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Reactor physics studies for steam generating heavy water reactors

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THE STEAM GENERATING HEAVY WATER REACTOR

A 100 MW(e) prototype Steam Generating Heavy Water (SGHW) reactor is now under construction at Winfrith. The fuel assemblies are large clusters of low enrichment UO_2 fuel pencils, clad in Zircaloy and cooled by pressurized boiling light water. Each cluster is housed in a zirconium pressure tube, surrounded by a thermally insulating gas gap and an aluminium calandria tube. Unpressurized heavy water moderator fills the spaces between the calandria tubes. Moderator displacement tubes, which may be empty or flooded with D_2O, provide an adjustment of the degree of undermoderation which controls the void coefficient. A typical lattice cell is shown in Fig. 1. In addition to these boiling channels, the prototype will include a few special channels for investigating nuclear superheat. These are of generally similar design to the boiling channels except that the fuel is canned in stainless steel and this material is also used to line the pressure tubes.

BRIEF REVIEW OF THE REACTOR PHYSICS PROBLEMS OF THE SGHW REACTOR

From the reactor physics point of view, special problems arise in the SGHW since about 30 per cent of the moderation occurs in the light water coolant. Calculation schemes must be capable of representing the two moderators, at different physical temperatures, one of which is in intimate contact with the fuel.

Preliminary experiments on one typical SGHW lattice in the DIMPLE reactor [1, 2] had shown the need to develop new methods of calculation for such lattices and had indicated the need for a more comprehensive series of tests of the influence of lattice geometry on various parameters such as power peaking in the fuel bundle, but with particular emphasis on the void coefficient of reactivity arising from changes in the steam content of the coolant. This void coefficient influences the dynamic behaviour of the plant, and a value near zero is required. It is the net result of physical effects having different algebraic signs; the main positive contribution comes from the gain in reactivity from loss of absorbing coolant, whereas the main compensating negative components arise from an increase in ^{238}U resonance absorption and increased leakage with reduction in coolant density. The variation of void coefficient with lattice geometry is a second differential effect, requiring high precision in experimentation and elaboration in methods of lattice calculation.

The detailed experimental investigation of the neutron balance in typical SGHW lattices, and the theoretical methods used to account for these measurements are the main topics of this paper.

After a period of operation of SGHW at power the core will contain a mixture of fuel elements with different fissile and fertile isotopic compositions. Suitable methods for the synthesis of the over-all reactor behaviour from the individual lattice properties are therefore required, backed by check experiments. This phase of the work is at present in progress.

THE THEORETICAL METHODS TESTED BY EXPERIMENT

Two distinct theoretical models have been developed for SGHW reactor calculations. For survey and initial design purposes, a five neutron group scheme has been used, which together with burn-up calculations has been organized into the METHUSELAH code. This scheme necessarily involved some intuitive features, and because of the sensitivity of, say, the void coefficient of reactivity to the details of the physical model, METHUSELAH was backed by the most exact calculations feasible at the time, with particular attention being given to checking the intuitive features of METHUSELAH. This elaborate scheme of calculation is known as THULE, and had as its objective exploration of the validity of METHUSELAH in selected situations only; therefore no particular effort was put into reducing the machine time used by THULE.

The METHUSELAH code

The METHUSELAH code is based on the four neutron group diffusion theory scheme of calculation of the Bettis laboratory of Westinghouse, the data for which is condensed from multi-group MUFT-SOFOCATE calculations. A detailed description of the physics underlying METHUSELAH has been presented elsewhere [3]. The main developments required

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to apply the four-group scheme to lattices of the SGHW type were to revise the treatment of resonance escape in clustered fuel geometry, using a simple extension of the Bell [4] approximation, and the replacement of the single thermal neutron group (condensed from SOFOCATE) by two overlapping groups covering the same energy range. The relative intensities of these two groups vary within a lattice cell and so represent spatial variations in the thermal neutron spectrum. The cell geometry is represented approximately by concentric rings. The leakage is calculated by volume and flux weighting of the diffusion coefficients of the individual regions of the cell, with a correction for the air gap.

METHUSELAH will also perform burn-up calculations following the isotopic changes in each ring of fuel pins separately. Fuel management problems are being studied by both one and two dimensional space dependent burn-up programmes based on METHUSELAH, which has a two neutron group output option from the cell calculations. Particular emphasis has been placed on a two-dimensional representation of the reactor in which each lattice cell is represented by its individual parameters and followed through burn-up and fuel shuffling procedures.

The THULE code

The features of METHUSELAH whose validity is not obvious are:

(*a*) The use of diffusion theory, particularly at fast fission energies;
(*b*) The extension of the Bell approximation for resonance escape calculations in rod cluster geometry;
(*c*) The treatment of thermal events with two overlapping thermal groups;
(*d*) The treatment of neutron leakage.

The THULE programme [5] was built round the existing multi-group Carlson transport theory code, Winfrith DSN [6], written for cylindrical geometry. The top two groups coincided in energy with those of METHUSELAH, but the third group covering the resolved ^{238}U resonances was cut off at 4 eV (compared with the METHUSELAH value of 0·625 eV). In the THULE scheme ^{240}Pu resonance absorption was therefore dealt with in the thermal group. Thermal events were represented by an existing structure of 42 interacting but non over-lapping thermal groups below 4 eV.

In the original version of THULE, resonance events were taken from the Monte Carlo code MOCUP [7] which can deal explicitly with SGHW geometry. Uncertainties are associated with the statistical variations in MOCUP. An alternative non-statistical method, of acceptable accuracy, based on collision theory and the equivalence theorem, has been developed and is now incorporated in THULE.

The many thermal group structure of THULE can accept any model of neutron thermalization. In the calculations reported here oxygen and all heavier atoms are represented as free gases. For the hydrogen of the coolant, the Effective Width Model I [8] has been used with an allowance for anisotropy. With a value of 3.7 for the parameter q, which specifies the width of the approximate phonon spectrum distribution selected, the observed room-temperature total cross section and diffusion coefficient are closely reproduced. The corresponding mean energy of the scattering molecule (denoted $\bar{K}$) is 6.6 kT, which agrees well [9] with the value deduced from the scattering law measurements [10]. Later measurements on asymptotic spectra in H_2O systems suggest $\bar{K} = 4.6\ kT$. A revised Effective Width Model, having an additional vibrational peak, is therefore being examined to see whether the additional parameter makes it possible to reduce the mean energy without affecting the total cross section. For D_2O, the present calculations use a free gas model with no anisotropy correction for scattering from deuterium. The correct diffusion coefficient is given, but the mean energy is too low. Improvement of this model is in hand, but is difficult since the calculation of total cross section must allow for the highly coherent nature of the scattering in D_2O.

Leakage is represented in THULE by the most up-to-date methods of Benoist [11, 12].

THE EXPERIMENTAL PROGRAMME

The programme of experiments was designed to test these methods of calculation with particular emphasis on the dependence of void coefficient on the D_2O/UO_2 volume ratio, on the H_2O/UO_2 volume ratio and on the fuel enrichment. These parameters had been identified by a METHUSELAH survey as those having most influence on the void coefficient. The variation of channel flux peaking with coolant density, and to a lesser extent with enrichment and channel size, was also identified for particular study. To provide a detailed check of the predictions of fast, resonance and thermal neutron events, it was necessary to make measurements of the following reaction rates:

(*a*) The lattice cell distribution of the activation of bare manganese and of the fission rate of ^{235}U.

(*b*) The lattice cell variation of the ratio of ^{239}Pu to ^{235}U fission, and of ^{176}Lu to Mn activation to check the variation of thermal spectrum.

(*c*) The relative conversion ratio, viz.

$$\frac{(^{238}\text{U capture}/^{235}\text{U fission})_{\text{fuel}}}{(^{238}\text{U capture}/^{235}\text{U fission})_{\text{thermal fission}}}$$

to check resonance capture in ^{238}U.

(*d*) The fast ratio, viz.

$$^{238}\text{U fission}/^{235}\text{U fission}$$

to check fast neutron predictions.

Measurements of the material buckling of the lattices provided an integral check on the reactivity deduced from these reaction rate measurements (see below under buckling measurements) and checked that the leakage was predicted adequately.

For ^{235}U enriched cores, the void coefficient of

Table 1. SGHW lattice investigated in experimental physics programme

Lattice serial no.	Pitch (inches)	UO_2 fuel pencils: Enrichment %	Cladding od (inches)	Cladding[a] thickness (inches)	Number of fuel pencils/cluster	Pressure tube[a] id (inches)	Pressure tube[a] od (inches)	Aluminium calandria tube id (inches)	Aluminium calandria tube od (inches)	Vol. coolant UO_2	Vol. mod. UO_2
SGHW prototype[b]	10.25[f]	—	0.627	0.026	36	5.14	5.53	6.99	7.25	1.1	6.8
SGHW prototype[c]	10.25[f]	—	0.627	0.026	36	5.14	5.53	6.99	7.25	1.1	5.3
SG1	9.5[g]	1.14	0.378	0.036	90	5.25	5.50	6.24	6.625	1.8	6.9
SG2	9.5[g]	0.91	0.578	0.036	37	5.25	5.50	6.24	6.625	1.7	6.3
SG3[d]	9.5[g]	0.91	0.578	0.036	37	4.52	4.776	6.24	6.625	0.9	6.3
SG4	9.5[g]	0.91	0.578	0.036	37	4.52	4.776	5.520	5.900	0.9	7.3
SG5[d]	9.5[g]	0.91	0.578	0.036	37	4.52	4.776	5.520	5.900	0.9	6.3
SG6[d]	9.5[g]	0.91	0.578	0.036	37	4.52	4.776	5.520	5.900	0.9	6.3
SG11	9.5[f]	1.78	0.378	0.036	90	5.25	5.496	6.240	6.690	1.8	8.8
SG12	9.5[f]	0.91	0.592	0.020	outer 20	5.25	5.496	6.240	6.690	1.2	6.6
		1.78	0.592	0.020	inner 23						
SG13	9.5[f]	1.35	0.451	0.020	71	5.25	5.496	6.240	6.690	1.2	6.3
SG14[e]	9.5[f]	1.35	0.451	0.020	71	5.25	5.496	6.240	6.690	1.2	6.1

[a] The prototype will use Zircaloy; aluminium was used in the experiments.
[b] Without moderator displacement tubes.
[c] With moderator displacement tubes.
[d] Lattices SG3, 4 and 6 represent three different methods of making the void coefficient more negative by reducing the vol. mod/vol. UO_2 ratio. Single moderator displacement tubes (as in SG6) were ultimately chosen for the SGHW prototype design.
[e] Lattice SG14 is identical with SG13, except for the addition of displacer tubes in the bulk moderator.
[f] Square lattice.
[g] Triangular lattice.

reactivity arises from changes in the density of the coolant rather than from changes in its physical temperature. It was therefore decided to conduct the experiments in relatively simple unpressurized assemblies operating generally at room temperature, with a few check measurements with coolant at 90°C. To conduct such cold experiments it was necessary to change the effective coolant density in the channels. A variety of different coolants was therefore used in each lattice tested. These were selected from (i) air, (ii) light water, (iii) an H_2O/D_2O mixture with the same value of $\zeta\Sigma_s$ as H_2O of density 0.4 g cm^{-3} and (iv) a polystyrene-bead-H_2O mixture producing a density of 0.6 g cm^{-3} with no more than 0.04 g cm^{-3} of polystyrene [13].

The METHUSELAH survey of possible lattices was used to select the lattices shown in Table 1 for detailed examination. Design parameters for the prototype SGHW are given for comparison. Reactivity considerations showed that it was impracticable to study this range of lattices in the zero energy critical reactor DIMPLE, nor did the limited time scale of this stage of the project allow the use of only one test assembly. Two sub-critical assemblies, known as SGHW I and II, were therefore constructed having features summarized elsewhere [14, 15].

Measurements on lattices SG1 to SG6 of Table 1 were conducted subcritically, the buckling and some of the manganese cell distribution measurements being performed in SGHW I, and the remaining detailed cell measurements in the higher flux SGHW II assembly. Lattices SG11 to SG14 of Table 1 were studied in DIMPLE. The availability of a critical assembly was used to extend the scope of the measurements to include:

(*a*) A direct check of the effect of moderator displacement tubes on the void coefficient, by comparing the critical sizes of the radially bare reactor with and without H_2O coolant, both with the displacement tubes voided and flooded with D_2O.

(*b*) A study of a few simulated superheat channels distributed in the central region, and then at the edge of the core, to observe the effects on local power peaking and on thermal spectrum.

Brief review of the experimental techniques used and their accuracy

Attention has been drawn already to the need for high precision in experimentation for void coefficient determination. The void coefficient of reactivity, K_v, is defined as $\delta k/\delta V_f$ where V_f is the void fraction. At the operating condition of the prototype this reduces to:

$$K_v = 0.72/k.\ d(k)/d\rho$$

where ρ is the coolant density.* Stability studies had shown that the K_v for a large SGHW power reactor should lie within the approximate range ± 0.02. Experiments were therefore designed to determine the void coefficient of the lattice tested to better than ± 0.01. The void coefficient was determined in two separate ways. The more accurate and informative way was to make use of the detailed reaction rate (R.R.) measurements, listed in the previous section, in the same lattice with different coolant densities to yield values of k_{inf} for the lattice. The variation of this value of k_{inf}, referred to as k_{inf} (R.R.) with coolant density then gave the void coefficient. As a check, buckling measurements were made at different coolant

* K_v may be defined in terms of either k_{inf} or k_{eff}. In analysing the critical and sub-critical experiments, we have preferred to work in terms of k_{inf}.

densities; together with a calculation of the leakage, these measurements yield values of k_{inf}, referred to as k_{inf} (L), and thus provide an independent estimate of void coefficient.

The technique and accuracy of the determination of k_{inf} (reaction rate) [16]

It is important to note that in testing the theoretical models by experiment, the quantity to be observed is calculated for direct comparison. For example, observed lattice cell distributions with manganese are not corrected for resonance absorption to give $1/v$ activation rates; instead the theory is used to predict the total manganese activation. This technique then ensures that a consistent theoretical model and nuclear data are used at all stages of comparison.

The quantity k_{inf} (R.R.) is derived by correcting the THULE calculations so that observable reaction rates agree with the measured values. Thus the lattice cell flux distributions with manganese and ^{235}U are used to correct for errors in the calculated pressure and calandria tube thermal absorption rates, while the relative conversion ratio and fast ratio are used to correct errors in resonance capture and fast fission in ^{238}U.

It will be recalled that a target was set to measure K_v to better than ± 0.01. The principal observable quantities contributing to experimental uncertainty in K_v, using the k_{inf} (R.R.) concept, are the fluxes in the pressure and calandria tubes relative to that in the fuel, the relative conversion ratio and the fast ratio. Making some allowance for uncertainties in prediction which were not checked by these experiments, a contribution of ± 0.005 in K_v was allowed from the observable quantities. If equal contributions to the error in K_v are assigned to these quantities, then the allowable error in K_v arising from a pair of measurements of each of these quantities at different coolant densities is about ± 0.0025. For values of the two densities of 0.4 and 1.0 g cm^{-3} an error in K_v of ± 0.0025, at a density of 0.4 g cm^{-3}, will arise from a 2 per cent error in the pressure and calandria tube flux relative to the UO_2 flux, or a 0.5 per cent error in the relative conversion ratio, or a 1.5 per cent error in the fast ratio. Acceptable accuracies in individual observable quantities are thus set; these are mainly limits on random errors, because systematic errors generally change only the absolute value of k_{inf} (R.R.) and not its dependence on coolant density. The repeatability of the results of the measurements showed that these accuracies were achieved and brief comments on the experimental techniques adopted are now presented.

Lattice cell distributions [17]

The accuracy demanded of the lattice cell distribution in determining K_v is clearly greater than the 5 per cent accuracy required for channel power peaking studies. Effort was concentrated upon the pressure and calandria tube flux levels relative to the UO_2 fluxes although full distributions including moderator and coolant values were observed. Manganese foils were generally used, either sandwiched between fuel pellets or as sector foils in the coolant or as discs suspended in the moderator. Attention was paid to the azimuthal variation in flux at the pressure tube wall due to the discrete spacing of the outer row of fuel pins.

Relative conversion ratio [18]

In these measurements, made in each representative fuel pin in the cluster, ^{238}U capture is characterized by ^{239}Np activity, and ^{235}U fission is determined from fission product activity as a by-product of the fast ratio measurement (see paragraph below). The ^{239}Np production was monitored in every measuring position using a technique developed by Tunnicliffe [19]. Coincident counts were observed between γ- and X-rays at 104 keV from 0.005 in thick UO_2 discs, ground to dimensional accuracies of ± 0.001 in, after irradiation between pellets of oxide of similar accuracy. At every fifth position an additional chemical check was used by irradiating a 0.200 in thick pellet and then separating and β-counting the ^{239}Np [20]. The results from chemical separation were repeatable to 1 per cent to 2 per cent, and agreed with the results of coincidence counting. Automatic counting equipment was designed for this experiment and over 1 000 measurements made with it during this 18 month programme. Random errors of 0.5 per cent were achieved with systematic errors of about 1.5 per cent.

Fast ratio [21]

Two foils of different but known enrichments sandwiched between fuel pellets were used, and the fission product γ-activity counted at a bias level set to reject bremsstrählung from ^{239}U decay. A significant systematic error may arise in calibrating γ-activity in terms of fission rate in ^{238}U and ^{235}U. Two methods have been used. In one, a back-to-back fission chamber, carrying foils of the same two enrichments, is irradiated; the fission rate is observed during a carefully controlled exposure, and the fission product γ-activity of the foils is then measured. In the second calibration, the yields of the fission product ^{140}La from ^{238}U and ^{235}U are assumed, and the intensity of the ^{140}La peak adopted as a measure of the fission rate. An 8 per cent difference was found between the two techniques which is being investigated further.

Technique and accuracy of the buckling measurements

Both the large exponential assembly, SGHW I, and DIMPLE were used, and radial and axial components of the buckling determined from BF_3 or fission chamber scans. Typical accuracies were ± 0.002 m^{-2} in axial and ± 0.003 m^{-2} in radial bucklings measured in SGHW I [22], for a lattice with a material buckling in the range -1 to $+5$ m^{-2}. Such errors are small compared with uncertainties in the prediction of migration area, which may amount to 5 per cent in each direction. Thus the absolute error in k_{inf} (L) may amount to

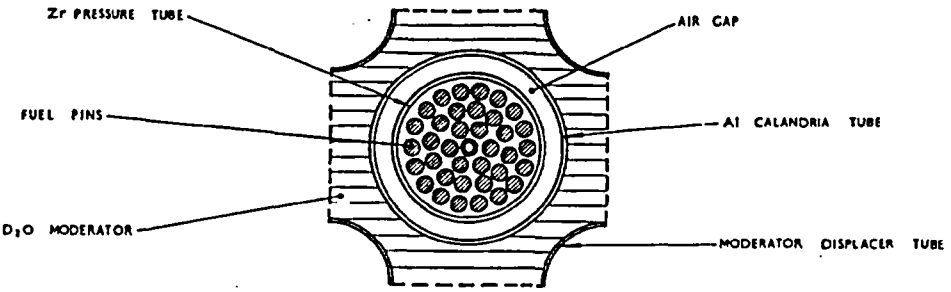

Figure 1. Typical SGHWR lattice cell

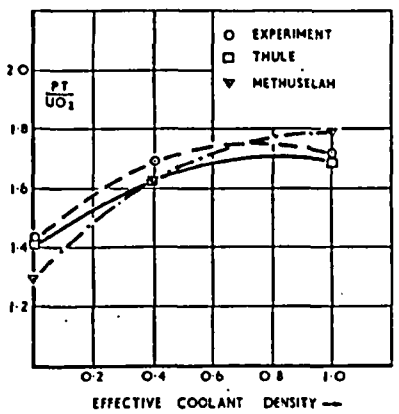

Figure 2. Ratio of mean Mn activation in pressure tube to mean Mn activation in UO_2 as a function of coolant density in the SG13 cores

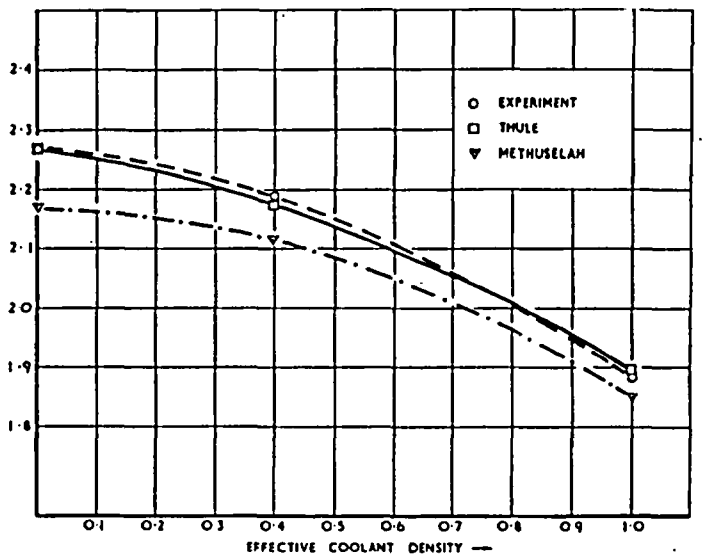

Figure 3. Relative conversion ratio in SG13 cores

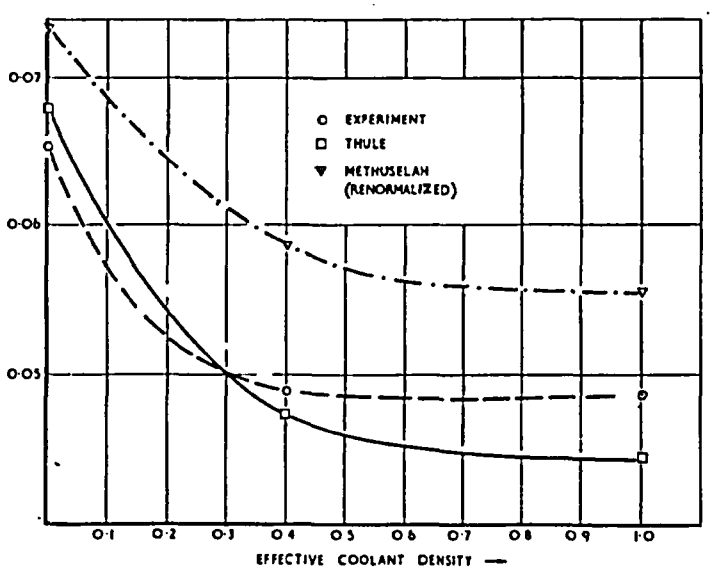

Figure 4. Fast ratio in SG13 cores

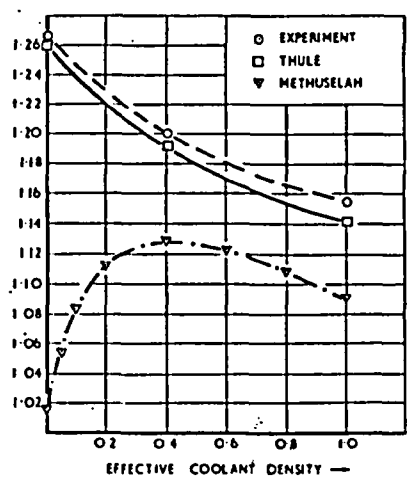

Figure 5. Mean Lu/Mn ratio in fuel in SG13 cores

about 1 per cent. It is reasonable to suppose that a large part of this error is systematic and that the uncertainty in the change in k_{inf} between two coolant densities is much smaller, say about 0.4 per cent, corresponding to an uncertainty in K_V of 0.005.

Thermal spectrum sensitive reaction rates [23]

The ^{239}Pu to ^{235}U fission ratio was determined from ¼ in diameter fission chambers inserted into the D_2O moderator of the lattice and from foils of Pu/Al and U/Al sandwiched between the fuel pellets. The ^{176}Lu to Mn ratio was found by irradiating composite discs between the pellets and then γ-counting the discs. These measurements were included to check the ability of the theoretical methods to predict the variation of thermal spectrum across the cell.

COMPARISONS OF THEORY AND EXPERIMENT

For each lattice tested, the variation with coolant density of each quantity observed was compared with METHUSELAH and THULE predictions. In addition, the values of k_{inf} (R.R.) and k_{inf} (L) were compared with the METHUSELAH and THULE estimates. Owing to shortage of space only a selection of the experimental results is presented. A complete discussion is given elsewhere [24].

Analysis of the SG13 cores

Figure 2 shows the variation of the ratio (mean Mn reaction rate in pressure tube)/(mean Mn reaction in UO_2) with the coolant density ρ. It will be seen that THULE reproduces satisfactorily the trend of this quantity with ρ. METHUSELAH over-estimates the variation, giving too little fine structure in the air core and too much in the H_2O core. This effect will cause METHUSELAH to over-estimate K_V in this case.

Figure 3 shows the variation of the relative conversion ratio with ρ. THULE is in very good agreement with experiment (the THULE calculation is based on the new method of determining resonance integrals described above). METHUSELAH is not as accurate as THULE.

It will be seen from Fig. 4 that both theoretical methods reproduce the trend of the fast ratio with coolant density reasonably well, though both show a drop with coolant density persisting to high coolant densities, which is not shown by the experiment. THULE is in rather better agreement with the experimentally determined absolute level than is METHUSELAH, which is rather too high. The discrepancy between METHUSELAH and experiment cannot be accounted for by the calibration uncertainties referred to in the preceding section.

Figures 5 and 6 show the variation of the Lu/Mn and Pu/U ratios with coolant density (the ratio is normalized by irradiating the detectors in a thermal spectrum). THULE reproduces the experimental trend very well, and is in adequate agreement with the measured absolute level: this suggests that the scattering laws used in THULE are quite satisfactory. The general level of the METHUSELAH predictions is too low, showing that the Wigner-Wilkins or free proton model of thermalization is too soft a representation of H_2O and D_2O. METHUSELAH gives the trend with coolant density quite well unless the coolant density is very low (when the two-thermal-group model becomes unsatisfactory).

Figure 7 shows the variation with coolant density of the four values of k_{inf} defined above. The METHUSELAH k_{inf} is 1 per cent to 2 per cent higher than the reaction rate value, and its increase with diminishing coolant density is too rapid. The general level of the THULE prediction is more satisfactory but it, too, shows the wrong variation with coolant density. The values of K_V calculated by differencing the various k_{inf} values between the mixture and water cores are:

Reaction rate . . .	-0.003 ± 0.004
Leakage	-0.003 ± 0.005
THULE	$+0.006$
METHUSELAH .	$+0.011$

The discrepancy between THULE and experiment stems from the flatness of the fast ratio measurements as the coolant density is increased, compared with the fall in this ratio predicted by the theories. The existence of this effect is confirmed by the value of K_V deduced from k_{inf} (L).

Results from the SG3, 5 and 6 cores

Figures 8, 9 and 10 show the variation of the four values of k_{inf} with coolant density in the SG3, 5 and 6 cores. These cores, which are described in Table 1, are designed primarily to test methods of calculating the leakage. In them the moderator volume is reduced by enlarging the gas gap (SG3) or by inserting large displacer tubes (SG5) or small displacer tubes (SG6) in the moderator. A number of trends are evident from these figures:

(*a*) k_{inf} (R.R.) and k_{inf} (L) agree to within 1 per cent at any given density, and the agreement on variation with ρ is considerably better than this, particularly for ρ greater than 0.4 g cm^{-3}. In conjunction with the similar behaviour shown on Fig. 7, this is strong confirmation of the estimates of accuracy given earlier in this paper and gives some confidence in the leakage model.

(*b*) The absolute level of the METHUSELAH k_{inf} is too high, but the METHUSELAH K_V values are in good agreement with experiment over the range of coolant density between 0.4 g cm^{-3} and 1.0 g cm^{-3}.

(*c*) The absolute values of the THULE k_{inf} are in good agreement with experiment, but the THULE K_V is about 0.010 low over the range of coolant density 0.4 g cm^{-3} to 1.0 g cm^{-3}. This is in marked contrast to the behaviour found in the SG13 cores, where THULE over-estimated K_V. The detailed reaction rate measurements show that this effect arises from deficiencies in the prediction of the variation of thermal fine structure with coolant density. It is believed that this difficulty is connected with the problem of smearing the coolant and the relatively large fuel pins used in these lattices. Since METHUSELAH and THULE use the same smearing model, any change in the model

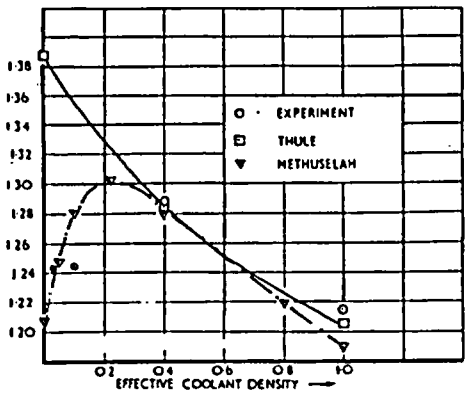

Figure 6. Mean $^{239}Pu/^{235}U$ fission ratio in fuel in SG13 cores

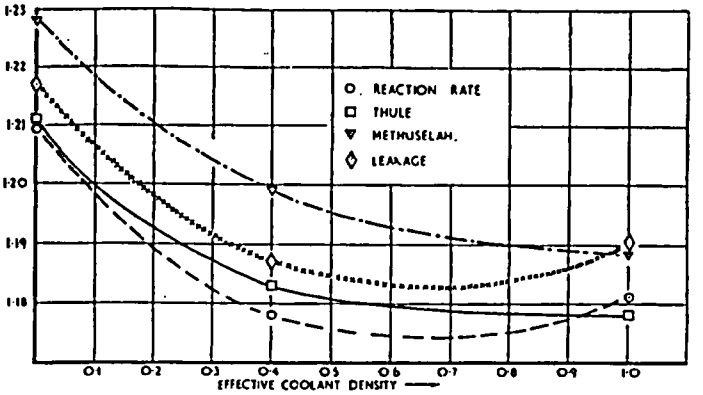

Figure 7. k_{inf} in SG13 cores

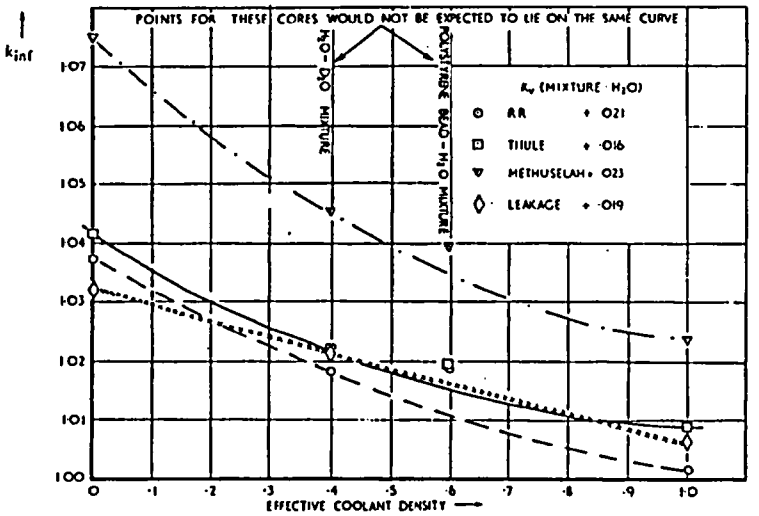

Figure 8. k_{inf} in SG3 cores

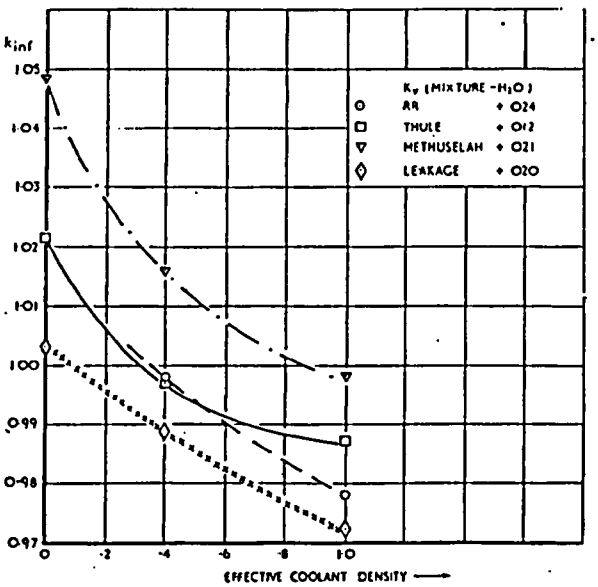

Figure 9. k_{inf} in SG5 cores

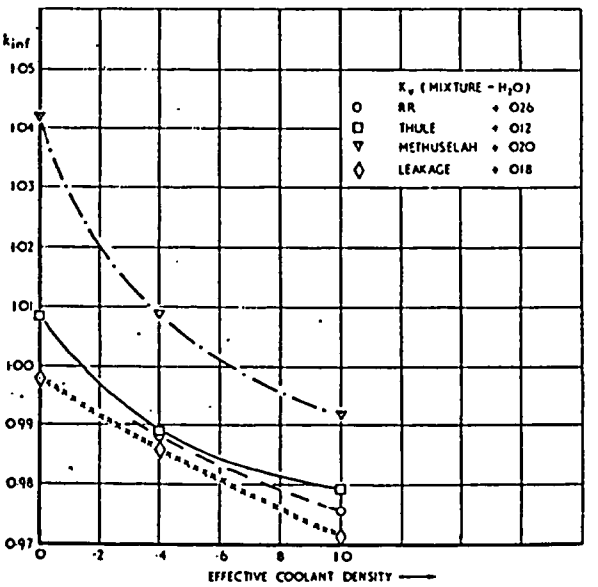

Figure 10. k_{inf} in SG6 cores

to improve agreement with experiment will then tend to spoil the good agreement between METHUSELAH and the void coefficients deduced from the experiments. The problem is being investigated further.

CONCLUSIONS

The experimental programme has provided data of sufficient accuracy to test the neutron balance in both METHUSELAH and THULE calculations, with the result that both of these methods have been developed to give an adequately accurate prediction of the initial void coefficient of reactivity over a range of lattices. The close agreement between the new theory of resonance capture now used in THULE and the detailed relative conversion ratio measurements is particularly noteworthy. The METHUSELAH code is rather less accurate than THULE in its predictions of absolute reactivity and is significantly in error in its prediction of spectrum at low coolant densities. Nevertheless it is quite adequate for the survey and initial design purposes for which it was intended. Further development of theory is required in the methods of representing clusters of relatively large fuel rods with light water coolant, where hyperfine structure in the thermal flux is not well represented by the present ring-smearing technique. The absolute level of the fast ratio is uncertain to about 10 per cent, and the predictions of the variation in this quantity with coolant density require further development. The good agreement between k_{inf} (L) and k_{inf} (R.R.) shows that the methods of calculating leakage incorporated in THULE are satisfactory.

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ABSTRACT—RÉSUMÉ—АННОТАЦИЯ—RESUMEN

A/174 Royaume-Uni

Études physiques des réacteurs générateurs de vapeur modérés à l'eau lourde

par C. G. Campbell et *al.*

Les problèmes spéciaux de physique des réacteurs soulevés par l'utilisation d'un mélange bouillant eau/vapeur comme réfrigérant dans un réacteur à tubes de force modéré à l'eau lourde sont discutés avec référence particulière au réacteur prototype SGHW maintenant en construction à Winfrith. Ce concept de réacteur a conduit au développement du code METHUSELAH à cinq groupes neutroniques pour évaluer le fonctionnement du cœur dans des projets et pour les calculs de physique au schéma THULÉ, qui utilise le meilleur modèle physique de cellule de réseau actuellement disponible.

La phase du travail expérimental décrite se rapporte à l'étude détaillée du bilan neutronique dans une gamme de réseaux à eau lourde du type à tubes de force. Le travail a été concentré sur l'étude du coefficient cavitaire de réactivité et a exigé la simulation du refroidissement à eau légère dans une gamme de densités dans des ensembles non pressurisés. On décrit les techniques expérimentales pour la caractérisation des événements dans les régions d'énergie des neutrons rapides, de résonance et thermiques. La haute précision demandée à ces techniques est fixée par le fait que le coefficient cavitaire, qui est un effet différentiel, doit être mesuré avec une précision acceptable. Outre ces mesures détaillées de cellules de réseaux, une gamme de mesures du laplacien dans des réseaux, avec et sans eau légère de refroidissement, a été utilisée pour servir de vérification du coefficient cavitaire de réactivité et des effets de fuite sur le bilan neutronique. On décrit l'utilisation de deux ensembles sous-critiques et du réacteur critique DIMPLE pour ces études.

Une attention particulière a été apportée à la comparaison des résultats expérimentaux avec les prédictions calculées à l'aide de THULÉ et de METHUSELAH avec commentaires sur la précision de ces méthodes de calcul dans des buts d'étude détaillée et d'évaluation d'un réacteur.

Проводится детальное сравнение экспериментальных результатов в области решеток с последними предсказаниями расчетов по программам THULE и METHUSELAH; причем отмечается точность этих методов расчета для целей детального конструирования и оценок реактора.

A/174 Соединенное Королевство

Исследование физики тяжеловодных реакторов для производства пара

К. Г. Кемпбелл *et al.*

В докладе рассмотрены специальные проблемы физики реакторов, возникающие в связи с использованием кипящей паро-водяной смеси в качестве теплоносителя в тяжеловодном реакторе с трубами под давлением; особое внимание уделено прототипному реактору SGHW, строящемуся в Уинфрите. В связи с разработкой этой концепции реактора потребовалось создать специальную пятигрупповую программу — код METHUSELAH для счетной машины для оценки характеристик активной зоны, необходимых при проектировании, и схему THULE для расчетов по физике реакторов, в которой использованы лучшие современные физические модели ячеек решетки.

На описываемой стадии экспериментальных работ детально изучается баланс нейтронов для ряда решеток из труб давления с тяжелой водой. Работа сосредоточена на изучении пустотного коэффициента реактивности; для ее проведения потребовалось имитировать теплоноситель — обычную воду в диапазоне плотностей, приемлемых для сборок, работающих без компенсации давления. Описана экспериментальная методика для характерных явлений в энергетических областях быстрых, резонансных и тепловых нейтронов. Высокая точность, необходимая при проведении измерений по этой методике, определяется требованием, чтобы пустотный коэффициент, который является дифференциальным эффектом, был измерен с приемлемой точностью. В дополнение к этим детальным измерениям ячеек решетки для проверки пустотного коэффициента реактивности и влияния утечки на баланс нейтронов была проведена серия измерений лапласиана в решетках с обычной водой в качестве теплоносителя и без воды. Описано использование для этих измерений двух подкритических сборок и критического реактора DIMPLE.

Особое внимание уделено сравнению экспериментальных результатов с предсказаниями расчетов по программам THULE и METHUSELAH, представлен метод подразумевающихся ошибок в теоретических предсказаниях путем определения значения бесконечного коэффициента размножения на основании детальных измерений ячейки решетки.

A/174 Reino Unido

Estudios de física de reactores aplicados a los de agua pesada, generadores de vapor

por C. G. Campbell *et al.*

Se discuten los problemas especiales de física de reactores que se derivan del uso de una mezcla hirviente de agua y vapor para refrigerar un reactor con tubos de presión moderado por agua pesada. Se hace referencia particular al reactor prototipo SGHW que se está construyendo en Winfrith. Este tipo de reactor ha llevado al desarrollo del código METHUSELAH para 5 grupos de neutrones con el fin de evaluar el comportamiento del núcleo con fines de diseño desde el punto de vista del proyecto y al esquema THULE de cálculo de la física del reactor, esquema que utiliza el mejor modelo físico de la celda de red de que se dispone en este momento.

La fase del trabajo experimental aquí descrita se refiere al estudio detallado del balance neutrónico en una serie de redes de las del tipo de tubo de presión y agua pesada. El trabajo se ha concentrado en el estudio del coeficiente de reactividad de huecos y ha hecho necesario simular el refrigerante de agua ligera en un intervalo de densidades, en montajes no sometidos a presión. Se describen técnicas experimentales que sirven para distinguir los sucesos en las regiones de neutrones rápidos, de resonancia y térmicos. Si se pide a estas técnicas gran precisión es para que el coeficiente de huecos, que es un efecto diferencial, se mida con exactitud aceptable. Además de estas medidas detalladas acerca de la celda de red, se ha usado una serie de medidas de laplacianas en redes con y sin agua ligera como refrigerante para comprobar el coeficiente de reactividad de huecos y los efectos de las fugas sobre el balance neutrónico. Se describe el uso de dos conjuntos subcríticos y del conjunto crítico DIMPLE para estos estudios.

Se ha prestado atención particular a las comparaciones de los resultados experimentales con las predicciones de THULE y METHUSELAH y se presenta un método de inferir errores en las predicciones teóricas deduciendo un valor del factor de multiplicación infinito a partir de las medidas detalladas de la celda de red.

Se dan comparaciones detalladas de los resultados experimentales obtenidos de la serie de redes, con las últimas predicciones de THULE y METHUSELAH así como comentarios acerca de la exactitud de estos métodos de cálculo con fines de diseño detallado y evaluación de reactores.

Lattice studies and critical experiments in D_2O moderated systems

By J. L. Crandall,* W. L. Brooks,** I. Kaplan,*** F. L. Langford, Jr.,**** and L. C. Schmid*****

Heavy water reactors play an important role in the United States atomic energy program. Five Savannah River production reactors and nine other D_2O reactors are built or being built for operation at powers over one megawatt [1]. Most of these installations are research and test reactors using the high neutron fluxes and large irradiation areas achievable with the heavy water designs. However, also included is the first US heavy water power reactor, the Carolinas-Virginia Tube Reactor (CVTR) [2], which went critical on 30 March 1963. The excellent neutron economy and low fueling costs associated with the D_2O power reactors have generated many development programs and design studies [3]. This paper reviews the US work performed on the physics of these systems since 1958. The major programs considered are the plutonium recycle studies at the Hanford Laboratories, the natural and slightly enriched uranium lattice studies at the Massachusetts Institute of Technology, the conceptual power reactor design studies at the Savannah River Laboratory and United Nuclear, and the CVTR studies at the Westinghouse Atomic Power Division.

LATTICE STUDIES

Buckling and reactivity measurements

Studies at Savannah River in the Process Development Pile (PDP), an unreflected D_2O critical 495 cm in diam [4], have provided reference buckling data on uniform lattices of natural uranium metal and oxide fuel in D_2O [5–10]. These data, given in Table 1, are believed free of systematic error and accurate to better than $\pm 7\ \mu B$.

The PDP has also been used for lattice substitution measurements in which the effect of substituting 1 to 19 test fuel assemblies for the reference fuel assemblies in a uniform critical loading is evaluated in terms of the change in the critical moderator height. Two techniques have been employed. In one, a single substitution measurement is made and the results analyzed by two-group, two-region theory [11]. In the other, successively larger substitutions are made and the results analyzed by one- or two-group, three-region perturbation theory (Persson method [12]) or two-group, three-region diffusion theory [13] to eliminate boundary effects. As shown by Table 2 [14], the two methods give good results when the test lattice is similar to the reference lattice. However, a more comprehensive study [13] intercomparing the lattices of Table 1 showed that the single substitution method broke down when differences in resonance capture between the regions were large.

For substitution measurements on irradiated fuel the Hanford Laboratories built a special critical facility, the PRCF [15]. It consists of a tank of D_2O (or H_2O) 6 ft in diam and 9 ft high in an underground cell adjacent to the PRTR storage basin. Although the PRCF can be used for a variety of critical and exponential studies, immediate plans are to compare single irradiated and unirradiated fuel assemblies in the central lattice position shown in Fig. 1. In a typical study, the reactivities of three fuel assemblies composed of 19 rods of 1.8 w/o Pu–Al 0.504 inches in diam [16], which had been irradiated to 30, 56, and 87 MWd in the PRTR [17], were measured in the PRCF as 67 per cent, 45 per cent and 25 per cent respectively of the unirradiated element reactivity worth of 14.4 mk [18, 19]. The reactivity values agree within −4 per cent (−0.6 mk) at 0 MWd and +12 per cent (+0.4 mk) at 87 MWd with those calculated by three-group diffusion theory [20]. The cross-section changes for the calculations were computed by MELEAGER [21].

The critical buckling measurements have been supported by exponential studies, particularly at MIT [22] and Savannah River [3, 9, 10]. Sample results from MIT are given in Table 3 [23–28]. A particular point has been to verify the validity of the exponential techniques. The MIT effort has concentrated on determining the criteria for equilibrium conditions [29], while the Savannah River effort has involved extensive comparisons of exponentials and criticals [9, 30]. The latter studies demonstrated that the radial buckling of the exponential could vary with the lattice loading by as much as $60\ \mu B$ in a manner not adequately explained by theory. Studies at the Hanford Laboratories [31], MIT [26], and Savannah River [3] also demonstrated small nonseparabilities of the heterogeneous cell and

* E. I. du Pont de Nemours Company, Savannah River Laboratory.
** United Nuclear Corporation.
*** Massachusetts Institute of Technology.
**** Westinghouse Electric Corporation.
***** General Electric Company, Hanford Laboratories.

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Table 1. Studies of one region D_2O lattices in the PDP

A. Clusters of 0.998″ natural uranium rods, 0.032″ Al clad, 1.5″ C-C rod pitch (99.75 mol % D_2O, 22 °C) [5,6]

Rods in cluster	Triang. lattice pitch (in.)	δ_{28}	δ_{25}	ρ_{28}	ϕ mod/ϕ fuel Mn Sub Cd	η Sub Cd	B^2, μB Meas.	B^2, μB ROC A [34]
1	7.00	0.053	0.024	0.181	1.815	1.312	648	646
	8.08	0.054	0.016	0.132	1.865	1.318	529	534
	9.33	0.050	0.012	0.114	1.916	1.315	408	420
	11.22	0.048	0.009	0.086	2.016	1.318	230	243
	∞	—	0.0087	0.075	—	—	—	—
3	7.00	0.063	0.063	0.538	2.154	1.318	759	759
	9.33	—	0.031	0.336	2.302	1.306	711	712
	12.12	—	0.023	0.212	2.497	1.315	505	500
	14.00	0.062	0.021	0.242	2.608	1.315	381	379
	∞	—	0.021	0.207	—	—	—	—
7	9.33	—	0.072	0.662	2.689	1.308	602	555
	12.12	0.071	0.045	0.457	—	1.311	611	590
	14.00	0.067	0.039	0.401	3.229	1.306	516	490
	18.52	0.068	0.035	0.347	3.794	1.317	286	274
	21.00	—	0.033	0.354	3.902	—	210	197
	∞	—	0.033	0.368	—	—	—	—
7[a]	18.52	0.057	0.024	0.339	3.236	1.313	326	324
19	14.00	0.100	0.091	0.949	4.294	1.311	341	156
	18.52	0.102	0.062	0.733	5.150	—	302	194
	∞	—	0.062	0.712	—	—	—	—

B. Clusters of 0.500″ natural uranium rods, 0.020″ Al clad, 0.650″ C-C rod pitch (99.59 mol % D_2O, 22 °C) [7, 13]

Rods in cluster	Triang. lattice pitch, in.	Al housing dim. (in.)	Coolant	Measured B^2, μB B_r^2	B_z^2	B_m^2	Calc. B_m^2 ROC A	M_z^2 Kinetics, cm^2	D_z/D_r Subst. Exp.
19	8.08	none	D_2O	339	263	602	607	270	—
	9.33	none	D_2O	352	234	586	587	316	—
	9.33	4 × 0.054	D_2O	213	289	502	494	316	—
	9.33	4 × 0.054	Air	213	235	(448)[b]	(457)	—	1.06
	9.33	5 × 0.056	D_2O	214	255	469	462	316	—
	9.33	5 × 0.056	Air	176	146	(322)	(302)	515	1.20
	12.12	none	D_2O	219	194	412	422	—	—
	12.12	4 × 0.056	D_2O	218	145	363	364	—	—
	12.12	4 × 0.056	Air	215	163	(377)	(383)	—	—
	14.00	none	D_2O	154	150	304	313	—	—
31[c]	9.33	none	D_2O	208	330	538	541	258	—
	9.33	5 × 0.056	D_2O	211	238	449	427	258	—
	9.33	5 × 0.056	Air	202	139	(341)	(340)	360	1.09
	11.10	none	D_2O	244	281	525	—	332	—
	12.12	none	D_2O	218	278	496	498	—	—
	12.12	5 × 0.056	D_2O	217	211	428	418	—	—
	12.12	5 × 0.056	Air	213	215	(428)	(422)	—	—
	14.00	none	D_2O	223	161	384	389	—	—
	16.17	none	D_2O	88	194	282	282	—	—
48[d]	12.12	none	D_2O	215	265	480	466	291	—
	12.12	6 × 0.058	D_2O	217	199	416	383	291	—
	12.12	6 × 0.058	Air	207	185	(392)	(363)	383	—
	14.00	none	D_2O	153	276	429	412	—	—
	14.00	6 × 0.058	D_2O	152	222	374	348	—	—
	14.00	6 × 0.058	Air	148	256	(404)	(379)	—	—

[a] 2″ C-C Rod Pitch.
[b] Bucklings in ( ) are anisotropic.
[c] 0.250″ × 0.035″ Al tubes in six corner rod positions.
[d] C-C rod spacing 0.656″.

Table 1. Studies of one region D_2O lattices in the PDP (*continued*)

C. Natural uranium metal tubes (99.59 mol % D_2O, 22 °C) [8]

Tube dimensions, in. Uranium OD	Uranium ID	6063 Al OD	6063 Al ID	Triang. lattice pitch (in.)	Coolant	Measured B^2, μB: B_r^2	B_z^2	B_m^2	Calc. B_m^2 BSQ μB	M_z^2 Kinetics, (cm²)
3.500	2.860	3.890	3.640	9.33	D_2O	385	286	671	655	256
				9.33	Air	375	195	(570)	(529)	357
				12.12	D_2O	217	341	558	542	—
				12.12	Air	212	312	(524)	(515)	—
				14.00	D_2O	171	269	440	420	—
				14.00	Air	168	270	(438)	(426)	—
0.998	—	1.090	1.026	9.33	D_2O	375	178	553	531	—
2.120	1.760	—	—	9.33	Air	378	212	(590)	(543)	—
3.500	2.860	3.890	3.640	12.12	D_2O	214	330	544	539	—
				12.12	Air	213	386	(599)	(593)	—
				14.00	D_2O	166	282	448	437	—
				14.00	Air	164	341	(505)	(500)	—

over-all flux distributions which slightly affected the radial buckling but which could be calculated by finite heterogeneous theory [31].

The reactivity measurements placing minimum requirements on the size of the lattice samples are the null reactivity techniques originally developed in the PCTR at Hanford [32]. United Nuclear built a special facility, the PLATR [33], to perform these measurements on heavy water systems; Fig. 2 shows a schematic view. As listed in Table 4 the PLATR has been used to study almost 200 lattices of oxide rod clusters and tubes [34, 35]. Agreement between PLATR results and SRL buckling measurements (transformed into k_∞ numbers by the two-group critical equation) is evidenced by Fig. 3. A joint study of null reactivity methods at the Hanford Laboratories and MIT [36] has shown that mismatches in the epithermal flux between the buffer and test regions in the PCTR can lead to erroneous results under the usual one- or two-group analyses. These errors can be corrected by a three-group analysis or by a new technique that separately measures the thermal and epithermal multiplication factors.

Another technique with minimal lattice sample requirements is the pulsed-source die-away measurements. This technique is under development at MIT [22], North Carolina State College [37], Savannah River, and the University of Florida. Results from the Florida program were presented at the IAEA Symposium on Exponential and Critical Experiments [38].

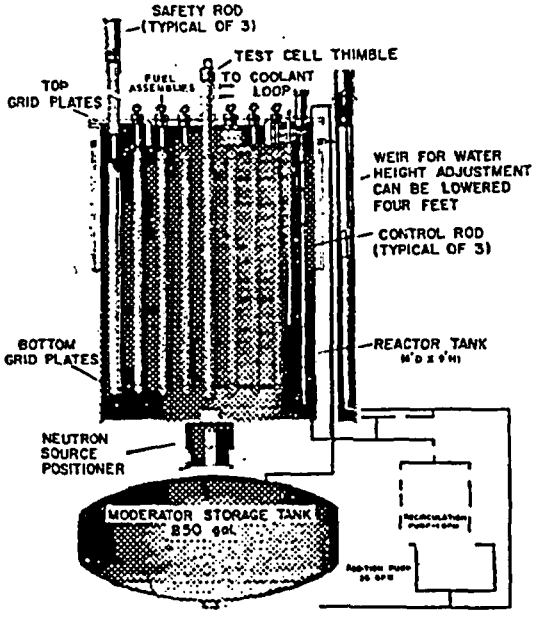


Figure 1. Plutonium recycle critical facility

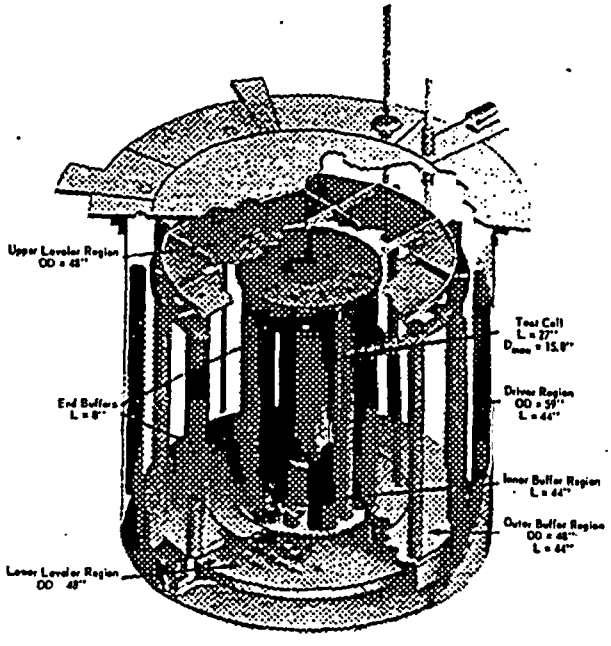


Figure 2. Artist's conception of PLATR

Table 2. Substitution measurements on UO_2 rod cluster lattices

9.33" Triangular lattice pitch, 99.58 mol % D_2O, 22.8 °C

Rods in cluster	Rod spacing C-C (in.)	6063 Al housing dimensions (in.) OD	Wall	Coolant	Buckling, μB Perturbation	Two-group	Average
19	0.607	3.080	0.030	H_2O	206	207	207
	0.607	3.080	0.030	Organic	353	353	353
	0.607	3.080	0.030	Air	542	548	545
	0.598	2.679[a]	0.030	Organic	454	460	457
	0.598	2.679[a]	0.030	Air	560	560	560
31	0.598	4.000	0.050	H_2O	117	108	113
	0.607	4.000	0.050	Organic	289	296	293
	0.607	4.000	0.050	Air	491	490	491
37	0.607	4.350	0.050	H_2O	64	75	70
	0.607	4.350	0.050	Organic	245	258	252
	0.607	4.350	0.050	Air	436	433	435
	0.607	4.746	0.050	Organic	28	33	31
	0.650	4.746	0.050	Organic	105	116	111
	0.650	4.970	0.162	Organic	−90	−70	−80
	0.650	4.746	0.050	D_2O	399	401	400
	0.650	4.746	0.050	Air	364	365	365

Oxide rods: 0.500" nat. UO_2 (10.4 g/cm³) clad in 6063 Al 0.547" OD × 0.020" wall.
Host lattice: 31 rod UO_2 cluster, no housing, on 9.33" pitch (see Table 1).
Organic coolant was 73.5% diphenyl oxide, 26.5% diphenyl.
[a] Outer dimension across the flats of an hexagonal housing.

Table 3. Studies of D_2O lattices in the MIT exponential (99.75 mol % D_2O, 22 °C) [23–28]

A. Single 1.010" natural uranium rods, 0.028" Al clad

Triangular lattice pitch (in.)	δ_{28}	δ_{25}	ρ_{28}	C^*	$\bar{N}$ mod[a] / $\bar{N}$ fuel	B^2, μB
4.50	0.0597	0.0479	0.507	1.017	1.72	848
5.00	0.0596	0.0340	0.401	0.948	1.72	865
5.75	0.0583	0.0268	0.301	0.859	1.80	815
∞	0.0559	0.0086	—	—	—	—

B. Single 0.250" enriched uranium rods (1.03% ^{235}U), 0.028" Al clad

Triangular lattice pitch (in.)	δ_{28}	δ_{25}	ρ_{28}	C^*	$\bar{N}$ mod[a] / $\bar{N}$ fuel	B^2, μB
1.25	0.0259	0.0522	0.842	0.813	1.18	1 157
1.75	0.0232	0.0303	0.434	0.642	—	—
2.50	0.0181	0.0184	0.224	0.553	1.21	883

[a] Neutron density ratio below 0.4 eV. Cross sections calc. by THERMOS.

Reactivity calculations

Two main approaches have been used in the US development of reactivity calculations for D_2O lattices. One approach, exemplified by the ROCLAND A [34, 39, 40, 41], BSQ [42], NDC [43], and IDIOT [44] codes, combines four-factor and diffusion parameter calculations into a two-group treatment. The second approach, exemplified in the Westinghouse CVTR calculations [45–54], uses the standard light water codes with suitable adjustments for the heavy water lattices. Figure 4 compares the two approaches. In general the CVTR codes offer better treatment of neutron energy variations while the ROCLAND type codes offer the better geometrical treatments. The calculations and experiments are compared in Tables 1, 4 and 5 and in Fig. 3. The agreement is generally excellent although enough normalization is implicit in the calculations to require that any large extensions to new types of lattices be again tested against experiments.

Table 4. Representative lattices studied in PLATR [34]

Type of fuel	Elements per cluster	Housing tubes	Coolant	Hexag. lattice pitch (in.)	D_2O purity (%)	k_∞ Meas.	k_∞ Calc.	Comments
Nat. U rods, 0.5" OD, 0.648", C-C spc.	19	None	D_2O	7.50	99.18	1.135	1.133	*a, b, c, d*
	19	None	D_2O	8.46	99.16	1.159	1.157	*a, b, c, d, e, f*
	19	None	D_2O	10.46	99.32	1.186	1.180	*b, c, d, f*
	19	None	D_2O	13.22	99.26	1.148	1.155	
	31	None	D_2O	8.46	98.82	1.096	1.094	
	31	None	D_2O	10.46	99.30	1.165	1.162	*c, d, e, g*
	31	None	D_2O	13.22	98.89	1.130	1.129	
	37	None	D_2O	10.46	99.49	1.144	1.143	
	37	None	D_2O	11.1	99.14	1.144	1.138	*a, c, f, g, h*
	37	None	D_2O	13.22	99.05	1.138	1.136	
Nat. UO_2 tubes	2	Yes	D_2O	7.50	99.38	1.058	1.060	*c, i, j, k, l*
	2	Yes	D_2O	8.46	99.53	1.097	1.100	*c, d, e, j, k, l*
	2	Yes	D_2O	10.46	99.70	1.150	1.152	*c, d, e, i, j, k, l, m*
	2	Yes	D_2O	13.22	99.45	1.130	1.144	*c, d, j, k, l*
Nat. UC rods, 0.5" OD, 0.580", C-C spc.	7	Yes	Air	7.50	99.75	1.213	1.217	*e, n*
	7	Yes	Air	8.46	99.75	1.218	1.224	*e, n*
	19	Yes	Air	8.46	99.75	1.172	1.180	*e, n*
	19	Yes	Air	10.46	99.75	1.210	1.209	*e, f, n*
	19	Yes	Air	13.22	99.75	1.205	1.212	*e, n*
	31	Yes	Air	10.46	99.75	1.176	1.164	*e, o*
	31	Yes	Air	13.22	99.75	1.199	...	*e, o*

[a] Also done with double cylindrical shroud tubes.
[b] Also done with single hexagonal shroud tubes.
[c] Also done with air coolant.
[d] Also done with H_2O coolant.
[e] Also done with organic coolant.
[f] Variations in center-to-center spacing measured.
[g] Also done with single cylindrical shroud tube.
[h] Variations in D_2O purity studied.
[i] Single tube element also measured.
[j] Cylindrical shroud tube and central filler tube.
[k] Void coefficient measured for individual coolant channels.
[l] Calculated by a method similar to ROCLAND A.
[m] 3 and 4 tube elements also measured.
[n] Cluster has single hexagonal shroud tube.
[o] Cluster has single cylindrical shroud tube.

Detailed parameter measurements and calculations

Experience has shown that a variety of calculational assumptions can fit the observed reactivity data. Accordingly it has proven essential to provide measurements and calculations of the detailed lattice parameters or their observable equivalents. The Westinghouse studies for the CVTR have been particularly notable in this respect, comparing parameter measurements made in the LRX critical [45] against many calculational schemes in order to arrive at an optimized model [49].

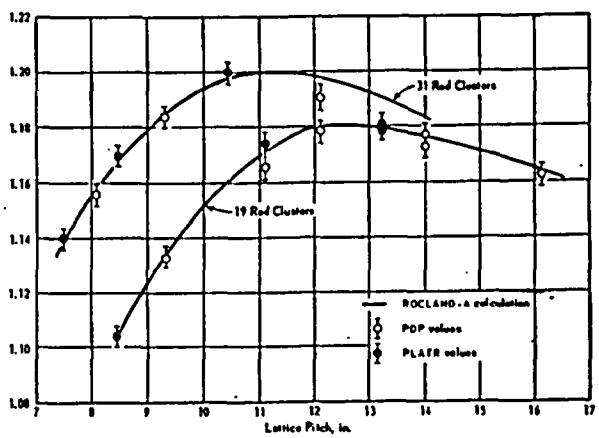


Figure 3. k_∞ vs lattice pitch for bare clusters of natural UO_2 rods in D_2O

In fast-fission measurements the observable quantity is δ_{28}, the ratio of ^{238}U to ^{235}U fissions. It has been measured in the MIT [23], Savannah River [6, 59, 60, 61], and WAPD [45, 47] programs by comparing the gamma activities of enriched and depleted foils exposed in the fuel. The primary experimental problem lies in determining $P(t)$, the ratio of the ^{238}U and ^{235}U fissions to their respective fission product activities. Factors influencing $P(t)$ were measured directly [23, 59, 60]. Also MIT [23, 62] developed an alternative approach based on counting the 1.6-MeV gamma ray of ^{140}La several days after the experiment to determine δ_{28} from the known yields of this isotope in the ^{238}U and ^{235}U fissions. The three-group formulation developed by Fleishman and Soodak [40] has given good results in calculating the fast fission effects, but (as shown by Table 5) MUFT [52] gives poor results probably because of the flat flux assumption. Over-all accuracy of the δ measurements is in the $\pm$3–5 per cent range.

In the resonance energy region the observables are ρ_{28}, the ratio of epithermal to thermal captures in ^{238}U and δ_{25}, the equivalent ratio for ^{235}U fissions. For the heavy water programs, the preferred method of measuring ρ_{28} has been based on cadmium ratio determinations of the 103 keV activity from ^{239}Np [6, 24, 45, 47]. However, Savannah River studies [9] showed that the direct Cd measurements for ρ_{28} could contain large errors. An alternative is to determine the

Table 5. 19-rod CVTR fuel clusters in rectangular lattices in D_2O

Lattice spacing C-C (inches) X	Y	Rod	Type data	δ_{28}	δ_{25}	ρ_{28}	C^*	ϕ mod/ϕ fuel Dy	Sub Cd	B^2 (μB)	M^2 (cm^2)
6.5	6.5	0.57	Meas.	0.050	0.055	0.72	0.752	—	—	—	—
6.5	6.5	0.60	Meas.	0.045	0.0535	0.74	0.747	1.52		770	215
			Calc.	0.016	0.0625	0.79	0.693	1.56		—	—
6.5	6.5	0.65	Meas.	0.044	0.052	0.76	0.764	—		—	—
6.5	8.0	0.60	Meas.	—	0.0483	0.67	—	1.57		782	238
			Calc.	0.013	0.0495	0.64	0.637	1.62		—	254
8.0	8.0	0.60	Meas.	0.045	0.0356	0.493	0.681	1.63		763	274
			Calc.	0.010	0.0396	0.504	0.595	1.69		—	280

Fuel: UO_2 pellets (10.2 g/cm³) 0.430" D × 0.860" L enriched to 1.1 % ^{235}U.
Clad: 6061 Al 0.500" OD, 0.032" wall.
Housing: 6061 Al 4.000" OD, 0.200" wall.
Moderator: 99.27–99.33 mol % D_2O, 20 °C.
Lattice: See Fig. 5.

subcadmium captures in ^{233}U by reference to cadmium ratios for a nearly $1/v$ absorber such as copper. Such problems are minimal in the measurements of the ^{235}U fission product activities to determine δ_{25}. An alternative to using cadmium ratio measurements at all is to determine C^*, the ratio of total ^{238}U captures to total ^{235}U fissions. Sample experimental results for ρ_{28}, δ_{25}, and C^* are given in Tables 1, 3 and 5.

Calculations of ρ_{28} require a detailed geometrical treatment. In the ROCLAND code, collision probability techniques are used with a nine-group energy formulation centered on the wide resonances of ^{238}U [41]. In the CVTR methods, auxiliary Monte Carlo calculations are made with the REPLICA code [46] to provide the self-shielding factor for MUFT. Both approaches match the measurements within the rather large experimental uncertainties, which average ±3–4 per cent excluding systematic errors due to cadmium. However, the MUFT-REPLICA calculations contain an implicit assumption that the slowing-down flux is flat over the lattice cell, and they may be expected to break down at very wide fuel pitches. Thus Table 1 shows ρ_{28} values as high as 0.368 for isolated fuel assemblies, whereas the flat flux assumption would require these values to tend towards zero.

All of the D_2O lattice programs have included thermal flux traverses. The observables are the responses of activating foils or other detectors, preferably with a variety of spectral characteristics, at different points in the lattice. It has been the conclusion of most of the US lattice programs that the best way to use this data is in comparisons with detailed multigroup

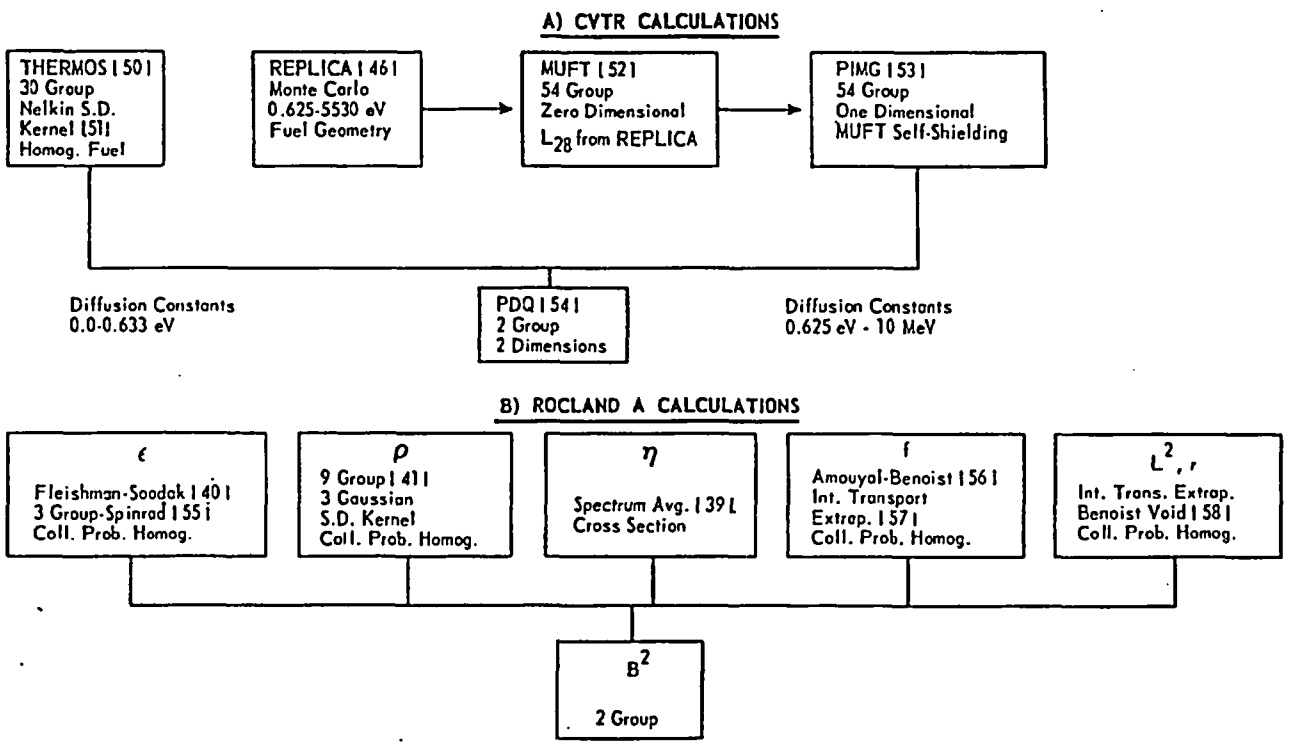


Figure 4. Block diagrams of D_2O lattice calculations

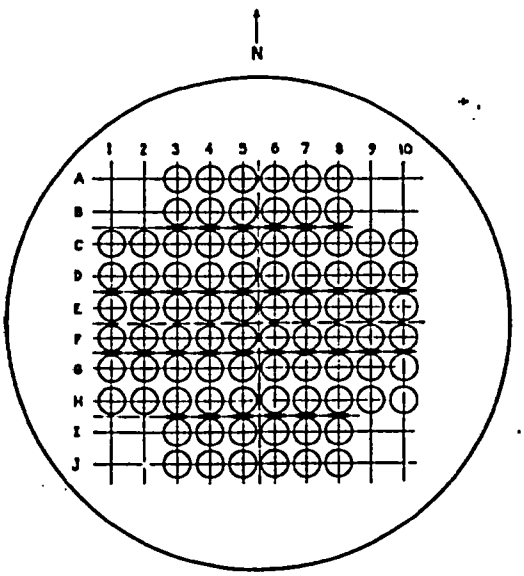
calculations. The THERMOS code [50], originally developed at MIT, is especially useful for this purpose. This code has provided excellent agreement with a wide variety of D_2O lattice traverses [3, 9, 25, 45, 47], although there is some evidence of systematic discrepancies of about 2 per cent in the moderator-to-fuel flux ratios. In the smaller lattice cells a two-dimensional or a modified THERMOS [63] has been used to avoid errors associated with cylindricizing the lattice cell [25, 64]. The preferred scattering kernel for these calculations is presently the Honeck modification of the Nelkin kernel [51]. However, MIT studies [64] comparing THERMOS calculations made with the Nelkin-Honeck, Brown-St. John, and Wigner-Wilkins kernels showed maximum variations of less than 3 per cent in any of the observable quantities.

^{176}Lutetium has been widely used as a thermal spectrum indicator both in comparison with THERMOS calculations and for neutron temperature determinations based on the Westcott formulism [65]. Detailed comparisons of the relative response of ^{176}lutetium and a $1/v$ detector in a purely Maxwellian spectrum were made at Hanford Laboratories [66], over a range of neutron temperatures and correction techniques [67], they were developed for subtracting the epithermal response of these foils in actual lattice measurements. Typical neutron temperature determinations in the CVTR criticals gave moderator values near the physical lattice temperatures but fuel values some 70° to 80° higher [47].

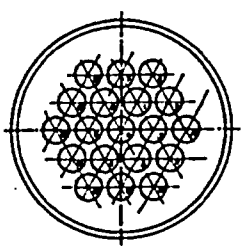
The thermal neutron distribution measurements are ordinarily combined with the appropriate cross sections and diffusion coefficients to provide values for the thermal diffusion area. The diffusion parameter most directly obtainable in the D_2O lattice measurements is the over-all migration area, M^2. This has been measured in the critical facilities at Savannah River [7, 8, 9] and Westinghouse [45, 47] by comparing buckling and k_{eff} changes through the two-group critical equation. In Tables 1 and 5 the actual measurements were of water height changes versus periods. Data have been obtained at Hanford Laboratories using this technique for irradiated lattices [20, 68, 69]. Calculational codes for evaluating the migration areas are the same as those discussed earlier for the bucklings.

Special lattice studies

A particular problem is met in the diffusion measurements and calculations whenever the lattices contain voids, since in these cases the diffusion coefficients will be different along and against the direction of the voids. The simplest method of observing the effects compares water height versus period measurements with and without void in the lattice. These measurements give the anisotropic migration area in the direction in which the water height changes. At Savannah River the Persson perturbation analysis [12] was used to determine the changes in diffusion coefficients resulting from creating voids at various locations in the test fuel assemblies. Results from both methods are shown in Table 1 [7, 13]. A study of voided UO_2 tubes by the Persson method [10] has



TOP VIEW OF CORE



CROSS SECTION OF A FUEL CLUSTER (SEE TABLE 5)

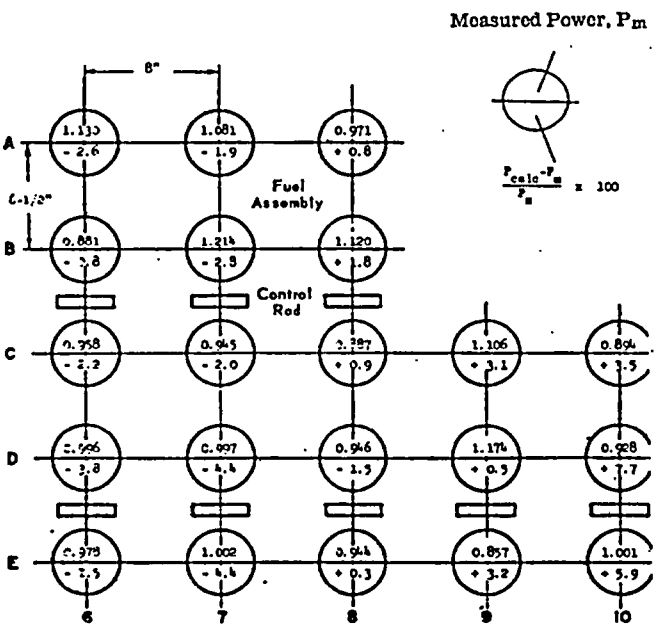


Figure 5. Measured power distributions in CVTR critical experiment compared with PDQ calculations

given satisfactory agreement with Benoist [58] calculations for void fractions up to 10 per cent of the cell.

Most of the power reactor lattices are extremely heterogeneous both in using large and widely spaced fuel assemblies and in using many different types of fuel assemblies and other lattice components. Studies with the CVTR (Fig. 5) and the very heterogeneous HWCTR reactor [70] have shown that the PDQ code can give good results for these lattices although some normalization is involved in the way the PDQ grid is chosen. Effort has also been put into the development of heterogeneous codes as exemplified by HERESY 1 [71] and 2 [72], and KERNMAT [73]. These codes take input data from the lattice parameter codes already discussed to determine neutron absorption and production coefficients for each lattice component. The neutron flux in the lattice is then calculated in terms of four kernels which give the slowing down and thermal diffusion distributions from the lattice components considered as line sources and sinks. HERESY 1 and KERNMAT are essentially three-group methods allowing for resonance absorption at a single energy; HERESY 2 permits a multigroup resonance treatment. Tests at Savannah River have shown good agreement between these codes and uniform lattice bucklings [10]. However, with actual heterogeneous lattices difficulties are experienced in defining the cells used to generate the input parameters. Such calculations are improved [74] by an iteration technique in which cell sizes determined from the zero current boundaries in the lattice calculations are fed back into the parameter calculations.

The effective delayed neutron fraction β must be considered a variable in a D_2O lattice. In part this is because these lattices operate with a variety of fissionable materials and in part because they have varying escape probabilities for the high energy γ's producing delayed photoneutrons. Measurements of β have been made in the Savannah River [75] and Westinghouse programs [45, 47] by comparing the kinetic response of the reactor to known changes in reactivity. The Westinghouse results indicated a γ escape factor of 0.10 for the CVTR lattice. Escape factors of 0.46 and 0.99 were obtained [76] from analogous experiments on the 19-rod UO_2 and Pu–Al fuel assemblies used in the PRCF and PRTR. The closely related quantity β/l, the ratio of the delayed neutron fraction to the neutron lifetime, was measured by transfer-function studies in the PRTR and by noise analysis and step reactivity changes in the PRCF to be 6.6 ± 0.6 sec^{-1} for the PRTR lattice [69, 77].

Most of the lattice measurements discussed have concerned natural and slightly enriched uranium systems. Limited work has also been done with other systems. The Hanford program, designed to investigate burn-up [78, 79] and plutonium recycle systems, included zero power critical studies of these lattices in the PRTR [69, 80, 81] plus the development of a variety of burn-up codes such as RBU [82] and MELEAGER [21]. Counterpart Westinghouse codes are CANDLE [83] and TURBO [84]. Investigations on thorium–^{235}U fueled D_2O reactors have been reported from the Argonne National Laboratory [85, 86, 87]. Finally, Savannah River is irradiating thorium and plutonium to produce some 120 kg of ^{233}U and 3 kg of ^{244}Cm and has reported studies of byproduct production of ^{232}U in the ^{233}U [88].

OPERATIONAL CHARACTERISTICS OF D_2O REACTORS

Means for incorporating the lattice physics information discussed above into the design of the fuel lattices for actual D_2O reactors are conveniently provided by techniques such as Savannah River's FAD code [89]. This code automatically designs tubular fuel assemblies by iterating BSQ physics calculations against engineering and economic calculations until an optimum is achieved under the input parameters of the problem.

The next stage is to add control capabilities to the lattice. The PRTR uses moderator level control with automatic sensors to maintain the reactor power within ± 0.25 MW at 70 MW output [90]. However, most US designs of D_2O power reactors use multiple control rods. (The PRTR has multiple shim rods.) The purpose is to provide xenon override capabilities and to allow for flux shaping in addition to criticality and shutdown control. Flux shaping is particularly important in the large D_2O power reactors because these reactors can develop large flux tilts which lead to power losses and xenon oscillations [91]. (In a typical experiment in the PDP a uniform displacement of the lattice by 0.4 cm in the total 495 cm diam produced a flux tilt of 7 per cent.) The methods discussed earlier for reactivity evaluations of the uniform lattices have proven adequate for determining control rod worths [9, 48, 61, 92]. Further, studies at Savannah River have shown that even one-group diffusion theory gives good results in computing the flux shaping produced by the control rods if the measured lattice cell bucklings are used as input. In more advanced calculations the four-group two-dimensional PDQ code [54] has matched the complex flux distributions in the HWCTR to within 3 per cent radially and 5 per cent axially [93].

Interesting physics parameters of the D_2O lattices are encountered in evaluating the operational coefficients. Such studies have been made both in the conventional zero-power experiments and in the actual operating D_2O reactors such as the CVTR, the HWCTR, and the PRTR. For example the measured moderator temperature coefficients for the PRTR and PRCF as determined from period measurements in the zero-power, zero-irradiation startup lattices were $(-8.8 \pm 0.3) \times 10^{-5}\ \Delta k/k(°C)^{-1}$ and $(-24.6 \pm 0.5) \times 10^{-5}\ \Delta k/k(°C)^{-1}$. Most of the measurements have involved either moderator heating or uniform lattice heating. However, the fuel temperature coefficients for the UO_2 and Pu–Al nineteen rod fuel assemblies [16] for the PRTR have been measured in the PCTR as $(-2.3 \pm 0.2) \times 10^{-5}(°C)^{-1}$ and $(-0.7 \pm 0.3) \times 10^{-5}(°C)^{-1}$ respectively [94]. In general both the ROCLAND and CVTR calculational techniques have proven adequate to handle the temperature coefficient calculations [47, 95].

Tables 1 and 2 show the bucklings of representative D_2O lattices with various fuel coolants. The effects influencing the measured reactivity changes are quite involved, but for lattices near the optimum moderator-to-fuel ratio the reactivity of the lattices with different coolants is in the order D_2O, gas, organic, H_2O. Thus the effects of void creation by coolant boiling is usually (but not always) negative for the D_2O coolants and positive for the organic and H_2O coolants. Similarly the coefficient for an inleakage of H_2O into the D_2O is negative in all but the most undermoderated lattices, but the effects of a D_2O leak into an organic or H_2O coolant would be expected to be positive. These variations have been studied as important safety coefficients in the D_2O reactors. Thus for the PRTR a value of -16×10^{-3} $\Delta k/k$ has been measured for total loss of coolant from the startup core [81]. The effects of the introduction of light water into the moderator have been measured in detail at Savannah River [3, 6, 30].

Calculations of the void and coolant interchange effects provide a severe test of the D_2O lattice calculation schemes. The agreement with experiment has been only fair (Table 1 [34, 47]). Streaming effects have been handled by the methods of Behrens [96] and Benoist [58]. As evidenced by the earlier anisotropy discussions, these methods are probably reasonably adequate, so the difficulty lies directly in the calculation of the lattice parameters, in particular p and η. The problems of void location have been treated by two- or three-group perturbation theory at Hanford [69, 97], Savannah River [98], and Westinghouse [45, 47] with uniformly good results.

The final stage of the physics evaluations of the D_2O reactor lattices has been to determine the over-all reactor stability. Transfer function measurements have usually proved unsuccessful in the large D_2O reactors because variations in moderator circulation patterns usually mask most other effects [99]. A more successful approach has been to combine the measured physics coefficients and other reactor parameters into an analytical transfer function. A new IBM 704 code LASS [100] has just been completed at Savannah River for comprehensive linearized analyses of D_2O reactor stability. The effects of xenon oscillations can be incorporated into this analysis by use of a second code OX [101].

True accident analysis of the D_2O reactors requires the use of non-linearized equations to follow serious reactor excursions. An alternative is being explored in the SPERT II [102] experiments which involve shutdown studies of heavy water reactors of the CP-5 type [103]. Shutdown mechanisms for initial asymptotic periods between 50 and 500 μs correlate with the formation of steam voids.

CONCLUSIONS

Although much work remains to be done in detailed reactor design and on determining the parameters for irradiated lattices and lattices operating on fuel cycles other than $^{235}U/^{238}U$, the studies described in this report are felt to provide a sufficient basis for the conceptual physics design of the present generation of D_2O power reactors in the US.

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ABSTRACT—RÉSUMÉ—АННОТАЦИЯ—RESUMEN

A/268 Etats-Unis d'Amérique

Études de réseaux et expériences critiques sur des ensembles modérés à l'eau lourde

par J. L. Crandall et *al.*

Au cours des cinq dernières années, les expériences critiques et exponentielles faites à Brookhaven, à Hanford, au Massachusetts Institute of Technology, à Savannah River et aux laboratoires de Westinghouse ont fourni une base expérimentale solide pour la physique des réacteurs concernant les réseaux uranium – eau lourde. Les efforts ont porté principalement sur la mesure des laplaciens de plus de 200 réseaux différents, représentant une vaste gamme de types de réseaux et de formes du combustible. On a également mis au point des méthodes d'activation pour mesurer certains paramètres: rapport du nombre de fissions

neutrons rapides/neutrons lents, capture de neutrons de résonance, taux de conversion, distribution spectrale des neutrons. Toutefois, les mesures par activation ont été appliquées à un plus petit nombre de réseaux que les mesures de laplaciens. Les études les plus récentes ont été notamment consacrées aux problèmes posés par les charges hétérogènes et les effets anisotropiques.

Les expériences critiques et exponentielles ont été complétées par des études de substitution, tant du type classique que du type à réactivité nulle appliqué à Hanford et par United Nuclear. Les mesures de substitution sont particulièrement utiles du fait qu'elles fournissent des valeurs différentielles et qu'elles permettent l'étude d'un réseau pour lequel on ne dispose que de quelques échantillons de combustible. L'assemblage critique de recyclage du plutonium (PRCF) est une nouveauté intéressante parmi les appareils de mesures de substitution qui permettent d'étudier le combustible irradié à Hanford. La méthode des neutrons pulsés dans les réacteurs à eau lourde servait d'abord essentiellement à l'étude de l'eau lourde elle-même, mais on l'utilise maintenant aussi pour les réseaux, en particulier dans les centres universitaires (Massachusetts Institute of Technology, Université d'Etat de Caroline du Nord et Université de Floride). Une installation expérimentale unique en son genre est le réacteur SPERT II, qui sert aux études d'arrêt et d'états transitoires dans les réseaux à eau lourde.

Bien que la plus grande partie des recherches aux Etats-Unis aient été concentrées jusqu'ici sur les réseaux à uranium, quelques travaux en cours portent sur les réseaux à plutonium et à thorium. On étudie les réseaux plutonifères pour obtenir des renseignements sur les effets de la formation de plutonium dans les réacteurs à uranium naturel, sur les réacteurs à recyclage du plutonium et sur l'irradiation du plutonium en vue d'obtenir des transuraniens. Les réseaux à thorium et eau lourde sont étudiés à la fois pour les réacteurs surgénérateurs à thorium et pour la production d'uranium 233.

Avec cette abondance de données expérimentales, la plupart des programmes de calculatrices servent essentiellement à faire des interpolations. Plusieurs méthodes de calcul électronique ont été mises au point pour l'étude détaillée de certains problèmes de physique des réacteurs à eau lourde. On peut citer divers programmes de Monte-Carlo pour l'étude des processus de ralentissement des neutrons et de capture par résonance, le programme THERMOS, établi par Honeck pour l'étude des effets du spectre des neutrons thermiques, et le programme HERESY, élaboré par Klahr pour l'étude d'effets hétérogènes. On a obtenu également de bons résultats en adaptant aux réacteurs à eau lourde les méthodes de calcul appliquées pour les réacteurs à eau légère; c'est notamment le cas des programmes CANDLE et PDQ de Westinghouse.

Comme les données concernant les réseaux à eau lourde se traduisent par la construction d'un nombre toujours croissant de réacteurs, on a combiné les données de physique aux données technologiques et économiques pour les calculs des caractéristiques techniques et des coûts; c'est le cas, par exemple, du programme FAD de Savannah River. Les aspects opérationnels des réseaux ont également une grande importance pour l'évaluation des dispositifs de commande, des possibilités d'aplatissement du flux et des coefficients d'exploitation du réacteur. On a mis au point des programme spéciaux, comme OX et LASS, qui combinent toutes ces données pour permettre des études de stabilité. La plupart des données expérimentales sur les coefficients d'exploitation proviennent des expériences à puissance nulle; toutefois, on obtient maintenant des renseignements importants grâce aux dives réacteurs à eau lourde qui sont en service aux Etats-Unis, notamment les réacteurs plutonigènes de Savannah River, le réacteur expérimental des Carolines et de la Virginie (CVTR), le réacteur d'essais technologiques à eau lourde (HWCTR), le réacteur d'essai à recyclage du plutonium (PRTR) et des réacteurs de recherches comme le CP-5 et le MITR.

A/268 США

Исследования решеток и критические опыты в системах с тяжеловодным замедлителем

Дж. Крандал *et al.*

Критические и экспоненциальные опыты, проведенные за последние пять лет в лабораториях в Брукхейвене, Ханфорде, Массачусетском технологическом институте, в Саванна-Риверской лаборатории и в лабораториях фирмы «Вестингауз» заложили основу для физики реакторов на уране и тяжелой воде. В опытах главным образом предусматривали измерение лапласиана. Было исследовано свыше двухсот решеток различного типа и форм топлива. Разработаны активационные методы для измерения таких величин, как отношение сечений деления на быстрых и тепловых нейтронах, резонансный захват нейтронов, коэффициенты воспроизводства и спектр нейтронов. Лапласиан измерялся с большим количеством решеток по сравнению с количеством решеток, на которых применялись активационные методы. В последнее время исследуются гетерогенные загрузки и влияние анизотропии.

Данные, полученные в результате критических и экспоненциальных опытов, были дополнены обычными стандартными исследованиями и измерениями при нулевой реактивности, которые проводились в Ханфорде и в лабораториях фирмы «Юнайтед ньюклеар». При ограниченном количестве образцов топлива для вывода дифференциалов или исследования решеток практически полезными являются методы замещения. Новым интересным добавлением к устройствам, в которых употребляются методы замещения, является критическая установка с повторным плутониевым циклом (PRGF)

для изучения облученного топлива в Ханфорде. Импульсная нейтронная методика для тяжеловодных систем, используемая главным образом при изучении тяжелой воды, в последнее время применяется и для исследований решеток, которые проводятся по университетской программе Массачусетским технологическим институтом и университетами штатов Северная Каролина и Флорида. Уникальным экспериментальным устройством является реактор SPERT-II, который используется для изучения остановки и переходных процессов тяжеловодных решеток.

Наряду с разработкой урановых систем в США в настоящее время проводятся опыты с плутониевыми и ториевыми системами. Решетки с плутонием изучаются для исследования эффектов накопления плутония в реакторах на природном уране, данных о системах с повторным плутониевым циклом и об облучении плутония для производства трансурановых элементов. Системы D_2O — Th изучаются с точки зрения ториевых реакторов-размножителей и производства U^{233}.

При наличии большого количества экспериментальных данных многие вычислительные программы являются главным образом рекомендациями для интерполяции экспериментальных результатов. Разработано также несколько методов расчета для более детального исследования частных вопросов физики тяжеловодных реакторов. Примерами программ вычислений по методу Монте-Карло для изучения процесса замедления и резонансного захвата являются программа THERMOS, разработанная Хонеком, для исследования спектра тепловых нейтронов и программа HERESY, разработанная Кларом, для изучения гетерогенных эффектов. Хорошие результаты также были получены при использовании методов расчета для обычной воды в расчетах тяжеловодного реактора, в частности такие программы, как GANDLE, и программы, разработанные фирмой «Вестингауз».

Разработанный в Саванна-Ривере код (FAD) является примером того, как полученные для тяжеловодных решеток экономические и технические данные могут быть использованы при проектировании и оптимизации стоимости реакторов. Большое внимание уделяется вопросам эксплуатации решеток, в частности оценке регулирующих систем, возможностей формирования потоков и эксплуатационных характеристик реактора. Для изучения устойчивости реактора на основе комбинации данных разработаны специальные программы OX и LASS. Большинство данных об эксплуатационных характеристиках было получено в опытах при нулевой мощности. Однако много данных теперь поступает от разных действующих в США тяжеловодных реакторов, включая производственные реакторы в Саванна-Ривере, опытный реактор фирмы «Каролинас-Вирджиния», опытный тяжеловодный реактор для испытания узлов, опытный реактор с повторным плутониевым циклом и исследовательские реакторы, такие, как CP-5 и MITR.

A/268 Estados Unidos de América

Estudio de redes y experimentos críticos en sistemas moderados por D_2O

por J. L. Crandall et al.

Durante los últimos cinco años las experiencias críticas y exponenciales realizadas en los laboratorios de Brookhaven, Hanford, Massachusetts Institute of Technology, Savannah River, y Westinghouse, han suministrado una firme base experimental a la física de reactores de uranio – D_2O. Se ha prestado la mayor atención a las medidas de laplaciana, habiéndose estudiado más de 200 redes diferentes que abarcan una amplia gama de tipos de reticulado y formas de combustible. Se han desarrollado también técnicas de activación con el fin de medir magnitudes tales como las razones de fisión rápida a térmica, captura por resonancia, razones de conversión y distribuciones espectrales. Sin embargo, el conjunto de redes a las que se les ha aplicado las técnicas de activación ha sido menor que aquel en que se han realizado medidas de laplaciana. Los estudios más recientes se refieren fundamentalmente a los problemas que plantean los sistemas heterogéneos y los efectos de anisotropía.

A los datos de los experimentos críticos y exponenciales deben añadirse los estudios de sustitución realizados tanto según el método clásico, como según la variante de reactividad nula, tal como se ha practicado en Hanford y United Nuclear. Las medidas de sustitución son particularmente útiles tanto para suministrar resultados por diferencia, como en el caso de disponer de una cantidad limitada de combustible a ensayar. Una nueva e interesante contribución a las instalaciones de sustitución es la instalación Plutonium Recycle Critical Facility (PRCF) que tiene por finalidad estudiar combustible irradiado en Hanford. Las técnicas de fuente pulsante en sistemas de D_2O encontraron su principal aplicación en el estudio del propio D_2O, pero ahora se está aplicando también para estudiar redes, principalmente en los programas del MIT, y de las universidades del estado de Carolina del Norte y de Florida. Una instalación experimental particularmente notable es el reactor SPERT 2, se está utilizando en estudios de transitorios y de parada en redes de D_2O.

Aunque hasta la fecha la mayoría de los trabajos llevados a cabo en los Estados Unidos se han concretado a sistemas de uranio, actualmente se están iniciando algunos trabajos relativos a sistemas de plutonio y torio. Las redes de plutonio se estudian a fin de obtener información sobre los efectos de acumulación del plutonio en reactores de uranio natural, sobre el ciclo del plutonio y sobre irradiaciones de plutonio para producir elementos transuránicos. Los

sistemas de D_2O – torio se estudian tanto con vistas a los reactores reproductores de torio, como a la producción de ^{233}U.

En razón a la enorme información experimental de que se dispone, la mayoría de los programas de cálculo son fundamentalmente formularios destinados a interpolar entre puntos experimentales. Otros métodos de cálculo se han diseñado con el fin de examinar detalladamente aspectos específicos de la física de los reactores de D_2O. Ejemplos son: los diversos programas de Monte Carlo, que estudian el proceso de la moderacion y la captura por resonancia, programa THERMOS, desarrollado por Honeck para examinar el espectro térmico, y el programa HERESY, desarrollado por Klahr para examinar efectos de heterogeneidad. Se han obtenido buenos resultados por la Westinghouse, con los programas CANDLE y PDQ, adoptando los métodos de cálculo para el agua ligera a los reactores de D_2O.

A medida que la información sobre redes de D_2O se ha ido incorporando a los proyectos, los datos físicos se han combinado con los técnicos y económicos en los cálculos de diseño y optimización de costos. Tal es el caso del programa FAD de Savannah River. Considerable ha sido también la importancia concedida a los aspectos operacionales del diseño de reactores en la evaluación de los sistemas de control de las posibles formas del flujo y de los coeficientes operacionales del reactor. Algunos programas especiales, como OX y LASS, se han desarrollado con el fin de combinar estos datos con vistas a los estudios de estabilidad. La mayor parte de la información experimental que existe sobre los coeficientes operacionales proviene de experimentos a potencia cero; no obstante, actualmente se está obteniendo mucha información merced a los diferentes reactores de D_2O que funcionan en los Estados Unidos, como los reactores de producción de Savannah River, el Carolinas Virginia Test Reactor, el Heavy Water Components Test Reactor y el Plutonium Recycle Test Reactor y los reactores de investigación CP-5 y MITR.