calculational results available for comparison with the measurements two different codes have been used. These are the U.K. lattice code WIMS-D [3] with its 1981 data library [4] and the code BOXER of the Swiss code system ELCOS [5], which was used to calculate reaction rate heterogeneities in the cell.

In WIMS-D the fuel zone cross sections in the resonance region from 4.0 to 9118.0 eV are shielded by table lookup methods with the background cross section being determined by applying equivalence theorems. Reaction rates in the non-fuel zones of the cell are calculated using unshielded cross sections. Thus, WIMS-D is unsuitable for the calculation of moderator-to-fuel reaction rate ratios. Whole reactor calculations necessary for the analysis of some of the measurements described in the introduction were made on the basis of 28-group cross sections for the individual reactor zones generated using WIMS-D. For each fuel containing region of the PROTEUS reactor the library cross sections were condensed to 28 energy groups and homogenized over the cell using the fine group neutron flux of an infinite lattice.

The ELCOS-code ETOBOX generates, starting from a basic neutron data library (JEF-1 for fissionable nuclides and ^{1}H, ENDF-B/IV for all other nuclides) a cross section library in 70 energy groups (the 69 groups of the WIMS library [4] with an added group from 10 to 15 MeV)

for smoothly varying cross sections. In the resonance region (between 1.3 and 907 eV) pointwise cross sections are produced for different temperatures. In the range above 907 eV condensed cross sections are tabulated as functions of temperature and dilution cross section. The flux in the resonance region is explicitly determined in two zones using about 8000 lethargy points for the fuel considered. This pointwise flux is then used for condensation into appropriate groups of the 70-group scheme. Interactions between resonances of any particular nuclide as well as of different nuclides are thus automatically taken into account.

ABSORBER ROD HETEROGENEITY

In the H_2O-moderated reference lattice (core 7) we examined the perturbation of the reaction rate distribution caused by introducing a central heterogeneity in the form of either an absorber rod (B_4C-pellets with a diameter of 8.3 mm and 93% ^{10}B-enrichment were canned in steel tubes of 9.60 mm outer diameter) or a water hole (created by removing the central MOX rod). In comparison to other absorber materials, B_4C is found to be more effective in the intermediate spectrum of an LWHCR. The reaction rate distributions measured were C8, F5, F8 and F9.

TABLE I

Ratio of Reaction Rates (Disturbed/Undisturbed) Near a Highly Enriched Boron Carbide Rod and a Water Hole. The Ratios are Normalized to an F9 Ratio of 1.0 at a Radius of 18.19 cm.

		Experiment[a]				Calculation[b]/Experiment			
	Radius (cm)	F9	F5	C8	F8	F9	F5	C8	F8
B_4C-Rod	1.07	0.815	0.814	0.838	0.937	0.989	0.982	0.978	0.974
	2.14	0.906	0.898	0.916	0.957	0.974	0.988	0.981	0.976
	3.21	0.944	0.934	0.949	0.967	0.973	0.991	0.981	0.979
	18.19	1.000	1.010	1.015	1.020	1.000	0.991	0.985	0.979
Water Hole	1.07	1.327	1.184	1.093	1.027	0.954	0.977	0.967	0.988
	2.14	1.057	1.042	1.012	1.024	0.988	0.989	1.004	0.989
	3.21	1.025	1.012	0.999	1.017	0.990	0.998	1.007	0.992
	18.19	1.000	1.005	0.995	1.010	1.000	0.995	1.005	0.990

[a]Experimental errors: $\pm 1.5\%$ (1σ)
[b]WIMS-D based

Table I gives experimental and calculated values of the ratio of perturbed to unperturbed reaction rates near the highly enriched B_4C-rod and the water hole, respectively. These ratios

were normalized to an F9 ratio of 1.0 at a radius of 18.19 cm for each individual heterogeneity. At this radial distance from the core center in the outer part of the test zone the reaction rates were nearly undisturbed by the central water hole, whereas a perturbation of about 3% remained in the case of the central absorber rod. Whole reactor calculations were made using an one dimensional transport theory program. Axial leakage was taken into account by a zone dependent leakage correction of the absorption cross sections. The WIMS-D multicell modell was used to produce homogenized cross sections of the absorber cell (B_4C-pellets, can and appropriate amount of moderator). Bearing in mind the experimental errors, the B_4C-rod heterogeneity is seen to be satisfactorily predicted by the calculations (Table I). Near the water hole F9, the main contribution to the power production, is underpredicted by about 5%. On the other hand, F5, C8 and F8 are underpredicted for both the B_4C-rod and the water hole by 1% – 2%. So the combined effect of withdrawing a control rod is well calculated for these reaction rates. A comparison of calculated values in the multizone PROTEUS reactor with corresponding calculations in a single zone reactor shows deviations of less than 0.7% for the B_4C-rod and even much smaller differences for the water hole, indicating the LWHCR-specific nature of the effects measured.

CORE/BLANKET HETEROGENEITY

Axial blanket regions of depleted UO_2 are located at the top and bottom of the central test zone. The MOX/blanket interface separates regions of different compositions and with axial changing neutron spectra. Current-day LWHCR designs have no axial blanket zone, but reaction rate trajectories give nevertheless valuable information on the predictability of resonance self-shielding effects in fuel constituents in regions adjacent to the MOX/blanket boundary. Reaction rate traverses were measured across the interface with foils activated between pellets in demountable fuel rods. Transport theory calculations were done using a very simple one dimensional slab-model with the same axial dimensions as the PROTEUS test zone. Criticality was achieved by adjusting the zone independent radial buckling. Though this model is rather crude several observations can be made from the comparison of the experimental and calculational values. Table II shows two typical reaction rate distributions through the MOX/blanket interface: C8, which sees similar ^{238}U nuclide concentrations in the two regions and F9, which is influenced by the large change of the ^{239}Pu density at the interface. The apparent global underprediction of all reaction rates may be explained by the fact that the axial buckling of the neutron flux in the MOX-zone given by the slab model is about 9.3 m^{-2} compared to a value of 8.2 m^{-2} measured in LWHCR-PROTEUS. Both reaction rates, C8 and F9, indicate an "increasing" underprediction of the low energy flux with increasing penetration depth in the blanket. It is possible that this effect reflects the neglection of the influence of the outer zones of PROTEUS (e.g. thermal driver zones) in our model. We observe further, that near the MOX/blanket boundary the C/E-values of F9 show large local differences while C8 varies smoothly. Possibly this discrepancies may be due to the calculation of resonance shielding. The neutron cross sections of the two regions were calculated with WIMS-D for independent infinite lattices.

REACTION RATE HETEROGENEITIES IN THE CELL

The calculation, as well as the measurement by foil activation of many reaction rates in the fuel of tightly packed, light water moderated lattices is affected by self and mutual shielding

TABLE II

Reaction Rate Distributions Through the Axial MOX/Blanket Interface and Comparison with Calculations. The Height is the Distance from the Interface. Experimental and Calculated Values are Normalized to Unity in the Core Center (-420 mm).

		Exp.[a]		Calc.[b]/Exp.	
	Height (mm)	F9	C8	F9	C8
	-420	1.000	1.000	1.000	1.000
MOX	-15	0.441	0.386	0.963	0.878
	-9	0.504	0.378	0.878	0.874
	9	2.236	0.397	0.894	0.849
	18	2.743	0.393	0.880	0.854
	25	2.938	0.387	0.884	0.845
Blanket	35	3.180	0.369	0.856	0.839
	41	3.180	0.358	0.860	0.830
	51	3.209	0.334	0.837	0.823
	58	3.087	0.316	0.846	0.817

[a]Experimental errors: ±2% (1σ)
[b]WIMS-D based

effects. As an extension of our routine experimental program, we measured the capture rates C8 and C2 and the fission rates F9 and F5 in the moderator, relative to the same reaction rates in the fuel. The investigations serve to improve the reliability of experimental results and to test calculational methods and data.

C8 and F9 dominate the conversion ratio and strongly contribute to the multiplication factor, and C2 is important for the build-up of higher actinides. The influence of these reactions on the void coefficient of reactivity is of particular interest. C8/F9 is larger in the hard spectrum of a voided lattice than at normal operational conditions. On the other hand, C2/F9 decreases with voidage. Thus, C8/F9 contributes to the void coefficient in the negative direction, while C2/F9 has a positive influence, in contrast to the fact that the cross sections of ^{238}U and ^{242}Pu exhibit similar structures over most of the energy scale. This different behaviour is explained, at least partially, by self-shielding. The resonance capture of ^{238}U is reduced at normal operational states by self-shielding which is expected to be larger than for the more dilute ^{242}Pu.

The experimental approach to investigate the shielding was to measure the reaction rates in the moderator, relative to the fuel-averaged reaction rates. These ratios are comparable with the thermal disadvantage factor. In the central region of the test lattice a fuel rod was replaced by a thin-walled stainless steel tube having an outer diameter of 11.23 mm which is equal to the diameter of the equivalent cylindrical cell. The spectrum and the magnitude of the neutron flux inside this tube closely correspond to those which exist at the boundary of an unperturbed

TABLE III

Nominal Properties of the Foils Used for the Determination of Moderator-to-Fuel Reaction Rate Ratios, and Experimental Results for the Excess Activation.

Reaction	Material	Composition	Thickness		Excess Activation	
			Total (mm)	Heavy Metal (mg/cm^2)	Eq. (1)[a] (%)	Exp.[b] (%)
C8	^{238}U 0.25% depl.	Oxide[c]	0.5	450	0.4 ±0.1	0.9 ±0.5[e]
	^{238}U 0.04% depl.	Deposit on[d] aluminum	0.1	1		
C2	^{242}Pu 99.9%	Deposit on aluminum	0.1	0.2	1.6 ±0.3	0.6 ±0.7
F9	^{239}Pu 97.3%	Pu-Al-alloy Ni-plated	0.08	2	1.9 ±0.4	1.4 ±0.4
F5	^{235}U 93%	U-Al-alloy Ni-plated	0.08	4	0.6 ±0.1	0.3 ±0.3

[a] Measured by applying Eq. (1)
[b] Measured by extrapolation from stacks of foils
[c] Foils inserted between fuel pellets only
[d] Foils mounted within the cell tube only
[e] Result produced by extrapolation from stacks of 0.02-mm-thick Ni foils forming the gap

cell. A small flux depression of -(3±1)%, probably due to streaming losses through the air-filled tube and to absorption in the steel, was determined by means of thin copper wires which were activated inside the tube and in the adjacent moderator. Thin samples or foils containing the appropriate nuclide in sufficient quantity were then inserted between the fuel pellets of a demountable fuel rod and into the cell tube. Table IV shows the moderator-to-fuel reaction rate ratios obtained by this technique.

Nominal properties of the foils are displayed in Table III. The probes with ^{239}Pu, ^{242}Pu, and ^{235}U contain additional material such as aluminum or nickel. They are neutron-optically less dense than the adjacent fuel, opening a gap between two pellets. Thus, the probes are partially exposed to an unshielded flux giving raise to stronger activation. The amount of excess activation is estimated according to a crude model including a solid angle integration and neglecting scattering and absorption in the sample:

$$C = 3\frac{t}{d}(R-1) \qquad (1)$$

C is the excess activation relative to the unperturbed reaction. t and d are, respectively, the gap width and the pellet diameter. The gap is given by the sum of the foil thickness and of two nickel platings of the adjacent pellets. For the UO_2 samples, the gap is 0.04 mm and stems from two nickel platings only. R is the moderator-to-fuel reaction rate ratio (see Table IV). The factor 3 results from an approximate solid angle integration over the foil area, limiting the validity of Eq. (1) to $t/d \ll 1$. Thus, $3t/d$ is the geometrical probability for foil exposure to unshielded flux from the moderator. In Table III the results of this alternative route to obtain activation corrections are compared with the standard method. For C8, the result from Eq. (1) has been multiplied by a factor of 0.08 due to the fact that the perturbation caused by the 0.04-mm-gap has to be smeared over the 0.5-mm-thick oxide foil. The uncertainty for the results of Eq. (1) is mainly due to the solid angle calculation. The effect of neglecting scattering and absorption in the foil is not considered in the uncertainty analysis. On the other hand, the extrapolation method takes scattering and absorption in the foil automatically into account. Therefore, the foil corrections given by the extrapolation method are applied in practice.

The probes irradiated in the cell tube are corrected for foil self-shielding. The corrections for C8 and C2 are calculated using a simple approximation for resonance self-shielding. For F9 and F5 the corrections are experimentally determined. Respectively, the corrections are (2 ±0.5)%, (1 ±0.5)%, (1 ±0.5)% and (0 ±0.5)%.

TABLE IV

Experimental Moderator-to-Fuel Reaction Rate Ratios, and Comparison with BOXER Calculations.

	Moderator-to-Fuel Ratio	
	Experiment	Calc./Exp.
C8	4.62 ± 3%	1.036
C2	1.330 ±1.4%	0.983
F9	1.437 ±1.5%	1.005
F5	1.142 ±1.5%	0.985

The measured moderator-to-fuel reaction rate ratios provide a useful test for lattice codes which are designed to calculate reaction rates in the moderator. In Table IV, the ratios are compared with results obtained with BOXER. Particularly noteworthy is that C2, a nearly pure single resonance reaction in this spectrum, is well predicted by the calculation. The large heterogeneity for C8 is obvious.

ACKNOWLEDGMENTS

The authors wish to thank H. Graf, P. Bourquin, T. Steiner, and P. Thomi for their help with the experiments and the smooth operation of the reactor. Helpful discussions with

R. Chawla are gratefully acknowledged. The help of D.R. Mathews with the editing of the manuscript is kindly appreciated. The work has been performed under a joint KWU/KfK/EIR co-operative agreement.

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