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Joint Research Centre
Ispra Establishment — Italy

MASTER

Preliminary

THERMAL BENCHMARK EXPERIMENT COMPILATION

84030001

THERMAL BENCHMARK EXPERIMENT COMPILATION

Forward

In the following is presented a Compilation of Neutron Experiments suited to check the basic data involved in the neutron balance and power distribution calculations of thermal reactor systems.

This compilation has been set-up according to a recommendation of the European American Reactor Physics Committee in view of facilitating the exchange of information on nuclear data libraries and methods in use at the various laboratories and industries for thermal reactor calculations.

The experiments have been selected in order to satisfy a number of requirements, namely:

– a simple and well defined geometrical configuration (homogeneous or lattice systems)
– a limited number of materials involved
– a high sensitivity, from the neutron balance point of view, to the nuclear data of one material of the test configuration, in order to facilitate the comparison with nuclear data files.

The experiments are listed according to this last criterion, with reference to fuel materials. First 235-U-systems are considered, then measurements involving Plutonium. In all cases, results for different moderators and temperatures have been included.

No measurements permitting a check of fission product data have been so far included in the compilation.

The description of the data of each experiment is presented in sheet-form; the results are given as reactivity, buckling and reaction rates; no detailed spectrum measurements are reported because these results are already covered by another EACRP Compilation.

The list has been drawn up primarily on the basis od information directly supplied by the national Representatives of EACRP. Any new proposal intended to cover an area not yet considered in this first list or considered more suited for the indicated use as compared to those already included will gladly be accepted and the compilation, if necessary, updated.

Ispra, May 1974

G. CASINI
J.R.C. EURATOM Ispra

THERMAL BENCHMARK EXPERIMENT COMPILATION

Sheet No. 1

Date submitted: January 1972

Status: Final

Laboratory: ORNL (USA)

Experimental identification: ^{235}U-Water Homogeneous (ORNL-1,2,3,4, 10)

Core configuration: Homogenous unreflected spheres of acqueous solutions

Geometrical data: Benchmarks ORNL-1 through ORNL-4 are of radius 34.595 cm; ORNL-10 has radius 61.011 cm

Composition data:

Case/ Material	Atom Density, 10^{24} atoms/cm^3 ORNL-1	ORNL-2	ORNL-3	ORNL-4	ORNL-10
^{10}B	0.0	1.029×10^{-6}	2.057×10^{-6}	2.532×10^{-6}	0.0
H	0.06623	0.006615	0.00661	0.00660	0.0664
O	0.03374	0.03380	0.00386	0.03390	0.3359
N	1.87×10^{-4}	2.13×10^{-4}	2.29×10^{-4}	2.55×10^{-4}	1.11×10^{-4}
^{234}U	5.38×10^{-7}	6.31×10^{-7}	7.16×10^{-7}	7.62×10^{-7}	4.09×10^{-7}
^{235}U	4.807×10^{-5}	5.621×10^{-5}	6.394×10^{-5}	6.80×10^{-5}	3.62×10^{-5}
^{236}U	1.38×10^{-7}	1.63×10^{-7}	1.84×10^{-7}	1.97×10^{-7}	2.20×10^{-7}
^{238}U	2.81×10^{-6}	3.29×10^{-6}	3.73×10^{-6}	3.97×10^{-6}	1.98×10^{-6}

Temperatures: 20°C

Type of experiment: Criticality as a function of the chemical concentration of the fissile isotope

Results:

Case	Measured k	'Corrected' Measured k
ORNL- 1	1.00118	1.00026
2	1.00073	.99975
3	1.00090	.99994
4	1.00028	.99924
10	1.00129	1.00031

The corrected values were evaluated (Ref.2) to account for newer β values, the thin aluminium shells, distortion of the spherical shape, fill tubes and room return

Auxiliary measurements: Spatial Neutron Flux Distribution, Kinetic Behaviour, Neutron Energy Deposition

Comments: Similar Experiments with the same type of solutions in cylinders of 5 and 9 ft diameter

References:

1. R. Gwin and D.W. Magunson, 'Eta and ^{233}U and ^{235}U for critical experiments, Nuc. Sci. Eng. 12, 364 (19620
2. A. Staub, et al. 'Analysis of a Set of Critical Homogeneous U-H_2O spheres', Nuc. Sci. Eng. 34, 263 (1968)

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THERMAL BENCHMARK EXPERIMENT COMPILATION

Sheet No. 2

Date submitted:	June 1972
Status:	Final
Laboratory:	JAERI (Japan)
Experimental identification:	$^{235}U-D_2O$ Homogeneous (JAERI–A,B,C)
Core configuration:	Homogeneous Reflected Spheres of Acqueous solutions

Geometrical data:

Case	Core Radius (cm)	Core Volume (liters)	Blanket Thickness (cm)
A	40.0	262.1±10.5	35.0
B	33.0	142.1±0.3	42.0
C	26.5	73.0±0.2	48.5

Composition data:

Core solution: Uranic sulfate–D_2O solution of 19.82°/o enriched uranium;
Blanket composition: Boron–poisoned–D_2O (99.75°/o purity)

Case	$D/^{235}U$	^{235}U-Conc. g/l	B-Conc. g/l
A	4468	6.81	182.6
B	1978	12.93	69.0
C	1080	20.41	18.11

Core Tante: 3S–Aluminium lined with polyethylene

Temperature:	20°C
Type of experiment:	Criticality as a function of fissile and boron concentration
Results:	k = 1.000 ± 0.5 °/o in all cases
Auxiliary parameters:	Cadmium ratios of ^{235}U and Au at the core center

References:

J. Hirota, et al: Experimental and Theoretical Studies of Heavy Water Homogeneous Two Region Systems, J. Nucl. Sci. Tech. 2 (1965) 132-140

THERMAL BENCHMARK EXPERIMENT COMPILATION

Sheet No. 3

Date submitted: June 1972

Status: Final

Laboratory: JAERI (Japan)

Experiment identification: ^{235}U–graphite ~~semi~~-homogeneous (SHE–1,5,6,7,8,9)

Core configuration: Graphite–modulated 20 °/o enriched uranium critical assembly. The core region is made up of UO_2 rods, surrounded by a radial reflector region composed of graphite rods (horizontal axis).

Geometrical data:

Configuration	$C/^{235}U$	Fuel Rods number	Critical Mass $U^{235}U$	Inner reflector radius(cm)	Core radius (cm)	Core length (cm)	Axial reflector thickness
SHE-1	5378	365	3.40	0	44.0	100	70
SHE-5	5378	219	5.29	0	35.7	240	0
SHE-6	4328	196	5.95	0	33.8	240	0
SHE-7	3276	173	6.99	0	31.8	240	0
SHE-8	2226	147	8.69	0	29.3	240	0
SHE-9	5378	298	7.20	44.4	60.8	240	0

Composition data: Fuel rods packed with fuel disks of a mixture of 19.85 °/o enriched UO_2 and pure graphite (C/UO_2 = 10/1), alternated with spacer disks of pure graphite

Temperature: 20°C

Type of experiment: Promt neutron decay measurement by pulsed neutron technique

Results:

Config.	SHE–1	SHE–5	SHE–6	SHE–7	SHE–8	SHE–9
a_c	5.54±0.11	5.73±0.07	5.86±0.15	6.48±0.29	6.74±0.35	3.86±0.34

a_c = promt neutron decay constant at delayed critical

References:

Y. Kancko, et al.: Measurement of Prompt Neutron Decay Constant in Delayed Critical State of Heavily Reflected Reactor, J. Nucl. Sci. Tech. 4 (1967) 462-467

THERMAL BENCHMARK EXPERIMENT COMPILATION
Sheet No. 4

Date submitted: January 1972

Status: Final

Laboratory: BNL (USA)

Experimental identification: ^{235}U-graphite, nearly homogeneous (BNL 1 through 5)

Core configuration: Five assemblies, each composed of highly enriched uranium-aluminium fuel, in rectangular strips or plates and graphite bars 24 in.long and 1 in.high by 1 in wide. Fuel and graphite are stacked in a V–shaped aluminium supporting structure to form a rectangular parallelepiped with upper and lower faces making an angle of 45° with respect to the horizontal and the end faces vertical. The two lower faces of the assembly (apart from 5) are lined with a thick cadmium sheet. In each assembly there are four control rods (three for assembly 5) completely withdra n during the reactivity measurements.

Geometrical data:

Assembly	Dimensions (cm)	^{235}U (kg)	$C/^{235}U \times 10^4$
1	162.45x183.36x182.88	7.904	2.290
2	139.67x168.31x182.88	8.302	1.718
3	133.19x132.95x182.88	9.355	1.145
4	105.06x121.13x182.88	13.866	5.724×10^{-1}
5	98.27x101.71x182.88	60.584	9.582×10^{-2}

Homogenized atomic density away from control rods
at/cm³

Assembly	$C(\times 10^{22})$	$^{235}U(\times 10^{18})$	$^{238}U(\times 10^{17})$	$A\ell(\times 10^{20})$
1	8.518	3.720	2.702	2.141
2	8.501	4.948	3.593	2.847
3	8.486	7.412	5.382	4.265
4	8.425	14.72	10.69	8.469
5	8.121	84.75	48.81	29.34

Temperatures:

Assembly	1	2	3	4	5
Temperature °C	25	22	20.7	22.5	18.9

Type of experiment: Criticality measurements

Results:

Assembly	keff(20°C)	$B^2 \times 10^{-4}$ (cm^{-2})	l^x/β(ugec)(+)	$d\rho/dT$(c/°C)
1	1.0058±0.0011	9.33±0.02	173±3	−4.43±0.07
2	1.0027±0.0011	11.02±0.03	143±3	−4.0 ±0.4
3	1.0033±0.0012	13.46±0.05	99±2	−2.7 ±0.5
4	1.9288±0.0020	16.88±0.08	54±1	−1.69±0.24
5	0.9935±0.0036	20.93±0.12	9.92±0.09	−0.45±0.02

(+) Ratio of neutron lifetime to effective delayed neutron fraction

Auxiliary parameters: Moderator to fuel disadvantage factors, cadmium ratios, activation rati os (Au, In, ^{176}Cu, Mn, ^{239}Pu, ^{235}U)

References:

J.P. Phelps and E.V. Weinstock: Criticality Measurements on Nearly Homogeneous Enriched Uranium–Graphite Systems, Nucl. Sc. and Eng. 34, 237-250 (1963)

THERMAL BENCHMARK EXPERIMENT COMPILATION

Sheet No. 5

Date submitted:

Status: Final

Laboratory: AEE – Winfrith (UK)

Experimental identification: Low Enrichment – UO_2 rods – H_2O lattices (UKAEA–RI–R2–R3)

Core configuration: Three core configurations using enriched UO_2 rods arranged in as near a cylindrical array as possible, the radial extent of each core being adjusted to give approximately equal buckling axially and radially

Core	Equivalent Radius (cm)	Number of fuel rods
R1	29.44	1565
R2	27.15	665
R3	37.47	2820

Composition data:

Fuel cladding.

Σ_a (2200 m/s absorption cross-section)	.0343±.0006 cm^2/g
Mass per unit length	.6983±.0035 g/cm
Outer diameter	1.0925±.0016 cm
Thickness	.0267±.00013 cm
Equivalent density	7.8 g/cm^3

Percentage weights of clad constituents

Fe	Cr	Ni	Mn	Ti	P	S	Si	C
67.48	19.11	10.80	1.87	0.5	0.028	.003	.20	.012

Fuel.

length of one pellet	1.0174±.0003 cm
weight	8.539 ± .008 g
Number of pellets per rod	68
Pellet diameter	1.012±.008 cm
Enrichment ^{235}U by total by weight	3.003±.011°/o
Density	10.44±0.011 g/cm^3

Pellets are wrapped in four groups of 8 and four of 9 in each pencil. Total weight of wrapping is 3.6±0.1 g; each fuel rod is 71.83 cm overall length and fitted with 1 cm thick aluminium bungs

Geometrical data:

Core	Lattice pitch (cm)	Moderator to fuel volume ratio
R1	1.320	1.001
R2	1.866	3.164
R3	1.251	0.775

Temperature: 20 – 80°C

Type of experiment: Criticality as a function of the fuel rods in the core

see overleaf

Sheet No. 5 (continued)

Results:

Core		R1–20°C	R1–80°C	R2	R3
$Bm^2 (cm^{-2})$		66.00 ±0.22	62.58 ±0.40	100.44 ±0.64	50.96 ±0.30
$\frac{^{238}U\ Fiss}{^{235}U\ Fiss}$		4.149± 0.030	4.263 ±0.047	–	4.789 ±0.053
$\frac{^{176}Lu}{^{55}Mn}$		1.160 ±0.006	1.241 ±0.008	–	1.122 ±0.009
^{235}U cell fine structure	Coolant to fuel	1.26 ±0.013	1.233 ±0.013	–	1.223 ±0.016
	Claddy to fuel	1.13 ±0.004	1.12 ±0.003	–	1.097 ±0.007
$\frac{^{239}Pu\ Fiss}{^{235}U\ Fiss}$		1.589 ±0.009	1.637 ±0.009	–	1.661 ±0.009
$\frac{d\rho}{dH}$ % (cm^{-1})		0.248 ±0.011	0.206 ±0.007	0.593 ±0.016	0.284 ±0.008

References:

Fox W.N. et al, Measurements of Material Buckling and Detailed Reaction Rates in a series of Low Enriched UO_2 fuelled cores moderated by light water. AEEW – R502 (1967)

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THERMAL BENCHMARK EXPERIMENT COMPILATION

Sheet No. 6

Data submitted:	January 1972
Status	Final
Laboratory	BAPL (USA)
Experiment identification	Enriched Uranium Metal Rod · H_2O lattices (TRX metal)
Core configuration	Uranium metal rods loaded according a triangular pitch, fully reflected radially
Geometrical data	Fuel dimensions (cm)

Geometrical data

Fuel dimensions (cm)

Fuel radius	0.4915
Clad I.R.	0.5042
Clad O.R.	0.5753
Fuel rod length	122
Cell	V_{mod}/V_{fuel}
TRX-1	2.35
TRX-2	4.02

Composition data

Fuel atom densities ($\times 10^{24} cm^{-3}$)

U^{238}	0.047205
U^{235}	0.0006253
Al^{27}	0.06025

Moderator atom densities ($\times 10^{24}$ cm^{-3})

H	0.06676
O^{16}	0.03338

Results

Cell	B^2 (cm^{-2})
TRX 1	0.0057 ± 0.0001
TRX 2	0.005469 ± 0.000036

Results

Lattice	σ^{28}	σ^{25}	σ^{28}
TRX 1	0.830±0.015	0.0608±0.0007	0.0667±0.0020
TRX 2	1.311±0.020	0.0981±0.0001	0.0914±0.0020

σ^{28} = ratio of epithermal-to-thermal U-238 captures
σ^{25} = ratio of epithermal-to-thermal U-235 fissions
σ^{28} = ratio of U-238 fissions to U-235 fissions.

Data adjusted to 0.625 eV thermal neutron energy cutoff.

see overleaf

Activation Disadvantage Factors

V_{Mod}/Fuel	Detector	Experiment
2.352	U-235	1.317 ± 0.013
4.023		1.378 ± 0.013
2.352	Dy-164	1.303 ± 0.013
4.023		1.360 ± 0.008
2.352	Lu-176	1.200 ± 0.011
4.023		1.236 ± 0.011

References

J. Hardy Jr., D. Klein and J.I. Volpe "A Study of Physics Parameters in Several Water Moderated Lattices of slightly Enriched and Natural Uranium", WAPD-TM-931, March 1970

Date submitted: January 1973

Status: Final

Laboratory: BAPL (USA)

Experiment identification: Natural Uranium metal slabs – H_2O-lattices (SLAB-1,2)
Critical flux mapping

Geometrical data:

Fuel dimensions.

Slab thickness (cm)	2.540
Slab length x width (cm^2)	61 x 61

Moderator dimensions.

	Vrod/V Fuel	Thickness
SLAB–1	1.09	2.7686
SLAB–2	0.50	1.2700

Composition data:

Fuel atom density ($\times 10^{24}$ cm^{-3})

U^{232}	0.047483
U^{235}	0.0003401

Cell atom density ($\times 10^{24}$ cm^{-3})

H^1	0.06676
O^{16}	0.03338

Temperatures: 20°C

Results:

Lattice	B^2 (cm^{-2})	ρ^{28}	δ^{25}	δ^{28}
SLAB-1	−0.0013 ±0.0001	1.21 ±0.03	0.124 ±0.002	0.190 ±0.004
SLAB-2	−0.0037 ±0.0003	2.63 ±0.06	0.267 ±0.006	0.264 ±0.006

ρ^{28} = ratio of epithermal to thermal ^{238}U captures
δ^{25} = ratio of epithermal to thermal ^{235}U fissions
δ^{28} = ratio of ^{238}U fissions to ^{235}U fissions
Data adjusted to 0.625 ev thermal neutron cut off

Thermal activation disadvantage factors

Lattice	^{235}U	^{164}Dy	^{176}Lu	^{239}Pu
SLAB-1	–	2.112 ±0.027	1.723 ±0.028	1.800 ±0.025
SLAB-2	2.843 ±0.025	2.856 ±0.020	2.119 ±0.03	2.368 ±0.03

Date submitted:

Status: Final

Laboratory: EIR – Wurenlingen (Switzerland)

Experiment identification: Natural uranium rods – D_2O lattices (MINOR – 8 through 16)

Core configuration: Uranium metal rods loaded according to a square pitch fully reflected radially (MINOR facility)

Geometrical and composition data: Fuel. (natural uranium metal)

Fuel radius	1.0 cm
Clad thickness	0.075 cm
Fuel density	18.94 g/cm^3

Cell.
Lattice pitch (square): for 8.0 to 16.0 cm (see Results)
D_2O concentration (see Results)

Temperature: 20°C

Type of experiment: Subcritical flux mapping

Results:

Lattice pitch (cm)	V_M/V_F	Number of rods	D_2O conc. (mol.°/o)	B_m^2 (μB)	B_m^2 corrected to 99.75°/o D_2O (μB)
MINOR 8	19.37	216	99.70	780±6	781±6
9	24.78	180	99.62	835±6	841±6
10	30.83	148	99.74	840±3	842±3
10	"	148	99.62	833±3	843±3
11	37.52	120	99.62	791±3	804±3
12	44.84	112	99.74	757±3	759±3
12	"	112	99.66	748±3	760±3
12	"	112	99.61	743±3	761±3
14	61.39	76	99.74	647±3	650±3
14	"	76	99.49	617±3	657±3
16	80.49	60	99.49	506±3	551±3

References:

Lutz H.R. et al. Proceedings of IAEA Conference on Exponential and Critical Experiments (1964) p. 85, M374/1+2+3

Data submitted:	September 1971
Status	Final
Laboratory	Kjeller (Norway)
Experimental Identification	Enriched Uranium Oxyde Rods - H_2O moderated lattices (NORA 1 through 3)
Core configuration	Uranium oxide rods loaded according a square pitch in water, radially reflected (90 cm of H_2O) Number of rods in the core : 240, 248, 384

Geometrical and Composition Data

Fuel

Fuel diameter (cm)	1.27
Clad thickness (cm)	0.048
Clad material	SAE-304 Stainless steel
Fuel enrichment	3.41 ± 0.02 %
Fuel density (g/cm^3)	10.4
Active fuel length (cm)	121.9 ± 0.6
End plugs : bottom steel plugs	11.4 cm length

Lattice cell

Core	V_m/V_f	Lattice pitch (cm)
NORA-1	4.51	2.687
NORA-2	3.03	2.314
NORA-3	1.66	1.900

Temperature 20 ± 3 °C

Type of Experiment Criticality of a function of the water height

Results

	B^2_m (m^{-2})	Thermal Disadvantage factor (cm)	ρ 28	Temper.coeff. pcm/°C
NORA-1	86.4	1.622	1.065	- 0.1857 . t°
NORA-2	98.8	1.572	1.542	- 0.2728 . t
NORA-3	91.8	1.441	–	- 0.3206 . t

**temperature (t) : 15-60°C*

References

Topics in light water reactor physics: Final Report of the NORA Project
Technical Report Series N.113-IAEA, Vienna (1971)

Date submitted: May 1973

Status final

Laboratory CEA-CADARACHE (France)

Experimental Identification Natural Uranium metal rod — Graphite lattices (CEA—UN -1-2)

Geometrical Data

	CEA-UN-1	CEA-UN-2
Fuel radius (cm)	1.535	2.50
Fuel density (gr/cm^3	18.92	18.54
Canning thickness		
Canning volume per unit height (cm^2)	4.50	3.38
Channel diameter	11.0	17.0
Lattice pitch (cm^2)	22.4	31.68

Composition Data

Fuel : Natural Uranium

Fuel Density : 18.92 g/cm^3

Canning : Mg-Zr Alloy

2200-n/s Absorption cross section: 0.0028 cm^{-1}

Graphite density: 1.7 g/cm^3

2200 m/s absorption cross section : 3,79 mb

Temperature 20°C

Type of Experiment Critical Flux Mapping

Results

	Parallel* buckling(m^{-2})	Transversal buckling(m^{-2})	Buckling (m^{-2})
CEA-UN-1	0.2590	0.7285	0.9875
CEA-UN-2	0.2535	0.5914	0.8449

* *parallel to the channel orientation*

References Private Communication (Dr. ROCHE, CEA, Saclay)

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THERMAL BENCHMARK EXPERIMENT COLLECTION

Sheet No. 11

Data Submitted:

Status: Final

Experiment Identification: Natural Uranium Rod — Graphite Lattices (BICEP — Nat)

Experimental Method: Exponential flux mapping

Geometrical Data and Composition Data

Stack	Uranium o.d. (in.)	Uranium S_u	Can o.d. (in.)	Can i.d. (in.)	Can S_{Al}	Graphite Pitch (in.)	Graphite Channel dia. (in.)	Graphite S_g	Material buckling K_R^2 (m^{-2})
Bicep I	1.64	0.9673	1.76	1.68	0.9944	14	4.5	1.052	0.595 ± .021
Bicep I	1.64	0.9673	1.76	1.68	0.9944	12	3.75	1.045	1.155 ± .037
Bicep I	1.64	0.9673	1.76	1.68	0.9944	8½	3.75	1.058	1.250 ± .018
Bicep I	1.64	0.9673	1.76	1.68	0.9944	7½	4.5	1.055	1.243 ± .017
Bicep I	1.64	0.9673	1.76	1.68	0.9944	8	3.75	1.052	0.646 ± .020
H	1.2	0.9968	1.445	1.365	1.022	7½	2.75	1.0406	1.100 ± .020
E	1.2	0.9968	1.445	1.365	1.022	7.875	1.5	1.0231	1.341 ± .041
Bicep I	1.2	0.9968	1.445	1.365	1.022	8	3.75	1.0622	1.052 ± .024
B	1.2	0.9968	1.445	1.365	1.022	8.25	3.77	0.9753	0.869 ± .033
E	1.2	0.9968	1.445	1.365	1.022	7.875	3.75	1.0368	1.101 ± .030
C	1.2	0.9968	1.445	1.365	1.022	7 x 8 1/6	1.75	0.9798	1.020 ± .023
H	1.2	0.9968	1.445	1.365	1.022	7	3.75	1.0423	0.721 ± .020
Bicep I	1.0	1.0004	1.264	1.06	0.9967	8	2.0	1.0573	1.093 ± .028
E	1.0	1.0004	1.264	1.06	0.9957	7.875	1.5	1.0231	1.032 ± .013
E	1.0	1.0004	1.264	1.06	0.9957	7.875	2.75	1.0245	0.982 ± .018
C	1.0	1.0004	1.445	1.365	1.022	7 x 8 1/6	1.75	0.9798	0.851 ± .030
E	1.0	1.0004	1.264	1.06	0.9967	7.875	3.75	1.0358	0.836 ± .020
Bicep I	1.0	1.0004	1.264	1.06	0.9967	8	4.5	1.0654	0.755 ± .012
E	1.0	1.0004	1.264	1.06	0.9967	7	1.5	1.0204	1.181 ± .018
E	1.0	1.0004	1.264	1.06	0.9967	7	2.75	1.0221	1.035 ± .017
E	1.0	1.0004	1.264	1.06	0.9967	7	3.75	1.0381	0.730 ± .035
F	1.0	1.0004	1.264	1.06	0.9967	6.2	1.5	1.0254	1.117 ± .060
A Mod	1.0	1.0004	1.445	1.365	1.022	6.3	1.83	0.9772	0.944 ± .050
Bicep I	1.0	1.0004	1.264	1.06	0.9967	6	3.75	1.0535	0.297 ± .030
F	1.0	1.0004	1.264	1.06	0.9967	5	1.5	1.0173	0.136 ± .069
F	1.0	1.0004	1.264	1.06	0.9967	5	2.75	1.0231	0.670 ± .055

* S_u = Fuel effective density relative to 18.9 gr/cm^3

** S_g = Graphite effective density relative to 1.65 gr/cm^3

*** S_{Al} = Aluminium effective density relative 2.7 gr/cm^3

Temperature: 20°C

Results: (see table)

Auxiliary parameters: Neutron fine flux structure measurements

References: BICEP — AEEW — R235 Volume I: Exponential Experiments on Rods and Tubes of Natural Uranium Metal; Volume III: Fine Structure Experiments — Exponential Experiments.

THERMAL BENCHMARK EXPERIMENT COLLECTION

Sheet no. 12

Data Submitted: May 1972

Status: Final

Laboratory: AEE-Winfrith (U.K.)

Experiment Identification: Enriched Uranium Rod – Graphite Lattices (BICEP–E)

Geometrical Data and Composition Data

Core	Uranium fuel rod			Graphite			Aluminium can		
	Fuel enrichment (Co)	Dia. (in.)	Relative density S_u	Pitch (in.)	Channel dia. (in.)	Relative density (S_g)	o.d. (in.)	i.d. (in.)	Relative density (S_{Al})
BICEP 1	1.606	1.196	.9909	8√2	4.5	1.0594	1.445	1.365	1.022
" 2				7	3.75	1.0584			
" 3				7	3.75	1.0584			
" 4				7	3.75	1.0584			
" 5				6	2.75	1.0486			
" 6	1.303	1.197	.9894	7√2	4.5	1.0548	1.445	1.365	1.022
" 7				7	3.75	1.0584			
" 8				6	2.75	1.0486			
" 9	0.589	1.193	.9884	8√2	2.75	1.0438	1.42	1.22	1.006
" 10				7√2	4.5	1.0548			
" 11				8	4.5	1.0654			
" 12				7	3.75	1.0584			
" 13				7	2.75	1.0544			

Temperature: 20°C

Type of Experiment: Exponential flux mapping

Results:

B_m^2 (m^{-2})	$\frac{M_Z^2}{M_R^2}$
2.835 ± .016	1.189 ± .011
3.168 ± .023	*1.229 ± .025
3.227 ± .033	*1.229 ± .025
3.347 ± .033	*1.229 ± .025
3.445 ± .023	1.127 ± .009
2.207 ± .015	1.257 ± .012
2.122 ± .016	1.213 ± .011
2.001 ± .023	1.054 ± .010
-2.410 ± .041	*1.045 ± .010
-1.715 ± .048	1.298 ± .014
-1.825 ± .043	*1.362 ± .028
-2.451 ± .028	1.297 ± .014
-2.997 ± .071	*1.094 ± .013

References:

BICEP – AEEW – R235 – Volume II: Exponential Experiments on Enriched Metal Rods and on Clusters of Uranium Metal and Uranium Oxide Rods (1963).

84030016

Date submitted	June 1973
Laboratory	JRC-EURATOM (ISPRA)
Status	Final
Experiment Identification	Natural Uranium rods – Heavy water moderated lattices (ECO-U-1, 2, 3) – Temperature coefficients

Geometrical Data

Fuel diameter	29.2 mm
Canning tube (I.D./O.D.)	29.5/ 31.5 mm
Separation tube	58/60 mm
Pressure tube	78.5/81.5 mm
Calandria tube	91.5/93.5 mm

Core	Lattice pitch (cm)
ECO-U-1	17.5
ECO-U-2	20.5
ECO-U-3	23.5

Composition data

Fuel : natural Uranium metal
density 18.94 g/cm^3

Canning, pressure and calandria tube, Zircalloy-2
Separation tube: Aluminium
Coolant and Moderator: (99%)
Insulation atmosphere between pressure and calandria tube: Nitrogen

Temperatures

Fuel channel	20-200°C
Moderator	20-60°C

Type of experiment

Criticality as a function of the water height by progressive heating of test fuel channels

Results:

Core	Moderator Temperature (°C)	Mean Coolant Temperature T_c(°C)	$\Delta B_m^2 / \Delta T_c$ (10^{-3} m^{-2}/°C)
ECO-U-1	22.3	41.7	- 1.955±0.209
		78.8	- 2.060±0.203
		122.9	- 2.002±0.190
		174.5	- 1.888±0.184
ECO-U-2	22.3	41.2	- 1.770±0.179
		79.4	- 1.585±0.159
		123.8	- 1.449±0.132
		175.5	- 1.249±0.116
ECO-U-3	60.3	82.0	- 1.299±0.12
		126.2	- 1.057±0.10
		175.5	- 0.865±0.08

See overleaf

Sheet No. 13 (continued)

References

Deutsches Atomforum, Reaktortagung - Karlsruhe 1973
"Messung von Kühlkanaltemperaturkoeffizienten des Buckling von Natrium- und Pu-haltigen Brennelementen und Vergleich mit der Theorie, W. HAGE et al.

Data Submitted: January 1972

Laboratory: ORNL (USA)

Status: Final

Experiment Identification: ORNL–U233 – Homogeneous Acqueous Spheres(ORNL - 233U-5, 7, 11)

Geometrical and Composition Data: Homogeneous Spheres (see also sheet no. 1)

Atom Densities (in units of $10^{24}/cm^3$)

	ORNL-233-U – 5	ORNL-233-U – 7	ORNL-233-U – 11
	Sphere Diameter (in.)		
	27.24	27.24	48.04
^{10}B	0.0	5.1230×10^{-7}	0.0
H	0.066360	0.066329	0.066467
O	0.033607	0.033633	0.033525
N	1.178×10^{-4}	1.274×10^{-4}	7.530×10^{-5}
233-U	4.328×10^{-5}	4.6798×10^{-5}	3.3460×10^{-5}
234-U	7.16×10^{-7}	7.72×10^{-7}	5.25×10^{-7}
235-U	1.8×10^{-8}	1.8×10^{-8}	1.0×10^{-8}
232-U	2.81×10^{-7}	3.01×10^{-7}	2.56×10^{-7}
232-Th	1.9641×10^{-7}	2.1333×10^{-7}	1.4757×10^{-7}

Temperatures: 20°C

Type of Experiment: Criticality as a function of the chemical concentration of the fissile isotope

Results:

Core	keff
ORNL–233-U – 5	1.00050
ORNL–233-U – 7	1.00109
ORNL–233-U – 11	1.00046

Auxiliary Experiments: Spatial Neutron flux distribution, kinetic behaviour, neutron energy distribution.

References:

R. Gwin and D.W. Magunson, Nuclear Science and Engineering 12, 364 (1962).

Sheet No. 15

Data submitted: June 1972

Status: Final

Laboratory: AB – Atomenergy, STUDSVIK, Sweden

Experimental Identification: Enriched Uranium Oxyde – H_2O Lattices – Temperature Coefficient (KZ–39,46)

Geometrical and Compilation Data:

Fuel

Material	UO_2 pellets
Enrichment	1.35 weight-% $U^{235}O_2$
Diameter	12.38 ± 0.03 mm
Density, effective	10.26 ± 0.02 g/cm^3
Total weight UO_2 per rod	2.71 ± 0.02 kg
Total length UO_2 per rod	2194 ± 3 mm
Weight UO_2 per unit length	1.235 ± 0.01 g/mm

Canning

Material	Zircaloy-2
Thickness	0.63 ± 0.08 mm
Diameter, inner	12.60 ± 0.05 mm
diameter, outer	13.86 ± 0.13 mm

Fuel rod

Rod length	2365 mm
Rod weight	3.23 kg
Total number available	2160 rods

Core data

Square lattice pitch	18.0 mm	
Spacer material	Stainless steel (SIS–2343)	
Spacer diameter	3.99 ± 0.01 mm	
Spacer plane separation	360 mm	
Number of lattice positions	39 x 39	46 x 46
Core side (Number of cells x 18.0 mm)	702 mm	828 mm
Water reflector thickness		
Southern and west. sides	75 mm	75 mm
Northern and estern sides	293 mm	167 mm

Temperatures: 20 – 245°C

Type of Experiment: Buckling Determination by activation and measurements of critical water level

Results:

Lattice	Boron[a)] conc. (ppm)	Temp. (°C)	B_z^2 [b)] (m^{-2})	B_r^2 [c)] (m^{-2})	B_m^2 (m^{-2})
KZ-39-1	0.8	41.2	11.83	27.44	39.27
		90.0	9.94	27.18	37.12
		120.4	8.49	26.96	35.45
		142.4	7.38	26.76	34.14
		183.1	4.99	26.32	31.31
		195.6	4.21	26.15	30.36
		207.2	3.44	25.99	29.43
		216.6	2.78	25.86	28.64
		225.6	2.12	25.72	27.84

See overleaf

46.3	90.4	13.88	20.63	34.51
KZ-46-2	206.0	7.53	19.86	27.39
	227.3	6.12	19.64	25.76
	245.8	4.72	19.43	24.15
175	22.4	9.02	20.83	29.85
	34.3	8.76	20.81	29.57
	50.3	8.35	20.78	29.13
	64.0	7.97	20.73	28.70
	79.4	7.50	20.67	28.17
KZ-46-2	89.4	7.17	20.63	27.80
	162.1	4.23	20.22	24.45
	171.7	3.78	20.15	23.93
	186.7	3.04	20.02	23.06
	201.3	2.32	19.90	22.22
	205.3	2.10	19.86	21.96

a) Accuracy ± 1.0 %
b) λ_z = 175 mm for the first three cases and λ_z = 165 mm for the last case
c) $B_r^2 = B_x^2 + B_y^2$; $2\lambda_x = 2\lambda_y$ = 145/θ (T)/θ (20°C) mm.

Auxiliary Parameters: Buckling effect of spaces ar 20°C: – 1.3 ± 0.2 m^{-2}

References: Private Communication (R. Persson, Stuswik, Sweden)

THERMAL BENCHMARK EXPERIMENT

Sheet No. 16

Date submitted:	12th June 1972
Status	Final
Laboratory	CSN Casaccia — CNEN - (Italy)
Experimental identification	"Reticolo ad Ossidi Misti PuO_2- UO_2"
Geometrical data	See References No. 1, 3, 4 and 5
Material data	See References No. 1, 3, 4 and 5
Temperatures	20°C
Type of Experiment	γ- scanning and foil activation (RITMO Reactor)
Results	Relative power distributions and relative values and spatial distributions for several spectral indices
Auxiliary parameters	See References No. 3
Comments	See References No. 5, 6 and 7
References	1) JNE, 25 (1971) 623 2) NSE, 46 (1971) 376 3) RT/FI(69)49 and RT/FI(70)51, CNEN, Rome (1969) and (1970) 4) RT/FI (70)45, CNEN, Rome (1970) 5) RTI/FCR (67)24, CNEN, Rome (1967) 6) RT/FI(70)48, CNEN, Rome (1970) 7) PL-447/34, IAEA, Vienna (1971)

84030022

Sheet No. 17

Data submitted:

Status: Final

Laboratory: PNL – Handford (USA)

Experiment Identification: Plutonium – Aluminium Rods – H_2O lattices (PNL – 1,2)

Geometrical Data:

Fuel Dimensions(cm)

Fuel Radius	0.635
Clad I.R.	0.64262
Clad O.R.	0.68072
Fuel Rod Length	60.96

Cell Dimensions

Core	V_{mod}/V_{rod}	Spacing (cm) (triangular pitch)
PNL-1	2.442	2.54
PNL-2	4.182	3.302

Composition Data:

Fuel atomic density (x 10^{24} cm^{-3}) and weight fractions (w/o)

Pu-239	3.3830×10^{-4}	94.09
Pu-240	1.9084×10^{-5}	5.33
Pu-241	1.9611×10^{-6}	0.55
Pu-242	1.0653×10^{-7}	0.03
Al	5.9120×10^{-2}	*
Ni	4.4680×10^{-4}	1.61
Fe	1.4294×10^{-4}	0.49
Zr (Clad)	4.3310×10^{-2}	—

* Silicon (0.69 w/o) combined with aluminium (97.21w/o)

Cell Atomic Density (x10^{24} cm^{-3})	
H^1	6.6799×10^{-2}
O^{16}	3.3399×10^{-2}

Temperatures: 20°C

Type of Experiment: Criticality

Results:

H/Pu Atom Ratio	Critical Number of Rods	Critical Buckling (10^{-6} m^{-2})	Reflector Savings and Extra-polation - Length, λ (cms)
583	170.1	108.5	7.73
1149	215.5	76.8	6.04

References:

1) V.I. Nelley, R.C. Lloyd and E.D. Clayton, "Neutron Multiplication Measurements with Pu-Al Alloy Rods in Light Water, HW-70944 Handford Laboratory, August 1961.
2) G.D. Trimble, et al., Vol. I of Lattice Physics Study, Summary Report throught June 30, 1969. USAEC Report GA-9658 July 1969.

Date submitted:	June 1973
Laboratory:	J.R.C. – EURATOM (ISPRA, Italy)
Status:	Preliminary
Experimental Identification:	Uranium – Plutonium Rods – Heavy water cooled and moderated lattices (ECO – Pu - 1, 2, 3)
Geometrical Data:	See sheet 13 for fuel dimensions

Core	Lattice pitch (cm)
ECO–Pu-1	17.5
ECO–Pu-2	20.5
ECO–Pu-3	23.5

Composition Data:

Fuel Material:	Uranium – Plutonium metal
235-U/238-U	0.22%
239-Pu/238-U	0.27%
240-Pu/239-Pu	8.53%
241-Pu/239-Pu	0.66%
Density (g/cm^3)	18.69
Other materials:	see sheet no. 16
Heavy water purity:	99.73 W %

Temperatures:

Channel: 20°C to 200°C
Moderator: 20°C to 60°C

Type of Experiment: Criticality as a function of the water level by progressive heating of the test fuel channels.

Core	Moderator Temperature (°C)	Coolant Temperat. Range (° C)	Mean Cool. Temp.	$\Delta B_M^2/\Delta TC$ (10^{-3} m^{-2}/°C)
ECO–Pu-1	22.3	22.3 – 58.3	40.3	– 0.231 ± 0.105
		58.3 – 97.3	77.8	+ 0.004 ± 0.112
		97.3 – 148.1	122.7	+ 0.127 ± 0.119
		148.1 – 198.7	173.4	+ 0.207 ± 0.148
ECO–Pu-2	22.3	22.5 – 60.1	41.3	+ 0.157 ± 0.064
		60.1 – 101.0	80.6	+ 0.168 ± 0.071
		101.0 – 146.7	123.9	+ 0.451 ± 0.089
		146.7 – 199.5	173.1	+ 0.508 ± 0.106
ECO–Pu-3	22.3	22.3 – 63.1	42.7	– 0.089 ± 0.036
		63.1 – 98.8	80.9	– 0.056 ± 0.038
		98.8 – 153.0	125.9	+ 0.355 ± 0.057
		153.0 – 304.0	178.5	+ 0.632 ± 0.086

References:

Deutsches Atomforium Reaktortagung 1973 – Karlsruhe "Messung von Kühlmitteltemp – eratur koeffizienten des Bückling von Naturuaen – und Pu-haltigen Brennelementen und Vergleich mit der Theorie. W. Hage et al.

THERMAL BENCHMARK EXPERIMENT COLLECTION

Sheet No. 19

Data Submitted: May 1973

Status: Final

Laboratory: CEA – CADARACHE (France)

Experimental Identification: Uranium – Plutonium Fuel – Graphyte - Lattice – Temperature Coefficient (CEA–T–UN,T– P2)

Geometrical Data:

Fuel Radius (cm)	1.46
Fuel Density (gr/cm^3)	18.94
Coming Value per unit height (cm^2)	
Channel Diameter (cm)	70
Lattice pitch (cm)	22.516

Composition Data:

Core	CEA-T-N	CEA-T-P2
Fuel	Natural Uranium	U-Pu Fuel
Fuel Density	18.85	18.69
$N_5/N_5+N_8\%$	0.72406	0.216
Pu tot/U tot N%	—	0.297
100 240-Pu/239-Pu	—	8.54

Canning Mg–Zr Alloy
2200 – m/sec Absorption Cross Section 0.0028 cm^{-1}
Graphite Density 1.7 g/cm^3
2200 m/sec Absorption Cross Section: 3.79 mb

Temperatures: 20 – 450°C

Type of experiment: Critical Flux Mapping and Reactivity Temperature Coefficient Measurements

Results:

Core	Temperature °C	Parallel Buckling m^{-2}	Trasversal Buckling m^{-2}	Total Buckling m^{-2}
CEA-T-UN	20	0.473 ± 0.006	0.340 ± 0.006	0.810
	200	0.4634 ± 0.003	0.257 ± 0.010	0.7204

Temperature Coefficients (pcm/°C)

	20–40 (°C)	100-120 (°C)	200-220(°C)	300-320(°C)	420-450°C)
CEA-T-UN	± 0.1 ± 0.8	– 1.5 ± 0.7	– 2.2 ± 0.7	—	—
CEA-T-P2	+ 13.6	+ 19.2	+ 22.9	+ 22.4	+ 20.8

References: Private Communication

84030025

THERMAL BENCHMARK EXPERIMENT
Sheet No. 20

Date submitted:	June 1972
Status	Final
Laboratory	CSN Casaccia — CNEN - Italy — Francesco V. Orestano
Experiment identification	"Simulazione Eterogenea del Plutonio"
Geometrical data	See References No. 1 and 2
Material data	See References No. 1 and 2
Temperatures	20°C
Results	Relative power distributions and spectral Index Eu-151 (activation)/U(fission)
Auxiliary parameters	See References No. 1 and 2
Type of Experiment	γ - scanning (RITMO Reactor)
Comments	See References No. 3 and 4
References	1) JNE, 24 (1970) 253 2) RT/FI(69)23, CNEN, Rome (1969) 3) RT/FI(70)48, CNEN, Rome (1970) 4) PL-447/34, IAEA, Vienna (1971)

THERMAL BENCHMARK EXPERIMENT COMPILATION

Sheet No. 21

Date submitted	June 1972
Status	Final
Laboratory	CSN – Casaccia – CNEN (Italy)
Experiment identification	"Simulazione Omogenea del Plutonio"
Geometrical data	See References No. 1, 2 and 4
Material data	See References No. 1, 2 and 4
Temperatures	20°C
Type of Experiment	Foil activation RITMO Reactor
Results	Relative Spectral Indices
Auxiliary Parameters	See References No. 3 and 4
Comments	See References No. 5 and 6

References

1) NSE, 40 (1970) 51
2) RTI/FCR(68)4, CNEN, Rome (1968)
3) RT/FI(61)15, CNEN, Rome (1967) or NASA-TT-F11, 671, Washington (1968)
4) Doc. Int. LFCR967)15, CNEN, Rome (1967)
5) RT/FI (70)48, CNEN, Rome (1970)
6) PL-447/34, IAEA, Vienna (1971)