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Lattice studies and critical experiments in D_2O moderated systems

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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

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Table I. Studies of one region D_2O lattices in the PDP 126-138

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.	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, cm²	D_2/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 I. 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.				Triang. lattice pitch (in.)	Coolant	Measured B^2, μB			Calc. B_m^2 BSQ μB	M_2^2 Kinetics, (cm^2)
Uranium		6063 Al								
OD	ID	OD	ID			B_1^2	B_2^2	B_m^2		
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	—
				9.33	Air	378	212	(590)	(543)	—
				12.12	D_2O	214	330	544	539	—
2.120	1.760	—	—	12.12	Air	213	386	(599)	(593)	—
3.500	2.860	3.890	3.640	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].

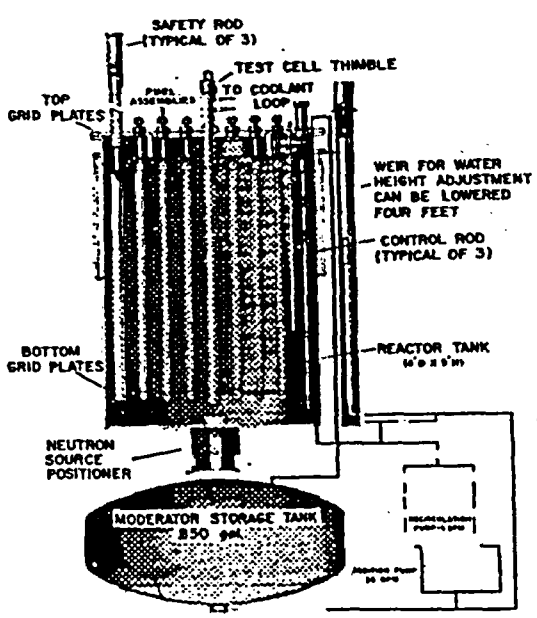


Figure 1. Plutonium recycle critical facility

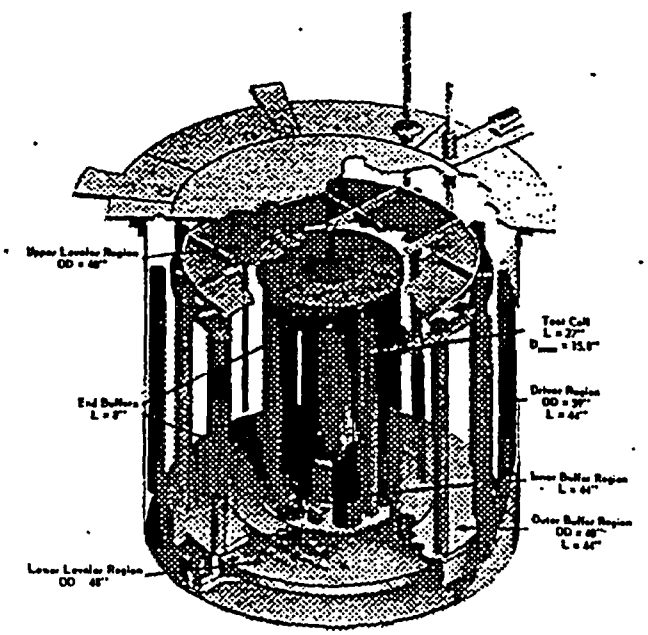


Figure 2. Artist's conception of PLATR

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^*	$\frac{\bar{N}\ \text{mod}^a}{\bar{N}\ \text{fuel}}$	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^*	$\frac{\bar{N}\ \text{mod}^a}{\bar{N}\ \text{fuel}}$	B^2, μB
1.25	0.0259	0.0522	0.842	0.813	1.18	1157
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].

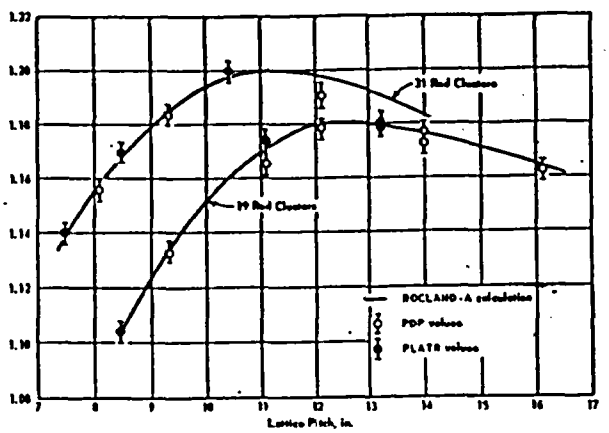


Figure 3. k_∞ vs lattice pitch for bare clusters of natural UO_2 rods in D_2O

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/cm^3) 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 ^{238}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

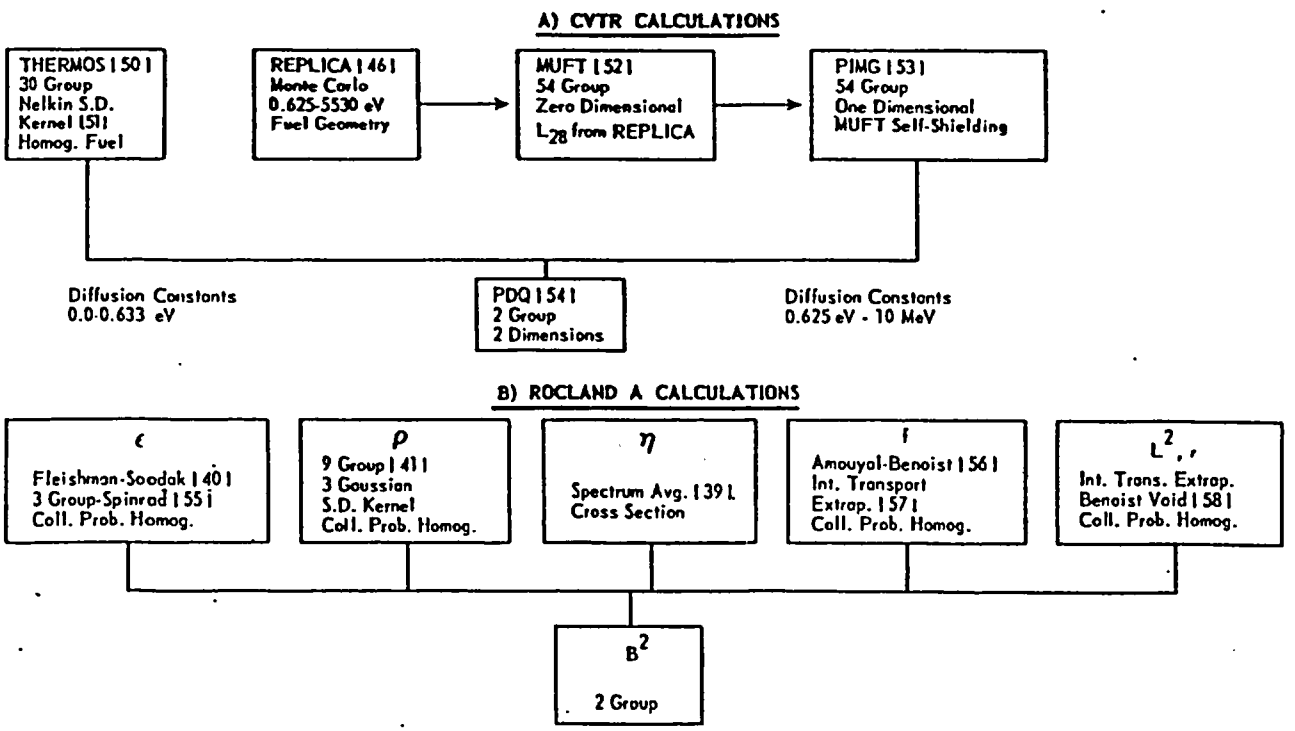


Figure 4. Block diagrams of D_2O lattice calculations

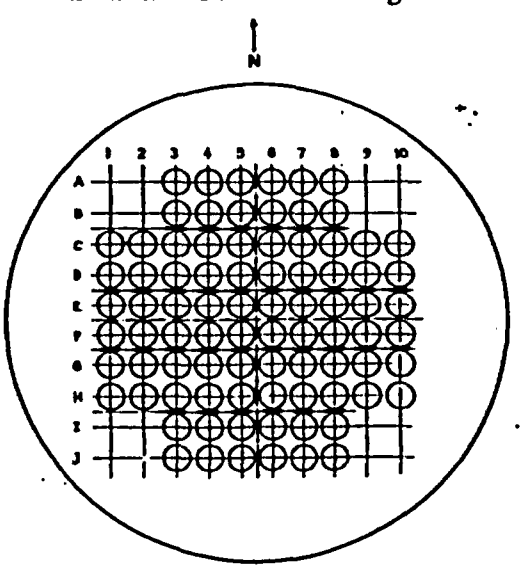
calculations. The THERMOS code [50], originally developed at MIT, is especially useful for this purpose. This code has provided excellent agreement with a wide variety of D_2O lattice traverses [3, 9, 25, 45, 47], although there is some evidence of systematic discrepancies of about 2 per cent in the 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].

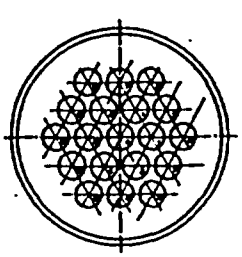
The thermal neutron distribution measurements are ordinarily combined with the appropriate cross sections and diffusion coefficients to provide values for the thermal diffusion area. The diffusion parameter most directly obtainable in the D_2O lattice measurements is the over-all migration area, M^2. 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)

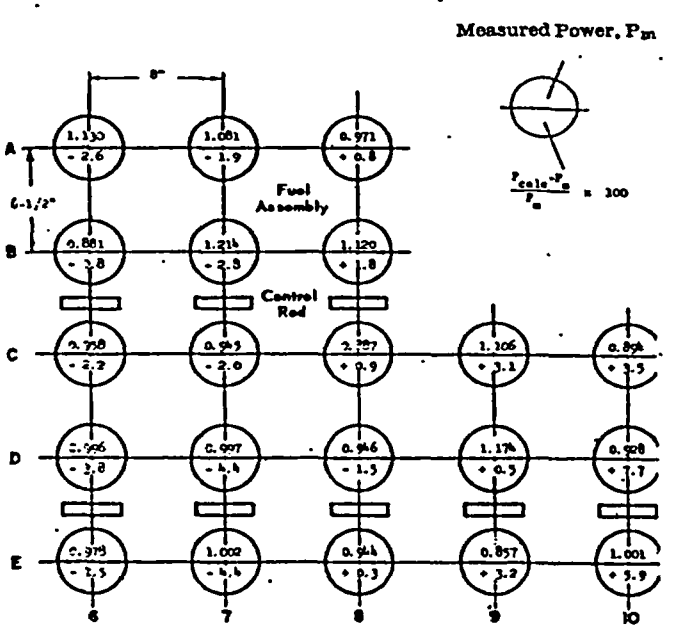


Figure 5. Measured power distributions in CVTR critical experiment compared with PDQ calculations

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.

REFERENCES

1. USAEC report TID-8200 (8th Rev.) (1963).
2. USAEC report CVNA-90 Vol. I–IV (1961). Also CVNA-40.
3. Crandall, J. L., USAEC report DP-787 (1962).
4. Dunklee, A. E., USAEC report DP-567 (1961).
5. Hurley, T. J., Jr., *et al.*, Nuclear Sci. Eng., *12*, 341–347 (1962).
6. Hurley, T. J., Jr., USAEC report EP-777 (1963).
7. Graves, W. E., *et al.*, Nuclear Sci. Eng., *16*, 186–195 (1963).
8. Dunklee, A. E., and Graves, W. E., Trans. Amer. Nuclear Soc. *3*, 244–245 (1963).
9. Crandall, J. L., Heavy Water Lattices Second Panel Report, IAEA Vienna 503–549 (1963). Also published as USAEC report DP-833.
10. Graves, W. E., and Hennelly, E. J., Symposium on Exponential and Critical Experiments, IAEA Vienna. To be published.
11. Baumann, N. P., USAEC report DP-407 (1959).
12. Persson, R., Symposium on Exponential and Critical Experiments, IAEA Vienna. To be published.
13. Graves, W. E., USAEC report DP-832 (1963).
14. Benton, F. D., USAEC report DP-873 (1964).
15. Anderson, J. K., and Winegardaer, W. K., USAEC report HW-69168 (1962).
16. Sharp, R. E., USAEC report HW-69200 Pt 2 (1961).
17. Wittenbrock, N. G., *et al.*, USAEC report HW-61236 (1959).
18. Uotinen, V. O., and Schmid, L. C., USAEC report HW-79960 (1963).
19. Uotinen, V. O., and Schmid, L. C., USAEC report HW-81435 (1964).
20. Schmid, L. C., and Bennett, R. A., USAEC report HW-81436 (1964).
21. Triplett, J. R., and Busselman, G. J., USAEC report HW-68100 (1961).
22. Kaplan, I., *et al.*, Symposium on Exponential and Critical Experiments, IAEA Vienna. To be published.
23. Wolberg, J. R., *et al.*, USAEC report NYO-9661 (1962).
24. Weitzberg, A., *et al.*, USAEC report NYO-9659 (1962).
25. Brown, P. S., *et al.*, USAEC report NYO-10205 (1962).
26. Palmedo, P. F., *et al.*, USAEC report NYO-9660 (1962).
27. Kim, H., MS Thesis MIT (June 1963).
28. Harrington, J., MS Thesis MIT (July 1963).
29. Peak, J., *et al.*, USAEC report NYO-10204 (1962).
30. Wingfield, E. C., and Hennelly, E. J., Nuclear Sci. Eng., *12*, 348–58 (1962).
31. Wood, D. E., USAEC report HW-70716 (1961). Also Trans. Amer. Nuclear Soc. *4*, 284 (1961).
32. Heineman, R. E., *Experience in the use of the physical constants testing reactor*, Proceedings of the Second International Conference on the Peaceful Uses of Atomic Energy, P/1929, Vol. 12, p. 650, United Nations (1958).
33. Schamberger, R., and Foster, G., USAEC report NDA-2109-3 (1959); NDA-2131-9 (1960).
34. Brooks, W. L., *et al.*, USAEC report UNC-5062 (1963).
35. Brooks, W. L., *et al.*, USAEC report NDA-2131-43 (1961); UNC-5019 (1962); UNC-5020 (1962); UNC-5022 (1962); UNC-5051 (1963).
36. Lanning, D. D., Ph.D. Thesis MIT (May 1963).
37. Westfall, R. L., and Waltner, A. W., Trans. Amer. Nuclear Soc. *5*, 387 (1962).
38. Parkinson, T. F., *et al.*, Symposium on Exponential and Critical Experiments, IAEA Vienna. To be published.

39. Soodak, H., and Sullivan, R., USAEC report NDA-2131-38 (1961).
40. Fleishman, M. R., and Soodak, H., Nuclear Sci. Eng., 7, 217-227 (1960).
41. Brooks, W. L., and Soodak, H., USAEC report NDA-2131-19 (1960).
42. Driggers, F. E., Heavy Water Lattices Second Panel Report, IAEA Vienna 551-572 (1963).
43. St. Paul, R. G., and Raje, S. A., USAEC report CVNA-146 (1962).
44. Richey, C. R., USAEC report HW-63411 (1960).
45. Jedruch, J., and Hamilton, G. N., USAEC report CVNA-114 (1962).
46. Dannels, R. A., and Hendley, S. M., USAEC report CVNA-130 (1962).
47. Jedruch, J., *et al.*, USAEC report CVNA-133 (1962).
48. Hecker, H. C., Jr., *et al.*, USAEC report CVNA-134 (1963).
49. Hecker, H. C., Jr., *et al.*, USAEC report CVNA-147 (1963).
50. Honeck, H. C., USAEC report BNL-5826 (1961).
51. Nelkin, M., Phys. Rev. *119*, 791 (1960).
52. Bohl, H., Jr., *et al.*, USAEC report WAPD-TM-72 (1957).
53. Bohl, H., Jr., *et al.*, USAEC report WAPD-TM-135 (1958).
54. Bilodeau, G. G., *et al.*, USAEC report WAPD-TM-70 (1957).
55. Spinrad, B. I., Nuclear Sci. Eng., 7, 217-227 (1960).
56. Amouyal, A., *et al.*, J. Nuclear Energy *6*, 79 (1957).
57. Kushneriuk, S. A., and McKay, C., AECL report AECL-137 (1954).
58. Benoist, P., CEA report SPM-522 (1958). Also SPM-710 (1962).
59. Futch, A. H., Jr., Nuclear Sci. Eng., *5*, 61-67 (1959).
60. Kinard, F. E., and Baumann, N. P., Trans. Amer. Nuclear Soc. *3*, 23-24 (1960).
61. Baumann, N. P., USAEC report DP-640 (1961).
62. Wolberg, J. R., *et al.*, Symposium on Exponential and Critical Experiments, IAEA Vienna. To be published.
63. Honeck, H. C., Trans. Amer. Nuclear Soc. *5*, 350 (1962).
64. Simms, R., *et al.*, USAEC report NYO-10211 (1963).
65. Westcott, C. H., *Effective cross sections and cadmium ratios for the neutron spectra of thermal reactors*, Proceedings of the Second International Conference on the Peaceful Uses of Atomic Energy, P/202, Vol. 16, p. 70, United Nations (1958).
66. Schmid, L. C., and Stinson, W. P., USAEC report HW-66319 (1960).
67. Schmid, L. C., USAEC report HW-81438 (1964).
68. Uotinen, V. O., and Schmid, L. C., USAEC report HW-81397 (1964).
69. Triplett, J. R., *et al.*, USAEC report HW-61900BA (1961).
70. Arnett, L. M., *et al.*, USAEC report DP-600 (1961).
71. Klahr, C. N., *et al.*, USAEC report NYO-2678 (1960).
72. Klahr, C. N., *et al.*, USAEC report NYO-2679 (1960).
73. Raje, S. A., USAEC report CVNA-107 (1961).
74. Suich, J. E., Trans. Amer. Nuclear Soc. *7* (1964).
75. Finn, B. S., Nuclear Sci. Eng., *7*, 369-76 (1960).
76. Schmid, L. C., USAEC report HW-79054 (1963).
77. Stinson, W. P., and Schmid, L. C., USAEC report HW-81437 (1964).
78. Henderson, W. B., and Dawson, F. G., USAEC report HW-77935 (1963).
79. Christensen, D. E., *et al.*, USAEC report HW-80067-RD (1964).
80. Triplett, J. R., *et al.*, Trans. Amer. Nuclear Soc. *4* (1961).
81. Triplett, J. R., and Peterson, R. E., Symposium on Power Reactor Experiments, IAEA Vienna 213 (1962).
82. Triplett, J. R., *et al.*, USAEC report HW-70049 (1961).
83. Marlowe, O. J., and Ombrellaro, P. A., USAEC report WAPD-TM-53 (1957).
84. Gelbard, E., *et al.*, USAEC report WAPD-TM-95 (1957).
85. Thie, J. A., *Heavy Water Exponential Experiments Using ThO_2 and UO_2*, Pergamon Press, New York (1961).
86. Redman, W. C., *et al.*, USAEC report ANL-6378 (1961).
87. Pennington, E. M., USAEC report ANL-661 (1962).
88. Towler, O. A., USAEC report TID-7650, Book 11, 614-629 (1962).
89. Jensen, J. C., USAEC report DP-784 (1962).
90. Fletcher, J. F., and Smith, R. I., USAEC report HW-79669 (1963).
91. Randall, D., and St. John, D. S., Nucleonics *16*, No. 3, 82-87 (1958).
92. Sakata, H., and Schmid, L. C., USAEC report HW-81398 (1964).
93. Rusche, B. C., Conference on Problems of Oper. Res. and Power Reactors, ANS (1963).
94. Engesser, F. C., USAEC report HW-56919 (1958).
95. Baumann, N. P., USAEC report DP-758 (1962).
96. Behrens, D. J., UKAEA report AERE T/R 877 (1952).
97. Smith, R. I., USAEC report HW-79929 (1963).
98. Hood, R. R., and Isakoff, L., USAEC report DP-605 (1962).
99. Swanson, C. D., USAEC report HW-81211 (1964).
100. Katz, S. M., USAEC report DP-894 (1964).
101. Haefner, R. R., Private communication.
102. Bright, G. O., and Grund, J. E., USAEC report IDO-16491 (1959).
103. Davis, T. H., and Grund, J. E., Trans. Amer. Nuclear Soc. *6*, 99 (1963). Also USAEC report IDO-16893 (1963) and USAEC report IDO-16920 (1963).

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 Хэнфорде. 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 торневыми 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 торневых 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 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.