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Research reactor core conversion guidebook

Volume 5: Operations (Appendices L–N)

INTERNATIONAL ATOMIC ENERGY AGENCY IAEA

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FOREWORD

In view of the proliferation concerns caused by the use of highly enriched uranium (HEU) and in anticipation that the supply of HEU to research and test reactors will be more restricted in the future, this guidebook has been prepared to assist research reactor operators in addressing the safety and licensing issues for conversion of their reactor cores from the use of HEU fuel to the use of low enriched uranium (LEU) fuel.

Two previous guidebooks on research reactor core conversion have been published by the IAEA. The first guidebook (IAEA-TECDOC-233) addressed feasibility studies and fuel development potential for light-water-moderated research reactors and the second guidebook (IAEA-TECDOC-324) addressed these topics for heavy-water-moderated research reactors. This guidebook, in five volumes, addresses the effects of changes in the safety-related parameters of mixed cores and the converted core. It provides an information base which should enable the appropriate approvals processes for implementation of a specific conversion proposal, whether for a light or for a heavy water moderated research reactor, to be greatly facilitated.

This guidebook has been prepared at a number of Technical Committee Meetings and Consultants Meetings and coordinated by the Physics Section of the International Atomic Energy Agency, with contributions volunteered by different organizations. The IAEA is grateful for these contributions and thanks the experts from the various organizations for preparing the detailed investigations and for evaluating and summarizing the results.

EDITORIAL NOTE

In preparing this material for the press, staff of the International Atomic Energy Agency have mounted and paginated the original manuscripts and given some attention to presentation.

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This text was compiled before the unification of Germany in October 1990. Therefore the names German Democratic Republic and Federal Republic of Germany have been retained.

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PREFACE

Volume 5 consists of detailed Appendices L-N, which contain a variety of useful information on the operation of research reactors with reduced enrichment fuels. Summaries of these appendices can be found in Chapters 12-14 of Volume 1 (SUMMARY) of this guidebook. Appendix L contains a summary of necessary and recommended experiments for reactor startup. Appendix M provides information on the procedures and experiences of several reactor operators with both mixed and full cores with reduced enrichment fuels. Appendix N contains information on transportation of both fresh and spent fuel elements, on spent fuel storage, and on the US Department of Energy's receipt and financial settlement provisions for nuclear research reactor fuels.

The topics which are addressed in Volume 5, the appendices in which detailed information can be found, and the summary chapters in Volume 1 are listed below.

Topic	VOLUME 5 APPENDIX	VOLUME 1 SUMMARY Chapter
Startup Experiments	L	12
Experience with Mixed and Full Core Operation	M	13
Transportation, Spent Fuel Storage, and Reprocessing	N	14

CONTRIBUTING ORGANIZATIONS

Argonne National Laboratory	ANL	United States of America
Chalk River Nuclear Laboratories	CRNL	Canada
Commissariat a l'Energie Atomique	CEA	France
GA Technologies Inc.	GA	United States of America
GKSS-Forschungszentrum Geesthacht GmbH	GKSS	Federal Republic of Germany
Japan Atomic Energy Research Institute	JAERI	Japan
Kyoto University Research Reactor Institute	KURRI	Japan
Netherlands Energy Research Foundation	ECN	Netherlands
Oak Ridge National Laboratory	ORNL	United States of America
Österreichisches Forschungszentrum Seibersdorf	ÖFZS	Austria
Transnuklear GmbH	TN	Federal Republic of Germany
United Kingdom Atomic Energy Authority	UKAEA	United Kingdom
University of Michigan - Ford Nuclear Reactor	FNR	United States of America
U.S. Department of Energy	USDOE	United States of America

The IAEA is grateful for the contributions volunteered by these organizations and thanks their experts for preparing the detailed investigations and for evaluating and summarizing the results presented in this Guidebook.

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Appendix L
STARTUP EXPERIMENTS

Appendix L-1

STARTUP EXPERIMENTS WITH REDUCED ENRICHMENT FUELS

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Abstract

A summary of necessary and recommended experiments for startup of a reactor with reduced enrichment fuels is provided.

1. General remarks

Recommended startup procedures and experiments depend on a number of factors such as:

- the completeness of nuclear and thermodynamical calculations,
- the completeness of dynamic and safety related calculations,
- the comparison of the actual old (HEU) and new (LEU) core design,
- the operation with mixed (HEU + LEU) or only new (LEU) reactor cores,
- old and new U-5 content,
- changed or unchanged fuel element design,
- changed or unchanged control rod design,
- knowledge of thermal flux distribution (power distribution),
- knowledge of burnup values,
- the trip values and safety margins.

Having most of these calculations and the needed knowledge one has to perform startup experiments for two reasons

- to check the calculations,
- to learn from experiments as flux and power distributions and reactivity values will change and as normally the core configuration for research reactors is not fixed.

In the following, proposals for necessary and recommended startup experiments are given together with a short commentary (see also /1/)

2. Necessary startup experiments

2.1 Critical experiments for the standard and for modified core configurations

C.: Only for a few reactors the increase in U-235 content will be only of a small amount (15 - 20 %) for compensating the reactivity effects. Normally for many reasons (e.g. economics) the U-5 content will be much higher. Therefore careful critical experiments are necessary to learn the reactivity behaviour of the new fuel elements and control rod worths for different core configuration.

2.2 Neutron flux measurements

A) Local thermal neutron flux distributions

a. in fuel elements parallel to the fuel plates,

b. fuel elements plus control fuel elements vertical to the fuel plates,

c. near irradiation positions.

C.: The local thermal neutron flux influencing the formfactors will change especially when using mixed cores. One has to be very careful to assure that no unallowed formfactors may occur.

B) Global thermal neutron flux distributions

C.: For the safety analysis it is necessary to have in addition to the local power distribution the global power distribution with its changes for different core configurations, control rod heights and burnup.

C) Neutron spectra for different positions

C.: The enrichment reduction will lead to a harder neutron spectrum and in many cases to changes in core configuration. The new spectra and ratios are necessary for planning and discussing the experiments in and outside the reactor core.

D) Neutron fluxes and Gamma-fluxes in the irradiation positions

C.: The enrichment reduction will lead to a harder neutron spectrum and in many cases to changes in core configuration. The new spectra and ratios are necessary for planning and discussing the experiments in and outside the reactor core.

2.3 Reactivity values

A) Control rod worths, reactivity speed

C.: For different and new core configurations surprising results can be obtained. Normally, limitations on the reactivity speed exist in safety reports when considering startup accidents.

B) Reactivity worth of some elements and reflector elements

C.: These reactivity measurements are of interest for the reactor operator to have a practical knowledge when changing core configurations using different type of loops for different positions and for e.g. replacing parts of the reflector by fuel elements.

C) Reactivity worth of loops, rigs, capsules etc.

C.: These reactivity measurements are of interest for the reactor operator to have a practical knowledge when changing core configurations using different type of loops for different positions and for e.g. replacing parts of the reflector by fuel elements.

D) Reactivity values when replacing oval type control rods with forked type control rods.

C.: As the controlable reactivity by one control rod may change by more than 50 %, many new experiences will be obtained. And it is necessary to make them stepwise.

E) Control rod drop time

C.: Changes may be possible if other types of control rods are used.

2.4 Fuel elements with thermocouples

Fuel elements instrumented with thermocouples will be very helpful in the licensing procedure. The fuel plate temperature measurements are an additional check of the nuclear and thermodynamical calculations. Such instrumented fuel elements are only necessary during startup and first fuel cycles.

A) Actual fuel plate temperature in selected positions

C.: To be sure that nuclear and thermodynamical design calculations are correct, fuel plate temperature measurements for selected positions are recommended. As these will be only point measurements, special hot spot factors have to be used when comparing the measured temperatures with the theoretically allowed temperatures.

B) Loss of flow experiments with different trip values and for different experimental conditions (position of the instrumented fuel element, different failure considerations)

C.: Of interest is the temperature behaviour between stopping the primary coolant and the scram of the reactor and the flow inversion. Of importance for the first case are the trip values and for the second case the considerations from the failure tree especially for those reactor operators who up to now never performed such experiments.

2.5 Coolant flow distribution

C.: Of main importance if the fuel element geometry and the control fuel element geometry will be changed.

3. Recommended startup experiments

3.1 Isothermal temperature coefficient

C.: Necessary for safety calculations. Measurements, e.g. by cooling the primary water.

3.2 Power coefficient of reactivity

C.: Of interest for the reactor operator and easy to measure.

3.3 Pressure drop

C.: Of main importance if the fuel element geometry and the control fuel element geometry will be changed.

3.4 Criticality of fresh and spent fuel storage (0,95)

C.: If a higher U-5 content will be used one has to assure that k 0,95 in any case. Measurements e.g. with pulsed neutron technique.

3.5 Reactivity values for Xe-equilibrium, Xe-peaking and burnup

C.: For long term fuel cycle planning and for restarting a reactor after a scram, these reactivity values for actual core configurations are of great importance to the reactor operator.

4. Finally

More detailed recommendations for the needed startup experiments can only be given for an actual case.

These recommendations are influenced by the experiences of a research reactor operator. From other standpoints (designer, theorist, independent experts or safety authorities) more or less of other startup experiments may be recommended or required.

REFERENCE

/1/ IAEA-TECDOC-304
Core Instrumentation and pre-operational procedures for core conversion HEU to LEU, IAEA 1984

Appendix M

EXPERIENCE WITH MIXED AND FULL CORE OPERATION

Abstract

Experience with mixed and full cores of HEU fuel and reduced enrichment fuels are described for several reactor designs that range from a coupled-core critical facility to high power reactors with both rodded and plate-type fuels.

MIXED CORES

Appendix M-1

CRNL EXPERIENCE WITH RESEARCH REACTOR FUEL CONVERSION AND MIXED CORE OPERATION

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Abstract

The NRX and NRU reactors at CRNL, both originally fuelled with natural uranium metal, have undergone a number of fuel conversions before arriving at the present highly enriched uranium-aluminum alloy fuel designs. The history of the changes and the regulatory approvals required to make the changes are summarized.

1. INTRODUCTION

The Chalk River Nuclear Laboratories (CRNL) in Canada have had a considerable amount of experience in fuel conversion and mixed core operation of the two high power, heavy water research reactors located there. Both NRX, built in 1947, and NRU, built in 1957, were originally fuelled with natural uranium metal. Both reactors have gone through several fuel conversions before arriving at the present design of highly enriched (HEU) uranium-aluminum alloy pin-type fuel loadings.

All major fuelling changes required some form of safety and hazards assessment, and approval by the appropriate regulatory body. None of the changes created any major problems or significantly affected the safety of the reactors.

The various fuel changes which occurred and resulting mixed cores, and the regulatory approvals that were necessary are summarized below, first for NRX, then NRU.

2. NRX EXPERIENCE

Both NRX and NRU are large volume tank-type reactors. NRX is D_2O moderated, H_2O cooled, and graphite reflected. With the current HEU loading it operates at about 25 MW although it has operated at up to 42 MW in the past. Vertical through tubes in the reactor vessel (tank) form 199 lattice sites in a hexagonal arrangement. Fuel assemblies, shut-off rods, and various experiments are installed in these through-tubes, suspended inside individual flow tubes.

With the initial U-metal loading, almost all lattice sites other than shut-off rods, were occupied by fuel assemblies, although by the mid-1950's a number of Pu-Al or enriched U-Al alloy assemblies had been installed as "boosters" to free some lattice sites for experiments. By 1958 the loading consisted of about 130 to 140 U-metal fuel assemblies, about 30 "boosters", and about 20 miscellaneous experiments and isotope irradiations.

Table 1 summarizes year-end reactor loadings from 1958 through 1981. Plans to convert NRX to natural UO_2 fuel were begun in the late 1950's and in 1960 a test irradiation of some 40 to 50 UO_2 fuel assemblies, installed in place of U-metal fuel assemblies, was begun. No formal safety and hazards assessment was performed at this time, but the change was reviewed extensively in the normal process for approval of experimental irradiations.

Beginning in 1962 a gradual conversion to a reactor loading with natural UO_2 fuel plus 7 pin U-Al alloy (93% U-235, 0.90 g/cm^3) was carried out in order to increase the neutron flux in the reactor and provide more room for experiments. A safety and hazards assessment of the new fuel loading was carried out and approval for the change obtained from the regulatory body. The loading after this change consisted of about 80 UO_2 assemblies, 50 U-Al (7 pin) assemblies, 30 miscellaneous experiments and isotope irradiations, and a number of vacant positions.

Between 1968 and 1970 the UO_2 fuel assemblies were gradually removed leaving the reactor fuelled entirely with 93% enriched U-Al fuel about 70 assemblies. At the same time the thermal power of the reactor was reduced from 42 MW to 30 MW while maintaining the same neutron flux levels. Coincident with this change a complete safety and hazards assessment of the NRX reactor was performed and approved by the regulatory body.

A further change occurred between 1972 and 1974 when a slightly different design of 7 pin U-Al alloy fuel assembly was introduced. The new design featured a thinner cladding, smaller flow area, and a longer fuel length (2.74 m versus 2.44 m). This required an addendum to the previous safety and hazards analysis and regulatory approval. Since 1974 the reactor has remained entirely fuelled by 2.74 m long 7 pin U-Al fuel assemblies (93% enriched), although the number of assemblies has varied.

Note that each of the above changes was made gradually so that the reactor in fact went through a series of mixed loadings intermediate between the old and new. At no time did any of the loading changes cause any significant problems in the operation of the reactor.

3. NRU EXPERIENCE

The NRU reactor is D_2O moderated and cooled, and has an H_2O reflector. While it initially operated at 220 MW, with the current loading it operates at 125 MW. Unlike NRX, the NRU reactor vessel (tank) does not have through tubes, but has a hexagonal array of 227 lattice sites formed by tubes extending upwards from the top of the tank through the upper shields of the reactor. Like NRX, the assemblies installed in each lattice position are suspended within their own individual flow tubes.

Initially NRU was fuelled with about 190 natural uranium metal fuel assemblies each consisting of five 3.05 m long plates (or "flats") in an aluminum flow tube. During the early 1960's some HEU assemblies were installed, primarily as test irradiations. Table 2 summarizes NRU year-end loadings from 1961 to 1981.

TABLE 1: NRX mixed core history

Year	Core Loading at End of Calander Year Number of Assemblies of Each Type Occupying Lattice Sites										
	U-Metal Fuel	UO_2 Solid Fuel	UO_2 Annular Fuel	U-Al (HEU) Slug Fuel	Pu-Al Slug Fuel	7-Pin U-Al Fuel (HEU) 2.44 m	7-Pin U-Al Fuel (HEU) 2.74 m	Fast Neutron	Expt., Isotope and Miscellaneous	S.O. Rods and Adjuster Rods	Vacant or Blocked*
1958	136	1	-	6	22			5	19	10	0
1959	128	4	1	17	13			4	22	10	0
1960	86	43	-	29	-			5	26	10	0
1961	91	42	-	29	-	-		6	21	10	0
1962	26	41	31	16	-	24	-	7	21	10	23
1963	15	38	34	-	-	52	-	8	26	10	16
1964	8	32	47			52	-	5	31	10	14
1965	1	24	59			51	3	9	28	10	14
1966	-	23	59			53	-	8	28	10	18
1967	-	7	63			40	13	9	27	10	30
1968	-	-	44			45	14	8	23	10	55
1969	-	-	20			60	5	9	29	10	66
1970	-	-	-			66	3	5	21	10	94
1971						65	4	9	23	10	88
1972						36	36	9	31	10	77
1973						9	63	11	31	10	75
1974						-	78	10	27	10	74
1975							77	8	24	10	80
1976							81	7	22	10	79
1977							54	6	15	10	114
1978							56	5	19	10	108
1979							52	3	18	10	116
1980							50	2	13	10	124
1981							60	1	14	10	114

Fuel Assembly Designs - NRX

U-Metal Fuel: Natural U cylinder - 34.5 mm OD x 3.06 m long - Al clad (2 mm)

UO_2 Solid: Natural UO_2 pellets - 35.8 mm OD x 3.05 m long (total) - Al clad (1.27 mm)

UO_2 Annular: Natural UO_2 pellets - 35.8 mm OD x 15.3 mm ID x 3.05 m long (total) - Al clad (1.27 mm)

U-Al Slug (HEU): U-Al alloy - 93% U-235 in U - 0.20 g/cm^3 U-235 - Al clad (2 mm) - 12 slugs 34.5 mm OD x 203.2 mm long - 25.4 mm Al spacer between each

Pu-Al Slug: Pu-Al alloy - 0.10 g/cm^3 Pu - 12 slugs 34.5 mm OD x 230 mm long - Al clad (2 mm)

7 Pin U-Al: U-Al alloy - 93% U-235 in U - 0.90 g/cm^3 U-235 - cluster of 7 pins - (a) 6.35 mm OD x 2.44 m long - 1.14 mm Al clad. (b) 6.35 mm OD x 2.74 m long - 0.76 mm Al clad.

Fast Neutron: Annular fuel rod with dry central cavity - may be UO_2 or U-Al

*Vacant or blocked lattice sites are generally peripheral, low flux positions, although a few high flux positions are blocked due to tube leakage.

TABLE 2: NRU mixed core history

Year	Core Loading at End of Calander Year Number of Assemblies of Each Type Occupying Lattice Sites											
	U-Metal Fuel	Double Annular (HEU) Fuel	Single Annular (HEU) Fuel	12 Pin U-Al Fuel 0.18 g/cm^3 U-235	12 Pin U-Al Fuel 0.27 g/cm^3 U-235	12 Pin U-Al Fuel 0.39 g/cm^3 U-235	12 Pin U-Al Fuel 0.50 g/cm^3 U-235	12 Pin U-Al Fuel 0.63 g/cm^3 U-235	Fast Neutron and Flux Peaking	Expt., Isotope and Miscellaneous	Control Rods and Adjuster Rods	Vacant or Blocked*
1961	192	4	-	-	-					3	18	10
1962	187	5	-	5	-				1	4	18	7
1963	184	6	5	2	-				1	4	18	7
1964	-	-	33	67	-				10	17	22	78
1965	-	-	9	-	75	-			11	30	22	80
1966			-	-	-	81	-		12	49	22	63
1967					-	78	4	-	12	54	22	57
1968						-	81	-	13	50	22	61
1969						-	58	29	15	42	22	61
1970							-	86	14	68	22	37
1971							-	83	10	67	22	45
1972												
1973			S h u t d o w n f o r V e s s e l R e p l a c e m e n t									
1974							-	73	6	49	22	77
1965							16	64	10	46	22	69
1976							54	28	12	44	22	67
1977							7	79	10	43	22	66
1978							1	86	9	44	22	65
1979							-	88	10	41	22	66
1980								88	9	42	22	66
1981								88	10	42	22	65

Fuel Assembly Designs - NRU

U-Metal Fuel: 5 natural U metal plates 3.05 m long - (2 at 31.1 mm x 4.5 mm; 2 at 49.8 mm x 4.3 mm; 1 at 54.5 mm x 4.3 mm) - Al clad 0.62 mm

Double Annular HEU: U-Al Alloy - 93% U-235 in U - 0.36 g/cm^3 U-235 - 2 concentric tubes 3.05 m long - inner 35.6 mm OD x 1.78 mm thick, outer 52.3 mm OD x 1.52 mm thick - Al clad 0.76 mm

Single Annular HEU: U-Al alloy - 93% U-235 in U - less than or equal to 0.36 g/cm^3 U-235
- Fuel annulus 51.5 mm OD x 0.76 mm thick x 2.74 m long

12 Pin U-Al: U-Al alloy - 93% U-235 in U
- 12 pins - 5.48 mm OD x 2.74 m long - clad in Al (0.76 mm)
- various U-235 densities have been used by varying alloy content:
0.18 g/cm^3
0.27 g/cm^3
0.39 g/cm^3
0.50 g/cm^3
0.63 g/cm^3

Fast Neutron: 15 fuel pins form an annulus with a dry central cavity
- fuel may be UO_2 or ThO_2-UO_2

Flux Peaking: Used to increase neutron flux in certain regions of core
- 19 pin ThO_2-UO_2

*Vacant or blocked lattice sites are generally peripheral, low flux positions of limited value.

During 1964 the reactor was shutdown and converted directly to highly enriched (HEU) U-Al alloy fuel. The reactor thermal power was reduced from 220 MW to 70 MW while maintaining the same neutron flux levels. A complete safety and hazards assessment of the reactor was carried out at this time and regulatory approval was obtained. Initially two types of U-Al fuel assembly (about 100 assemblies in total) were installed; an annular U-Al alloy design with U-235 density of about 0.36 g/cm^3, and a 12 pin U-Al alloy design with 0.18 g/cm^3 U-235. As experience was gained with operation of the enriched reactor, the fuel was gradually changed to the 12 pin design (about 80 assemblies) with a slightly higher U-235 density (0.27 g/cm^3). This change had been addressed and approved with the initial safety and hazards assessment.

Between 1966 and 1970 the reactor loading was changed three more times. Each change simply involved an increase in the U-235 content of the U-Al fuel by increasing the quantity of uranium in the alloy. The U-235 density increased to 0.39 g/cm^3 in 1966, to 0.50 g/cm^3 in 1968 and to 0.63 g/cm^3 in 1970. For each of these changes an addendum to the previously approved safety and hazards assessment of the reactor was required and regulatory approval was obtained. As the fuel loading increased, reactor power was gradually increased to 125 MW.

Note that these later changes occurred gradually with the reactor loading going through a number of intermediate mixed stages. None of the loading changes caused any significant problems.

Appendix M-2

EXPERIENCE WITH MIXED CORES IN THE ASTRA REACTOR

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Abstract

Core configurations composed of different MTR-type fuel elements were operated in the ASTRA-reactor. A thermal hydraulics analysis was performed by calculating the safety margins. Results of measurements of LEU-test elements in the ASTRA-core are presented.

1) Introduction

The ASTRA-reactor, a pool type research reactor with a thermal power of 8 MW, is in operation since 1960. In the course of reactor operation several modifications of MTR-type fuel elements have been used. Reasons for the change of the fuel element design were mainly the incentive to improve the performance of the core and to decrease the cost of reactor operation. Later on concerns on proliferation became important. The main changes of the fuel element design affected element geometry, the number of plates per fuel element, the amount of uranium per plate and the enrichment of fuel. New fuel elements with different design were loaded into the existing core gradually step by step so that mixed cores arose and were in operation for a longer time period. Two important mixed core configurations are described in this paper. The first type of mixed core configurations described is related to the transition phase from curved to straight fuel element geometry, the second type is related to the test of LEU-fuel elements in the ASTRA-core. For all mixed core configurations a thermal-hydraulic safety analysis is performed.

2) Fuel Element Types

Table 1 gives a description of the essential characteristics of all fuel elements used in the ASTRA-reactor since start of reactor operation in 1960. As can be seen from the table the most important change ocurred in 1969 when the geometry of

TABLE 1. FUEL ELEMENT DESCRIPTION

Year of Delivery	1960	1965	1969	1974	1982	1982
Plates/Standard F.E.	16	19	23	23	23	20
Plates/Control F.E.	7	9	17	17	-	-
Shape of Plate	curved	curved	straight	straight	straight	straight
Outer Plates	Aluminium	Fuel	Fuel	Fuel	Fuel	Fuel
Plate Thickness (mm)	1.27	1.27	1.27	1.27	1.27	1.60
Water Channel Thickness (mm)	3.12	2.95	2.23 (2.12*)	2.23	2.23	2.44
Uranium Enrichment (%)	90	90	90 (93*)	90	45	20
Fuel Meat Material	UAl	UAl	UAl	UAl	UAl_x-Al	U_3O_8-Al
U-235 (g)/Standard F.E.	193	197	263	285	320	350
U-235 (g)/Control F.E.	84	93	184	206	-	-
Gap for Control Rod (mm)	1 x 28.5	1 x 28.5	2 x 6.2	2 x 6.2	-	-

* Data for Control Fuel Element

fuel plates changed from curved to straight plates, the number of plates per standard fuel element from 19 to 23, U-235 weight from 197 g to 263 g and the water gap decreased from 2.95 mm to 2.23 mm. The change of the control fuel element and the control rod was even yet more important. The central bar-type absorber-rod consisting of boron carbide was replaced by fork-type absorber blades made of a Ag-In-Cd alloy. By eliminating the central water gap in the control fuel element a strong reduction in flux and power peaking could be achieved.

In 1982 the first test element with 20 % enriched uranium, 350 g U-235 and 20 fuel plates was loaded into the core. The design of this LEU-test fuel element was based on calculations performed in cooperation with the Argonne National Laboratory in the RERTR program of the I.A.E.O.

3) General Characterization of Mixed Cores

A core can be characterized as mixed, if fresh fuel elements of different design are loaded and are simultaneously together in the core. MTR-type fuel elements may differ in many aspects as number of plates, thickness and shape of plate, uranium weight

per plate, enrichment, fuel meat material, area of meat and so on. But even if fresh fuel elements with the same design and uranium weight are introduced in the core, fuel elements became quickly different if burnup is becoming effective. In the equilibrium core of the ASTRA-reactor fuel elements are simultaneously present which differ in U-235 weight by more than a factor 3. If one looks at the list of the fuel elements in the ASTRA-reactor (Table 1) one will notice that the normal equilibrium core (Fig. 9) is a mixed core because of the different water gaps in the control fuel element and standard fuel element. This means that mixed cores are rather the rule than the exception in the ASTRA-reactor.

4) Thermalhydraulic Analysis of Mixed Cores

The thermalhydraulic analysis of mixed core configurations was carried out with methods outlined in several parts of the guidebook for core conversion [1]. Peak heat flux at ONB, peak heat flux at DNB using the Mirshak corellation and peak heat flux at onset of flow instability were calculated for each fuel element type in the core using the measured pressure drop across the core at a flow rate of 14 m^3/min and a water inlet temperature of 38° C.

The measured pressure drop across the core agreed fairly well with the calculated values. The pressure drop across the fuel element was calculated for different fuel element types and is shown in Fig. 1.

5) Mixed Core Configurations with Curved and Straight Type Fuel Elements

In 1969 the fuel elements in the ASTRA-reactor were changed in several aspects (Table 1). The most limiting aspect from the standpoint of operation was the transition from curved to straight fuel plate shape. It demanded a specific strategy for core conversion. It was no longer possible to load only one fresh fuel element in the core but it was necessary to replace at once a complete row of fuel elements. Typical core configurations in the transition phase are shown in Fig. 2. Fig. 3 indicates

Type of Fuel Element	Number of Fuel Plates	Water Channel Thickness [mm]
C.F.E.	17	2.12
ST.F.E.	23	2.23
LEU-Test F.E.	20	2.44
ST.F.E.(curved)	19	2.95

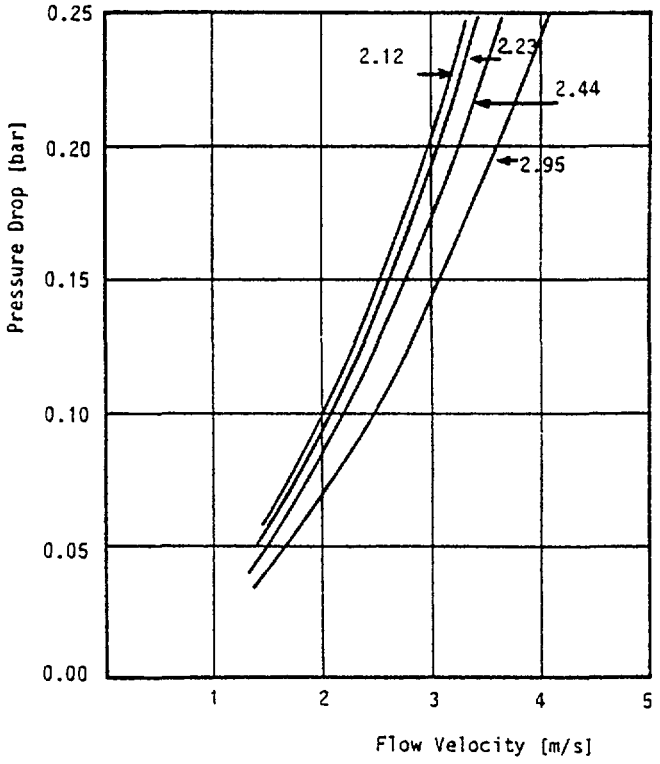


Fig. 1 Pressure Drop across Fuel Element for Different Fuel Element Types

the flux and power distribution in a core configuration with one row of fuel elements replaced. The thermal hydraulic analysis for standard fuel elements with 19 plates, 23 plates and a control fuel element with 17 plates is given in Fig. 4. The result of the analysis is summarized in Table 2 for three core configurations:

VI/2 a core configuration before start of change
VI/8 a core configuration with one row of S.T.F.E. replaced
VII/8 a core configuration with two rows of fuel elements replaced including also two new control elements

The peak heat fluxes at DNB or flow instability of the three core configurations analysed are indicated in Fig. 4 and are connected with a dashed line.

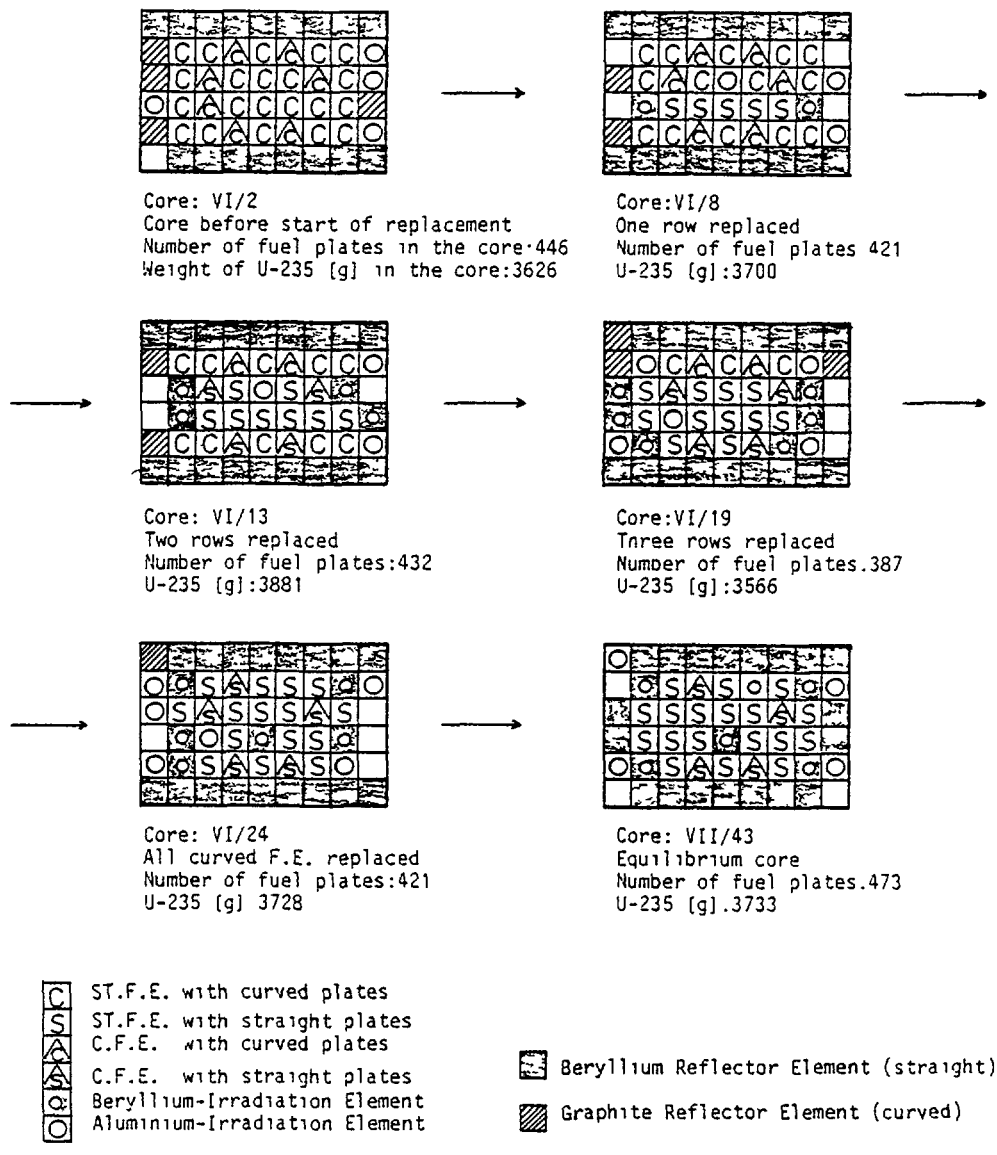


Fig. 2 Mixed Core Configurations in the Transition Period from Curved to Straight Type Fuel Elements

One can see that the peak heat flux at onset of flow instability decreased remarkably for the new standard fuel element and even more in the new control fuel element. However this effect was by far compensated by the additional number of fuel plates as can be seen from the margins to DNB and flow instability in the mixed core configurations VI/8 and VI/13. Unexpectedly the margins are higher in the mixed cores. After the core conversion from curved to straight type fuel elements a new equilibrium core was obtained. In the new equilibrium core the fuel shuffling pattern was reversed. New fuel elements are introduced at the edge of the core.

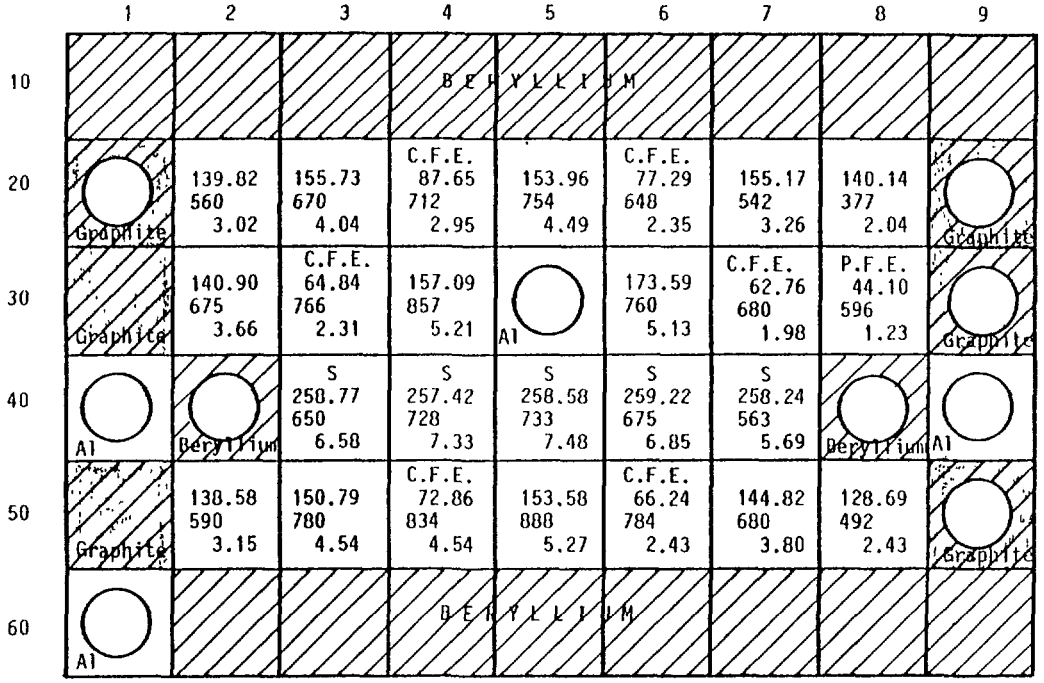
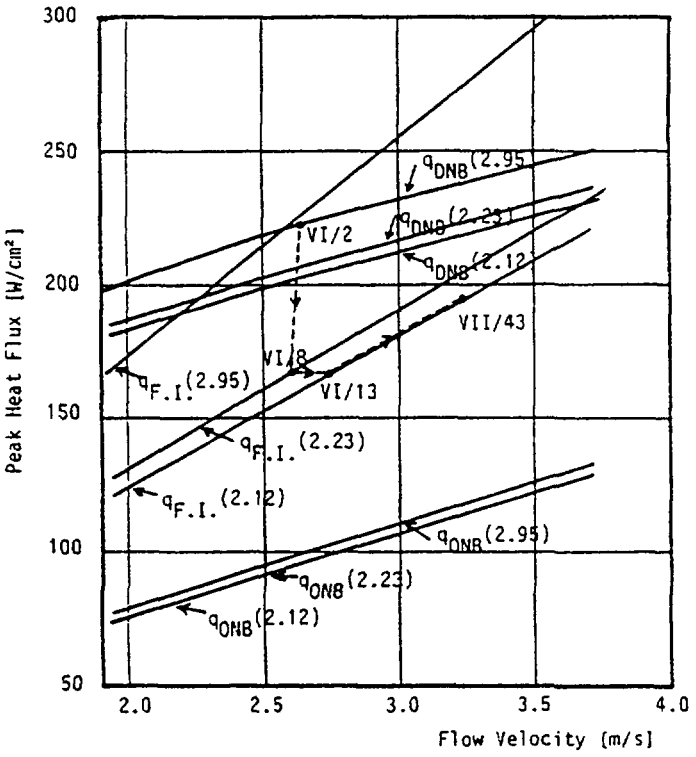


Fig. 3 Mixed Core Configuration VI/8 with Curved and Straight Type Fuel Elements Thermal Neutron Flux and Power Distribution



Type of Fuel Element	Number of Fuel Plates	Water Channel Thickness [mm]
C.F.E.	17	2.12
ST.F.E.	23	2.23
ST.F.E.(curved)	19	2.95

Fig. 4 Peak Heat Flux at ONB, DNB (Mirshak) and Flow Instability for Fuel Element Types in the Transition Period from Curved to Straight Type Fuel Elements

TABLE 2. THERMAL HYDRAULIC ANALYSIS OF MIXED CORE CONFIGURATIONS WITH CURVED AND STRAIGHT TYPE FUEL ELEMENTS

Reactor Power: 6 MW Flow Rate: 14 m^3/min. Inlet Temperature: 38° C

Core Configuration		Number of Fuel Plates	Pressure Drop across Core [bar]	Flow Velocity [m/s]	Total P.P.F.	Average Heat Flux [W/cm^2]	Peak Heat Flux [W/cm^2]	q_{ONB} [W/cm^2]	q_{DNB} (Mirshak) [W/cm^2]	$q_{F.I.}$ [W/cm^2]	Margin to ONB	Margin to DNB	Margin to F.I.
VI/2	curved S.T.F.E.	385	0.111	2.64	2.25	18.68	42.04	99	222	225	2.35	5.28	5.35
	curved C.F.E.	61		2.64	3.38	18.68	63.15	99	222	225	1.57	3.52	3.56
VI/8	curved S.T.F.E.	252	0.147	3.09	1.76	19.79	34.83	113	233	260	3.24	6.69	7.46
	curved C.F.E.	54		3.09	2.62	19.79	51.85	113	233	260	2.18	4.49	5.01
	straight S.T.F.E.	115		2.60	2.04	18.91	38.58	94	205	167	2.44	5.31	4.33
	straight C.F.E.	-		-	-	-	-	-	-	-	-	-	-
VI/13	curved S.T.F.E.	187	0.185	3.35	1.91	19.28	36.82	121	238	281	3.29	6.46	7.63
	curved C.F.E.	36		3.35	3.21	19.28	61.89	121	238	281	1.96	3.85	4.54
	straight S.T.F.E.	175		2.83	1.85	18.44	34.12	102	211	181	2.99	6.18	5.30
	straight C.F.E.	34		2.75	1.87	18.44	34.48	102	206	167	2.96	5.97	4.84

S.T.F.E. = Standard Fuel Element

C.F.E. = Control Fuel Element

TABLE 3. THERMAL HYDRAULIC ANALYSIS OF MIXED CORE CONFIGURATIONS WITH CURVED AND STRAIGHT TYPE FUEL ELEMENTS

Reactor Power: 8 MW Flow Rate: 14 m³/min. Inlet Temperature: 38° C

Core Configuration	Number of Fuel Plates	Pressure Drop across Core [bar]	Flow Velocity [m/s]	Total P.P.F.	Average Heat Flux [W/cm²]	Peak Heat Flux [W/cm²]	q_{ONB} [W/cm²]	q_{DNB} (Mirshak) [W/cm²]	$q_{F.I.}$ [W/cm²]	Margin to ONB	Margin to DNB	Margin to F.I.
Equilibrium core VII/43 ST.F.E.	405	0.227	3.32	1.91	22.46	42.90	117	224	210	2.73	5.22	4.90
C.F.E.	68		3.22	2.50	22.46	56.15	117	218	193	2.08	3.88	3.44
VIII/6 ST.F.E.	382	0.227	3.32	2.26	22.60	51.08	117	224	210	2.29	4.39	4.11
C.F.E.	68		3.22	1.40	22.60	31.64	117	218	193	3.70	6.89	6.10
LEU-Test F.E.	20		3.51	2.38	22.60	53.79	125	233	236	2.32	4.33	4.39
VIII/9 ST.F.E.	359	0.227	3.32	1.98	22.75	45.05	117	224	210	2.60	4.97	4.66
C.F.E.	68		3.22	2.57	22.75	58.47	117	218	193	2.00	3.73	3.30
LEU-Test F.E.	40		3.51	2.79	22.75	63.47	125	233	236	1.97	3.67	3.72

ST.F.E. = Standard Fuel Element

C.F.E. = Control Fuel Element

LEU-Test F.E. = Low Enriched Uranium Test Fuel Element

Thermalhydraulic analysis was performed for a typical configuration VII/43 of the equilibrium core. The results are summarized in Table 3.

Comparing the results with those in Table 2 one can see that this core conversion caused a remarkable increase in core performance allowing the increase of reactor power from 6 to 8 MW without decreasing the safety margins.

6) Mixed Core Configurations with LEU-Test Fuel Elements

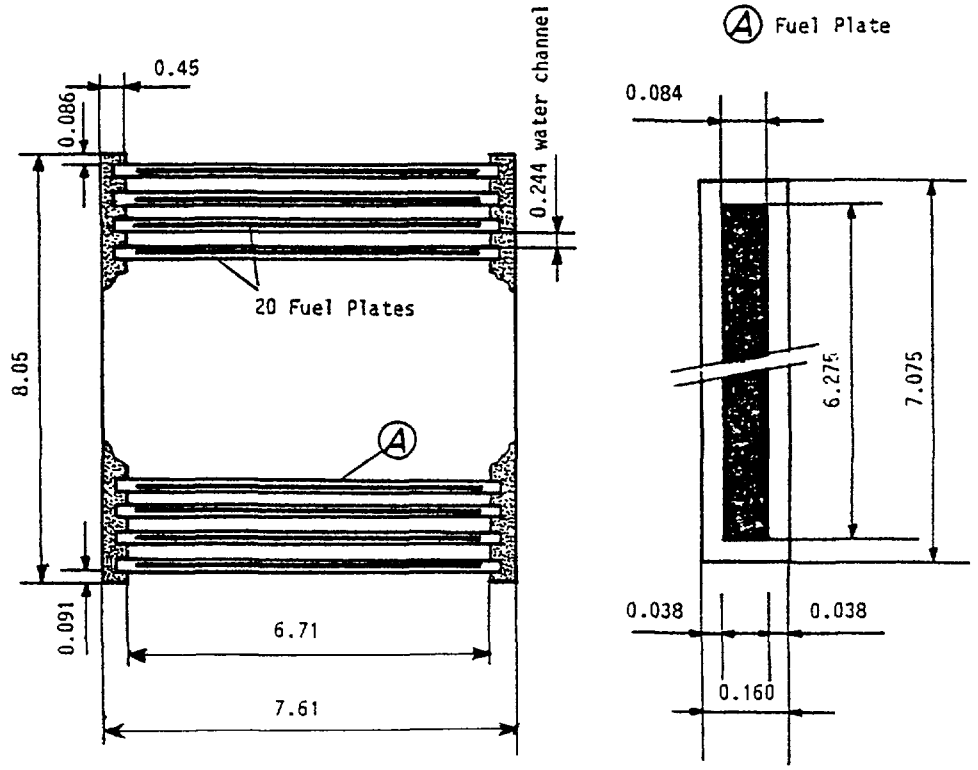
Based on neutronic and thermalhydraulic calculations performed in close cooperation with ANL it was decided to choose a 20 plate type LEU-fuel element (Fig. 5) with 350 g U-235 as a prototype test element for the ASTRA-reactor. Three LEU-test fuel elements were fabricated by NUKEM. The first LEU-test fuel element was loaded into the core in April 1982, the second in November 1982. The resulting mixed core configurations are shown in Fig. 7 and Fig. 8.

The thermalhydraulic analysis for a standard fuel element with 20 and 23 plates and a control fuel element with 17 plates is given in Fig. 6. The results of the analysis is summarized in Table 3 for three core configurations:

VII/43 equilibrium core of the ASTRA-reactor based on 23 plates ST.F.E. and 17 plates C.F.E.
VIII/6 one LEU-test fuel element in the core. The control fuel elements in the core have all high burnup values.
VIII/13 two LEU-test fuel elements in the core, one MEU-ST.F.E. (45 % enrichment), one fresh C.F.E.

The following conclusions can be drawn from the results in Table 3:

- Safety margins of LEU-test elements are by approximately 25 % lower than for HEU-ST.F.E. in the same position due to the higher uranium contents and the reduced number of fuel plates.
- Safety margins of LEU-ST.F.E. are still higher than those of HEU control fuel elements.



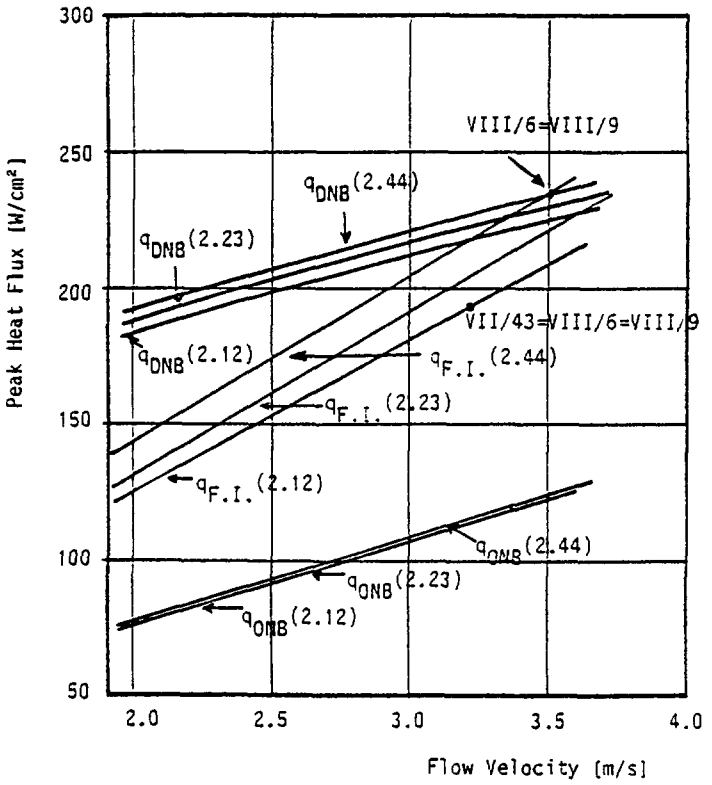
All dimensions in cm
The two outer fuel plates have a clad thickness of 0.048 cm
Lattice pitch 8.1 x 7.71

Fig. 5 ASTRA-Reactor
LEU-Test Fuel Element (20 Plates)
Fuel Material: U_3O_8-Al
U-235 Weight [g] in Fuel Element: 350
Enrichment [%]: 19.50
Uranium density [g/cm³]: 2.84

This conclusions are in agreement with those drawn from theoretical calculations made by ANL and Ö.F.Z.S.

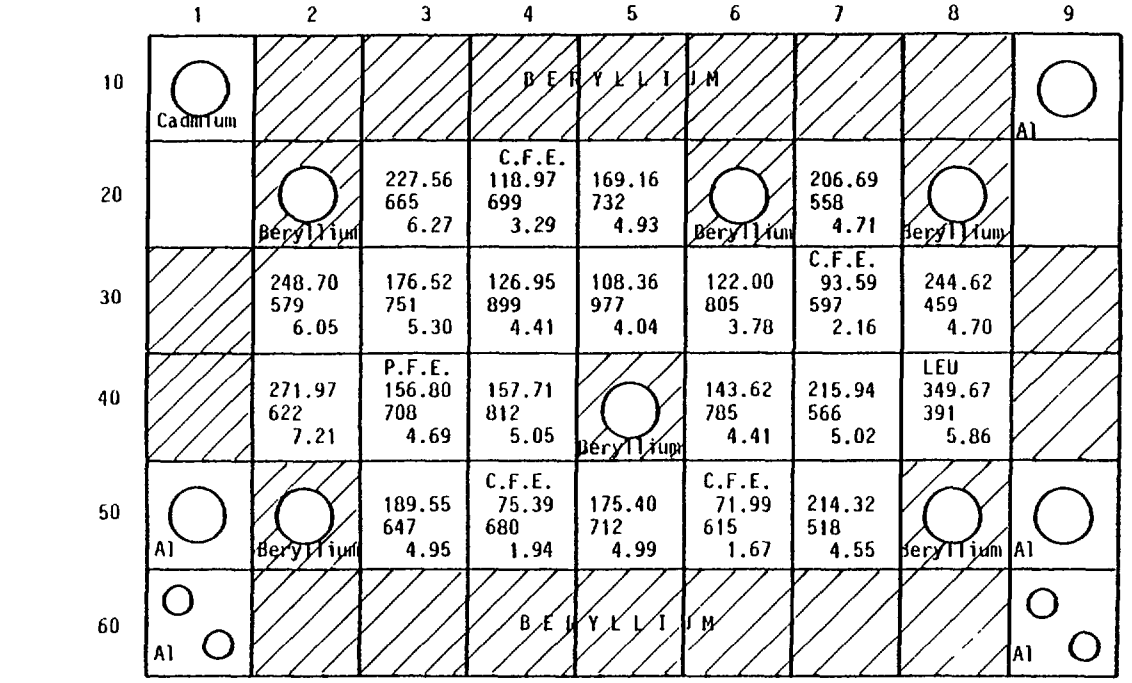
7. Reactivity and Flux Measurements with LEU-Test Fuel Elements

Before the LEU-test fuel elements were loaded into the core for power operation a series of measurements and comparison with HEU-S.T.F.E. and MEU-ST.F.E. (45 % enrichment) were carried out.



Type of Fuel Element	Number of Fuel Plates	Water Channel Thickness [mm]
C.F.E.	17	2.12
ST.F.E.	23	2.23
LEU-Test F.E.	20	2.44

Fig. 6 Peak Heat Flux at ONB, DNB (Mirshak) and Flow Instability for Fuel Element Types in Mixed Core Configurations with LEU-Test F.E.



LEU
349.67
391
5.86

LEU = LEU-test fuel element C.F.E. = Control fuel element P.F.E. = Partial fuel element
U-235 weight [g] in fuel element at begin at cycle
Relative average thermal neutron flux in the fuel element
Thermal power of the fuel element in percentage of reactor power [%]

Fig. 7 Mixed Core Configuration VIII/6 with one LEU-Test Fuel Element Thermal Neutron Flux and Power Distribution

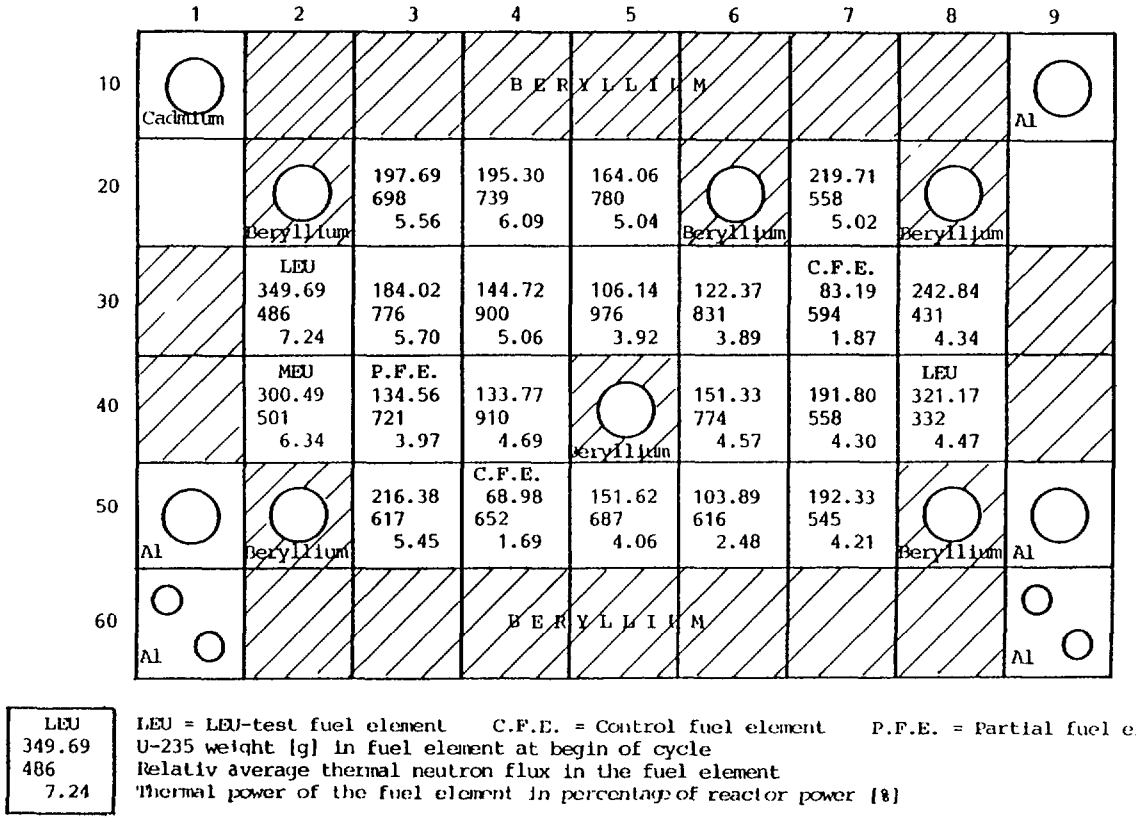


Fig. 8 Mixed Core Configuration VIII/9 with two LEU-Test Fuel Elements
Thermal Neutron flux and Power Distribution

7.1 Reactivity Measurement

The reactivity of LEU-test fuel elements was measured in comparison to HEU-ST.F.E. and MEU-ST.F.E. in two different positions of the core, in Pos. 48 on the edge of the core and in Pos. 35 in the centre of the core.

	MEU to HEU [Δk/k %]	LEU to HEU [Δk/k %]
Pos. 48	+ 0.01	- 0.16
Pos. 35	+ 0.02	- 0.33

7.2 Flux Measurements

The thermal neutron flux was measured with copper wires in the HEU-ST.F.E., MEU-ST.F.E. and LEU-test fuel element in a position (Pos. 48) at the core edge.

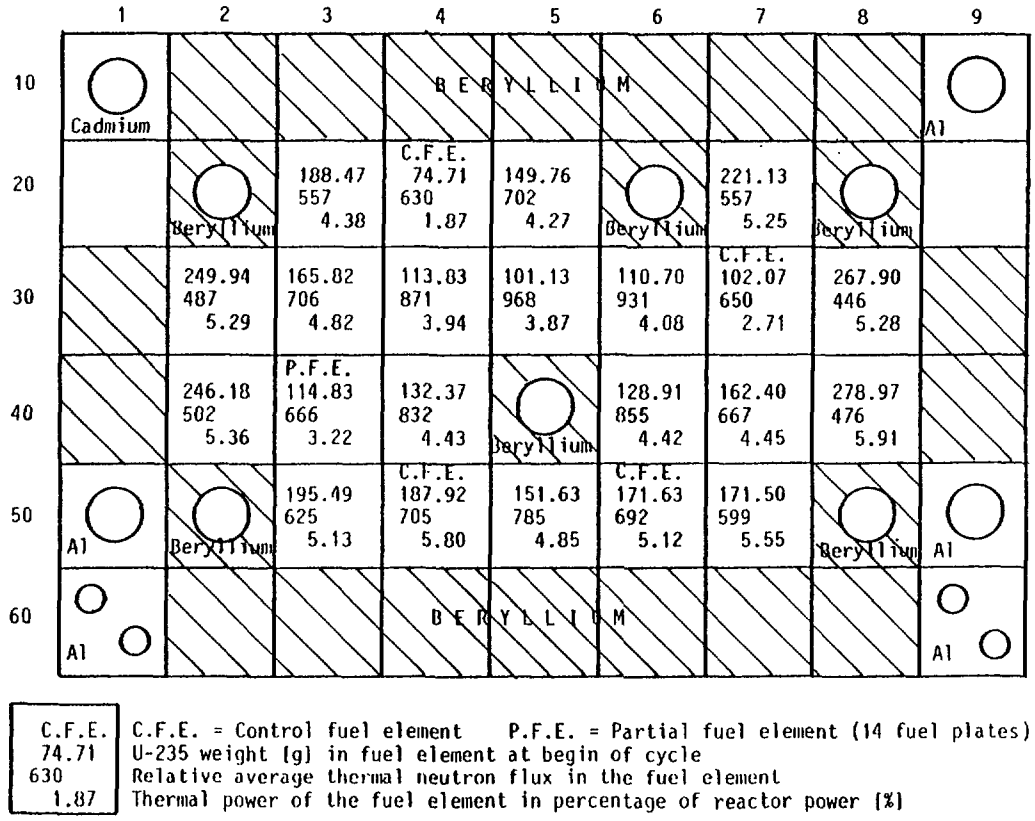


Fig. 9 Equilibrium Core VII/43
Thermal Neutron Flux and Power Distribution

From this measurements the following power peaking factors were derived:

	HEU	MEU	LEU
Radial P.P.F.	1.18	1.18	1.38
Axial P.P.F.	1.42	1.40	1.38
Local P.P.F.	1.14	1.20	1.25
Total P.P.F.	1.91	1.98	2.38

Thermal neutron flux measurements were also performed in irradiation elements close to the position (Pos. 48) of the LEU-test fuel element, MEU-ST.F.E. and HEU-ST.F.E. No remarkable difference could be found.

7.3 Burnup of LEU-Test Fuel Elements

At the end of March 1983 the LEU-test fuel elements in the reactor achieved the following burnup values

ST-31 introduced in April 1982 ..... 13.5 %
ST-32 introduced in November 1982 ..... 8.5 %

8. Conclusions

According to the experience made with the core conversion in 1969 and the recent experience with LEU-test fuel elements the following conclusions can be drawn from the standpoint of thermalhydraulics

- Operation of the ASTRA-reactor with mixed cores is possible without reducing the safety margins
- By choosing a suitable strategy for replacement of fuel elements a gradual transition to the reduced enrichment cycle is feasible.

Reference

[1] Research Reactor Core Conversion from the Use of Highly Enriched Uranium to the Use of Low Enriched Uranium Fuels Guidebook, IAEA-TECDOC-233

Appendix M-3

HEU-MEU MIXED CORE EXPERIMENTS IN THE KUCA

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Abstract

In response to a request from the consultant meeting of IAEA, the HEU-MEU mixed-core experiments in the KUCA were started in April 1984. The HEU-MEU mixed-core employed in the KUCA experiments was a light-water-moderated and heavy-water-reflected coupled-core.

Several patterns of HEU-MEU mixed-cores employed in the KUCA coupled-core experiments were broadly classified into two categories. The first was called as "Separate Core" in which one cylindrical core consisted of only HEU fuel and the other MEU fuel. The second was called as "Mixed Core" in which each cylindrical core consisted of both HEU and MEU fuels. For these cores, the critical mass and the reactivity worth of the control rod were measured. For "Separate Core", the effect of boron burnable-poison and the neutron flux distribution were also investigated. In both "Separate Core" and "Mixed Core", the number of fuel plates in each cylindrical core of the coupled two cores was maintained as the same number.

The imbalance of neutron importance between the two coupled cores was observed through the present KUCA mixed-core experiments, since the MEU fuel plate had a slightly higher reactivity effect than the HEU fuel plate. The reactivity worth of each control rod varied from case to case depending on the mixed-core configuration. In other words, the worth depended on the balance of neutron importance between the two coupled cores. However, the total reactivity worth of the control rods gave approximately the same value in any mixed-core configuration.

INTRODUCTION

In accordance with the joint ANL-KURRI [Argonne National Laboratory - Kyoto University Research Reactor Institute] study concerning the RERTR [Reduced Enrichment for Research and Test Reactors] program, the critical experiments using MEU [Medium-Enriched-Uranium] fuel in the KUCA [Kyoto University Critical Assembly] were started in May 1981.[1-3] The KUCA core employed in the MEU experiments was a light-water-moderated and heavy-water-reflected cylindrical core (single-core). The KUCA MEU experiments have been providing useful data with regard to the RERTR program.[4-18]

The consultant meeting of IAEA [International Atomic Energy Agency] concerning the RERTR program requested KURRI to perform a HEU-MEU mixed-core experiment. It is important to investigate the nuclear characteristics of the

HEU-MEU mixed-core through the critical experiments, since most of reactors cannot hardly avoid installing a mixed-core on the way to reduce the enrichment of uranium fuel for reasons of economy and reactor performance.

In response to the request from IAEA, an application for a safety review (Reactor Installation License) of the HEU-MEU mixed-core installed in the KUCA was submitted to the Science and Technology Agency of Japan [STAJ] in July 1983. This application was revised in December 1983 and a license was issued in February 1984. Subsequently, an application for "Authorization before Construction" was submitted and was approved in March 1984. Then, the critical experiments of the HEU-MEU mixed-core were started in April 1984, as a part of "Inspection before Operation" by STAJ for which a certificate was issued in May 1984. The HEU-MEU mixed-core employed in the present KUCA experiments was a light-water-moderated and heavy-water-reflected coupled-core.

This paper provides some results of the HEU-MEU mixed-core experiments in the KUCA. The HEU-MEU experiments included the measurements of (1) criticality, (2) control rod worth, (3) boron burnable-poison effect and (4) neutron flux distribution.

EXPERIMENTAL

Core Configuration

Figure 1 shows a view of a heavy-water tank made of aluminum for the present KUCA mixed-core experiments. The fuel elements were assembled in a cylindrical form as shown in Fig. 2. Two cylindrical assemblies of fuel elements such as shown in Fig. 2 were installed in the heavy-water tank to form a coupled-core.

Since the amount of both HEU and MEU fuels in the KUCA was just equal to that required for one cylindrical assembly of fuel elements, patterns of HEU-MEU coupled-cores employed in the present mixed-core experiments were broadly classified into two categories. The first was called as "Separate Core" in which one cylindrical assembly consisted of only HEU fuel and the other MEU fuel. The second was called as "Mixed Core" in which each cylindrical assembly consisted of both HEU and MEU fuels.

The thickness of the heavy-water reflector is 30 cm and the minimum thickness of the heavy-water layer between the coupled two cores is 15 cm. Each core has a cylindrical center island of light-water, and each fuel region is divided into two parts by the space for the control rods. Each inner fuel region consists of 6 fuel elements which are numbered as IN-01, IN-02 and so on for both HEU and MEU fuel elements. Each outer fuel region consists of 12 fuel elements numbered as OUT-01, OUT-02, etc. for HEU fuel elements and as EX-01, EX-02, etc. for MEU elements. The maximum number of fuel plates which can be loaded in a fuel element is 15 fuel plates per element for an inner fuel element and 17 fuel plates per element for an outer fuel element.

A typical core configuration is shown in Fig. 3. The criticality of the core was controlled by three rods, namely C1, C2 and C3 rods, because all safety rods (S4, S5 and S6) were withdrawn to their upper limit at every operation. The detectors were arranged around the heavy-water tank, and the neutron source was located under the heavy-water tank.

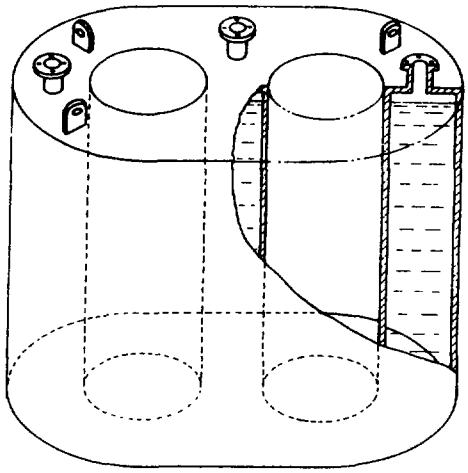


Fig. 1. View of the Heavy-Water Tank for a Coupled-Core.

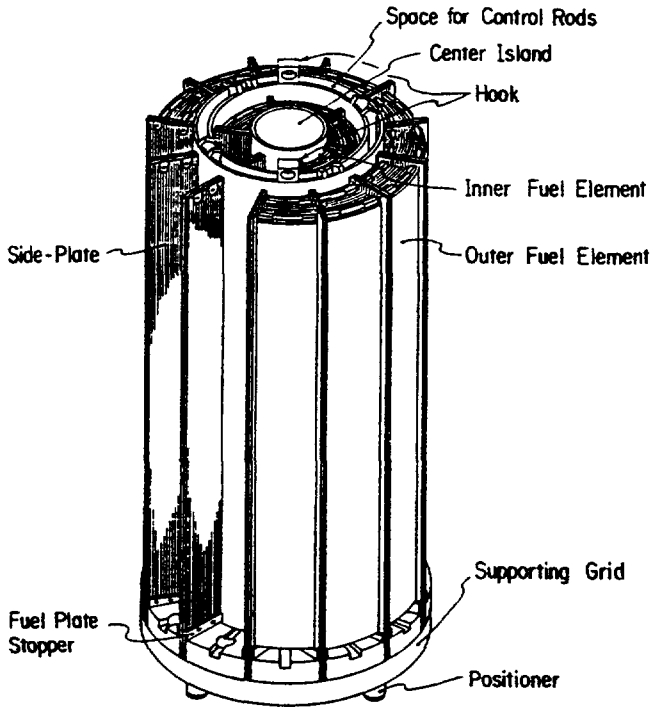
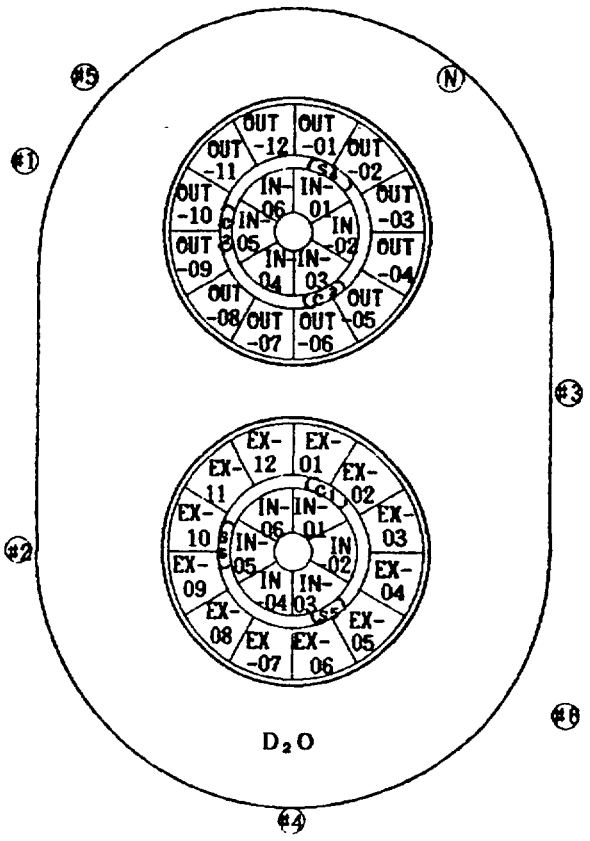


Fig. 2. View of the Assembled Fuel Elements.

It should be noted that HEU and MEU fuel plates cannot be mixed in one fuel element, since the dimensions of HEU and MEU fuel plates differ each other (see Table 1-1 and 1-2). Although a 3.80 mm fuel pitch was employed in the present mixed-core experiments, HEU fuel plates were originally designed for a 3.84 mm fuel pitch, while MEU fuel plates for a 3.80 mm fuel pitch. It should be also noted that an aluminum pipe (see Fig. 2) which separates the center island of light-water from the inner fuel elements was not utilized in the present experiments.

In both "Separate Core" and "Mixed Core", the criticality was adjusted by the number of fuel plates inserted into the inner part of the inner fuel



#1 ∿ #3 : Start-up Channels (Fission Chambers),
#4 ∿ #6 : Lin-N, Log-N and Safety Channels, Respectively (Uncompensated Ionization Chambers),
C1 ∿ C3 : Control Rods,
S4 ∿ S6 : Safety Rods,
N : Neutron Source,
D_2O : Heavy-Water Reflector,

OUT-01 ∿ OUT-12 : Outer Fuel Elements for HEU Fuel,
EX-01 ∿ EX-12 : Outer Fuel Elements for MEU Fuel,
IN-01 ∿ IN-06 : Inner Fuel Elements.

Fig. 3. Typical Core Configuration of a Mixed-Core in the KUCA.

elements, and the number of fuel plates in each cylindrical assembly of the coupled-core was maintained to be the same number.

Criticality Measurement

As the first step for the critical approach of the coupled-core, all outer fuel elements were fully loaded with 17 fuel plates. Then, the critical approach was performed by inserting fuel plates into the inner fuel elements from the outside toward inside in order. The inverse multiplication method was adopted for the critical approach. The detectors used in this measurement were three fission chambers utilized for the start-up channels, namely #1, #2 and #3.

For "Separate Core", the criticality measurements were performed for 4 patterns of the coupled-cores. These patterns were the cores (1) containing no boron burnable-poison [BP] at all, (2) containing BP only in the inner fuel region of the MEU side core, (3) containing BP only in the outer fuel region

Table 1-1. Specification of the HEU Fuel Plate.

plate no.	inner fuel plate					outer fuel plate				
	width of fuel (mm)	width of meat (mm)	curvature radius (mm)	Uranium (gr)	U-235 (gr)	width of fuel (mm)	width of meat (mm)	curvature radius (mm)	Uranium (gr)	U-235 (gr)
1	51.71	42.95	56.17	8.10	7.54	62.60	53.84	133.83	10.22	9.52
2	55.74	46.98	60.01	8.83	8.22	64.61	55.85	137.67	10.54	9.82
3	59.76	51.00	63.85	9.70	9.03	66.62	57.86	141.51	10.97	10.22
4	63.78	55.02	67.69	10.49	9.77	68.63	59.87	145.35	11.33	10.55
5	67.80	59.04	71.53	11.22	10.45	70.64	61.88	149.19	11.83	11.02
6	71.82	63.06	75.37	12.06	11.23	72.65	63.89	153.03	12.13	11.30
7	75.84	67.08	79.21	12.84	11.96	74.66	65.90	156.87	12.57	11.71
8	79.86	71.10	83.05	13.60	12.67	76.67	67.91	160.71	12.98	12.09
9	83.88	75.12	86.89	14.37	13.38	78.69	69.93	164.55	13.54	12.61
10	87.91	79.15	90.73	15.07	14.04	80.70	71.94	168.39	14.03	13.07
11	91.93	83.17	94.57	15.68	14.60	82.71	73.95	172.23	14.17	13.20
12	95.95	87.19	98.41	16.46	15.33	84.72	75.96	176.07	14.73	13.72
13	99.97	91.21	102.25	17.41	16.22	86.73	77.97	179.91	14.90	13.88
14	103.99	95.23	106.09	18.32	17.06	88.74	79.98	183.75	15.28	14.23
15	108.01	99.25	109.93	18.96	17.66	90.75	81.99	187.59	15.54	14.47
16	—	—	—	—	—	92.76	84.00	191.43	15.91	14.82
17	—	—	—	—	—	94.77	86.01	195.27	16.42	15.29

enrichment 93.14 w% plate length = 650 mm
fuel plate pitch = 3.84 mm meat length = 600 mm

Table 1-2. Specification of the MEU Fuel Plate.

plate no.	inner fuel plate					outer fuel plate				
	width of fuel (mm)	width of meat (mm)	curvature radius (mm)	Uranium (gr)	U-235 (gr)	width of fuel (mm)	width of meat (mm)	curvature radius (mm)	Uranium (gr)	U-235 (gr)
1	48.70	39.50	54.4	20.00	8.99	61.16	51.96	133.3	25.96	11.67
2	52.68	43.48	58.2	21.64	9.72	63.15	53.95	137.1	26.94	12.11
3	56.66	47.46	62.0	23.67	10.64	65.14	55.94	140.9	28.51	12.81
4	60.64	51.44	65.8	25.67	11.54	67.13	57.93	144.7	28.99	13.00
5	64.62	55.42	69.6	27.57	12.39	69.12	59.92	148.5	30.12	13.54
6	68.60	59.40	73.4	29.57	13.29	71.11	61.91	152.3	31.04	13.91
7	72.58	63.38	77.2	31.87	14.21	73.10	63.90	156.1	31.92	14.36
8	76.56	67.36	81.0	34.26	15.41	75.09	65.89	159.9	32.85	14.76
9	80.54	71.34	84.8	36.18	16.24	77.08	67.88	163.7	33.89	15.23
10	84.51	75.31	88.6	38.27	17.16	79.07	69.87	167.5	35.55	15.99
11	88.49	79.29	92.4	40.34	18.10	81.06	71.86	171.3	36.49	16.41
12	92.47	83.27	96.2	43.09	19.26	83.05	73.85	175.1	37.10	16.61
13	96.45	87.25	100.0	44.49	19.98	85.04	75.84	178.9	38.25	17.10
14	100.43	91.23	103.8	46.74	20.89	87.03	77.83	182.7	39.68	17.76
15	104.41	95.21	107.6	48.30	21.64	89.02	79.82	186.5	40.69	18.27
16	—	—	—	—	—	91.01	81.81	190.3	41.45	18.63
17	—	—	—	—	—	93.00	83.80	194.1	42.69	19.11

enrichment 44.87 w% plate length = 650 mm
fuel plate pitch = 3.8 mm meat length = 600 mm

of the MEU side core, and (4) containing BP both in the inner and outer fuel regions of the MEU side core. For "Mixed Core", the criticality measurement was performed only for one pattern of the coupled-core containing no BP at all.

For all patterns of mixed-cores listed above, the excess reactivities were measured by the positive period method.

Control Rod Worth

After achieving the criticality, the reactivity worths of all control and safety rods were measured by the integral count technique of the rod drop method. Three fission chambers of the start-up channels (#1, #2 and #3) were used in the measurements.

The measurements were performed for all patterns of mixed-cores listed before, excluding "Separate Core" containing BP only in the inner fuel region of the MEU side core.

Boron Burnable-Poison [BP] Effect

For "Separate Core", the reactivity effect of BP was investigated. From the criticality measurements for 4 patterns of "Separate Core" described before, the BP reactivity effect could be estimated.

It should be noted that special side-plates containing burnable-poison made of natural boron were prepared only for MEU fuel. An inner side-plate with BP contained 104 mg ^{10}B, while an outer side-plate 117 mg ^{10}B. It should be also noted that a fuel element consisted of two side-plates.

For "Separate Core" containing no BP, the spatial dependence of the BP effect was also investigated. The procedures were as follows: (1) to measure the excess reactivity of the coupled-core containing no BP, (2) to substitute one of MEU fuel elements with BP for that without BP and to measure the excess reactivity, and (3) to change the position of the substitution and to measure the excess reactivity until the spatial dependence was obtained.

Neutron Flux Distribution

For "Separate Core", the neutron flux distribution was measured by the activation method using gold wires. From the activity measurements of gold wires with and without cadmium covers, the cadmium ratio and the thermal neutron flux were obtained.

RESULTS AND DISCUSSION

The results of criticality measurements are tabulated in Table 2 and the loading patterns of fuel plates are shown in Fig. 4. The measured results of control and safety rod worths are tabulated in Table 3. Figure 5 shows the spatial dependence of the BP effect in the inner or outer fuel region of "Separate Core". The vertical neutron flux distribution in "Separate Core" containing no BP is shown in Fig. 6.

Table 2. Critical Mass and Excess Reactivity.

Coupled-Core		"Separate Core" No BP[1]	"Separate Core" Inner BP[2]	"Separate Core" Outer BP[3]	"Separate Core" All BP[4]	"Mixed Core" (No BP)
Number of Fuel Plates	HEU	241	250	254	258	243
	MEU	241	250	254	258	243
	Total	482	500	508	516	486
^{235}U Mass (g)	HEU	3132.66	3238.66	3287.92	3335.76	3147.84
	MEU	3853.66	3996.50	4055.74	4112.58	3886.14
	Total	6974.74	7235.16	7343.66	7448.34	7033.98
Excess Reactivity (%Δk/k)		∿0	0.26	0.35	0.26	0.23
Fuel Loading Pattern		Fig.4(a)	Fig.4(c)	Fig.4(d)	Fig.4(e)	Fig.4(f)

Single-Core	MEU Core No BP	MEU Core Inner BP	MEU Core Outer BP	MEU Core All BP	"HEU Core" (No BP)
Number of Fuel Plates	262	——	286	——	278
^{235}U Mass (g)	4165.74	——	4438.62	——	3542.52
Excess Reactivity (%Δk/k)	0.04	——	0.13	——	0.17

[1]Boron burnable-poison [BP] was not contained at all.
[2]BP was contained only in the inner fuel region of the MEU side core.
[3]BP was contained only in the outer fuel region of the MEU side core.
[4]BP was contained both in the inner and outer fuel regions of the MEU side core.

Table 2 shows that, for "Separate Core" containing no BP, the number of fuel plates required to achieve criticality was less than that for "Mixed Core". This fact indicates that there was an imbalance of neutron importance between the coupled two cores of "Separate Core". In other words, MEU fuel was not completely equivalent to HEU fuel in reactivity.

Table 3 clearly shows the imbalance of neutron importance mentioned above. The reactivity worth of each control rod varied case by case depending on the mixed-core configuration. For "Separate Core" containing no BP, the control rod located in the MEU side core had a larger worth than that in the HEU side. Even for "Mixed Core" employed in the present experiments, the control rod located in the MEU dominant side core had a larger worth than that in the HEU dominant side. Such a tendency was caused by the fact that MEU fuel had a slightly higher reactivity effect than HEU fuel. On the other hand, for "Separate Core" containing BP, the control rod located in the MEU side core was of less worth than that in the HEU side.

Table 3. Reactivity Worth of Control Rods.

		"Separate Core"			"Mixed Core" (No BP)	MEU Single Core (No BP)[4]
		No BP[1]	Outer BP[2]	All BP[3]		
Reactivity Worth of Control Rods[5] (%Δk/k)	S4	0.13	0.34	0.49	0.22	0.73
	C2	0.21	0.46	0.63	0.31	0.71
	C3	0.29	0.50	0.65	0.38	0.71
	C1	0.63	0.32	0.17	0.52	——
	S6	0.45	0.24	0.11	0.38	——
	S5	0.37	0.18	0.07	0.28	——
	Total	2.07	2.04	2.12	2.08	2.16
Fuel Loading Pattern		Fig.4(b)	Fig.4(d)	Fig.4(e)	Fig.4(f)	——

[1]Boron burnable-poison [BP] was not contained at all.
[2]BP was contained only in the outer fuel region of the MEU side core.
[3]BP was contained both in the inner and outer fuel regions of the MEU side core.
[4]Single-core with a 3.84 mm fuel pitch, other coupled-cores ("Separate Core" and "Mixed Core") with a 3.80 mm fuel pitch.
[5]The relative experimental error was estimated to be 2 ∿ 3 %.

Table 3 also shows that the control rod located near the gap of heavy-water between the two coupled cores had a larger worth than the others. This fact indicates that, the closer to the gap of heavy-water, the higher the neutron importance becomes. Such a distribution of neutron importance is one of the most typical characteristics in a coupled-core.

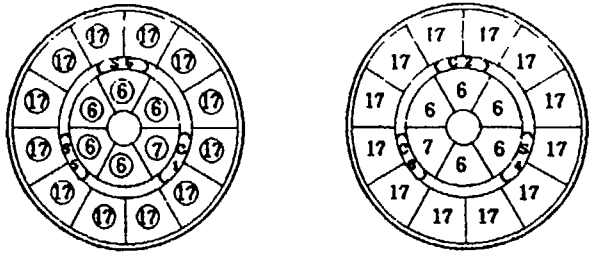
However, Table 3 shows that the total reactivity worth of the control and safety rods gave approximately the same value in any mixed-core configuration. Furthermore, the total worth of the control rods in the mixed-core (coupled-core) was approximately equal to that in the MEU single-core.

Table 2 shows that, for "Separate Core", the BP reactivity effect of the outer fuel region was larger than that of the inner fuel region when side-plates containing BP were fully loaded in each region.

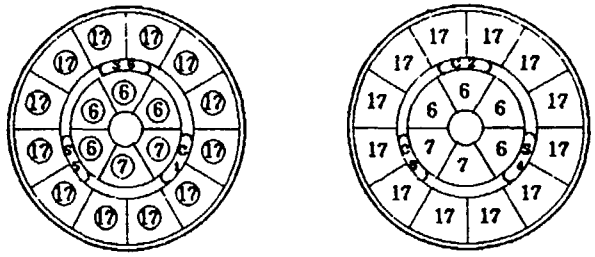
Figure 5 shows that, the closer to the gap of heavy-water between the two coupled cores the position of the BP substitution became, the larger (more negative) the reactivity effect became. This fact also demonstrates that the closer to the gap of heavy-water, the higher the neutron importance becomes.

Figure 6 shows that, in "Separate Core" containing no BP, the thermal neutron flux in the MEU side core was higher than that in the HEU side. This fact clearly demonstrates that MEU fuel had a slightly higher reactivity effect than HEU fuel.

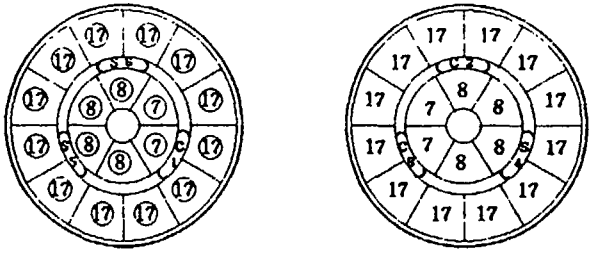
Figure 6 also shows that the cadmium ratio in the MEU side core was lower than that in the HEU side, in other words, the neutron spectrum in the MEU side core was harder than that in the HEU side. The reason was that the H/U ratio in MEU fuel was smaller than that in HEU fuel.



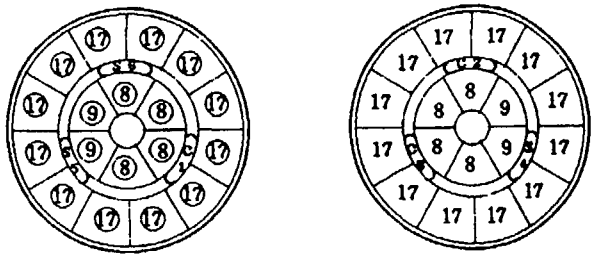
(a) "Separate Core" Containing No BP (employed in the criticality measurement).



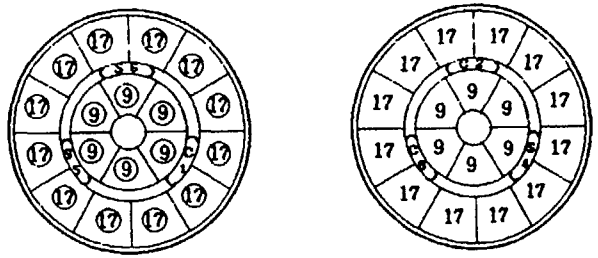
(b) "Separate Core" Containing No BP (employed in the measurement of the control rod worth).



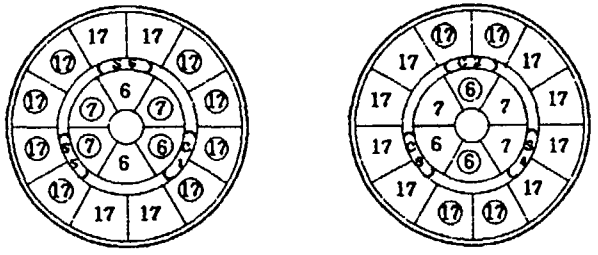
(c) "Separate Core" Containing BP Only in the Inner Fuel Region of the MEU Side Core.



(d) "Separate Core" Containing BP Only in the Outer Fuel Region of the MEU Side Core.



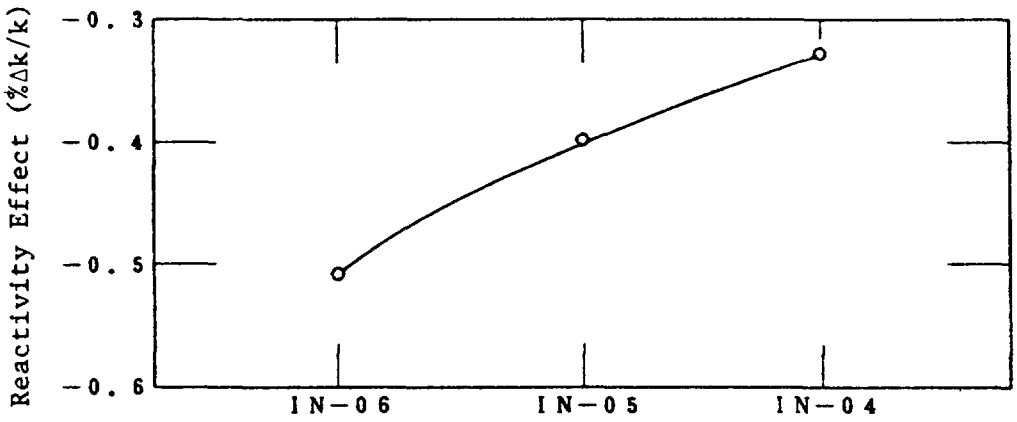
(e) "Separate Core" Containing BP Both in the Inner and Outer Fuel Regions of the MEU Side Core.



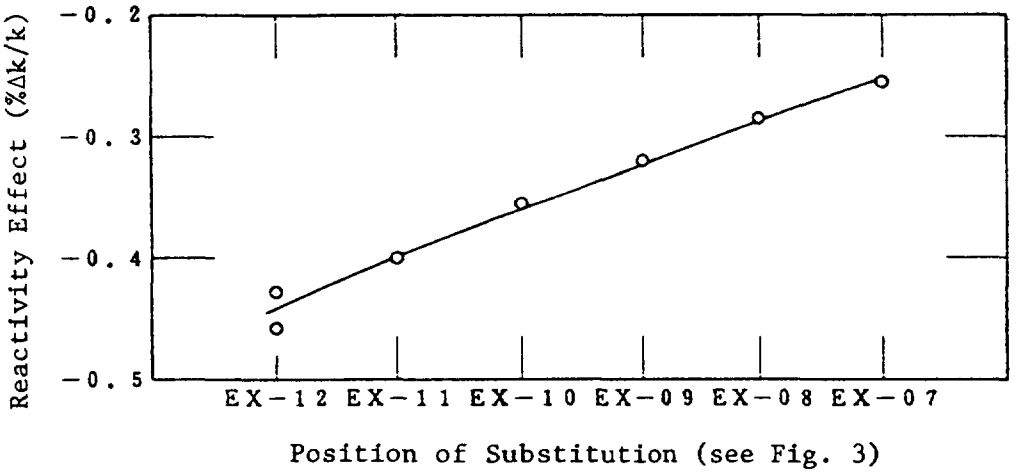
(f) "Mixed Core" Containing No BP.

Note that an Arabic numeral enclosed with a circle shows a number of MEU fuel plates loaded in a fuel element, while an Arabic numeral without a circle indicates a number of HEU fuel plates.

Fig. 4. Fuel Loading Patterns Employed in the Mixed-Core Experiments.



(a) Spatial Dependence of the BP Effect in the Inner Fuel Region.



(b) Spatial Dependence of the BP Effect in the Outer Fuel Region.

Fig. 5. Spatial Dependence of the BP Reactivity Effect in the MEU Side of "Separate Core".

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The present work was specially supported by the Ministry of Education, Science and Culture, Japan.

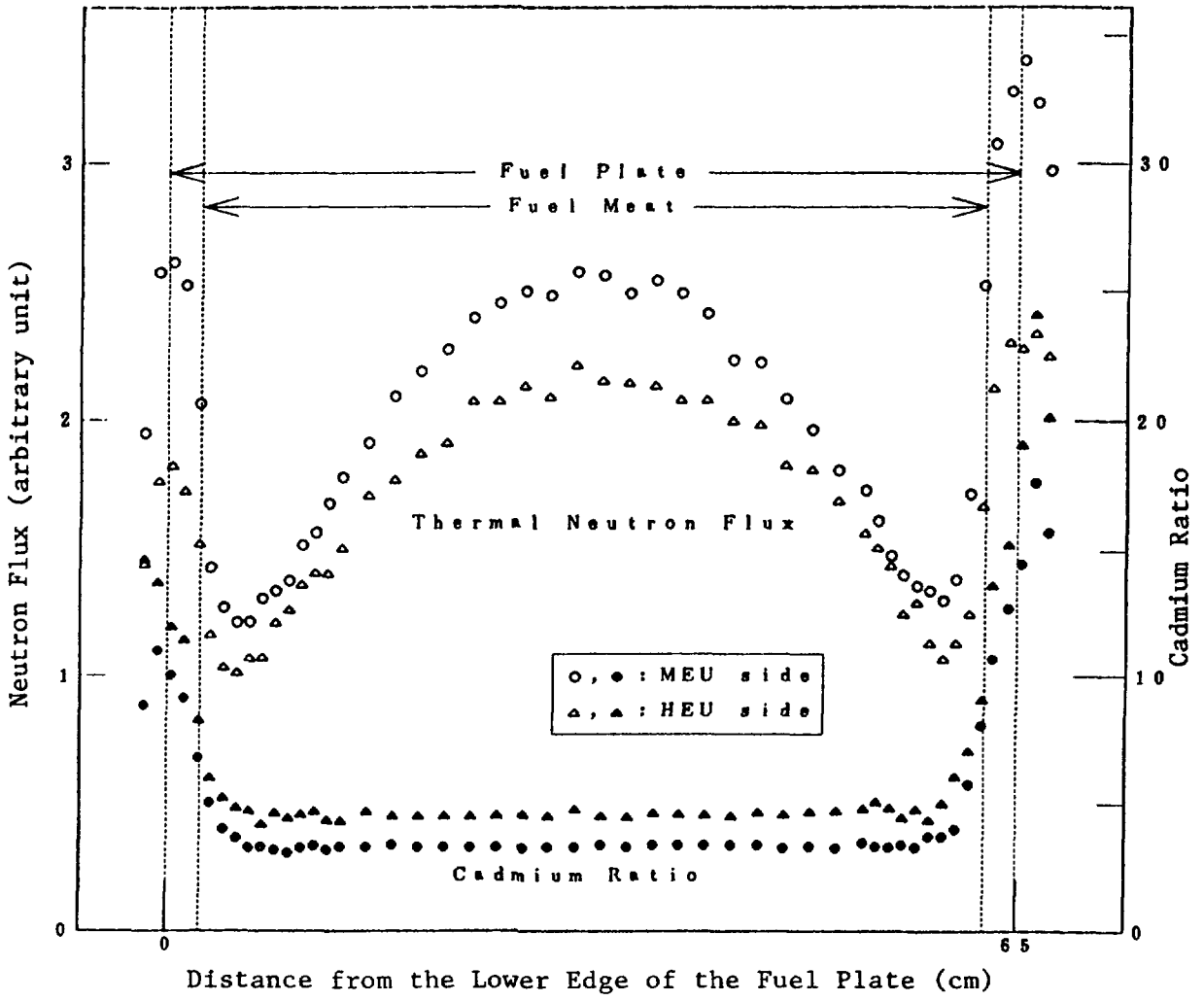

Fig. 6. Vertical Distributions of the Thermal Neutron Flux and the Cadmium Ratio in "Separate Core" Containing No BP.

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Appendix M-4

THE TRANSITION PHASE OF THE WHOLE-CORE DEMONSTRATION AT THE OAK RIDGE RESEARCH REACTOR

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Abstract

The transition from operation of the Oak Ridge Research Reactor with high-enrichment uranium (HEU) fuel to operation with low-enrichment uranium (LEU) fuel is nearing completion. The systematics of the replacement of the HEU fuel with the LEU fuel are discussed. The results of the core physics measurements that have been conducted during the transition phase are described.

INTRODUCTION

The Oak Ridge Research Reactor (ORR) has been selected as host for a full-core demonstration of the newly developed U_3Si_2 low-enrichment uranium (LEU) fuel. LEU fuel elements were first introduced into the reactor in January of this year. Currently, only two high-enrichment uranium (HEU) control rod elements remain in the operating core, and it is expected that operation with a full LEU core will begin in late December. The transition has been accomplished by a gradual phase-in of the LEU fuel in a manner consistent with normal operation of the ORR. The method and schedule followed during this transition are presented in this paper. During the phase-in period, various core physics measurements have been conducted to provide data to validate the core neutronics calculations being performed by Argonne National Laboratory (ANL). In addition, measurements to determine fuel element burnup and to verify that required safety margins were met were also performed. Details of these experiments and a summary of the results are presented.

BRIEF DESCRIPTION OF THE OAK RIDGE RESEARCH REACTOR

The operating configuration of the ORR prior to the introduction of any LEU elements is shown in Figure 1. The 9x7 core matrix is beryllium reflected on three sides and water reflected on the fourth. It contains 27 HEU oxide elements and six control rod elements with fueled followers. The fuel element is a box type containing 19 curved Al-clad plates and a total of 285 g of ^{235}U as U_3O_8 when new. The fueled control rod followers are constructed with 15 fuel plates and contain a total of 167 g ^{235}U each.

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* Operated by Martin Marietta Energy Systems, Inc., for the US Department of Energy under contract DE-AC05-84OR21400.

The LEU fuel elements and control rods are geometrically identical to the HEU elements and control rods. Only the fuel meat section has been changed. LEU fuel elements contain a total of 340 g of ^{235}U each as U_3Si_2 when new, and the shim rods contain 200 g ^{235}U as U_3Si_2 each when new. Experiments are located in core positions B-1, B-9, C-3, C-7, E-3, E-5, E-7, F-1, and F-9. During the phase-in period, the ORR operating configuration has slowly changed in response to the removal of experiments and various safety reviews. The current operating configuration, 177-D, is shown in Figure 2.

The operating cycle time at 30 MW for configurations similar to those shown in Figures 1 and 2 is approximately three weeks. At the end of the three-week operation, all fuel elements are removed from the core and

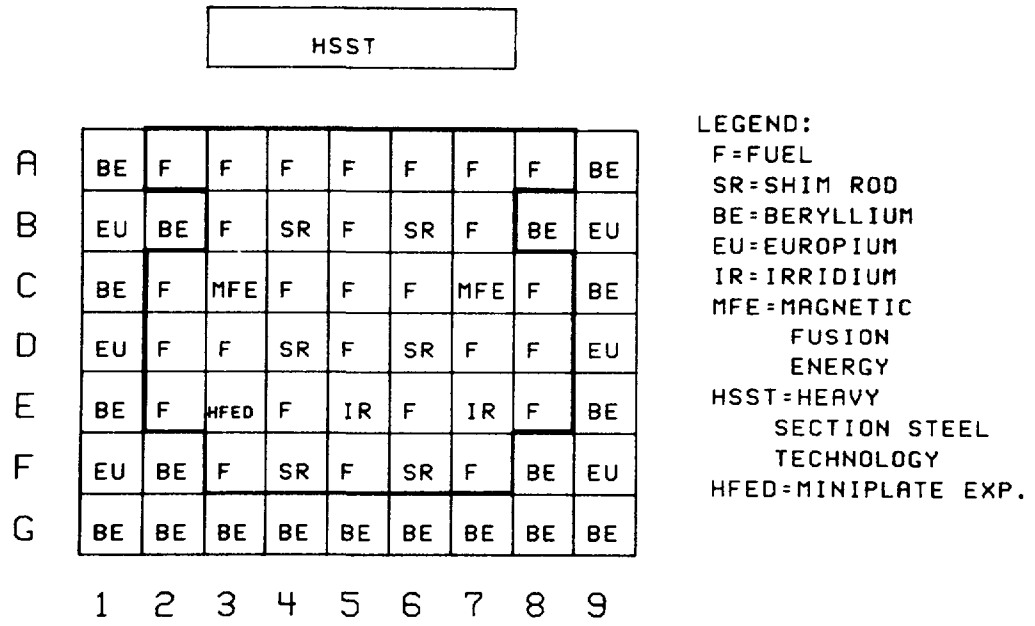


FIG. 1. ORR core lattice configuration prior to phase-in of LEU.

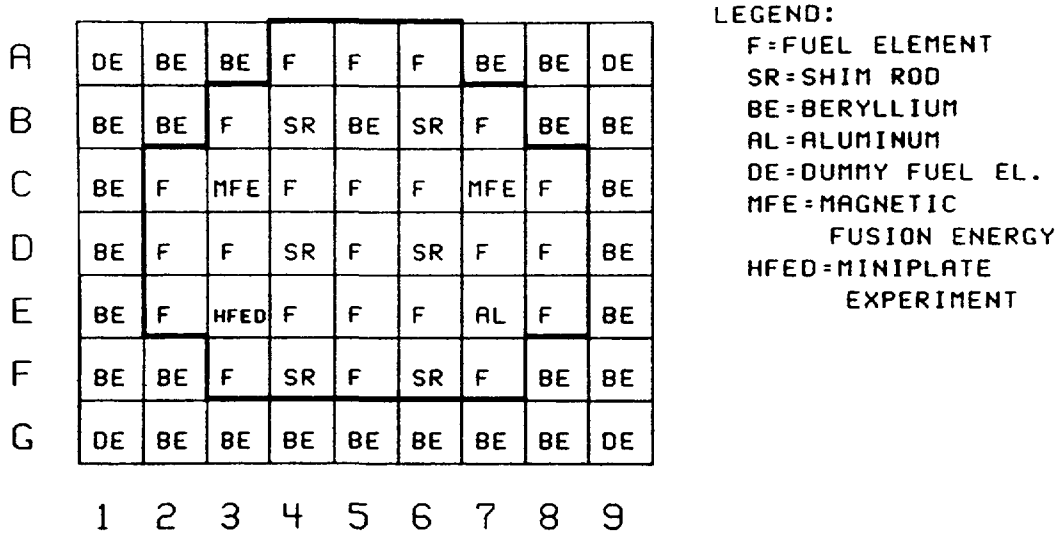


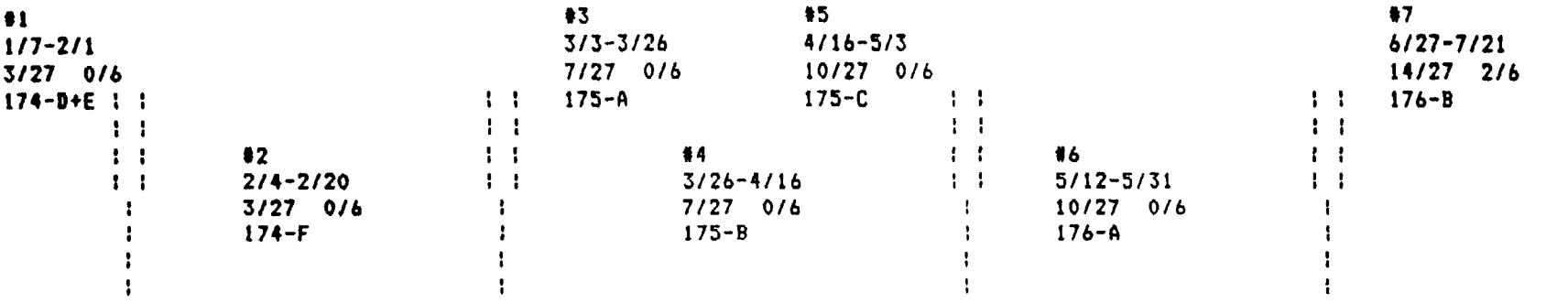
FIG. 2. ORR core configuration 177-D.

stored in the reactor pool to allow for Xe decay. Typically, three to four of these elements have achieved 50% burnup and are declared spent. The reactor is then refueled with three to four new elements and fuel elements of various ^{235}U content from the inventory of irradiated fuel stored in the pool. The six control rod elements located in positions B-4, B-6, D-4, D-6, F-4, and F-6 follow a different refueling scheme, since they are not returned to the pool after each operating cycle. The scheme is, that after approximately three months of operation, two new unirradiated control rods are introduced into positions D-4 and D-6. The control rods from positions D-4 and D-6 which have been irradiated for three months are moved to positions B-4 and B-6, and the control rods from positions B-4 and B-6 are moved to positions F-4 and F-6. The control rods from positions F-4 and F-6 which have been in the reactor for approximately nine months of operation and have achieved approximately 70% burnup are declared spent and are moved to pool storage.

PHASE-IN OF LEU FUEL

The objective of the transition period is to replace spent HEU fuel elements and control rods with fresh LEU elements and control rods and to irradiate this fuel to obtain an inventory of elements and control rods with the ^{235}U burnups typical for the ORR. This phase-in has been accomplished by installing three to four unirradiated LEU elements rather than the three to four unirradiated HEU elements normally introduced during each refueling. Thereafter, the LEU fuel is used in the same manner as the HEU fuel, i.e., after each cycle of irradiation, the LEU fuel is placed in pool storage for one operating cycle to allow for Xe decay and then returned to the operating core. Thus, an inventory of LEU elements with various ^{235}U burnup becomes available and an all-LEU core is established. As with the HEU fuel elements, the phase-in of the LEU control rods has been accomplished by introducing two LEU control rods rather than the two unirradiated HEU control rods at approximately three-month intervals and shuffling the rods as previously described.

The sequence of fuel management followed during the phase-in period is shown in Figures 3 and 4. Examination of these figures shows that two LEU cores were being developed; the first core consisted of elements irradiated in the odd cycles and the second consisted of elements irradiated in the even cycles. The first cycle of operation, 174-DE, used three LEU elements and 24 HEU elements for a total of 27 fuel elements and operated from January 7, 1986, to February 1, 1986. All six control rods in this core were HEU. After irradiation in cycle 1, these three LEU elements were stored in the pool during the operation of cycle 2 and then returned to the core along with four fresh LEU elements to give a total of seven LEU elements in cycle 3 (175-A). Cycle 5 used the seven LEU elements irradiated in cycle 3, plus three additional fresh LEU elements for a total of ten LEU elements. Similar comments hold for the remaining odd cycles. The even-numbered cycles are configured similar to the odd cycles containing the same number of LEU elements and follow the same fuel management scheme just described. The first pair of LEU control rods were loaded into core positions D-4 and D-6 at the start of the cycle 7 (176-B) so that two of the six control rods were LEU. Two additional control rods were added in cycle 11 (177-B) to bring the number of LEU control rods to 4 out of 6. It is expected that the final two control rods will be loaded into the core at the beginning of cycle 15 (178-A) scheduled to start in mid-December. This will be the first all-LEU core and will mark the end of the transition phase of the demonstration.



Measurements on Core 174-FX
(174-FX is similar to 175-A)
7 LEU Fuel Elements
0 LEU Shim Rods

1. Flux Mapping 174-FX
2. β_{eff}/l - unsuccessful
3. Shim Rod Calibrations 174-FX

Approach to critical measurements
Cores:
HEU-1
HEU-2
LEU-1
LEU-2

Measurements on Core 176-AX1
(176-AX1 is similar to 176-B)
14 LEU Fuel Elements
2 LEU Shim Rods

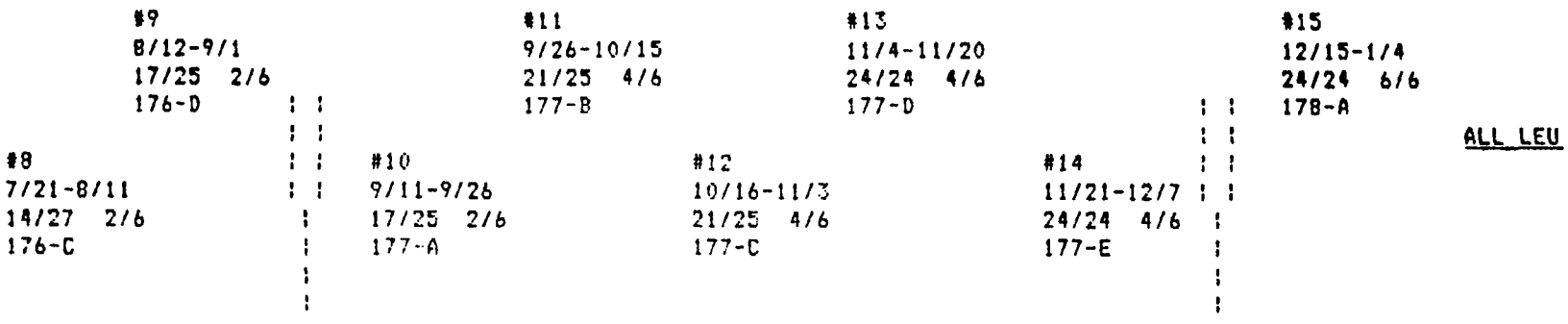
1. Flux Mapping 176-AX1
2. β_{eff}/l - unsuccessful
3. Shim Rod Calibrations 176-AX1
4. HSST Reactivity Measurement

Reactor shutdown due to ORR seismic review and subsequent siphon-break modification

1. Shim Rod Calibrations 176-B
2. β_{eff}/l, 176-B and 176-BX2 (All HEU)
3. Gamma Heating Measurements - unsuccessful

DATA FORMAT
Cycle number
Date
LEU elements/total elements LEU shim rods/total shim rods
Core configuration

FIG. 3. Schedule for transition to LEU (cycles 1 through 7).



Measurements on Core 177-AX1
(177-AX1 is similar to 177-B)
21 LEU Fuel Elements
4 LEU Shim Rods

1. Flux Mapping 177-AX1
2. β_{eff}/l 177-AX2
3. Shim Rod Calibrations
   177 AX1
4. Gamma Heating
   Measurements-
   unsuccesful

Measurements on Cycle #15 (all LEU)
Core 178-A
24 LEU Fuel Elements
6 LEU Shim Rods

1. Flux Mapping
2. β_{eff}/l
3. Shim Rod Calibrations
4. Gamma Heating
   Measurements

DATA FORMAT
Cycle number
Date
LEU elements/total elements LEU shim rods/total shim rods
Core configuration

FIG. 4. Schedule for transition to LEU (cycles 8 through 15).

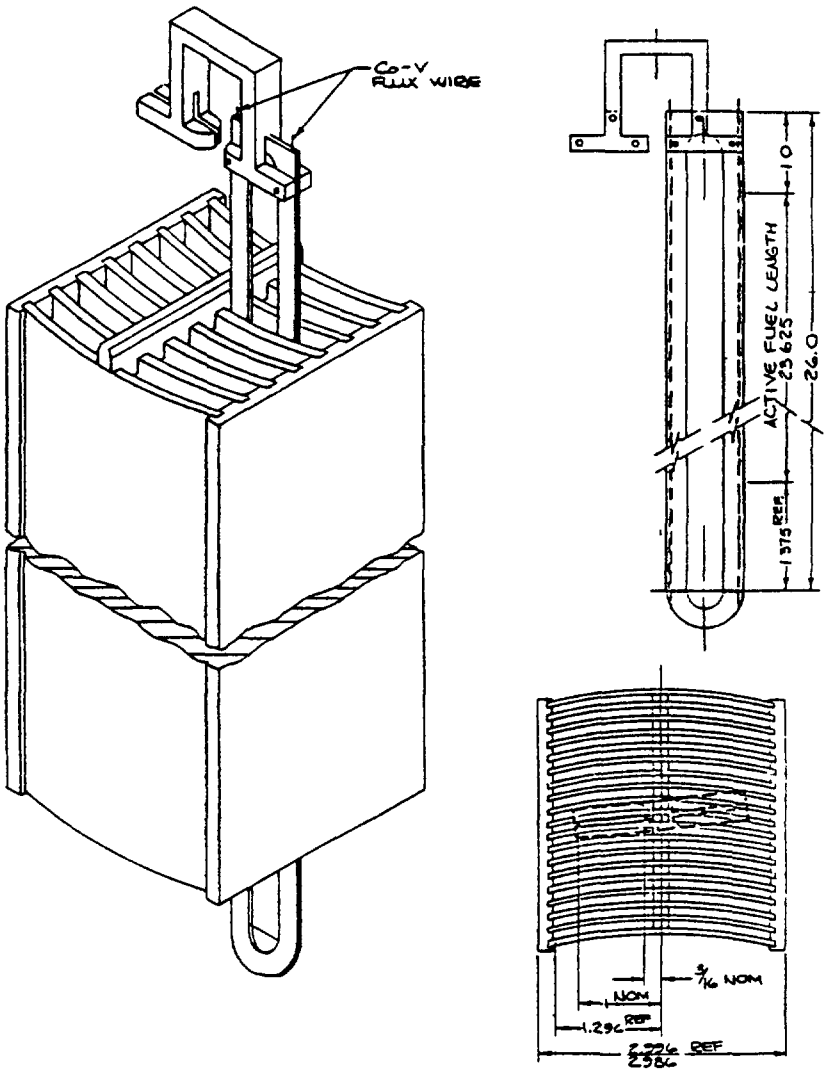


FIG. 5. Flux monitor holder.

Figures 3 and 4 show that there were periods during which the reactor was shut down for scheduled maintenance and quarterly inspections. During these periods (ends of cycles 1, 2, 5, 6, and 9), the various core physics measurements listed were made to provide data for neutronics code validation and to obtain information necessary to verify that the required safety margins were maintained. These measurements are discussed in the following sections.

FLUX MEASUREMENTS

Determinations of the thermal neutron flux distribution in the various core configurations have been made using 0.020-inch diameter Co-V wire (2.0 wt % Co in V) and measuring the induced ^{60}Co activity. The Co-V wire is inserted axially into the water channel between two fuel plates using the aluminum holder shown in Figure 5. This holder accommodates two full-length Co-V wires and allows each of the wires to extend approximately one inch above and below the active fuel region. Each fuel element in the core is monitored with one flux monitor assembly inserted into the ninth water channel counting from the concave side of the element. Additional monitors have been inserted into channels three and fifteen in the elements with the

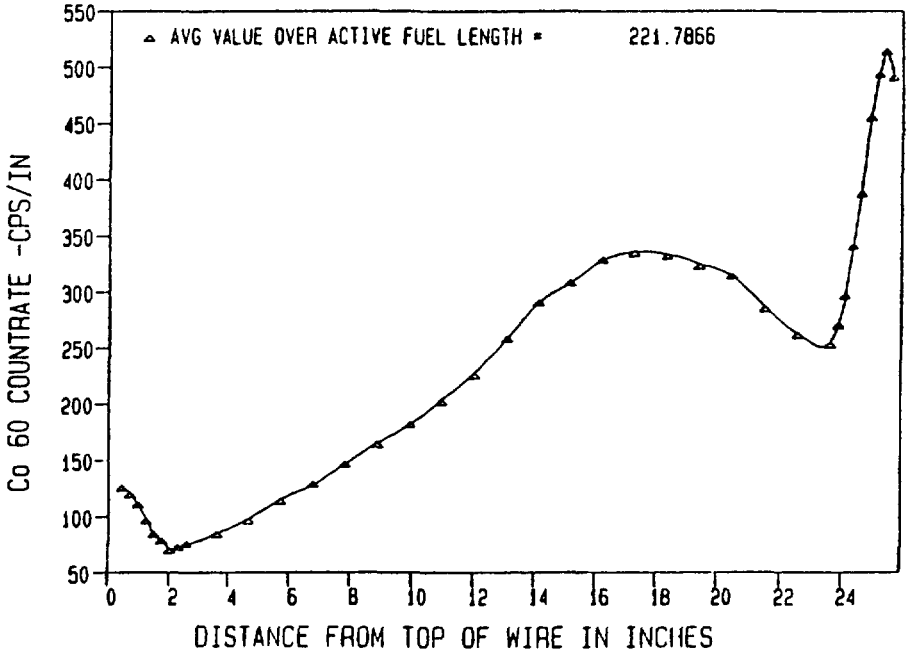


FIG. 6. Typical axial flux profile.

highest power density. An underwater camera is used to ensure both the location and alignment of the monitors in the elements. Control rods, which are difficult to access, have been monitored by inserting the Co-V wire into an Al tube (0.080-inch diameter) which is then inserted into the water channel between the fuel plates in the fuel followers of the control rods. The precise location and alignment of these monitors are uncertain. A similar arrangement (Al tube with Co-V wire) has been used to monitor fluxes in several Be reflector pieces that surround the fuel.

All of the Co-V wires in a given test configuration are irradiated simultaneously for six hours at approximately 300 kW. After irradiation, the wires are removed and counted on a computer-controlled wire scanning system. In this system, the Co-V wire is wrapped around a circular disk. The disk is indexed to determine axial position along the wire, and the computer controls a stepping motor which moves the disk in predetermined increments behind a tungsten collimator and initiates the predetermined counting interval.

A discriminator window is set on the output signal from the NaI crystal detector so that only the two ^{60}Co photopeaks are recorded. The measured ^{60}Co counting rates versus axial positions are forwarded to ANL to be compared with calculated values of the relative thermal flux distributions in the fuel elements.[1] For most of the core, the agreement between measured and calculated values is quite good. A typical plot of these data is shown in Figure 6. Note the reflector flux peak near the the bottom of the fuel. In this analysis, only relative values (counting rates) have been determined. Absolute measurements require an accurate determination of the reactor power during the irradiation. This is difficult to obtain with sufficient accuracy at the low power level maintained during the irradiation of the flux monitors.

For the Co-V wires closest to the fuel element center in each element, an axially averaged value of the ^{60}Co counting rate may be determined. The power produced by a fuel element is assumed to be related to the product of

Table 1. Calculated safety margins

Core	Configuration				Core position	Element power (MW)		Safety margin[a]
	Fuel elements		Control rods					
	LEU	HEU	LEU	HEU		Meas.	Calc.	
174-C	0	27	0	6	C-6	1.30	1.29	2.12
174-FX	7	20	0	6	D-7	1.27	1.24	2.52
176-AX1	4	13	2	4	E-4	1.31	1.36	2.13
177-AX1	1	4	4	2	E-4	1.40	1.39	1.87

this average counting rate times the ^{235}U weight (adjusted for burnup) in the fuel element.[2] The element power can then be determined by normalizing the sum of the above products to the reactor power. As mentioned earlier, several of the elements producing highest power contained additional flux wires. The measured neutron flux gradients, determined by comparing the activity of the multiple wires in a single fuel element, are used to determine power density peaking factors within the elements. From these data, one may determine the maximum thermal heat fluxes during operation and the safety margin, which is the ratio of the critical heat flux to the maximum heat flux at the limiting conditions of operation (43.5 MW and 14,100 gpm flow). The methodology used is given in Reference 3. Results for the limiting element in each of the four cores measured to date are given in Table 1. Note that, for core 177-AX1, the increase in fuel element power and reduction in safety margin are the results of decreasing the core size and are not attributed to the use of LEU fuel. If the core size had been maintained at 27 elements, it is estimated that the safety margin would be 2.10, which is comparable to the margin in the all-HEU core 174-C. The minimum allowed margin is 1.6 at 43.5 MW and 14,100 gpm coolant flow.

It should be noted that the flux measurements and safety margin analyses are made prior to 30-MW operation of a new configuration. As an example, at the end of cycle 1, configuration 174-FX was loaded and run at low power for the flux measurements. Core 174-FX contained seven LEU elements (four new LEU elements and the three irradiated in core 174-DE) and was similar to cycle 3 (175-A). After the measurement, this core was removed and core configuration 174-F installed. The analysis of the flux measurements was made and approval obtained prior to operation of core 175-A. Similar comments hold for cores 176-AX and 176-B and cores 177-AX1 and 177-B.

β_{eff}/ℓ

Measurements of the prompt neutron decay constant $\alpha = \beta/\ell$, where β is the effective delayed neutron fraction and ℓ the prompt neutron lifetime, have been made using noise analysis techniques.[4] Signals from two fission

chambers located on opposite sides of the core periphery and near the core midplane are processed by a Fourier analyzer to obtain the cross-power spectral density (CPSD) as a function of frequency. The CPSD is then fitted, using least-squares techniques, to obtain the break frequency, F_b, where $\alpha = 2\pi F_b$. The measurements are made at very low reactor power (approximately 3 kW).

Measurements on small, cold, clean water-reflected cores (Figures 9 and 10) consisting of all fresh HEU elements (285 g) and shim rods (167 g) and all fresh LEU elements (340 g) and control rods (200 g) show the prompt neutron decay constant to be about 14% larger in the LEU core than in the HEU core. This agrees very well with the ANL calculations. These measurements were made with the fission chamber operating in the pulse mode.

Measurements of α in operating cores have been made on configurations 176-BX2 (all HEU), 176-BX1 (14 LEU of 27 fuel elements and 2 LEU of 6 shim rods) and 177-AX1 (21 LEU of 25 fuel elements and 4 LEU of 6 shim rods). Results have not yet been obtained. To conduct these measurements, special amplifiers were designed to allow the fission chamber to operate in the current mode. Additional measurements of α are planned on full HEU and LEU cores.

GAMMA SCANNING OF FUEL ELEMENTS

After each cycle of operation, the fuel elements are removed from the core for one cycle to allow for Xe decay. During this inter-cycle time, each fuel element irradiated in the previous operating cycle is scanned along its centerline to detect γ-rays from the fission products. The experimental apparatus is shown schematically in Figure 7. The element is placed horizontally, convex side up, in a tray located 16 feet under water. The tray translates the element below a dry tube 1/2 inch in diameter and is computer controlled to stop at predetermined counting locations for a specified time. The dry tube (which acts as a collimator) extends up through the 16 feet of pool water to the bottom of a lead collimator with a 1/16-inch-diameter hole. A Ge-Li crystal detector is centered over the lead collimator and connected to a Nuclear Data 6600 data gathering system. An IBM PC is interconnected to the Nuclear Data System, and the results of a γ-peak fitting routine for each axial location scanned are stored on a separate floppy disk for each element.

For each isotope, the measured counting rate for a given energy line is related to its irradiation history and decay time by an equation of the form:

$$\text{Activity} \propto [F_1 D_1(\lambda, \text{tirrad}_1, \text{tdecay}_1) + F_2 D_2(\lambda, \text{tirrad}_2, \text{tdecay}_2) + \ldots\ldots\ldots F_n D_n (\lambda, \text{tirrad}_n, \text{tdecay}_n)] \quad (1)$$

where,

Activity = measured counting rate

F_n = the average fission rate during operating cycle n,

D_n = the production and decay factor for cycle n,

tirrad_n = the irradiation time during cycle n, and

tdecay_n = the decay time measured from the shutdown of cycle n to counting time.

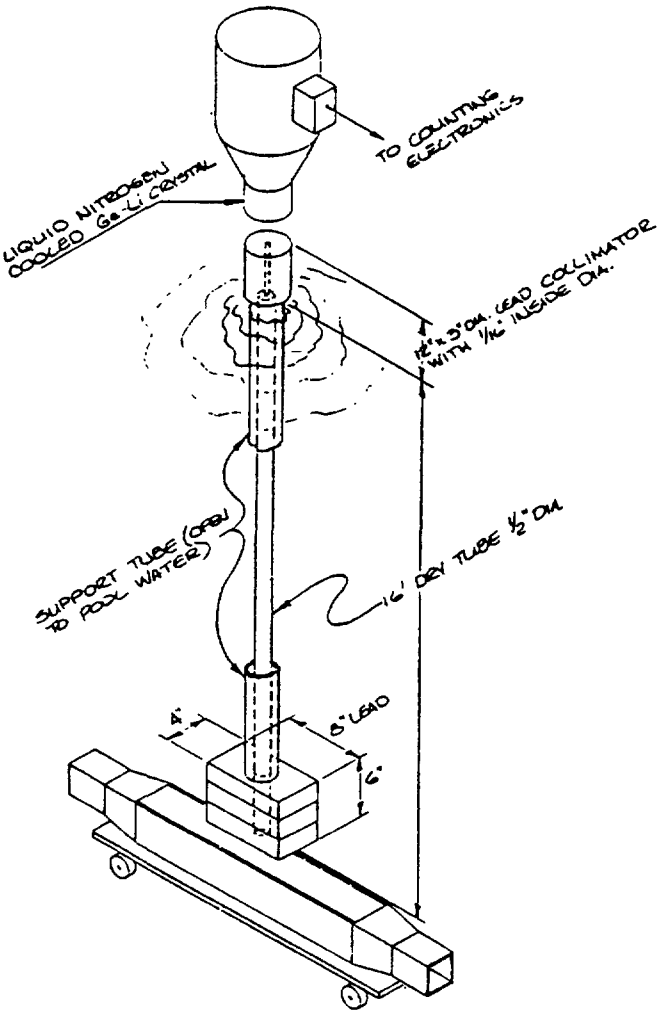


FIG. 7. Fuel element gamma scanning.

For a long-lived isotope such as ^{137}Cs, the measured counting rate is proportional to the total number of fissions. For a short-lived isotope, all contributions from irradiation cycles other than the most recent are negligible, and the measured count rate is proportional to the fission rate in the last cycle of operation. Intermediate-lived isotopes can be used to determine the fission rate during the last cycle of irradiation (F_n) if values of F_iD_i for the other cycles of irradiation are known.

Initially, it was hoped that ^{137}Cs could be detected and used to integrate the fissions to determine the burnup of each element. Due to the abundance of short-lived fission products and the small elapsed time between irradiation and counting of the element (typically two to three days after irradiation), ^{137}Cs cannot be unambiguously detected. The predominant isotopes remaining after approximately two days decay are ^{140}Ba, ^{140}La, ^{132}I, ^{95}Zr, ^{99}Mo, and ^{95}Nb. The 1596 KeV line of ^{140}La, which decays with an effective half-life of 12.8 d, has been been selected and sorted from the data to use in the analyses due to its high yield and energy. By determining the element fission rate in each cycle, the ^{235}U burnup can be estimated, and in addition, confirming information on the core power distribution may be obtained.

Table 2 gives a comparison of calculated and measured fuel element powers for core 176-A as determined from the gamma scanning data in the following manner:

Table 2. Comparison of calculated and measured fuel element powers core 176-A

Core position	Element power (MW)		C/E[a]
	Calc.	Meas.	
A-2	0.641	0.701	0.914
A-3	0.767	0.846	0.906
A-4	0.924	1.031	0.897
A-5	1.074	1.124	0.955
A-6	0.928	1.029	0.902
A-7	0.794	0.823	0.965
A-8	0.605	0.659	0.918
B-3	0.925	1.007	0.919
B-5	0.987	1.032	0.957
B-7	0.867	0.888	0.977
C-2	1.227	1.216	1.009
C-4	1.122	1.112	1.009
C-5	1.109	1.241	0.894
C-6	1.372	1.411	0.972
C-8	1.179	1.158	1.018
D-2	0.792	0.827	0.958
D-3	1.318	1.274	1.034
D-5	1.064	1.061	1.003
D-7	1.233	1.152	1.070
D-8	0.790	0.765	1.033
E-2	1.035	0.983	1.052
E-4	1.241	1.177	1.055
E-6	1.234	1.137	1.086
E-8	1.067	0.958	1.114
F-3	0.758	0.676	1.122
F-5	0.867	0.740	1.171
F-7	0.766	0.661	1.158

[a]Calculated to experimental values.

1. Spectral data were acquired at 12 axial positions along each fuel element.

2. The counting rate of the 1596 KeV peak for each axial location along the fuel elements is sorted from the spectral file. These counting rates are then corrected (reduced) for the residual La remaining from previous cycles of irradiation. The corrected counting rate (proportional to the La formed in the last cycle of irradiation only) is then corrected for decay to the end of shutdown of the last cycle.

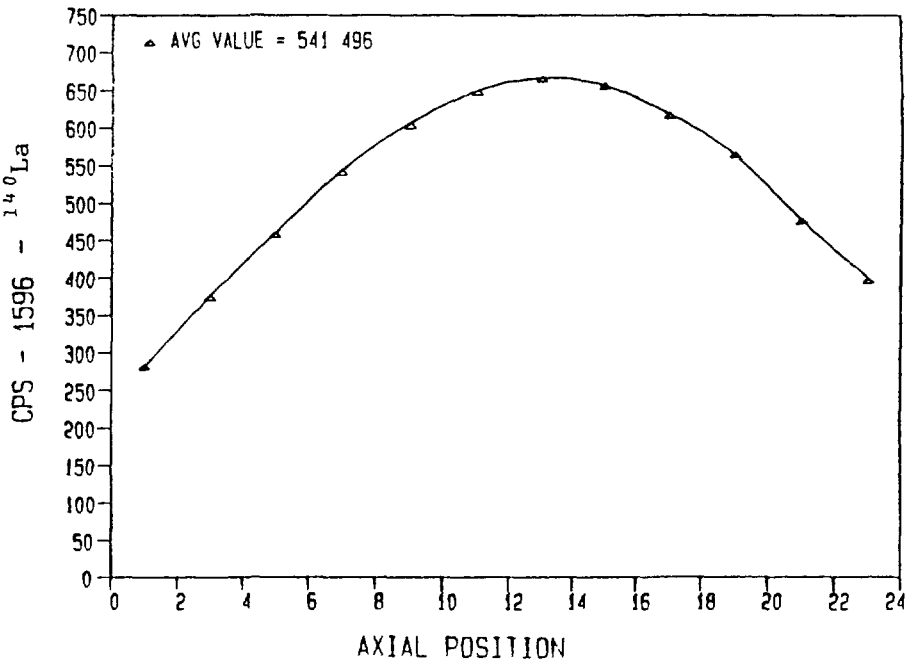


FIG. 8. Typical data from fuel element gamma scanning.

3. The trapezoidal rule was then used to integrate the area under the curve generated from the 12 points, and an axially averaged value of the ^{140}La activity at shutdown was determined. This value is then proportional to the fuel element power during the last cycle.

4. The absolute fuel element power was then determined by normalizing the axially averaged counting rates to the total power generated by the elements as determined by averaging the beginning-of-cycle and end-of-cycle element powers calculated by ANL.

A typical plot of the counting rate data from one element is shown in Figure 8. Note that calculated to experimental (C/E) ratios are high in the F-row core positions and low in the A-row core positions. This is discussed in others papers given at this meeting.[1,5]

To determine an element's ^{235}U burnup, the accumulated MWd of irradiation on the element are determined from the measured element power and operating time. This value (MWd) is then multiplied by a burnup constant ($g^{235}U/MWd$) to give the ^{235}U burnup. Note the value of the burnup constant ($g^{235}U/MWd$) varies (decreases) with the total MWd due to the contribution of Pu to the fission rate. A comparison of ^{235}U burnup determined by this method and by ANL diffusion calculations is given in Table 3 for element C-023. For the elements irradiated to date, calculated and measured ^{235}U burnups are normally agreed to within two percent.

CONTROL ROD WORTHS

The worth of each of the control rods is measured using the positive period method. Table 4 lists the measured integral rod worths for three core configurations measured. To accomplish this measurement one of the six control rods is fully inserted and the remaining five are ganged to

Table 3. Fuel element C-023 irradiation history

Cycle	Full power days of operation (d)	Calc. power	Exp. power	Calc. MWd	Exp. MWd	^{235}U mass calc. (g)	^{235}U mass exp. (g)	C/E[a]
174-D	12.86	1.16	1.12	14.89	14.36	321	322	1.00
174-E	10.62	1.11	1.13	26.72	26.38	307	307	1.00
175-A	18.52	1.01	0.94	45.42	43.72	285	286	1.00
175-C	17.39	1.36	1.37	69.14	67.56	256	257	1.00
176-B	21.86	0.99	0.83	90.72	85.60	231	235	0.98
176-D	19.45	0.72	0.81	104.63	101.25	214	217	0.99

[a]Calculated to experimental mass.

Table 4. Results of control rod calibrations

Control rod position	Rod worth %ΔK/K		
	Cycle 174-C[a] 12-15-85 ^{235}U wt = 6449 g	Cycle 176-B[b] 06-16-86 ^{235}U wt = 7235 g	Cycle 177-AX1[c] 09-08-86 ^{235}U wt = 6948 g
F-4	1.254	1.936	1.909
F-6	1.289	1.896	1.786
B-4	4.181	5.624	3.045
B-6	4.301	5.241	3.271
D-4	3.984	6.852	4.687
D-6	4.682	6.118	4.749
TOTAL	19.691	27.667	19.447

[a] All-HEU core with 27 fuel elements and six control rods.

[b] This core contains 14 LEU of 27 fuel elements and two LEU (D-4 and D-6) of six control rods.

[c] This core contains 21 LEU of 25 fuel elements and four LEU (D-4, D-6, B-4, and B-6) of six control rods.

maintain reactor criticality. The rod 'on-seat' is then withdrawn a measured amount and the resulting positive period determined. From tabulated values of period versus reactivity addition, the differential rod worth at this location is determined. The rod 'on-seat' is then withdrawn 2 inches and the others ganged at critical. The differential worth is measured at this location, and the measurement is repeated in 2-inch increments until the upper limit of rod travel is reached. Integral rod worths are obtained from the differential worths. This procedure is repeated for each of the remaining five rods. In addition to these data, tabulations of the control rod position versus time into the operating cycle are provided to ANL for comparsion with fuel cycle calculations. Currently, calculated and measured values of control rod worths are not in good agreement[5] and further investigation into this is planned.

COLD CLEAN CRITICAL CORE MEASUREMENTS

At the end of cycle 2, the reactor was shut down to conduct measurements on cold, clean, critical cores consisting of all fresh HEU and LEU fuel elements and shim rods. These measurements were intended to provide the simplest benchmark data for the ANL calculations (i.e., no fission products, well-known fuel distribution, and few in-core experiments).

Four different critical configurations were established using standard approach-to-critical procedures (based on subcritical multiplication). These four configurations and the loading sequence are shown in Figures 9 through 12. The two Magnetic Fusion Energy Experiments (MFE) in core positions C-3 and C-7 were not removed from the cores for fear of possible damage to the experiments.

The following measurements were conducted on the water-reflected cores LEU-1 and HEU-1.

- Approach to critical
- Shim rod calibrations
- Reactivity worth of MFE-6J in C-7
- Core flux mapping by activation of Au wires
- β_{eff}/ℓ

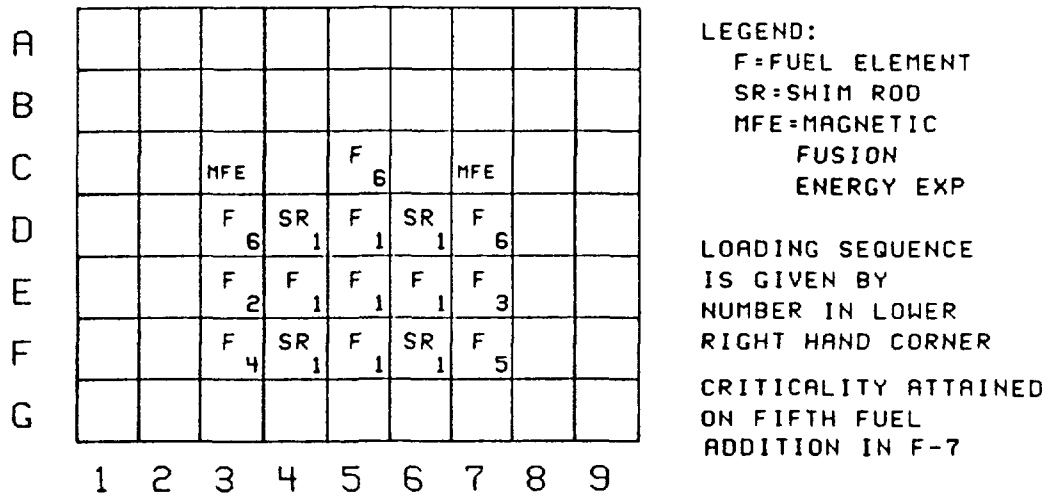


FIG. 9. Water reflected core HEU-1.

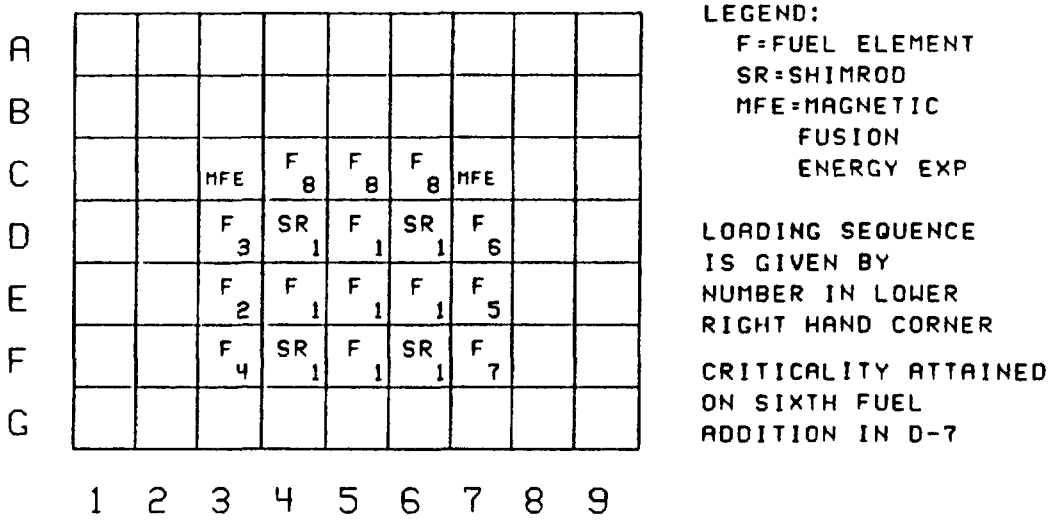


FIG. 10. Water reflected core LEU-1.

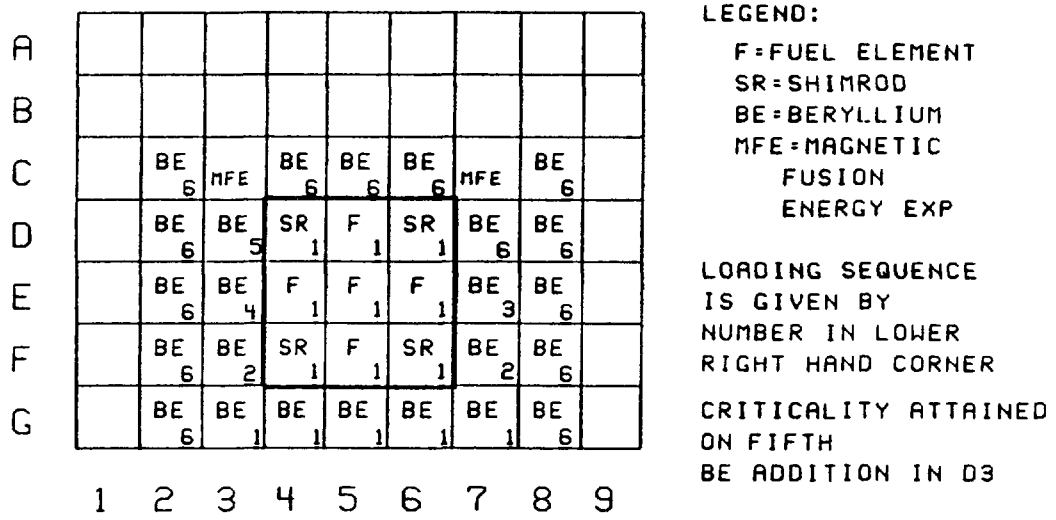


FIG. 11. Beryllium reflected core LEU-2.

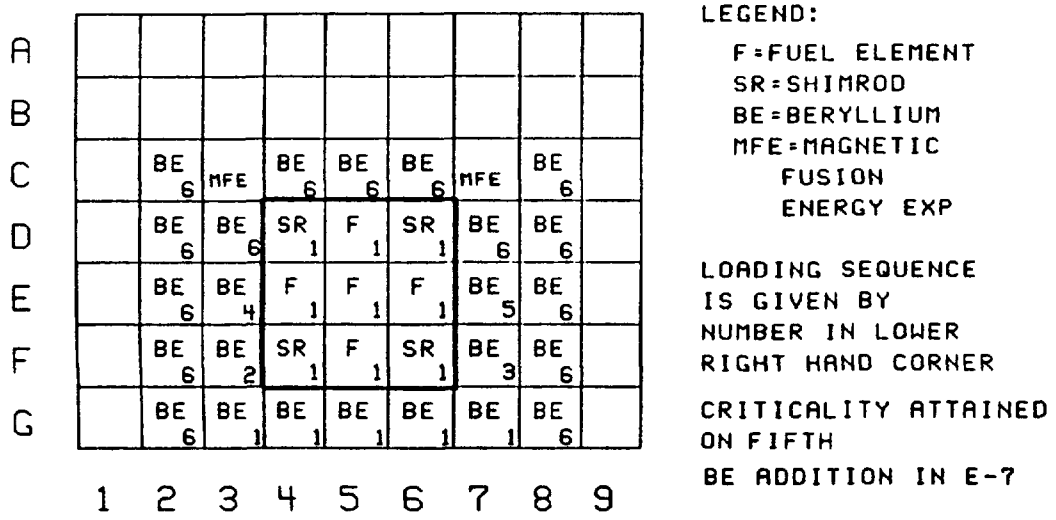


FIG. 12. Beryllium reflected core HEU-2.

Measurements conducted on the Be-reflected core include:

- Approach to critical
- Shim rod calibration
- Reactivity worth of MFE-6J experiment

Prior to assembling these cores, calculations were made by ANL to determine when a given configuration would achieve criticality and to determine the additional amount of fuel or beryllium required to attain sufficient excess reactivity for control rod calibrations. During the course of measurements, these calculations were shown to be quite accurate. Comparison of calculated and measured quantities are given in Reference 5.

CONCLUSIONS

The transition phase of the Whole-Core Demonstration at the ORR is proceeding smoothly. No changes to normal operating procedures have been required. With the exception of the control rod calibrations, the agreement between measurements and calculations is good. Safety margins greater than that required have been maintained throughout this period of mixed core operation. These margins are comparable to those that have existed in HEU cores operated prior to the beginning of this demonstration.

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FULL CORES

Appendix M-5

OPERATIONAL IMPACTS OF LOW ENRICHMENT URANIUM FUEL CONVERSION ON THE FORD NUCLEAR REACTOR

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Abstract

As part of the RERTR Program, a whole-core demonstration using LEU fuel began in the Ford Nuclear Reactor in December 1981. Numerous core performance measurements were made on a full HEU core, full LEU cores, and mixed cores of LEU and HEU fuel. The measurements included control rod worths, temperature and void coefficients, full core flux maps, and in-core and ex-core spectral mesurements. Overall, no significant operational impacts resulted from conversion of the FNR from HEU to LEU fuel.

INTRODUCTION

The University of Michigan Department of Nuclear Engineering and the Michigan Memorial-Phoenix Project have been engaged in a cooperative effort with Argonne National Laboratory to test and analyze low enrichment fuel in the Ford Nuclear Reactor (FNR). The effort was begun in 1979, as part of the Reduced Enrichment Research and Test Reactor (RERTR) Program, to demonstrate on a whole-core basis the feasibility of enrichment reduction from 93% to below 20% in MTR-type fuel designs.

The first low enrichment uranium (LEU) core was loaded into the FNR and criticality was achieved on December 8, 1981. Critical loading was followed by a period of about six weeks of low power testing and 3 months of high power testing during which control rod worths, full core flux maps, and in-core and ex-core spectral measurements were made.

FUEL DESIGN

The LEU fuel was designed to be similar to the high enrichment uranium (HEU) fuel, hence all existing fuel handling equipment and procedures can be used with the LEU fuel. The similarity in the fuel design greatly simplified the HEU to LEU fuel conversion.

The original HEU aluminum alloy fuel used in the FNR for twenty-one years had an overall plate thickness of 0.060 in. (0.020 in. clad-0.020 in. meat-0.20 in. clad). In subsequent HEU aluminide fuel, utilized after 1978, overall plate thickness was reduced to 0.050 in. by reducing clad thickness to 0.015 in. (0.015 in. clad-0.020 in. meat-0.015 in. clad). The enrichment reduction from 93% to 19.5% was accomplished by increasing the ^{238}U loading from 8.0 g to 691 g per element and by increasing the ^{235}U loading from 140.6 g to 167.3 g

per element to overcome the resultant reactivity loss caused by resonance absorption. The extra uranium loading was accommodated by increasing the meat weight fraction from 14.2% to 42.0%, increasing the uranium density from 0.4 g/cc to 1.6 g/cc, and increasing meat thickness to 0.030 in. The resultant overall plate thickness was restored to 0.060 in. (0.015 in. clad-0.30 in. meat-0.015 in. clad). The water gap thickness and number of plates per element are identical to those in fuel that had been utilized in the FNR for over twenty years. Thermal hydraulic performance of LEU fuel was not an issue in obtaining a license for its use.

CONVERSION SCHEDULE

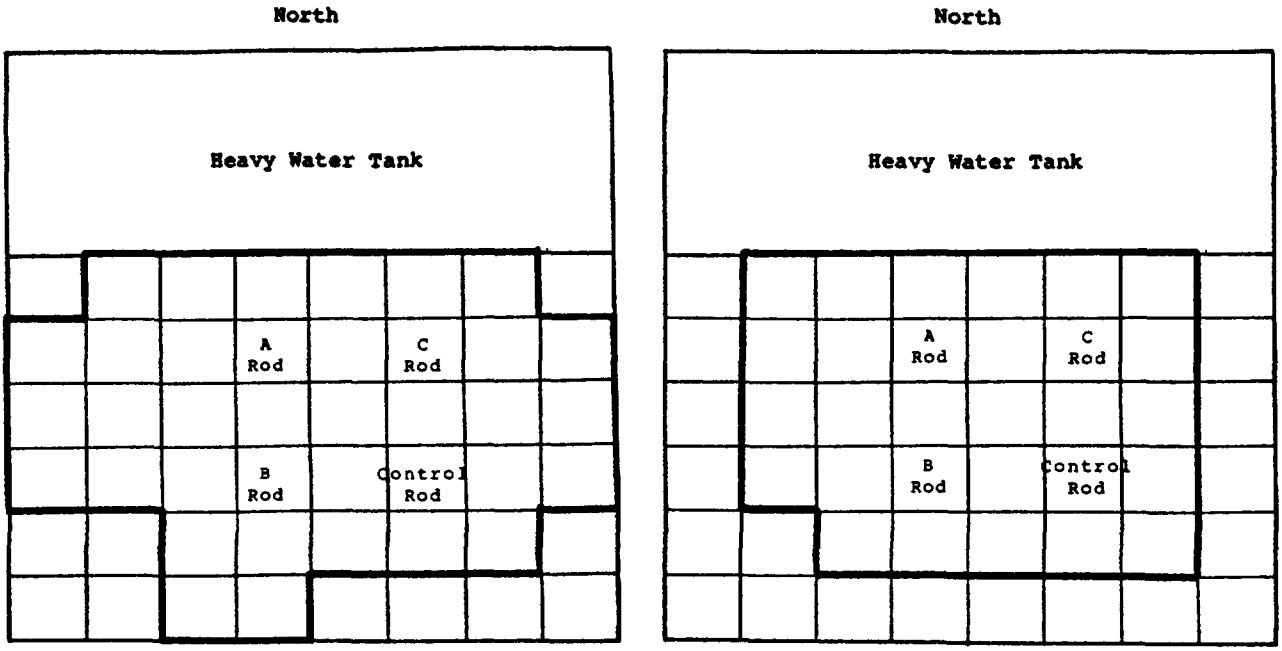
Initially a single LEU element was installed in the reactor to test its integrity. A single element was followed by a whole core, critical loading experiment. Transitions between LEU, HEU, and mixed cores occurred over the next three years, in accordance with the following schedules, as numerous core performance measurements were completed.

October 22, 1981	Installed first LEU element. Performed integrity test.
December 8, 1981	Loaded initial whole LEU core. Performed critical experiment.
May 10, 1982	Restored full HEU core. Remeasured HEU parameters.
December 6, 1982	Began phased transition to LEU core.
June 7, 1983	Loaded whole LEU core. Verified LEU core measurements.
September 30, 1983	Restored mixed LEU-HEU core. Completed HEU burnup.
October 11, 1984	Achieved full LEU core. Removed last HEU element from core.

CORE PERFORMANCE CHARACTERISTICS

It is difficult to precisely compare operating parameter measurements for HEU and LEU cores. Core sizes (number of elements) and configurations (element arrangement in the core grid) vary between cores. The amount of fuel burnup of specific elements, particularly control elements within which control rods are inserted, causes significant variations in parameter measurements. In general, HEU measurements for a large, equilibrium core were compared to LEU measurements for a smaller core with almost fresh, uniform burnup fuel during the three year HEU to LEU conversion. Two typical core configurations are shown in Figure 1. The equilibrium HEU core contained 37 elements. The LEU core contained 29 elements for some measurements and 33 elements for others. Rod positions within the cores are shown on the figure.

Thermal flux levels were measured using a self-powered rhodium neutron detector.



(a) Equilibrium 37 Element HEU Core (b) 29 Element LEU Core

Figure 1. Ford Nuclear Reactor Core Configurations

	Thermal Flux ($n/cm^2/sec \times 10^{-13}$)	
	HEU Core	LEU Core
Center Element	2.15	1.70
Peripheral Element	1.49	1.25
In-Core Water Gap	3.56	3.64
Heavy Water Tank	2.41	2.64

Were the two fuels identical, one would have expected a higher in-core thermal flux with LEU fuel because of the smaller core volume. In fact, the reverse was true, indicating that the LEU core has a harder in-core flux and a higher percentage of power generation results from fast fission. The harder flux increases fast leakage, and the thermal flux in large in-core water gaps and external to the core actually increases with LEU fuel as the fast leakage neutrons thermalize. Since thermal neutron irradiations are generally conducted adjacent to the core, the use of LEU fuel may actually enhance a facility's irradiation capabilities.

Rod worths were measured in a 37 element, equilibrium HEU core; a 26 element, fresh LEU core; and the first 37 element, equilibrium LEU core.

Equilibrium HEU core, July 1, 1980 37 Elements

A Rod	2.2422 %Δk/k	Excess = 2.98 %Δk/k
B Rod	2.1354	
C Rod	2.3794	
Control Rod	0.3251	
TOTAL	7.0821	

Initial LEU Core With Excess Added, December 10, 1981 26 Elements

A Rod	2.2198 %Δk/k	Excess = 2.95 %Δk/k
B Rod	2.3203	
C Rod	2.2833	
Control Rod	0.3822	
TOTAL	7.2056	

Equilibrium LEU Core, October 11, 1984 37 Elements

A Rod	2.3211 %Δk/k	Excess = 3.20 %Δk/k
B Rod	2.0399	
C Rod	2.2565	
Control Rod	0.3718	
TOTAL	6.9893	

The total rod worth of the first, small LEU core was somewhat greater than the HEU total. The equilibrium LEU core total rod worth was slightly less than the HEU total. From an operational viewpoint, the differences, which may be caused by core size, overall core burnup pattern, and particularly the specific burnup of control elements that surround control rods, are of no consequence.

Power defect, the negative reactivity inserted by increasing power from zero to 2 MW, was measured experimentally for one LEU and two HEU cores.

HEU Core	September 1979	-0.21 %Δk/k
	May 1982	-0.31 %Δk/k
LEU Core	July 1983	-0.25 %Δk/k

Temperature coefficient is measured by establishing steady state power with the reactor under automatic control of the control rod. Cooling systems are secured, and reactor power heats the 50,000 gallon pool in which the core is immersed. The reactivity inserted by automatic adjustment of the control rod is in direct reaction to the reactivity loss produced by the increase in pool temperature.

HEU Core	March 1981	-7.5×10^{-3} %Δk/k
LEU Core	July 1983	-7.9×10^{-3} %Δk/k

The temperature coefficient could be expected to increase in magnitude because of increased resonance absorption as slowing down length increases at higher temperatures, but the change observed was insignificant.

Void coefficient is approximated by inserting a thin aluminum blade of known volume into various core locations, the relatively low cross section aluminum producing a void by displacing water. Void coefficient is extremely sensitive to core location and ranges from a maximum magnitude of -1.2 %Δk/k/%void in the center of the core to +0.2 %Δk/k/%void in overmoderated locations. An average value of +1.0 %Δk/k/%void is used at the FNR, and imperceptable differences were seen between HEU and LEU fuel.

CONCLUSIONS

No significant operational impacts have resulted from conversion of the FNR to LEU fuel. Thermal flux in the core has decreased slightly; thermal leakage flux has increased. Rod worth, temperature coefficient, and void coefficient have changed imperceptibly. Impressions from the operators are that power defect has increased slightly and that fuel lifetime has increased.

The FNR is fully converted to LEU fuel. Remaining fresh HEU is being shipped to Oak Ridge National Laboratory; spent HEU is being shipped to the Savannah River Plant. The FNR license and technical specifications are scheduled for renewal in 1985. The licensed use of HEU fuel is being eliminated.

Appendix M-6

FULL CORE MEU FUEL DEMONSTRATION IN THE JMTR

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Abstract

The joint ANL-JAERI program for the RERTR was started in January 1980. The final goal of this program is to achieve the full core conversion to LEU in the JMTR.

In 1980, the LEU Fuel with silicide had not been qualified yet. In this situation, the MEU fuel was selected for the first stage, and integrity of the MEU fuel was confirmed through the following three steps;

(1) hydraulic test,
(2) critical experiments in the JMTRC (Japan Materials Testing Reactor Critical Facility), and
(3) irradiation test in the JMTR.

In August 1986, the full core MEU fuel demonstration test had been successfully completed.

INTRODUCTION

The JMTR is a light water moderated and cooled 50 MW tank type reactor using ETR type fuel elements, and there are many irradiation facilities such as in-core capsules, hydraulic rabbit tubes, in-pile loops and a shroud facility as shown in Table 1, and the specification of the MEU fuel is shown in Table 2.

For converting to the MEU fuel, the guidlines of the design set up as follows:

a) the number of fuel elements loaded in the core should not be increased in order to maintain the fast neutron flux level,
b) the dimentions of fuel elements should be unchanged,
c) the fuel elements should be currently qualified up to 1.6 gU/cm^3 density, and
d) the U-235 content per fuel element should be sufficient to allow operation with the same cycle characteristics as the current HEU core.

The program of the use of the MEU fuel in the JMTR had been proceeded in compliance with the following steps;

(1) hydraulic test,
(2) critical experiments in the JMTRC, and
(3) irradiation tests of two MEU fuel elements in the JMTR.

The full core conversion to the MEU fuel was permitted by Japanese government based on the above experimental results.

Table 1 Characteristics of JMTR

Type	Tank type
Power	50 MW thermal
Moderator/coolant material pressure temperature coolant velocity	 H_2O 14 kg/cm^{2}G 47°C (Inlet), 55°C (Outlet) 10 m/s
Reflector	Be
Fuel material enrichment loading type	 UAl_x-Al 45% 8 kg of 235-U Modified ETR
Control rod	5 H_f rods with 5 fuel followers
Neutron flux ($\times 10^{14}$ n/cm^2·s),(max.) fuel region reflector region	fast (>1 MeV) thermal 4 4 1 4
Power density (ave.)	490 kW/ℓ
Experimental facility	Capsules Hydraulic rabbit tubes Loops Shroud

Table 2 Specification of the MEU fuels

		Standard fuel	Fuel follower
Meat	Material	UAl_x - Al	UAl_x - Al
	Enrichment (%)	45	45
	U-Density (g/cm^3)	1.6	1.6
	Dimension (mm)	0.5 × 62W × 760L	0.5 × 50W × 750L
Clad	Thickness (mm)	0.385	0.385
Fuel plate	Dimension (mm)	1.27 × 71W × 780L	1.27 × 60W × 770L
Fuel element	No. of plates	19	16
	U-235 content (g)	310	205
	Dimension (mm)	76 × 76 × 1200L	64 × 64 × 890L

The full core MEU fuel demonstration test began in July 1986, and had been successfully completed in August 1986.

Developments of the LEU fuel has been started as the next step for the full core with the LEU fuel.

Hydraulic Test

Hydraulic test using hydraulic test facility were planned to
(1) measure the coolant velocity distribution between fuel plates,
(2) confirm the strength of the standard fuel elements by exposing them to hydraulic forces developed with up to 140 percent design flow,
(3) determine the critical velocity, and
(4) confirm withstanding of fuel follower in drop tests.

The results of hydraulic tests are shown in Table 3.

Table 3 Results of Hydraulic Test

Test Item	Type of Fuel Element	Test Result
Coolant velocity distribution measurement	Standard & fuel Follower	Good equalization
Confirmation of the strength against 140% of design velocity (10 m/s)	Standard	Strong enough under the 6 hours test
Determination of critical velocity Calculated critical velocity ; 15 m/sec (in disregard of the strength of fuel core) ; 18 m/sec (on the assumption that fuel core has the same strength as the cladding material)	Standard	Enough withstanding against 20 m/sec (maximum velocity of the test facility)
Drop test	Fuel Follower	Strong enough under the following test conditions o 100 time drop-test at 140% (14 m/sec) of the design velocity o 20 time drop-test at 160% (16 m/sec) of the design velocity o 20 time drop-test at 180% (18 m/sec) of the design velocity

The critical velocity was estimated to be approximately 15 m/s in disregard of the strength of fuel meat. The results of the critical velocity test showed that the MEU fuel had withstanding against hydraulic forces of at least 20 m/s which is the maximum velocity of that facility.

The fuel follower was drop-tested up to 100 times at 140 percente (14 m/s) of average velocity. Further drop tests were conducted up to 20 times each at 160 and 180 percent of average velocity (16 m/s and 18 m/s, respectively).

After every test, coolant channel gaps of fuel elements were measured and no channel gap change was observed.

Critical Experiments

The purposes of the experiments are to obtain nuclear characteristics and to validate neutronics calculation performed by SRAC code system.

Critical experiments in the JMTRC are as follows;
(1) critical mass
(2) excess reactivity,
(3) control rod worths,
(4) flux distribution,
(5) β/ℓ,
(6) shut down margin, and
(7) void coefficient.

The results of main critical experiments are shwon in Tables 4-7 and the standard core in the JMTRC is shown in Figure 1. The validity of the neutronic calculations were confirmed through these experiments.

Irradiation Test

Irradiation tests of two MEU fuel elements were carried out in the JMTR and post irradiation examinations were conducted in the hot laboratory. Items of PIE's are as follows;
(1) sipping test,
(2) measurement of swelling
(3) oxide layer thickness measurements, and
(4) dimensional measurements.

The results of PIE's including burn up of the fuel elements are shown in Table 8 and 9. Measured oxide layer thickness was 5 ~ 8 μm and the calculated value was 21 μm. No swelling was observed at 0.45×10^{21} fission/cm^3 of burn up. The other examinations showed to be in good condition.

Table 4 JMTRC Fuel Element Loading

Kind of Element	Plates per Element	Uranium Density, g/cm^3	^{235}U Content, g
M E U			
Standard fuel			
A	19	1.6	310
B	19	1.4	280
C	19	1.3	250
Fuel follower	16	1.6	205
H E U			
Standard fuel			
A	19	0.7	279
B	19	0.6	237
C	19	0.5	195
Fuel follower	16	0.7	195

Table 5 Calculated Exess Reactivity, Control Rod Worths, Shut-Down Margin and Void Coefficient, Comparing with Measured Ones for JMTRC

	M E U			H E U		
	Measured	Calculated	Δρ (Cal-Meas)	Measured	Calculated	Δρ (Cal-Meas)
Exess reactivity %Δk/k	11.2	11.5	0.3	10.0	10.6	+0.6
Critical mass g, U-235	5077.4	5108	+30.6	4746.8	4741	-5.8
Control rod worths %Δk/k						
SH-1 & SH-2	11.3	11.7	+0.6	11.7	12.5	+0.8
SA-1	3.1	2.9	-0.2	3.2	3.1	-0.1
SA-2	5.9	6.0	+0.1	6.3	6.4	+0.1
SA-3	3.4	3.2	-0.2	3.4	3.3	-0.1
Shut-down margin %Δk/k	14.0	15.3	+1.3	16.4	18.2	+1.8
Void coefficient %Δk/k/void-%	-0.012	-0.013	-0.001	-0.012	-0.013	-0.001

Table 6 Calculated Kinetics Parameters, Effective Delayed-Neutron Fraction β_{eff} and Prompt-Neutron Life Time ℓ_p, Comparing with Measured Ones for JMTRC

	M E U			H E U		
	Measured	Calculated	C/M	Measured	Calculated	C/M
β_{eff}/ℓ_p, sec	111	125	1.13	103	118	1.15
β_{eff}	-	0.00766	-	-	0.00766	-
ℓ_p, μsec	-	61.1	-	-	64.8	-

Table 7 Calculated Thermal Flux Changes by Core Conversion from HEU to MEU Fuel Comparing with Measured Ones for JMTRC

	Measured	Calculated
Thermal neutron flux		
Fuel region	-8 ~ -12%	-8 ~ -13%
Be reflector region	-1 ~ -3%	0 ~ -3%
Fast neutron flux		
Fuel and Be reflector region	-	+2 ~ -2%

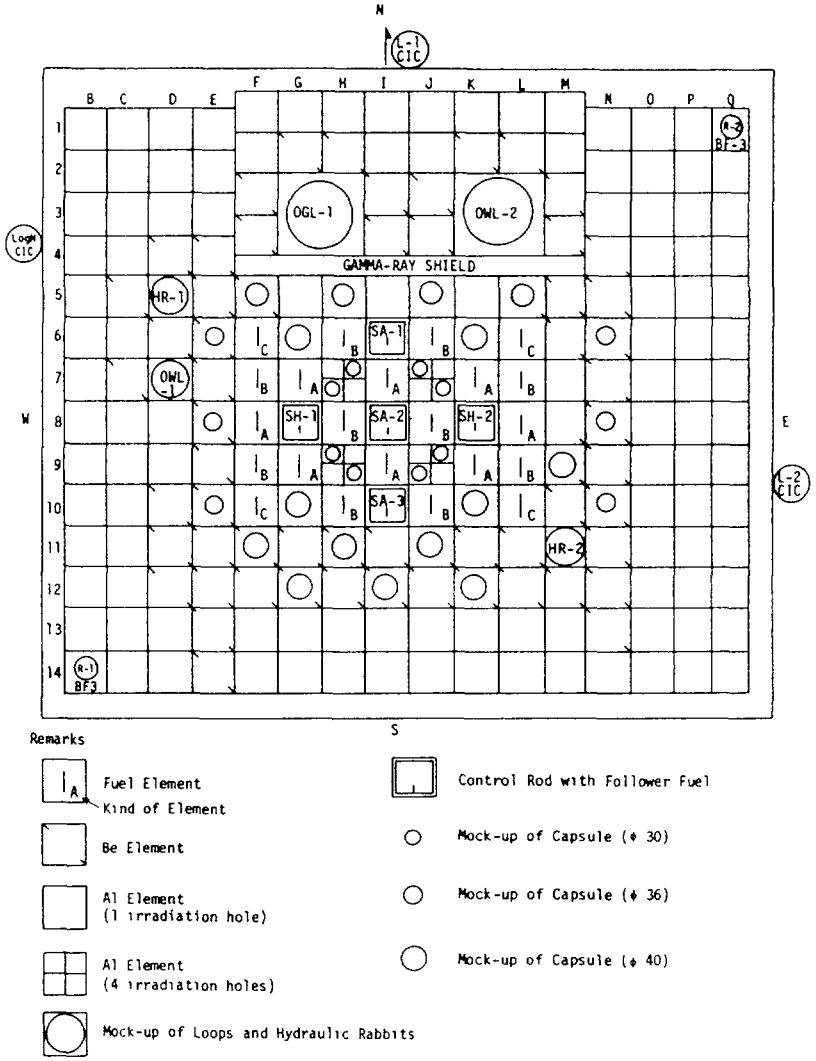

Fig. 1 JMTRC Standard Core

Full Core MEU Fuel Demonstration

The full core demonstration with MEU fuel was performed July 8, 1986 through August 2, 1986 with satisfactory results.

The important characteristics to convert to MEU fuel have been already measured as mentioned above. Following items were performed on the full core with MEU fuels.

(1) excess reactivity,
(2) one rod stuck margin,
(3) shut down margin,
(4) temperature coefficient,
(5) characteristics of burn up,
(6) radioiodine concentration in primary cooling water, and
(7) sipping test.

The main results are shown in Table 10 and Figures 2 and 3. The demonstration core configuration with MEU fuels is shown in Figure 4.

Table 8 Swelling of MEU Test Fuel Elements

Element	Irradiation Position	Uranium Density (g/cm^3)	Ave. Fission Density (fis/cm^3,Cal.)	Thickness Change (mm)	Swelling (%Δv/v)	
					Meas.	Cal.
SM-1	H-8	1.6	0.45×10^{21} (28.4% B.U)	0.0	0.0	2.8
SM-2	J-8	1.6	0.44×10^{21} (28.1% B.U)	0.0	0.0	2.8

Table 9 Oxide Layer Thickness of MEU Test Fuel Elements

Element	Oxide Layer Thickness (mm)	
	Meas.	Cal.
SM-1	0.005 ~ 0.008	0.021
SM-2	0.005 ~ 0.008	0.021

Table 10 Comparison of Measurements and Calculation Results of MEU Full Core Demonstration

	Measured	Calculated
Excess reactivity $\%\Delta k/k$	10.6	11.8
Shut-down margin $\%\Delta k/k$	21.5	20.5
Temperature coefficient $\Delta k/k/°C$ at 30°C	1.02×10^{-4}	1.25×10^{-4}

Concluding Remarks

The full core demonstration with MEU fuel had been successfully completed. Some neutronics data such as excess reactivity, shut down margin, temperature coefficient, etc were measured and compared with neutronics calculation. The results are satisfactory.

During the reactor operation, fission products leakage was carefully checked by the primary coolant analysis. Sipping tests were also performed to

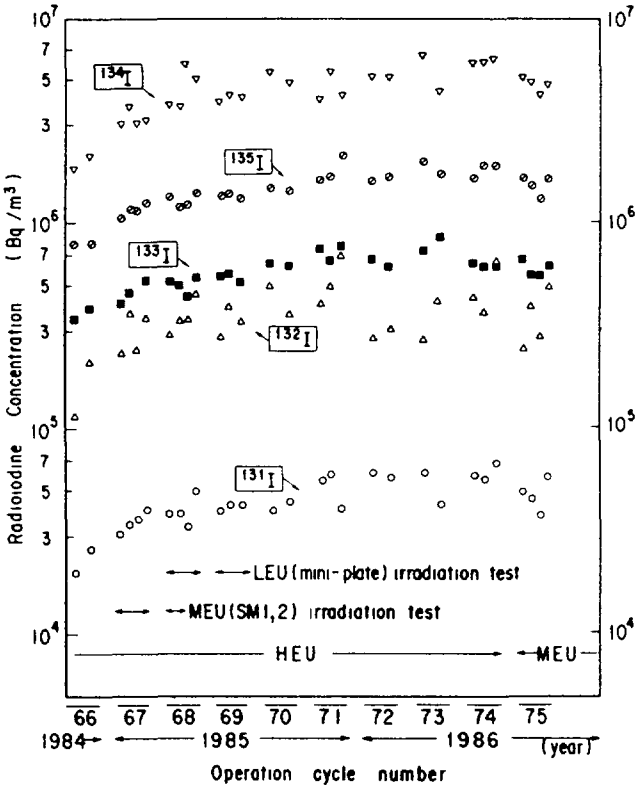

Fig. 2 Radioiodine Concentration in Primary Cooling Water of JMTR

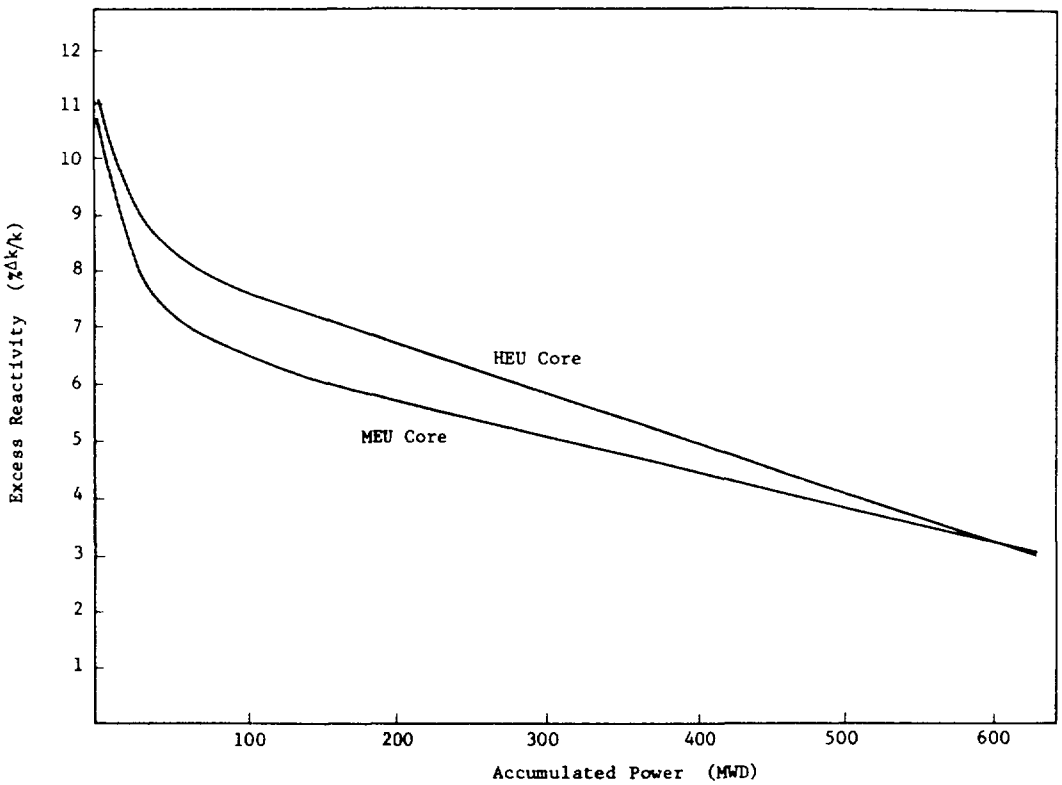

Fig. 3 Characteristics of Burn up for HEU and MEU Core

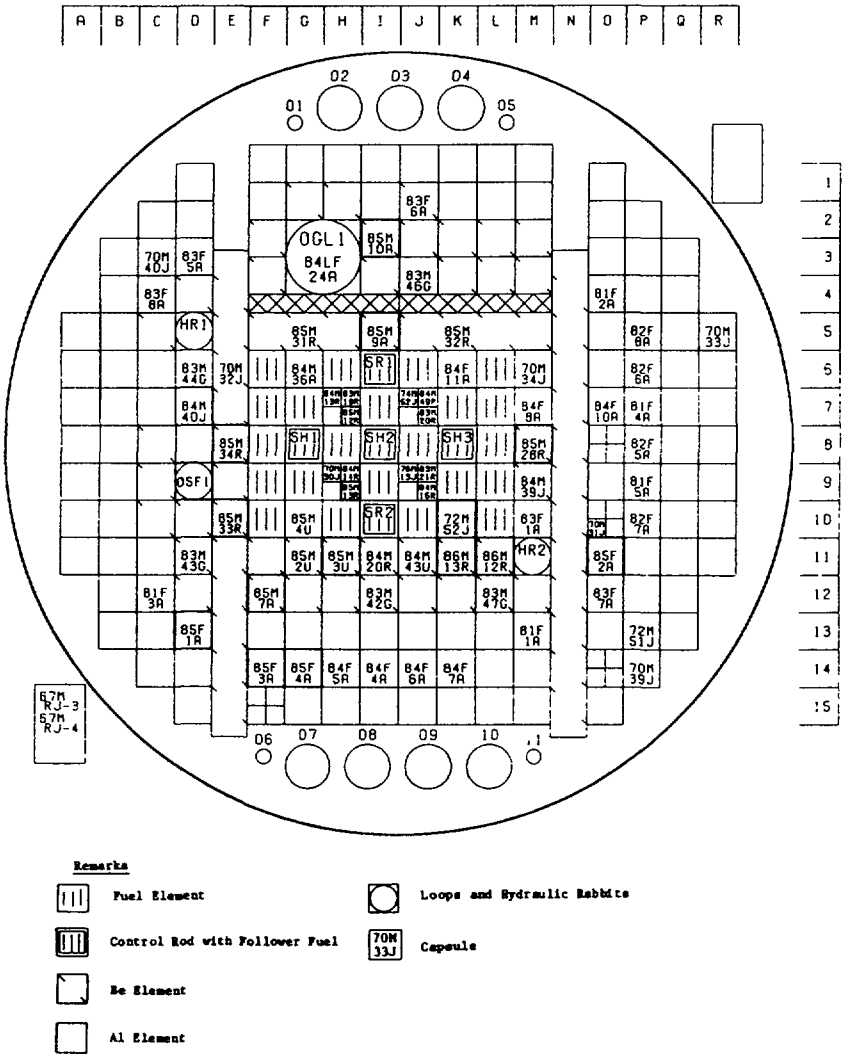

Fig. 4 MEU Fuel Demonstration Core Configuration

check if the MEU fuel was failed. As the results, no fission products were observed.

In addition, the brief review was made on the hydraulic tests and the critical experiments in the JMTRC, of which results were presented in the previous meeting.

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Appendix N

TRANSPORTATION, SPENT FUEL STORAGE, AND REPROCESSING

Abstract

Information is provided on transportation of fresh and spent fuel elements, spent fuel storage, and reprocessing in the U.S.

A variety of transport containers for fresh fuel and spent fuel are described along with certain contractual, transportation, reprocessing batch size, and economical considerations. Examples are provided of specific fresh fuel transport regulations in the FRG (as of August 1982) and administrative procedures in Japan for transport of fresh fuel elements.

Methods and results of criticality analyses for storage of HEU, MEU, and LEU fuels are presented. Results include fissile loading, fuel element geometry, and storage rack geometry considerations.

U.S. Federal Register Notices (as of 30 December 1987) on DOE's "Receipt and Financial Settlement Provisions for Nuclear Research Reactor Fuels" are provided.

Appendix N-1

TRANSPORTATION OF FRESH FUEL ELEMENTS

Appendix N-1.1

TRANSPORTATION OF MTR FUEL ELEMENTS WITHIN THE FEDERAL REPUBLIC OF GERMANY

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Hanau, Federal Republic of Germany

Abstract

Since April 1982, a regulation has been introduced in Germany that requires transports of HEU and Pu (inluding MTR fuel elements) to be performed by a special safety vehicle called "SIFA". A summary of transport regulations as of August 1982 is provided along with data on fresh fuel shipping containers and the SIFA safety vehicle.

1. Introduction

The transport of MTR fuel elements can generally be performed by road, by rail by sea or by air.
Transnuklear GmbH transported MTR fuel elements through Europe almost exclusively by road.

In the past TN (Transnuklear GmbH) has executed a series of transports of MTR fuel elements by road to almost every research reactor operator. In addition, TN has also transported MTR fuel elements in a combined road/air/road transport to the United States of America.

Since April 1982, in Germany a regulation has been introduced, that transports of HEU and Pu can only be performed by a special safety vehicle called SIFA (Sicherheitsfahrzeug). MTR-transports are also subject to this regulation. TN has in the mean time carried out three MTR fuel element-transports with the SIFA. In particular, transports of MTR fuel elements have been made to KfA (Kernforschungsanlage) in Julich and to Sweden.

2. Transport regulations

The transportation of radioactive materials by road in the Federal Republic of Germany is subject to the GGVS regulations. These regulations are closely based on the ADR, which cover the international road transport of radioactive materials.

For the transportation of rad. materials by air, the JATA regulations apply, and for transportation by sea the IMCO regulations apply.

These regulations are based on the IAEA-regulations for the Safe Transport of Radioactive Materials (1973).

3. History of new regulations (see point 1 above) for physical protection in Germany (SMC - Safety Measurement Catalogue)

In September 1976, TN was informed that a new SMC would be prepared by the BMI (Federal Ministry of the Interior).

In July 1977 the announced SMC draft was a subject for discussion in the "Association of Nuclear Fuel Cycle Companies". TN prepared objections to the draft -- without success.

The SMC was then put into effect by the German authority (PTB), responsible for, among other things, the transport of radioactive material, in December 1977, but was forwarded to TN only in April 1978. The transition period will occur as follows:

> 15 months transition status starting from the time of notification, i.e., valid until October 1980; thereafter assuming full responsibility of the SMC catalogue.

TN received a study in which a "SIFA" (Safety vehicle) was specified by the SMC.

The SMC and the study raised questions which in the technical sector can only be resolved by specialized companies. After initial thorough research it was established that the final specified regulations could not be reached in the 15 months indicated; this was due to

- administrative measures
- technical measures relating to the vehicle,
- the fact that a specific communication system did not exist.

Starting in the summer of 1977, TN informed companies with nuclear activities, for example, Alkem, Nukem and nuclear research centers, of the measures in this catalogue.

In a letter dated 16 April 1980, the PTB informed TN and a large number of other companies that, according to an order from BMI dated 3 April 1980, the Safety Measures Catalogue, edition dated December 1977, and that the repeatedly extended transition status will thereby be terminated.

Decision on construction of the "SIFA"

Following publication of the BMI order, TN decided to construct the vehicle required. Design and construction started in June, 1980. Construction took about 18 months. The vehicle came into force in April, 1982.

4. Execution of transports since the SMK 77 came into force

Use of the SIFA - vehicle

According to the SMC the transport of highly enriched uranium (more than 250 g U-235/transport) or plutonium (more than 100 g Pu/transport) have to be carried out with the SIFA-vehicle.

This armored vehicle which is protected against the use of firearms and explosives, is equipped with a communication system with redundancy. The armed crew is security cleared and specially trained. In addition the transport is kept under surveillance by a high security control center with a radio control system covering the whole Federal Republic of Germany. For Pu/U - quantities exceeding 2 kg/transport or 5 kg/ U-235/transport an additional escort vehicle with a communication system and an armed crew must be provided.

5. Shipping container

For the shipment of MTR-elements, 2 types of packaging is available, namely the model UK 1612 and the MTR-bird cage. Both packagings are approved by the PTB under approval no. D/4031/F (Rev. 4)
D/4033/AF (Rev. 2)

5.1 Description of MTR-bird cage

The MTR-bird cage is a type A packaging, gross weight 306 kg. It consists of a steel frame bird cage in special form made of tubes having the external dimensions of 770 x 770 x 1440 mm. In the bird cage itself there is a steel inner container with dimension: 358 x 350 x 1020 mm supported in central position by a steel tube container.

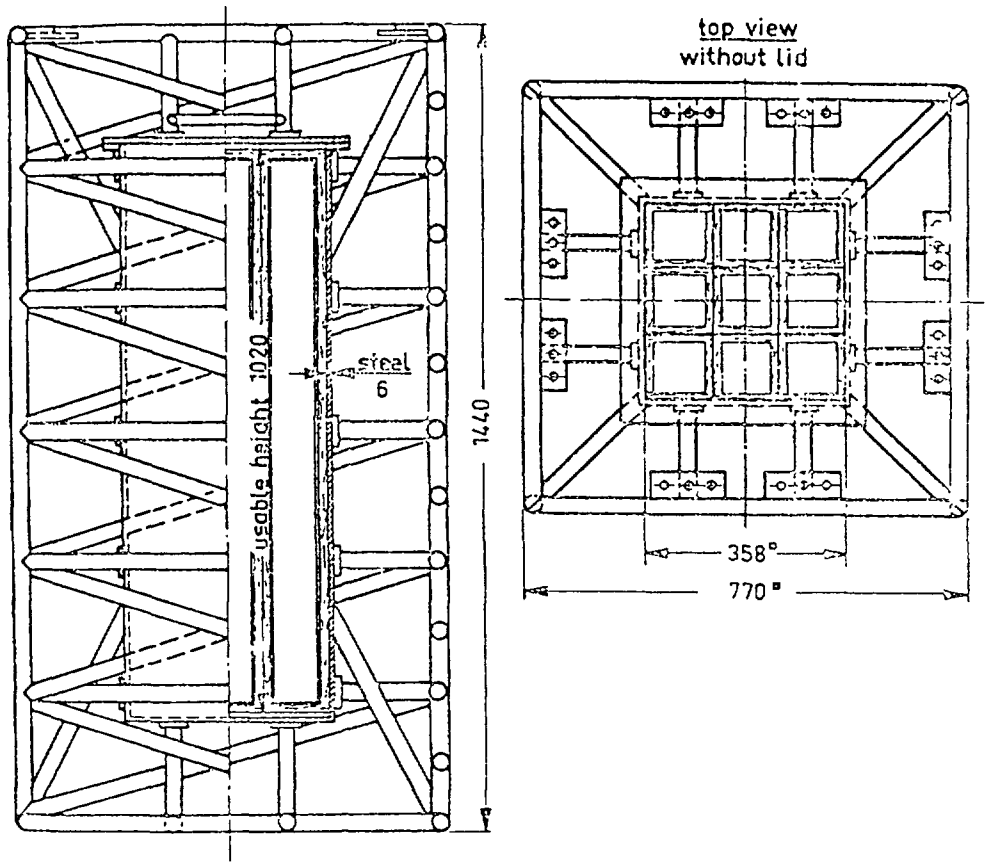
If there is any space in the inner container besides the element it must be filled out in appropriate way so that the elements will be locked and cannot move.

The authorized content of the container is max. 9 MTR-fuel elements not having more than 200 gr U-235 each.

In case the fissile content of elements is different from the above mentioned quantity, the allowable quantity is 1,8 kg U-235/packaging. Authorized enrichment is up to 93,5 %, authorized activity max. 150 mCi. Maximum number of container/shipment is 6.

5.2 Description of UK 1612

The container UK 1612 is a type A packaging. Gross weight is 282 kg. It consists of a rectangular steel box (Design no. 1612) with the exterior dimensions of about 530 mm x 762 mm x 20155 mm, with 8 inner transport positions.



Main Data:

Gross Weight: 300 daN
Authorization: Type A

CONTAINER FOR MTR-FUEL ELEMENTS

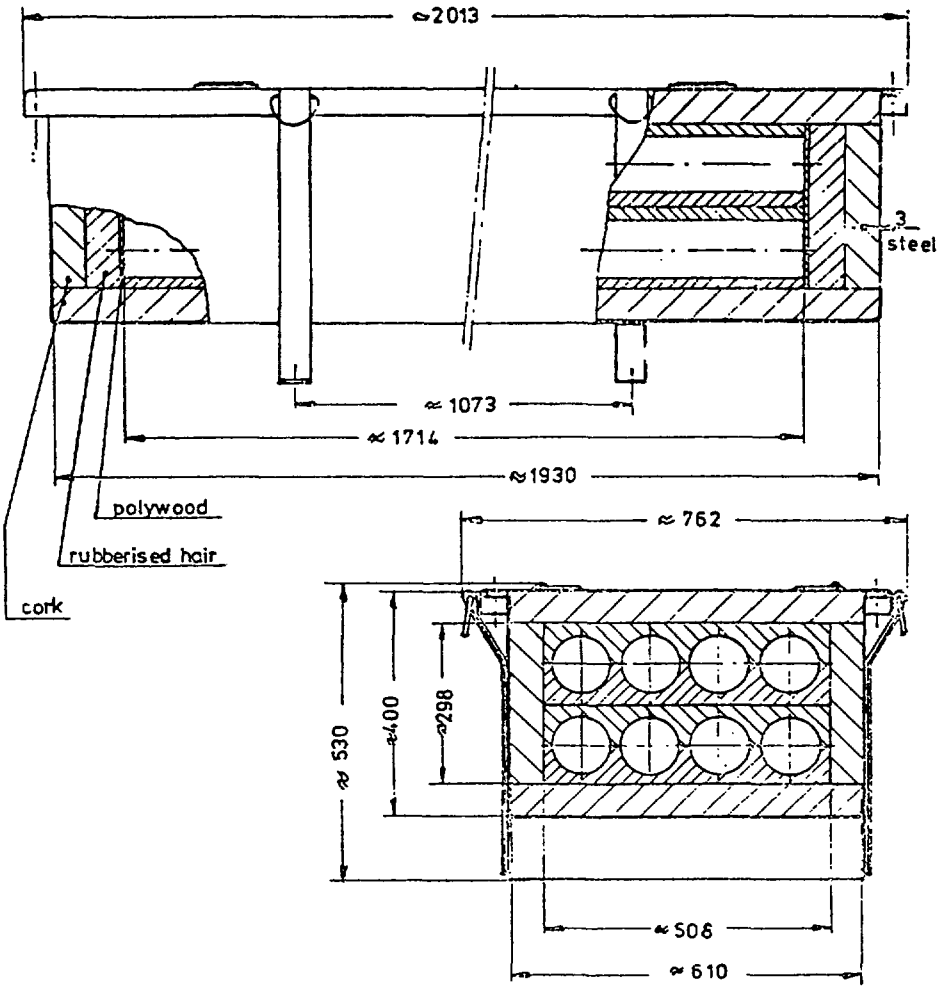
Assembly Drawing

Authorized content per packaging:

a) max. 6 nonirradiated tubic MTR fuel elements (active length: no less than 72 cm, per element containing 29 % on max. 93 % of U-235 enriched uranium with max. 403 g U-235 in form of U/Al-Alloy (altogether max. 2,418 kg U-235)).

b) max. 7 nonirradiated rectangular MTR fuel elements with 19 plates each, or MTR control elements with 16 plates each (type GE), containing per element up to 559,44 g on about 93 % of U-235 enriched uranium with max. 520 g U-235 in form of U/Al-Alloy (altogether max. 3,64 kg U-235).

c) max. 8 nonirradiated tubic MTR fuel elements (active length: no less than 60,96 cm, active diameter: max. 10,41 cm) in form of U/Al-Alloy, containing per element up to 30 % on about max. 81 % of U-235 enriched uranium with max. 210 g U-235 (altogether max. 1,68 kg U-235)



Main Data:

Weight: (calculated) empty 162 kg
full 282 kg

BNFL-CONTAINER DESIGN 1612

Assembly Drawing

d) max. 8 nonirradiated MTR fuel elements (active length: between 38,1 and 63,5 cm, active surface-cross-section: max. 79,03 cm^2) incorporating uranium-metal of 94 % U-235 max. enrichment, in form of U/Al-Alloy sandwiched in an Al-sheet with a U/Al mass ratio from max. 0,14. Containing per element max. 300 g U-235 (altogether max. 2,4 kg U-235)

e) max. 8 nonirradiated tubic or rectangular MTR fuel elements (active length: between 35,56 and 63,5 cm, active surface-cross-section: max. 87,1 cm^2) incorporating uranium-metal of 94 % U-235 max. enrichment, in form of U/Al-Alloy with a U/Al mass ratio from max. 0,14. Containing per element max. 300 g U-235 (altogether max. 2,4 kg U-235)

Package classification

authorized content mentioned under:	a)	b)	c)	d)	e)
fissile class:	II	III	II	II	II
transport index:	12,5	50	33	33	12,5
max. no. of container/shipment:	4	1	15	15	4

A) Registration No.: - motor tractor: HU-PK 850
- trailer: HU-PK 851

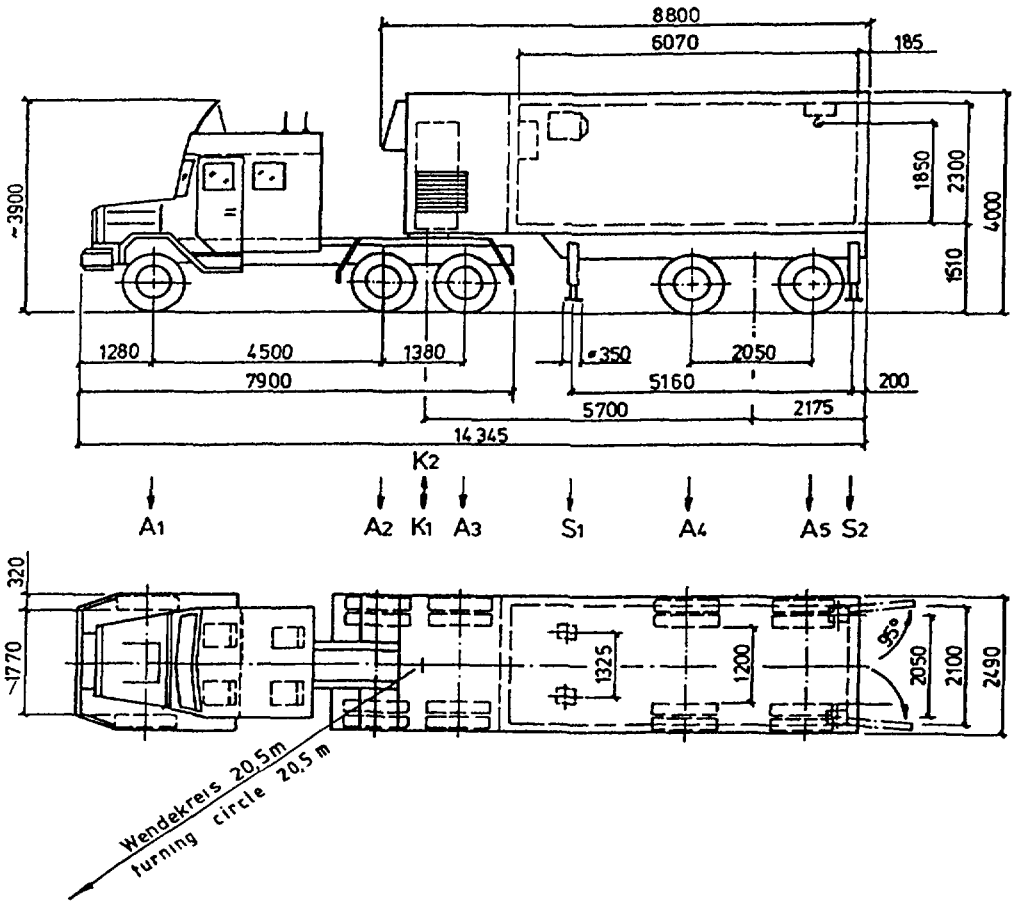
B) Owner: - Firma Transnuklear, Hanau

C) Dimensions:
- Sifa-total length: 14.385 mm
- Sifa-total height: 4.000 mm
- Sifa-total breadth: 2.490 mm
- loading space:
  - length: 6.070 mm
  - breadth: 2.100 mm
  - height: 2.300 mm

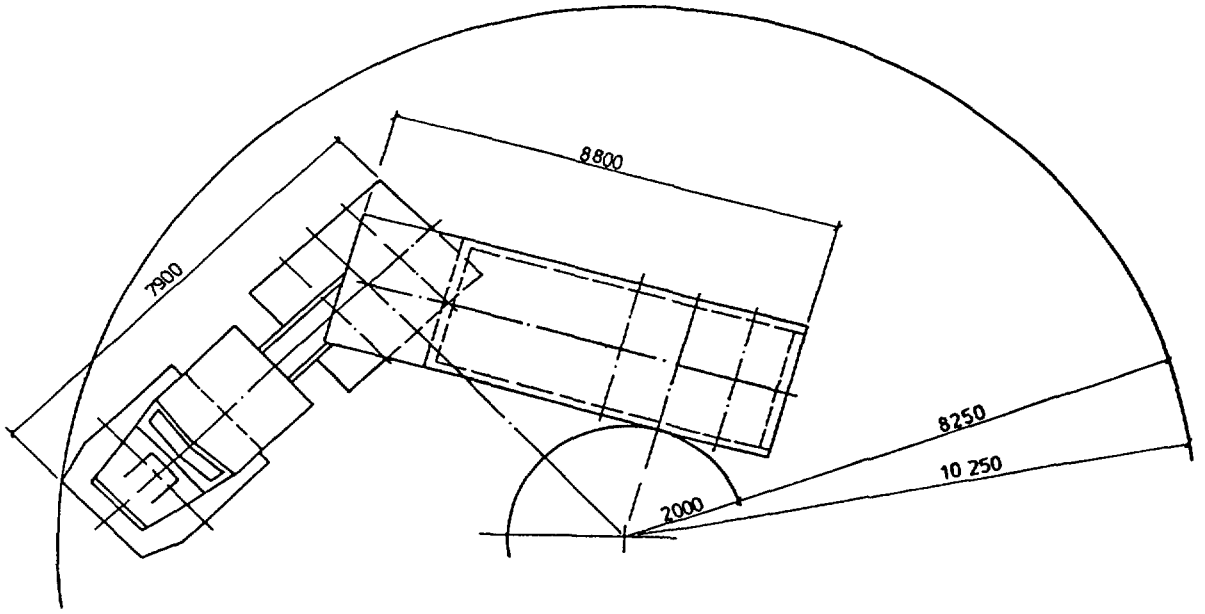
D) Weights:
- empty weight (total) : 33.730 kg
- usable weight: 4.270 kg for an allowable total weight of 38 t
  8.270 kg for an allowable total weight of 42 t
  13.770 kg for an allowable total weight of 47 t

Further details concerning dimensions/weights/axle weights/turning circle can be taken from the attached data sheets.

"SIFA" DATA



"SIFA" MAIN DIMENSIONS



"SIFA" TURNING CIRCLES

Description	Allowable weights (kg)	Allowable weights (kg)	Allowable weights (kg)
total weight Sifa	38000	42000	47 500
empty weight (total)	33 730	33 730	33 730
usable weight	4 270	8 270	13 770
axle weight A 1	7500	7500	7 500
axle weight A 2	8000	8000	10 000
axle weight A 3	8000	8000	10 000
axle weight A 4	8000	10000	10 000
axle weight A 5	8000	10 000	10 000
K 1	9 730	9730	13730
K 2	9 730	9730	13 730
S 1	14 500	14 500	14 500
S 2*	14 500	18 500	18500
empty weights	A 1	A2 + A3	GES
motor tractor	7 200	6375	13 575
	K1	A4 A5	GES
trailer	5730	14330	20060

* when crane load fully extended

"SIFA" WEIGHT SURVEY

Appendix N-1.2

TRANSPORTATION OF FRESH FUEL ELEMENTS FOR JAPANESE RESEARCH REACTORS

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Abstract

Administrative procedures in Japan for transportation of fresh fuel elements are described.

1. Introduction

The transportation of nuclear materials is generally assorted to three means; road transport, sea transport and air transport. In Japan, fresh fuel elements (enriched uranium fuels) for research reactors are ususally transported by vehicles from the fabrication plant to the reactor site when the plant is located in Japan.

In the case of foreign fabricators the transportation of fresh fuel elements is carried out by sea or by air. In the current status of nuclear materials transportation in Japan, air transport is more difficult than sea transport. Moreover, if the packages are type B it is very difficult to carry out the transportation by air. Only one case of air transport of fresh fuel elements was experienced in Japan, which was a transportation of type A fissile class II packages from France (CERCA) to Japan (Research Reactor Institute, Kyoto University).

In this paper we describe the administrative procedures in Japan for the transportation of fresh fuel elements.

2. Administrative Licensing Procedures

2.1 Regulations

Transportation of nuclear materials in Japan is regulated almost by the Science and Technology Agency (STA) and the Ministry of Transportation (MOT) regulations which are based on the Law for the Regulations of Nuclear Source Material, Nuclear Fuel Material and Reactors (for road transport), the Ship's Safety Law (for Sea Transport) and the Civil Aeronautics Law (for air transport). These Japanese regulations are based on the IAEA Regulations for the Safe Transport of Radioactive Materials (1973 revised edition). For road transport, the vehicle transportation is further regulated by the Police Agency Regulation if the packages exceed certain criteria.

2.2 Application and Approval

An example of actual administrative procedures for nuclear material transportation is shown in Fig. 1. It is the case for type B and/or fissile class nuclear material packages.

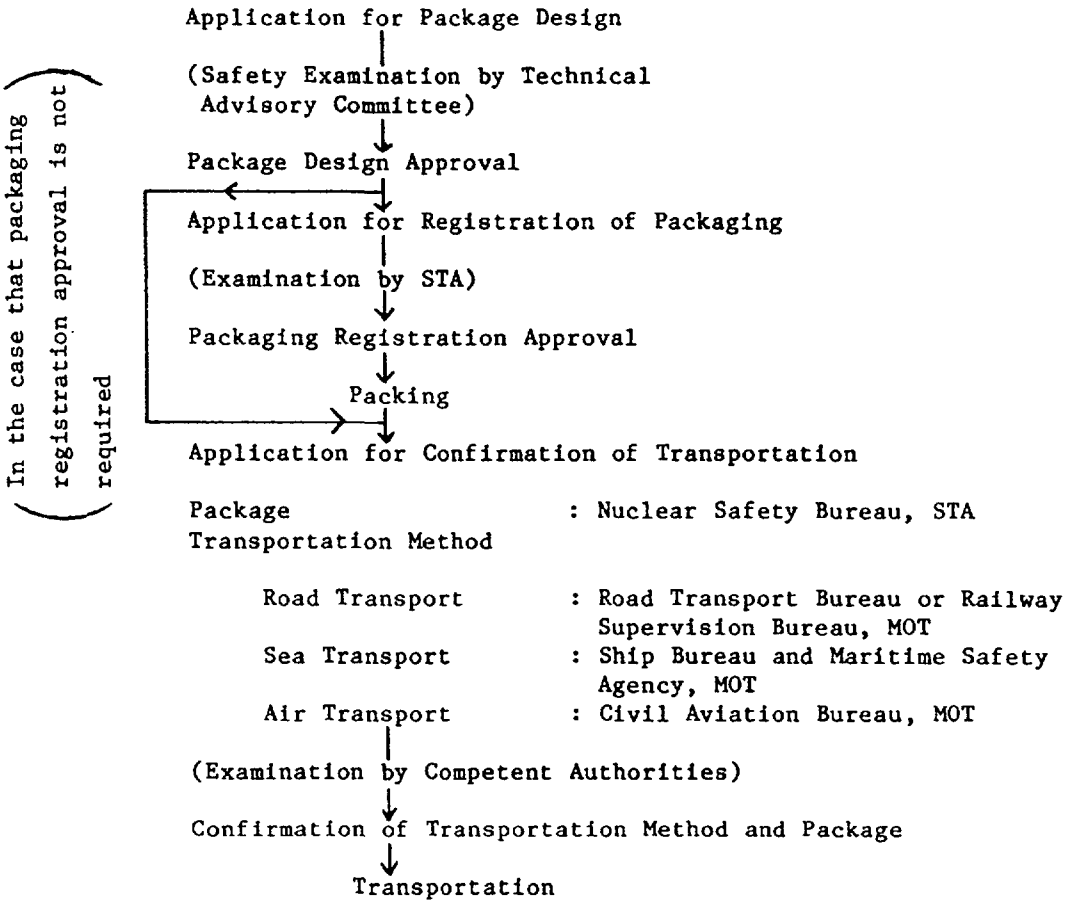


Fig. 1. Licensing flow chart for the transportation of type B and/or fissile class nuclear material package.

3. Experience

When we carried out the air transport of fresh fuel elements (45 % enriched uranium; MEU) from France to Japan in 1981, the IAEA certificate of competent authority of the United States was required because of transient stop of the cargo plane at Fairbanks Airport in Alaska.

Appendix N-1.3

THE UKAEA UNIRRADIATED FUEL TRANSPORT CONTAINERS

R. PANTER
United Kingdom Atomic Energy Authority,
Harwell, Didcot, Oxfordshire,
United Kingdom

Abstract

The UKAEA transport containers for unirradiated MTR fuel elements are briefly described.

MTR fuel elements manufactured in the U.K. are, prior to irradiation, transported by road or air using the type GB/1612A container or the shorter type GB/3104A container.

These containers are of similiar construction, being rectangular lidded steel boxes, lined with core slabs and using synthetically bonded hair packing to support eight elements in two layers. The type 1612 is illustrated in the attached diagram.

The containers have been approved in 1982 and 1983 respectively under the 1973 IAEA regulations, as type B(U)F fissile class II designs.

MTR Type fuel transit container

Design No. 1612

Reference Drawing No. FE 10758

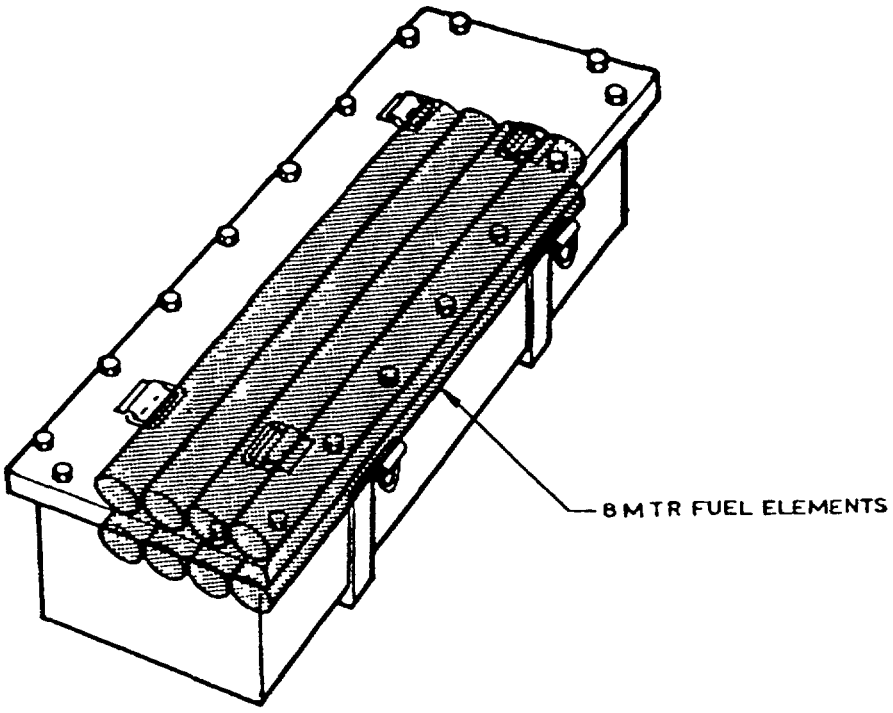
General Description
Non gas tight mild steel container with removable lid.

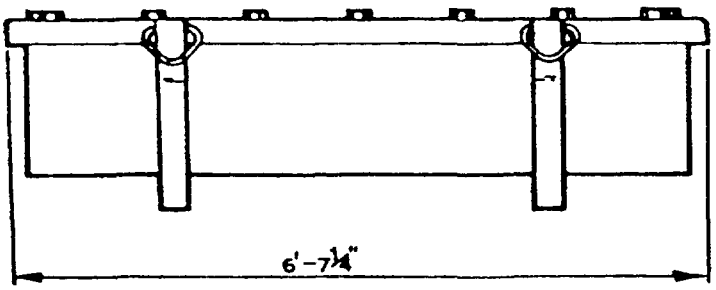
No. of Flasks
29.

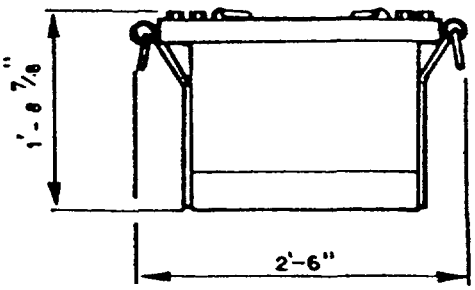
Unladen Weight
355 lbs (Calculated)

Materials (shielding)
0.128" mild steel.

Cavity size or capacity
6'—4" x 2'—0" x 1'—3¾"
8 MTR fuel elements.







Appendix N-1.4

TRANSPORTATION OF UNIRRADIATED TRIGA-LEU FUEL

GA TECHNOLOGIES, INC.
San Diego, California,
United States of America

Abstract

Shipping containers for unirradiated TRIGA-LEU fuel are described.

Unirradiated TRIGA fuel is shipped in licensed shipping containers designated as either TRIGA-1 or TRIGA-2. The TRIGA-2 container is designed for special elements such as fuel-followed control rods and temperature instrumented fuel rods.

Seven 1.5 in. nominal O.D. fuel elements or 25 of the 0.5 in. nominal O.D. rods fit in the TRIGA-1 container.

Descriptions of the shipping containers are as follows:

Model No. TRIGA-1.

Description: TRIGA fuel element shipping container. The outer packaging is fabricated to DOT Specification 6J requirements. The outer dimensions are approximately 22.5 in. in diameter by 36 in. high. The inner vessel is a 5-in. Schedule 40 carbon steel pipe. Dimensions of the inner vessel are approximately 31 in. in height with a 1/4-in. thick wall and a 5-in. inside diameter. The top of the inner vessel is a threaded pipe cap and the bottom is a welded 1/4-in. thick flat disc. The inner vessel is centered and supported within the outer packaging by eight, 3/8-in. diameter braced, support spacer rods. The void between the inner vessel and the outer packaging is filled with vermiculite tamped to a minimum density of 4.5 lb/ft^3. Maximum gross weight including contents is approximately 235 lb.

Model No: TRIGA-2.

Description: TRIGA fuel element shipping container. The outer packaging is fabricated to DOT Specification 6J requirements. The outer dimensions are approximately 22.5 in. in diameter by 55 in. high. The inner vessel is a 5-in. Schedule 40 carbon steel pipe. Dimensions of the inner vessel are approximately 50 in. in height with a 1/4-in. thick wall and a 5-in. inside diameter. The top of the inner vessel is a threaded pipe cap and the bottom is a welded 1/4-in. thick flat disc. The inner vessel is centered and supported within the outer packaging by eight, 3/8-in. diameter braced, support spacer rods. The void between the inner vessel and the outer packaging is filled with vermiculite tamped to a minimum density of 4.5 lb/ft^3. Maximum gross weight including contents is approximtely 330 lb.

Appendix N-2

TRANSPORTATION OF SPENT FUEL ELEMENTS

Appendix N-2.1

REMARKS ON THE TRANSPORTATION OF SPENT FUEL ELEMENTS

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Geesthacht, Federal Republic of Germany

Abstract

Information and data are provided on several aspects of the transportation of spent fuel elements. These aspects include contract, transportation, reprocessing batch size, and economical considerations.

1. Contracts

a) A contract for reprocessing the spent fuel elements with a US-Department of Energy (DOE) reprocessing plant (e.g. Savannah River, Idaho). Present contracts will be valid till 31.12.1987.

The attachment A of this contract has to be prepared for each transport and sent to the reprocessing plant. After this one receives the acceptance of the actual transport. The acceptance is valid for one year.

Remark: Standard contracts include only HEU (enrichment > 20 %) and UAl_x or U_3O_8 fuel. For other fuel and other enrichments, one has to contact US-DOE directly. LEU and silicide fuels will probably be accepted for reprocessing in the near future.

b) Transport company

The transport company coordinates the transportation for the whole route (e.g. at home and abroad, harbours, container ship, actions for physical security, reprocessing contract). Normally the transport company has subcontractors for the transportation.

c) Risk insurance

The needed risk insurance differs from country to country. This is of main importance if during shipment the container ships entry ports of third countries.

2. Transportation

a) Minimum decay time of spent fuel elements: normally 120 or 200 d.

b) Cranes

When comparing the max. laden weight of the casks with the needed allowable crane weight, one has to add some weights (e.g. water inside the casks, ropes, impact absorbers, fuel elements).

c) Subpressure

The casks have to be dried by air. After drying inside the container, the pressure should be ca. 6×10^4 Pa.

d) Contamination

The cask surface has to be free of contamination

$\beta + \gamma \leq 10^{-5}$ µCi/cm²

$\alpha \leq 10^{-6}$ µCi/cm²

e) Cask limitations

There exists cask limitations for U-5 content of the fresh fuel elements, decay heat, fission product inventory.

f) Physical protection

In many countries the physical protection demands for transportation of spent fuel elements are higher than the IAEA recommendations. In many cases guarded transportation and additional communication systems are required.

g) Shipping

The price for shipment with container ships depends mainly on the weight of the casks and not on the volume. On the other hand,US-regulations require that splittable cargos be transported independently. Therefore, it is recommended to ship two casks with two containers. Otherwise, one has additional cost in the US-harbour for separating the casks onto two containers.

3. Reprocessing

a) Minimum reprocessing batch

One reprocessing batch consists normally of that number of fuel elements which have been transported to the reprocessing plant within 60 days. On the other hand, the reprocessing price has two parts:

- ca. 1000 US-$/kg (U+Al) in 1985
- a minimum charge of 44.500 US-$.

Therefore a reprocessing batch should consist of sufficient spent fuel elements that the total weight is greater than a minimum weight (contact DOE or transport company). Otherwise, it is recommended to store the fuel elements in the reprocessing plant till the total weight is higher than this minimum weight. It is possible that smaller reactor stations can reprocess their spent fuel together. But in this case, the reprocessed fuel can only be separated by calculations. Additional fees need to be paid for storage of spent fuel, conversion to UF_6, shipment to the enrichment plant.

b) Spent fuel

Taking 1000 MWd/a, 40 % burnup, 200 g U-235/fuel element, a ratio of 1 : 5 for the number of control fuel elements and fuel elements, there will be produced around 17 spent fuel elements per year at the facility.

4. Economical considerations

a) Transportation

Transportation cost depend on: management cost, insurance, physical protection demands, amortisation for the casks and shipment. The shipment cost depend mainly on the freight rate of the container ships (see 2.e). If cranes allow transportation with heavier casks, it will be cheaper to use these casks, as the possible number of fuel elements which can be transported increase faster than the cask weight.

b) Reprocessing

Reprocessing is reasonable only if one can reach the minimum weight for a reprocessing batch (see 3.). If there are enough spent fuel elements at the reactor facility, the reactor operator has to check the possibility of cutting off the upper and lower parts (Al) of the fuel elements since the reprocessing price depends on kg (U + Al). Then there have to be at the facility cutting tools, special handling tools and storage possibilities for the Al-waste.

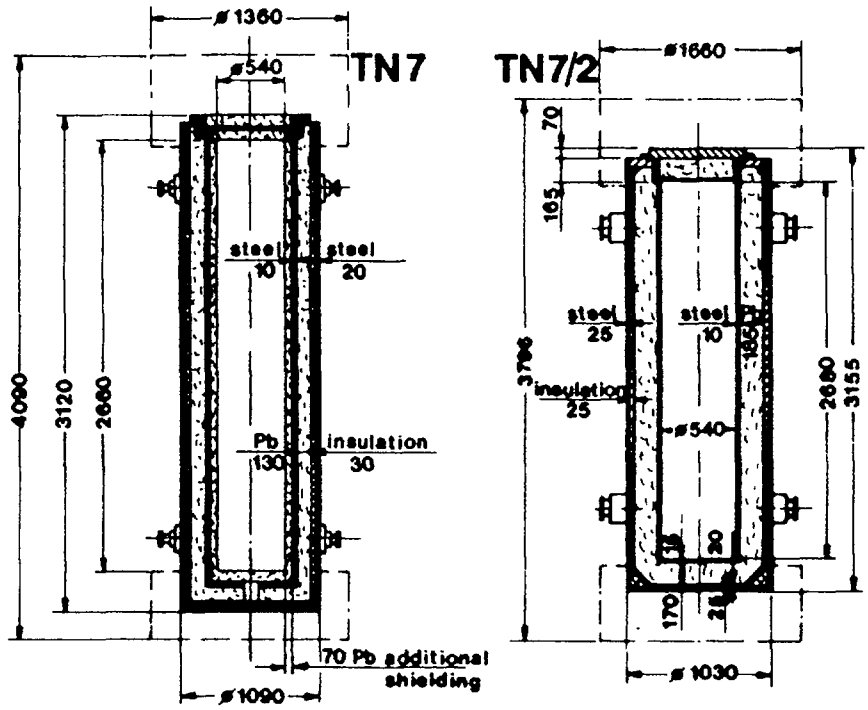
Appendix N-2.2

TRANSNUKLEAR SPENT FUEL SHIPPING CONTAINERS

TRANSNUKLEAR GmbH
Hanau, Federal Republic of Germany

Abstract

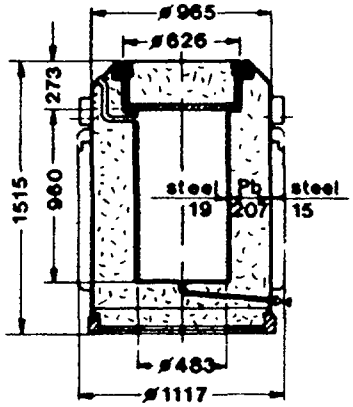
Detailed data are provided on the TN-7, TN-7/2, Goslar, and TN-1 spent fuel shipping containers for MTR-type fuel elements.



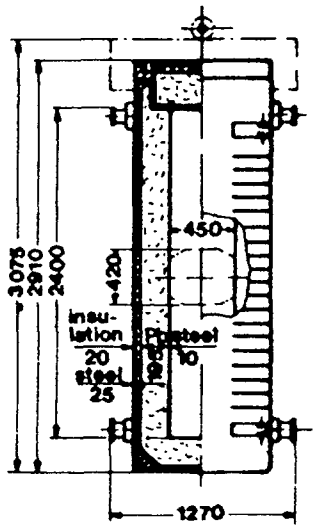
	TN-7		TN-7/2	
max. laden weight (tonnes)	25		22	
weight during handling (tonnes)	24		20	
interior dimensions (mm)	ø540 x 2650		ø540 x 2680	
authorisation	B (U)		B (U)	
package approval no. (PTB)	D/4001/B(U)F		(-)	
nuclear fissile class	II		II	
category	III - yellow		III - yellow	
no. of containers available	1		1 since mid. 1982	
no. of inner baskets per container	4		4	
method of loading/unloading	under water		under water	
MTR fuel elements:	DIDO	MERLIN	DIDO	MERLIN
- capacity per container	60 cut	64 cut	60 cut	64 cut
- element dimensions (mm)	ø95 x 632	80x86x658	ø95x632	80 x 86 x658
- orig. U-235 cont.per element (g)	185 (-)	270 (-)	185 (-)	270 (-)
- max. orig. enrichment (%)	93 (20)	93 (20)	93 (20)	93 (20)
- max. decay heat per container (kW)	4.5	4.5	4.5	4.5

MTR container details

Goslar



TN 1



	Goslar	TN-1
max. laden weight (tonnes)	11	18
weight during handling (tonnes)	10.2	17.5
interior dimensions (mm)	ø 483 x 960	min. ø 420 x 2400
authorisation	B(M)	B(U)
package approval no. (PTB)	D/4053/B(M)F	D/4004/B(U)F
nuclear fissile class	II	II
category	III - yellow	II - yellow or III - yellow
no. of containers available	2	1
no. of inner baskets per container	1	3
method of loading/unloading	under water	under water
MTR fuel elements:		
- capacity per container	13	42 cut
- element dimensions (mm)	80 x 86 x 950	79 x 86 x 670
- orig. U-235 cont. per element (g)	320 -	400 (-)
- max. orig. enrichment (%)	93 (20)	93 (20)
- max. decay heat per container (kW)	3.2	6.4

MTR container details

Appendix N-2.3

UKAEA'S 'UNIFETCH' IRRADIATED FUEL TRANSPORT CONTAINERS

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Harwell, Didcot, Oxfordshire,
United Kingdom

Abstract

The UKAEA "UNIFETCH" containers for transport of irradiated fuel from research reactors are described.

In the UK, irradiated fuel elements from research reactors are transported by road using the UNIFETCH containers.

These containers are finned cylindrical steel containers, of which there are two versions, the UNIFETCH 'H', type No. GB/1112A, suitable for cropped MTR type elements and the longer UNIFETCH 'L', type No. GB/1113. The type 'H' has a 26 element basket, while for the type 'L' there are alternative 24 and 40 element baskets.

The containers are loaded and unloaded under water, but the elements are transported in a dry state.

These containers were originally approved under the 1967 IAEA regulations as a Large Source Package design, but have in 1985 been reapproved as type B(M)F fissile Class II containers.

The attached illustrations and data sheets give the size, weight and handling details.

'Unifetch' Type 'H' Transport Container

Design No. 1112

Reference Drawing No. DH 1767

General Description
Gas tight (test pressure 150 p.s.i.g. hydraulic) vertical cylindrical finned M.S. container with removable lid. Primarily intended for the transport of irradiated M.T.R. type fuel elements, using inner container Design No. 1423.

No. of Flasks
1.

Unladen Weight
15 tons 3 cwts. 3 qtrs. (without inner basket. Design No. 1423).

Materials (shielding)
12½" mild steel.

Cavity size or capacity
2'–6" dia. x 2'–5½"

Safe Loading of Lifting Points
22.5 tons.

Approved Lifting Harness Drawing No.
EH 1767/005 (Lifting frame) and EH 1767/001 (Lifting ear).
Both lift flask complete with hold-down equipment. Total weight 18.52 tons.

Max. Loading of Harness
Lifting frame – 25 tons.
Lifting ear – 25 tons.

Lifting Harness Plant Item No.

Vehicle
Any suitable and approved vehicle also transported by rail and sea.

Approved Hold Down Equipment Drawing No.
By road – rail and sea – EH 1767/003.

Speed Restrictions
5 M.P.H. on site

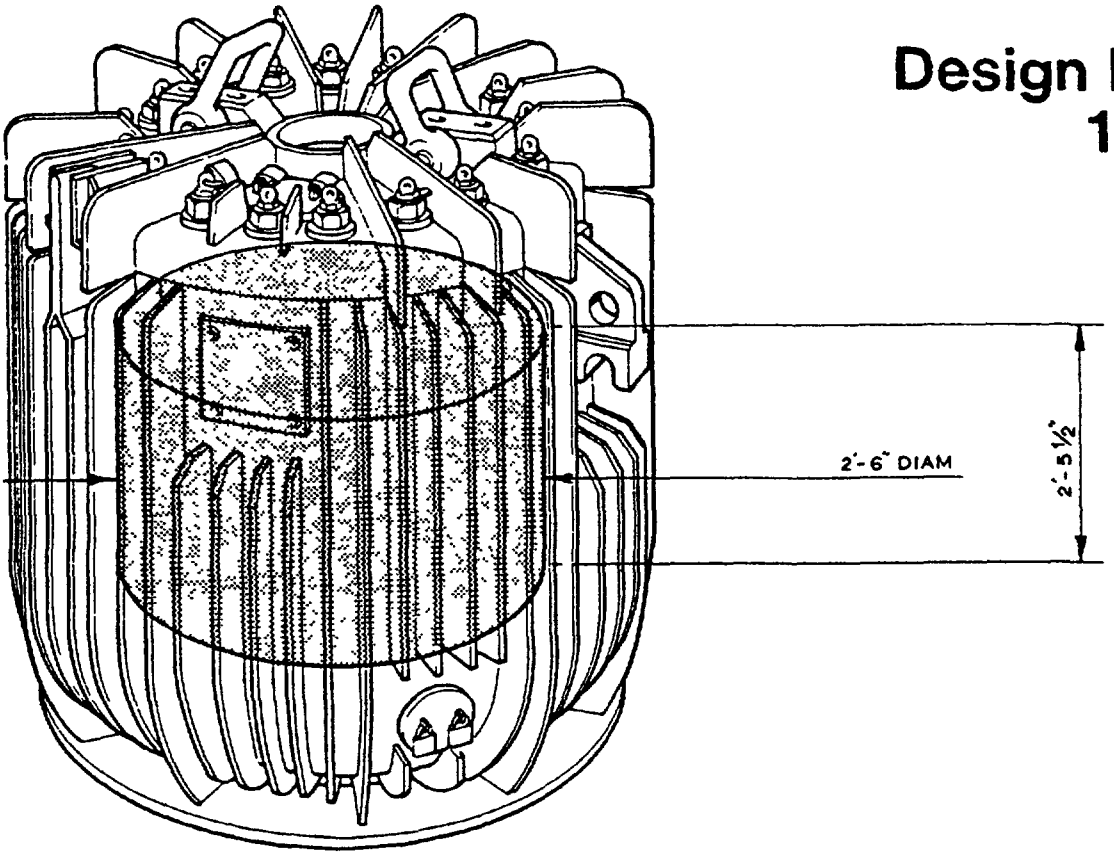
Normal Storage
Flask storage compound

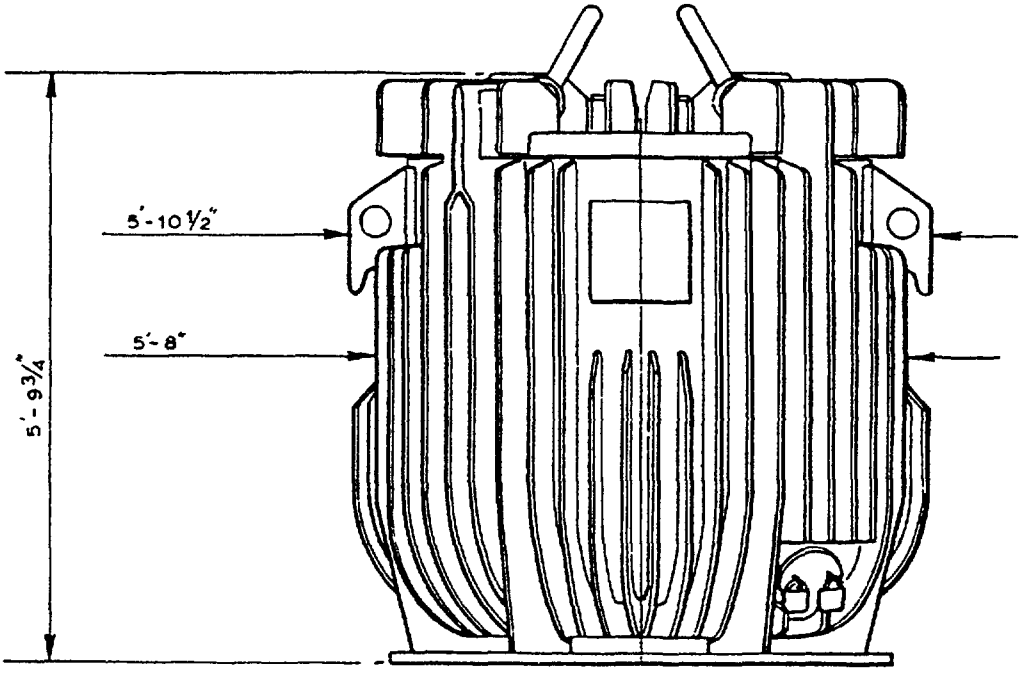
Routes
Primarily intended for the international traffic of M.T.R. type fuel elements.

Normal Usage
Transport of M.T.R. fuel elements from various sites in U K. and abroad to D.N.P.D.E.

Ancillary Equipment
Inner container, design No. 1432, drawing No. EH 1767/8

Design No. 1112





'Unifetch' Type 'L' Transport Container

Design No. 1113

Reference Drawing No. AE 231563

General Description

Gas tight (test pressure 150 p.s.i.g. hydraulic) vertical cylindrical finned M.S. container with removable lid. Primarily intended for the transport of irradiated M.T.R. type fuel elements, using inner container Design Nos. 1331, 1376, and 1753.

No. of Flasks

2.

Unladen Weight

16.8 tons (without inner basket).

Materials (shielding)

12½" mild steel.

Cavity size or capacity

2'–6" dia. x 3'–5" long (approx.).

Safe Loading of Lifting Points

22.5 tons.

Approved Lifting Harness Drawing No.

(a) AE 231580 (this harness lifts flask complete with hold-down equipment when being trans-shipped).
Total weight 21.1 tons (approx.).
(b) AE 231585 (lid removal).

Max. Loading of Harness

(a) 25 tons.
(b) 8 tons (lid removal).

Lifting Harness Plant Item No.

Vehicle

Off site - Any suitable and approved vehicle.
— also transported by rail and sea.
25 ton 'Carrimore' trailer (on site only).

Approved Hold Down Equipment Drawing No.

By road — rail and sea — EH 1767/003.
25 ton 'Carrimore' trailer (on site only) ZAE 61075.

Speed Restrictions

5 M.P.H. on site.

Normal Storage

D.E.R.E. flask storage area.

Routes

Primarily intended for the international traffic of M.T.R. type fuel elements.

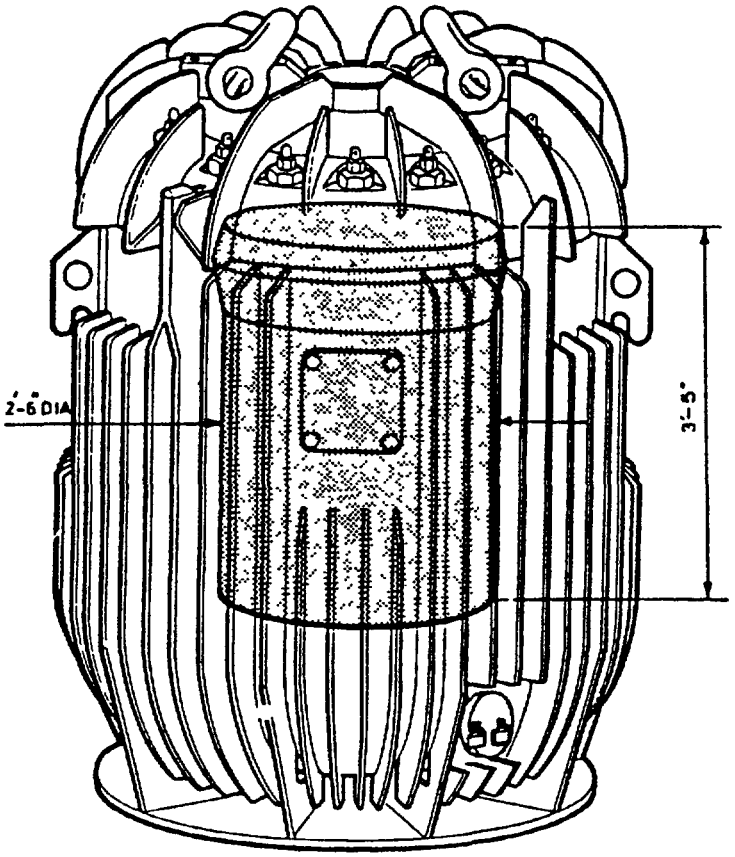
Normal Usage

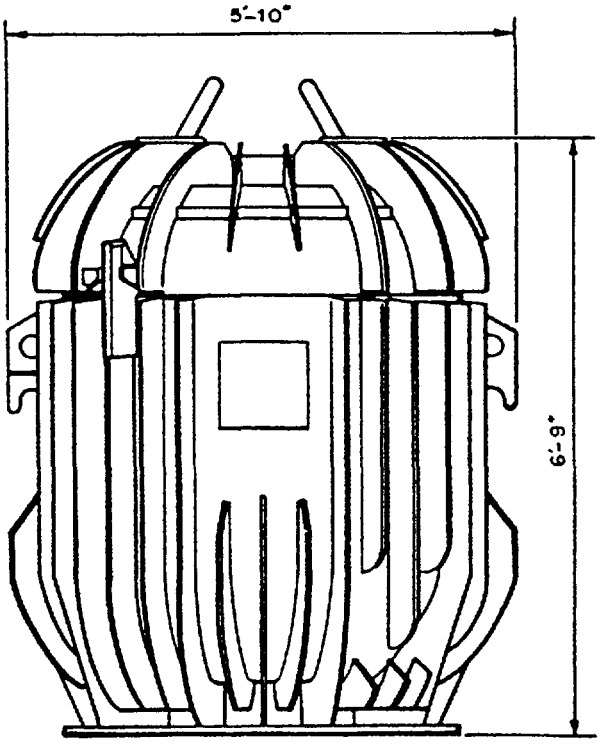
1. Transport of M.T.R. fuel elements between various sites in U.K. and abroad to D.N.P D.E.
2. Transport of F.R. breeder slugs from D 1206 to Windscale.

Ancillary Equipment

Inner container, design No. 1331 — Drawing No. ZAE 60705.
Inner container, design No. 1376 — Drawing No. AE 231573.
Inner container, design No. 1753 — Drawing No. ZAE 61218

Design No. 1113





Appendix N-2.4

THE TRANSPORT OF SPENT FUEL ELEMENTS OF RESEARCH REACTORS

COMMISSARIAT A L'ENERGIE ATOMIQUE
Institut de recherche technologique
et de développement industriel,
Division d'exploitation des réacteurs prototypes
et expérimentaux,
Service des piles de Saclay,
Centre d'études nucléaires de Saclay,
Gif-sur-Yvette, France

Abstract

The COGEMA 'IU 04' cask, frequently called the 'Pégase' cask, and some of its available internal containers are briefly described.

Spent fuel elements of research reactors are transported between the different French research centers, or between one of these centers and the reprocessing plants handling these fuels in Belgium, in France and finally in the USA, in a single model of transfer cask, of which many copies have been built. This cask is known as the 'IU 04', frequently called 'Pégase' cask, from the name of the first reactor for which it was commissioned.

Only the cask's internal arrangements or basket varies from one reactor to another.

The 'IU 04' casks are owned by COGEMA, which maintains them and makes them available to users.

Container AA49 for Osiris fuel elements, consisting of five 72° center-angle segments, capable of accommodating thirty elements.

Container AA50 for Siloë, Triton, Mélusine fuel elements etc., consisting of six 60° center-angle segments, capable of accommodating 36 elements.

Container AA77 for the EL3 reactor cells: 120 cavities per container.

Container AA117 for an RHF element of the Grenoble ILL.

This list is not exhaustive.

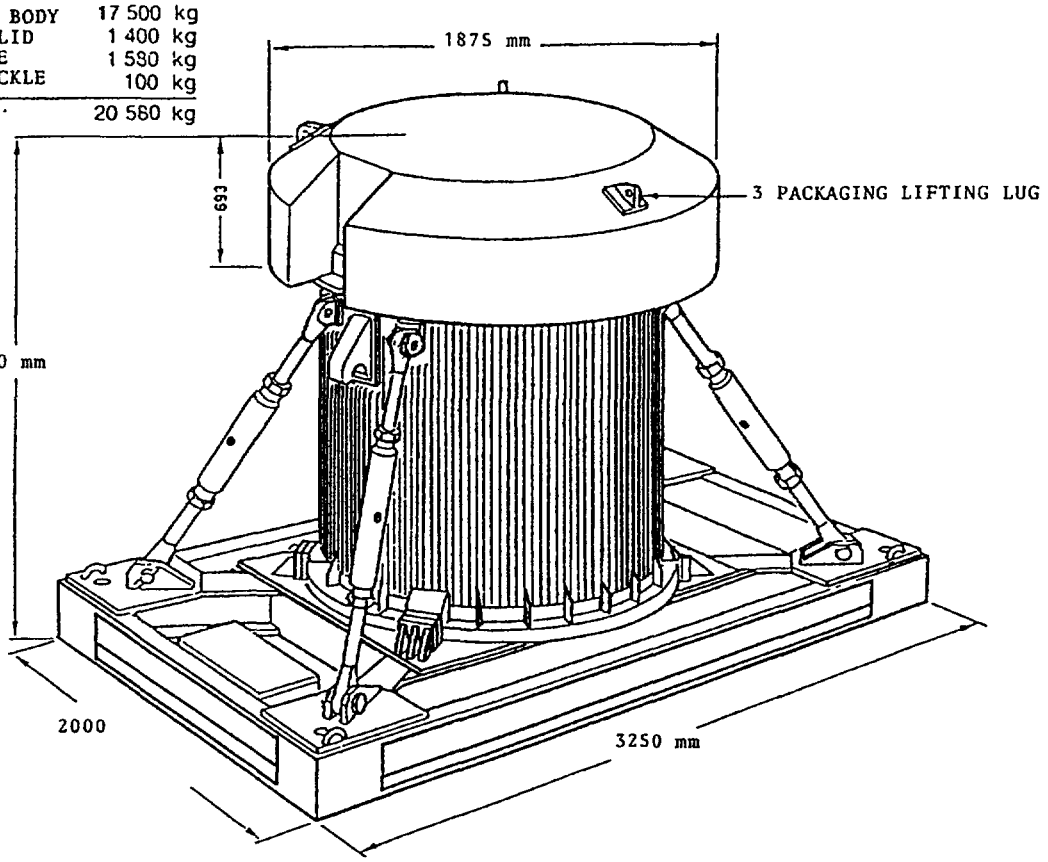
The weight of the assembly equipped for transport and loaded with fuel elements varies slightly with loading. The orders or magnitude are as follows:

- cask: 17,500 kg,
- transport equipment (cover and frame): 3100 kg,
- internal container and fuel element load: 3000 kg (2500 to 3500 kg).

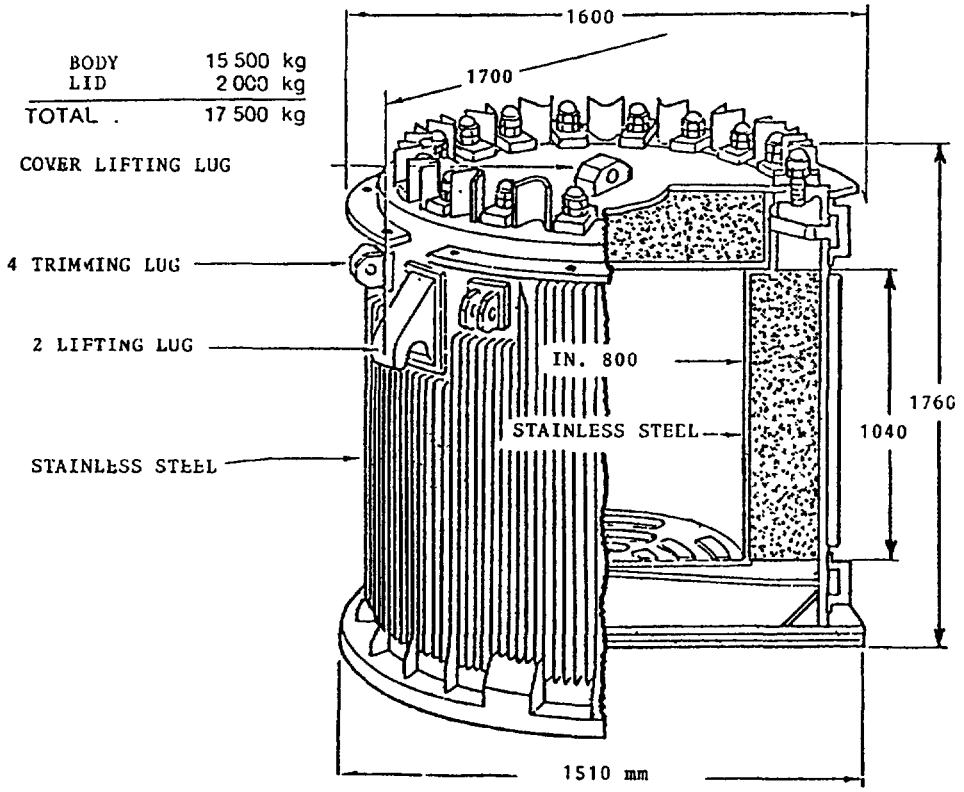
This makes a total weight of about 23,600 kg, and possibly as high as 24,000 kg.

WEIGHTS

MAIN BODY	17 500 kg
COVER LID	1 400 kg
FRAME	1 580 kg
TURNBUCKLE	100 kg
TOTAL	20 580 kg



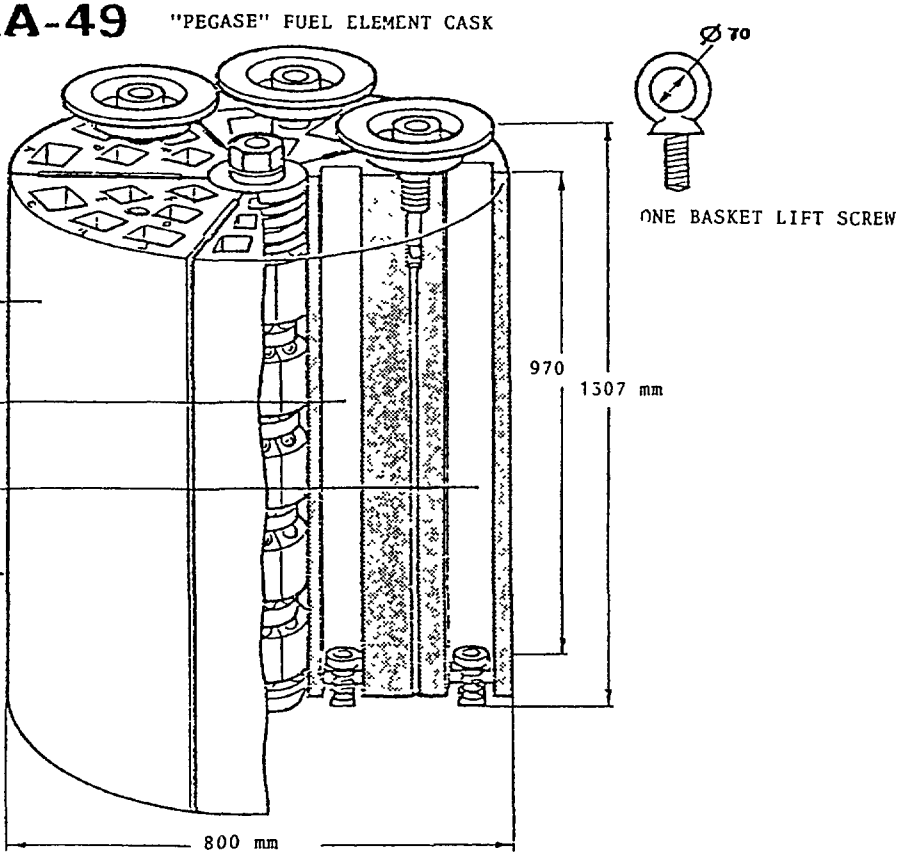
BODY	15 500 kg
LID	2 000 kg
TOTAL	17 500 kg



IU-04

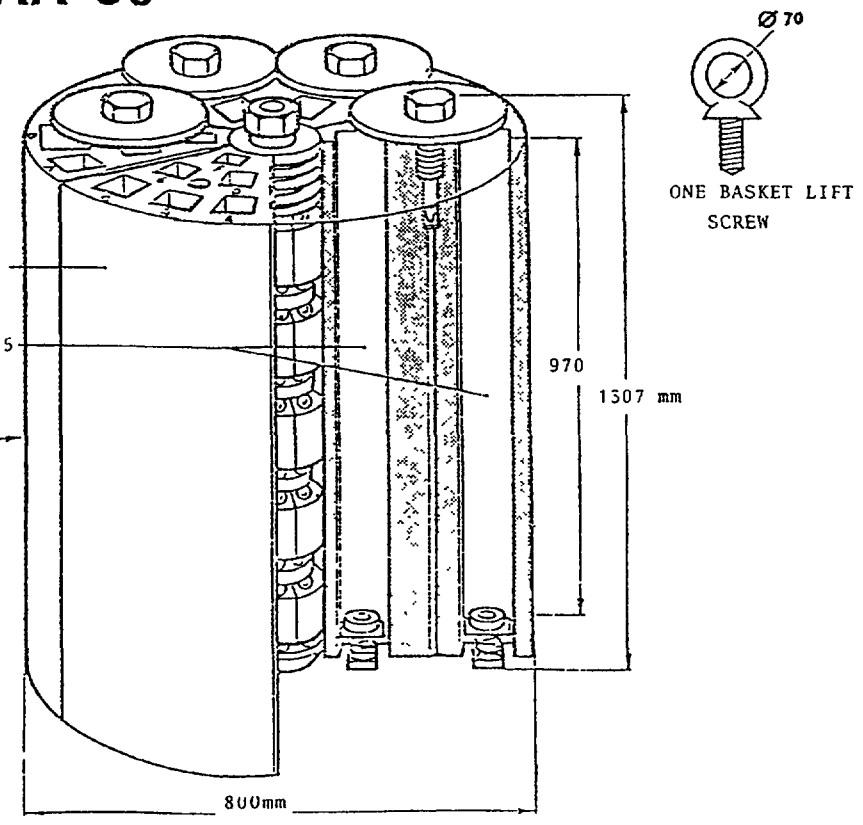
AA-49 "PEGASE" FUEL ELEMENT CASK

WEIGHTS
ONE BASKET : 480 Kg
CENTRAL BODY : 100 Kg
TOTAL : 2500 Kg



AA-50 "SILOE" FUEL ELEMENT CASK

WEIGHTS
ONE BASKET : 316 kg
CENTRAL BODY: 100 Kg
TOTAL : 1996 Kg



Appendix N-2.5

TRANSPORTATION OF IRRADIATED TRIGA-LEU FUEL

GA TECHNOLOGIES, INC.,
San Diego, California,
United States of America

Abstract

Containers for shipping irradiated TRIGA-LEU fuel in the United States and in Europe are described.

Shipments in the United States of irradiated 1.5 in. nominal O.D. TRIGA fuel have been made in the BMI-1 shipping cask currently owned by Cintichem, Inc. The fuel shipped in the cask has come from reactors which operated at power levels up to 2 MW and the container carried up to 38 elements. A description of the cask is as follows.

Model No: BMI-1.

Description: Steel-encased lead shielded shipping cask. The basic cask body is a cylinder 33.37 in. in diameter by 73.37 in. high formed by two concentric stainless steel shells whose annular region is filled with lead. The outer 1/2-in. thick shell has a 0.12-in. thick plate spot welded to it, providing a 0.06-in. thick air gap insulator. The inner shell is 15.5 in. inside diameter by 54 in. inside length. The cask lid is a stainless steel weldment having 7.75 in. of lead shielding. The cask lid is secured to the cask by twelve steel studs which are welded to the cask body. Cask appurtenances include a drain line with needle valve and plug, pressure gauge, and a pressure relief valve. The total cask weight, including maximum contents of 1,800 lbs, is 23,660 lbs.

Shipments in Europe of irradiated 1.5 in. nominal O.D. TRIGA fuel have been made by Transnuklear in the Goslar and TN6/3 type containers. The Goslar container weighs 10 tons and has a capacity of 60 fuel elements. The TN6/3 container weighs 6.3 tons and has six positions for special elements such as fuel-followed control rods and temperature instrumented fuel elements. The fuel shipped in these containers has come from reactors which operated at power levels up to 1 MW.

Appendix N-3

SPENT FUEL STORAGE

Appendix N-3.1

NUCLEAR CRITICALITY ASSESSMENT OF LEU AND HEU FUEL ELEMENT STORAGE

R.B. POND, J.E. MATOS
Argonne National Laboratory,
Argonne, Illinois,
United States of America

Abstract

Criticality aspects of storing LEU (20%) and HEU (93%) fuel elements have been evaluated as a function of ^{235}U loading, element geometry, and fuel type. Silicide, oxide, and aluminide fuel types have been evaluated ranging in ^{235}U loading from 180 to 620 g per element and from 16 to 23 plates per element. Storage geometry considerations have been evaluated for fuel element separations ranging from closely packed formations to spacings of several centimeters between elements. Data are presented in a form in which interpolations may be made to estimate the eigenvalue of any fuel element storage configuration that is within the range of the data.

INTRODUCTION

Criticality aspects of storing fuel elements is of concern to all reactor operators. Any change to the types of fuel elements approved for storage may require that the subcriticality of a storage rack be reconfirmed. As an insight into what might be expected, this report presents results of a study assessing the storage of HEU and LEU fuel elements with various fissile contents.

SCOPE

Fuel Element Storage Model

In assessing fuel element storage, the type of fuel element and the storage configuration must be defined. For purposes of this report, twenty fuel element types were considered and a generic storage rack was used.

The storage rack is defined as an unpoisoned aluminum framework within which partitions form individual fuel element storage compartments. The entire unit is immersed in a pool so that the fuel elements are moderated and reflected with water. An infinite-by-infinite array of fuel elements is assumed and the separation between storage compartments is adjusted to control the storage rack reactivity. On this basis, reactivity effects associated with various types of fuel elements in various storage rack configurations can be made. (The bases for this fuel element storage model are developed in Appendix A.)

Calculation Model

Three-dimensional (XYZ) diffusion theory is used in the calculations in this report. In many of the calculations a simplified representation of the storage rack and the fuel elements are used. These simplifications included neglect of the aluminum storage compartments and the fuel element end-fittings. Sensitivity studies to assess the reactivity effects of these simplifications were, however, made and validation of the diffusion theory calculations were made using Monte Carlo techniques.

CROSS SECTIONS

Microscopic cross sections for the fuel elements and the storage rack were calculated using the EPRI-CELL code[1] with ENDF/B-IV cross section data. The integral transport calculations in EPRI-CELL are performed for 69-fast groups and 35-thermal groups (<1.855 eV), and then collapsed to 5-broad groups with upper energy boundaries of 10 MeV, 0.821 MeV, 5.53 keV, 1.855 eV, and 0.625 eV. The fuel element geometry and the unit-cell models used in the EPRI-CELL calculations are shown in Fig. 1.

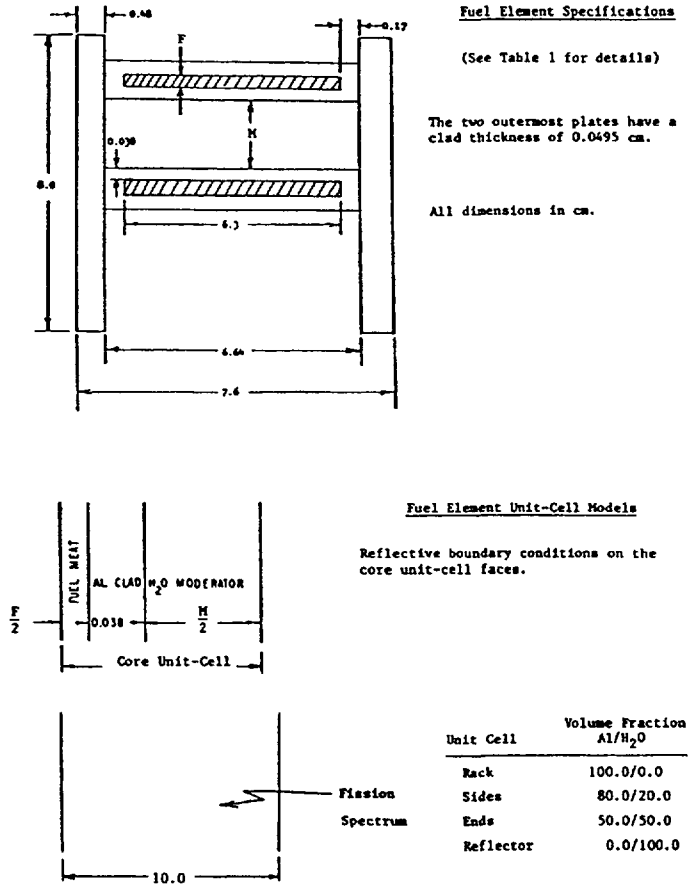


Fig. 1. Fuel Element Specifications and Unit-Cell Models.

Broad group cross sections were generated for each fuel element type using the flux spectrum of the core portion of the fuel element. The core portion of a fuel element included the fuel, clad and water channel regions as shown in Fig. 1.

Separate microscopic cross sections for the fuel element sideplates were generated using a pure ^{235}U fission spectrum on a 80/20 volume percent mixture of aluminum and water. The sideplate portion of a fuel element included the portion of clad on the fuel plates between the fuel meat and sideplates plus the same corresponding part of water in the water channels. Macroscopic cross sections appropriate for the sideplates of each fuel element type were used in the neutronic calculations.

The same methodology as used for the sideplates was used in generating cross sections for the fuel element end-fittings, the aluminum rack and the storage rack water reflector.

FUEL ELEMENT DESCRIPTION

Thirteen LEU (20% enriched) and seven HEU (93% enriched) fuel element types are used in this study covering a wide range of fuel densities, fuel types, and fuel element geometries. The choice of fuel element types are made based upon types currently in use in plate-type research and test reactors, and types which might be expected to be available as fuel material technology develops. The fuel element geometries considered contain between 16 and 23 plates per element.

Detailed specifications of the twenty fuel element types are listed in Table 1. The range of fuel densities, fuel types, and fuel element geometries are summarized below.

Fuel Type	Plates per Element	Fuel Meat Thickness, mm	Uranium Density, g/cc	^{235}U Loading, g
LEU U_3Si-Al	19	0.51	3.1-5.3	225-390
	23	0.51	3.2-7.0	280-621
LEU U_3O_8-Al	23	0.51	3.1	278
	16-22	0.76	3.1	288-396
	20	1.00	3.1	473
HEU UAl_x-Al	19	0.51	0.5-1.2	180-405
	23	0.51	0.4-1.3	180-530

Table 1. Fuel Element Loadings.

Fuel Element Loading Number	Fuel Type	Plates/ Element	^{235}U/ Element, g	Uranium Density, g/cc	Fuel Meat (F) Thickness, mm	Water Channel (M) Thickness, mm	Sideplate Volume Fractions, % Al	Sideplate Volume Fractions, % H_2O
1	LEU U_3O_8-Al	23	278	3.130	0.51	2.188	81.11	18.89
2	LEU U_3O_8-Al	16	288	3.130	0.76	3.451	79.56	20.44
3	LEU U_3O_8-Al	18	324	3.130	0.76	2.899	80.56	19.44
4	LEU U_3O_8-Al	20	360	3.130	0.76	2.457	81.51	18.49
5	LEU U_3O_8-Al	22	396	3.130	0.76	2.095	82.43	17.57
6	LEU U_3O_8-Al	20	473	3.130	1.00	2.217	83.04	16.96
7	LEU U_3Si-Al	19	225	3.071	0.51	2.916	79.49	20.51
8	LEU U_3Si-Al	19	350	4.778	0.51	2.916	79.49	20.51
9	LEU U_3Si-Al	19	390	5.324	0.51	2.916	79.49	20.51
10	LEU U_3Si-Al	23	280	3.157	0.51	2.188	81.11	18.89
11	LEU U_3Si-Al	23	320	3.609	0.51	2.188	81.11	18.89
12	LEU U_3Si-Al	23	390	4.398	0.51	2.188	81.11	18.89
13	LEU U_3Si-Al	23	621	7.000	0.51	2.188	81.11	18.89
14	HEU UAl_x-Al	19	180	0.528	0.51	2.916	79.49	20.51
15	HEU UAl_x-Al	19	280	0.822	0.51	2.916	79.49	20.51
16	HEU UAl_x-Al	19	405	1.189	0.51	2.916	79.49	20.51
17	HEU UAl_x-Al	23	180	0.437	0.51	2.188	81.11	18.89
18	HEU UAl_x-Al	23	280	0.679	0.51	2.188	81.11	18.89
19	HEU UAl_x-Al	23	405	0.982	0.51	2.188	81.11	18.89
20	HEU UAl_x-Al	23	530	1.285	0.51	2.188	81.11	18.89

All fuel elements are assumed to be fresh in accordance with standard practice for this type of criticality assessment. The presence of any burnable poison which might be required in many of the heavier loaded fuel elements is also neglected. These assumptions about fuel element poisoning effects are made in order that all calculated reactivities for the storage rack configurations will be conservative. The fuel elements are assumed to be 68 cm long with a 60 cm active fuel height and 4 cm above and below the fuel to simulate fuel element end-fittings.

STORAGE RACK CALCULATIONS

The first part of this section examines the reactivity trends that one fuel element type will have in various storage rack configurations. In the second part, the reactivity trends that one storage configuration will have with each of the twenty fuel element types are examined.

Overall, the results of this section are intended to provide the means of estimating eigenvalues for various fuel element types in various storage rack configurations. Based upon these data, a reactor operator will have a basis upon which to estimate the reactivity effect of substituting one fuel type for another in unpoisoned storage racks. (Examples illustrating the use of these data are provided in Appendix B.)

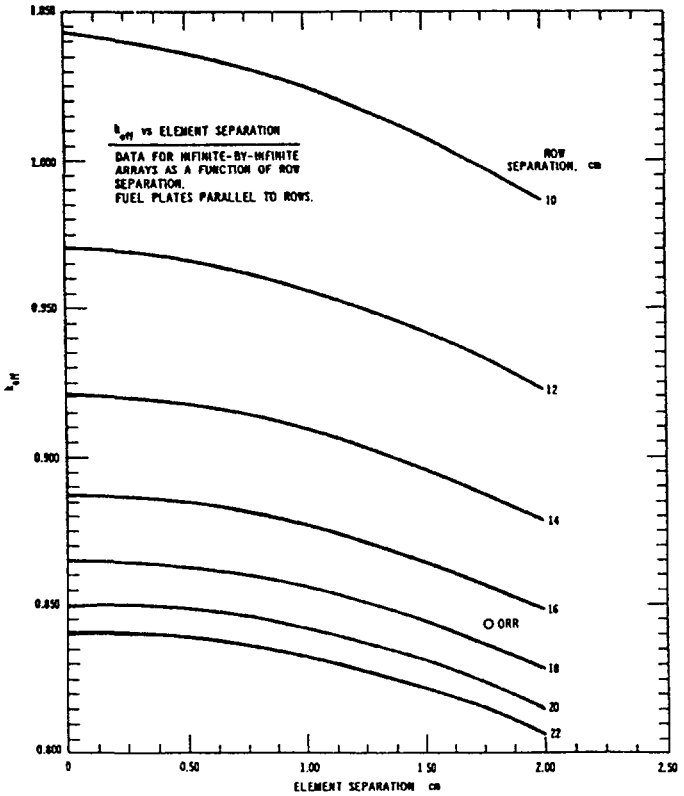
The storage rack model used in the calculations assumed an infinite-by-in-finite array of fuel elements in which there are an infinite number of fuel elements in a row and an infinite number of rows. The spacing between fuel elements in a row and the separation between rows are specified to define the storage rack configuration. The calculational models used in this study are shown in Appendix C.

Eigenvalues for a Given Fuel Element Type in Various Storage Rack Configurations

Table 2. Eigenvalue Calculations for an Infinite-by-Infinite Array of Number 13 Fuel Elements (LEU U_3Si 621 g ^{235}U) as a Function of Element Separation in a Row and Row Separation. (See Fig. D in Appendix C.)

Element Separation in a Row, cm	Row Separation, cm						
	10	12	14	16	18	20	22
0.0	1.0419	0.9704	0.9206	0.8868	0.8643	0.8494	0.8398
0.25	1.0398	0.9691	0.9198	0.8865	0.8642	0.8496	0.8401
0.50	1.0360	0.9662	0.9176	0.8847	0.8628	0.8483	0.8390
0.75	1.0309	0.9620	0.9141	0.8816	0.8600	0.8458	0.8365
1.00	1.0245	0.9565	0.9092	0.8772	0.8559	0.8419	0.8327
1.50	1.0078	0.9417	0.8958	0.8647	0.8441	0.8305	0.8216
2.00	0.9868	0.9226	0.8782	0.8481	0.8281	0.8149	0.8064

Eigenvalue results are shown in Table 2 for an array of LEU U_3Si 621 g ^{235}U fuel elements with row separations of 10 to 22 cm, and element separations in a row of 0.0 to 2.0 cm. These data are plotted in Fig. 2 as a function of element separation for various row separations. In these calculations, both the aluminum of the storage rack and the fuel element end-fittings were neglected.



The results in Fig. 2 show the relative reactivity effects for this fuel element type as a function of various storage rack configurations. Interpolations to determine the eigenvalue for any specific configuration can also be readily made.

The Oak Ridge Research (ORR) reactor storage rack[2], for example, has a row separation of 17.2 cm and an element separation of 1.77 cm. According to Fig. 2 the eigenvalue for this configuration with the LEU U_3Si 621 g ^{235}U fuel elements would be 0.8431. This (calculated) eigenvalue is plotted in Fig. 2 and is identified "ORR".

Fig. 2. Eigenvalues for Various Row Separations in Infinite-by-Infinite Arrays of LEU U_3Si 621 g ^{235}U Fuel Elements as a Function of the Separation Between Elements.

Eigenvalues for a Given Storage Rack Configuration with Various Fuel Element Types

Table 3. Eigenvalue Calculations for Twenty Fuel Elements Loadings, Each in an Infinite-by-Infinite Array and Assuming the ORR Fuel Storage Rack Spacing Specifications of 1.766 cm Element Separation and 17.24 cm Row Separation. (See Fig. E in Appendix C.)

Fuel Element Loading Number	Fuel Type	Plates/ Element	^{235}U/ Element, g	k_{eff} Eigenvalue
1	LEU U_3O_8	23	278	0.7410
2	LEU U_3O_8	16	288	0.7587
3	LEU U_3O_8	18	324	0.7695
4	LEU U_3O_8	20	360	0.7765
5	LEU U_3O_8	22	396	0.7803
6	LEU U_3O_8	20	473	0.7967
7	LEU U_3Si	19	225	0.7154
8	LEU U_3Si	19	350	0.7889
9	LEU U_3Si	19	390	0.8044
10	LEU U_3Si	23	280	0.7402
11	LEU U_3Si	23	320	0.7613
12	LEU U_3Si	23	390	0.7899
13	LEU U_3Si	23	621	0.8453
14	HEU UAl_x	19	180	0.6903
15	HEU UAl_x	19	280	0.7772
16	HEU UAl_x	19	405	0.8377
17	HEU UAl_x	23	180	0.6783
18	HEU UAl_x	23	280	0.7635
19	HEU UAl_x	23	405	0.8229
20	HEU UAl_x	23	530	0.8594

Eigenvalue results are shown in Table 3 for infinite arrays of the twenty fuel element types in a storage configuration having the fuel element spacing specifications of the ORR storage rack. These data are plotted in Fig. 3 as a function of the ^{235}U loading in each fuel element type. In these calculations, 1/8 in. (0.32 cm)-thick aluminum storage compartments were included and the fuel element end-fittings were neglected.

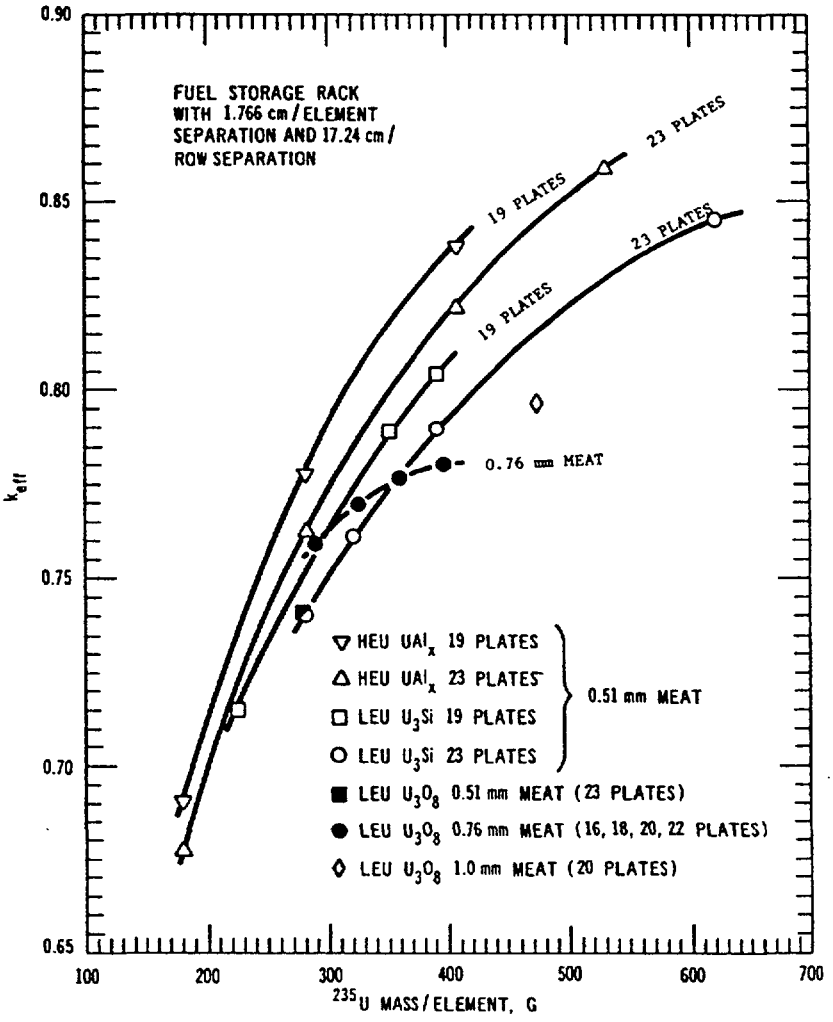


Fig. 3. Eigenvalues for Various LEU and HEU Elements in Infinite-by-Infinite Arrays With Separations of 1.766 cm Between Elements and 17.24 cm Between Rows as a Function of the ^{235}U Fuel Element Loading.

The results in Fig. 3 show the relative reactivity effects of fuel element storage in this configuration as a function of: (1) the number of plates per element, (2) the fuel meat thickness in an element, and (3) LEU vs. HEU fuel types. Interpolations to determine eigenvalues for other ^{235}U fuel element loadings can be readily made.

The data in Fig. 3 indicate that reactivity effects due to the fuel element geometry can be characterized as a function of the H/^{235}U atom ratio of the fuel element. For these fuel element geometries, the eigenvalues are inversely proportional to the number of plates per element and inversely proportional to the fuel meat thickness.

As would be expected, LEU fuel is less reactive than HEU fuel for a given fuel element geometry and ^{235}U loading. It is also evident that the eigenvalue results are not sensitive to the form of LEU fuel since the LEU U_3Si and LEU U_3O_8 results for 23-plate elements with 0.51 mm fuel meat are almost identical.

SENSITIVITY STUDIES

Configuration Model

The sensitivity of eigenvalue calculations to effects of the storage rack compartments, fuel element end-fittings, fuel element sideplates, and the diffusion theory model mesh have been evaluated. These results are listed in Table 4 for four storage rack configurations.

Table 4. Eigenvalue Sensitivity of an Infinite-by-Infinite Array of Number 13 Fuel Elements (LEU U_3Si 621 g ^{235}U) as a Function of Fuel Element and Storage Rack Representation, and the Calculational Model.

Configuration	k_{eff} Eigenvalue Assuming the ORR Fuel Storage Rack Geometry		k_{eff} Eigenvalue Assuming		Conclusion
	17.24 cm/Row 1.766 cm/ Element	17.64 cm/Row 1.366 cm/ Element	14 cm/Row 0.25 cm/ Element	14 cm/Row 0.75 cm/ Element	
1. As shown in Figs. G, H, I, and J in Appendix C	0.8453	0.8366	0.9185	-	-
2. With Al replaced by H_2O in the 1/8 in.-thick Al frame region	0.8463	-	-	0.9169	Effect of frame is small
3. Without the Al frame region[a]	0.8431	0.8332	0.9198	0.9141	Effect of mesh is small
4. With an assumed 4 cm-long end-fitting (45.53/55.47-Al/H_2O)	0.8450	-	0.9182	0.9181	Effect of ends is very small
5. Fuel only - no sides, no frame, no ends - all replaced by H_2O	0.8558	-	0.9536	0.9391	Effect of sides is significant
6. Fuel and ends only - no sides, no frame - all replaced by H_2O	-	-	-	0.9392	Effect of ends is very small
7. Fuel and frame only - no sides, no ends - all replaced by H_2O	-	-	-	0.9436	Effect of frame is small

[a]This is the same as #2 but with a reduced number of X- and Y-mesh points in the water reflector. Without the Al frame region, the Y-mesh water reflector boundaries were, respectively: 12.62(9), 12.62(9), 11.0(7), and 11.0(7).

The following table summarizes the reactivity effects listed in Table 4 for one of these storage configurations; in general, all configurations show the same reactivity trends. The data are for the ORR storage rack configuration with LEU U_3Si 621 g ^{235}U fuel elements.

Change in Configuration	Reactivity Effect, % δk/k
Include fuel element end-fittings	-0.03
Include storage rack compartments	-0.10
Nominal (~10%) increase in mesh points*	+0.32

These reactivity effects indicate that the diffusion theory eigenvalue uncertainties of the storage rack configuration models are less than 1% δk/k.

Infinite Array Versus Finite Array of Fuel Elements

As an example of the conservatism implied by assuming an infinite-by-infinite array of fuel elements in a storage configuration vs. a finite array, the eigenvalue for the LEU U_3Si 621 g ^{235}U fuel elements in the ORR storage rack was calculated. A plan view of the ORR storage rack is shown in Fig. 4.

*Mesh sensitivities shown in the validation studies section indicates that the maximum mesh reactivity effect is about 0.9% δk/k for a 100% increase in the number of mesh points. Further increases in the number of mesh points do not show a substantial additional reactivity increase.

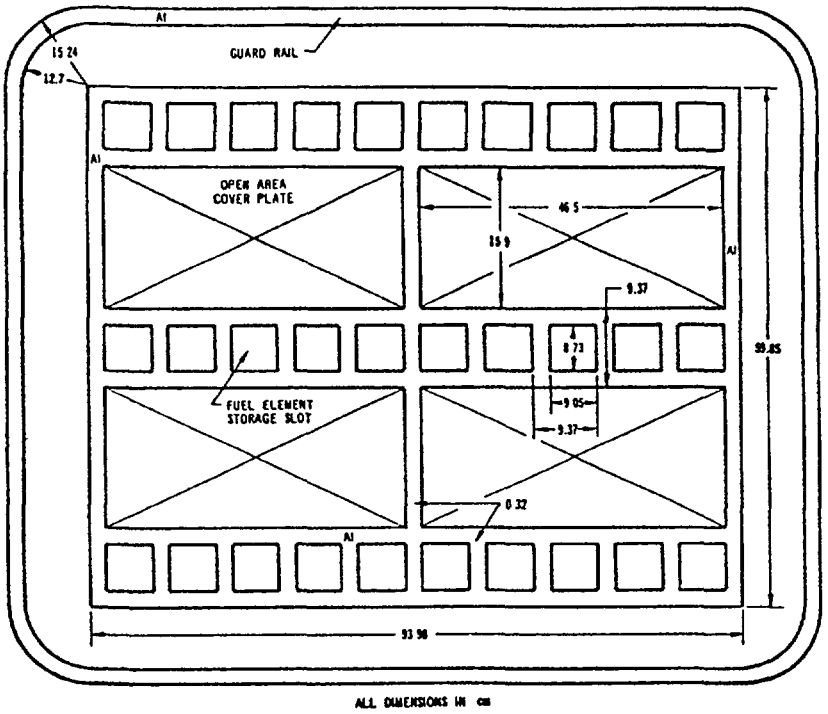


Fig. 4. The ORR Fuel Storage Rack Configuration.

The rack has three rows of storage compartments and ten compartments per row. When fuel elements are centered in the compartments with the fuel element plates parallel to the rows, the spacing between elements in a row is 1.77 cm and the separation between rows is 17.2 cm.

The calculated eigenvalue for this configuration is 0.7985. This eigenvalue compares with a k_{eff} of 0.8453 (see Table 4) for an infinite-by-infinite array of fuel elements. The reactivity difference is about 5% δk/k. Calculations performed using infinite arrays are, therefore, clearly conservative.

VALIDATION STUDIES

Because of the importance and often the necessity of relying upon calculations to determine safe fuel element storage configurations, some of the diffusion theory eigenvalues were compared with results using Monte Carlo techniques. These data are shown in Table 5. As shown in Table 5, calculations were performed for a critical configuration,[3] for an infinite row of fuel elements, and for an infinite-by-infinite array of fuel elements. The diffusion theory code used was DIF3D[4], and the Monte Carlo codes were VIM[5] and KENO[6].

Table 5. Validation of Calculational Methods.

Configuration	Diffusion Theory	Monte Carlo		
	DIF3D	VIM	KENO	
	ENDF/B-IV	ENDF/B-IV	ENDF/B-IV	Hansen Roach
1. Critical SPERT-D HEU UAl_x 306 g ^{235}U	0.9999	-	1.0217±0.0039[a]	0.9997±0.0048
2. One infinite row LEU U_3Si 621 g ^{235}U,				
0.25 cm/element, 8 cm reflector	0.8036[b]	0.8244±0.0049	0.8353±0.0052	0.8327±0.0052
1.77 cm/element, 8.62 cm reflector	0.7860	0.8113±0.0049	0.8144±0.0030	-
3. Infinite-by-infinite LEU U_3Si 621 g ^{235}U,				
1.77 cm/element, 17.2 cm/row	0.8431[c]	0.8642±0.0066	0.8617±0.0038	0.8899±0.0051

[a]Data are for 54K histories. Five batches of 30K histories each were: 1.0255±0.0055, 1.0249±0.0052, 1.0229±0.0046, 1.0096±0.0056 and 1.0274±0.0056.

[b]Doubling the xy mesh gives a k_{eff} of 0.8092. Mesh effect is of the order of 0.6% δk/k.

[c]Doubling the xy mesh gives a k_{eff} of 0.8521. Mesh effect is of the order of 0.9% δk/k.

In general, the eigenvalues calculated with Monte Carlo are systematically larger than with diffusion theory. The uncertainties quoted for the Monte Carlo results are $\pm 1\sigma$ and in most cases, at least two standard deviations would be required to cover the diffusion theory results.

The 2 to 3% $\delta k/k$ difference between diffusion theory and Monte Carlo is somewhat accounted for by a mesh reactivity effect in diffusion theory. For the two subcritical configurations, this reactivity effect is worth between 0.6 to 0.9% $\delta k/k$. Based upon these comparisons, diffusion theory eigenvalues for this fuel element storage study could be underestimated by 1 to 2% $\delta k/k$.

RESULTS AND CONCLUSIONS

The results of this study have shown that replacement of HEU fuel elements with LEU fuel elements will not have a significant reactivity effect in most storage racks. The magnitude of any reactivity effect will depend upon the change in ^{235}U loading and differences in the fuel element geometry.

As an aid to assess fuel element storage, curves are developed (Figs. 2 and 3) for reactivity effects as functions of LEU and HEU fuel element types for various unpoisoned storage rack configurations. The curves cover LEU and HEU fuel element loadings between approximately 200 and 600 g ^{235}U per element with various fuel element geometries, and storage rack configurations with various row and fuel element separations.

Relative to HEU fuel elements, reactivity increases associated with larger ^{235}U loadings in LEU fuel elements tend to be compensated for, simply by the reduced enrichment. Increases of about 50 grams of ^{235}U per element result in no net reactivity change when the fuel element geometries are the same. Reactivity effects due to fuel element geometry differences are slowly varying functions of the number of plates per element, the fuel meat thickness, and the water channel thickness.

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APPENDIX A

STORAGE RACK CALCULATIONS AS FUNCTIONS OF FUEL ELEMENT ORIENTATIONS, WATER REFLECTOR THICKNESSES, AND FUEL ELEMENT SPACINGS

FUEL ELEMENT ORIENTATION

Infinite Row of Fuel Elements

Eigenvalue calculations were performed using three dimensional (XYZ) diffusion theory for an infinite row of fuel elements as a function of the separation between fuel elements and the element orientation in the fuel storage compartments.

Three orientations of a fuel element in a storage compartment were considered. The first was as if the fuel elements were placed with the fuel plates perpendendicular to the row; the second, as if the fuel elements were square (no directional distinction); and the third, as if the fuel elements were placed with the fuel plates parallel to the direction of the row. Figure A1 depict these three fuel element orientation schemes in an infinite row of fuel elements.

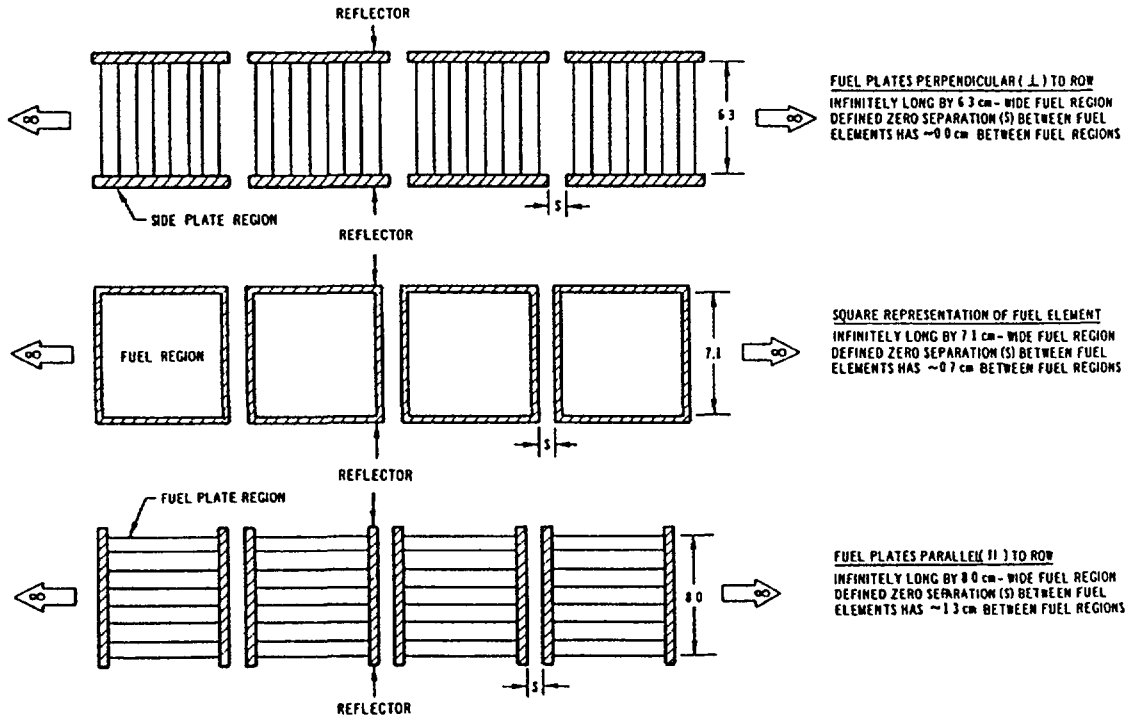


Fig. A1. Possible Fuel Element Orientations in a Row.

The calculations were performed over a range of 0.0 to 2.0 cm separation between fuel elements and with a 10 cm water reflector on each side of the row. In these calculations, the alumimum of the storage rack compartments was neglected and only the most reactive LEU fuel element type was used. In this case, that would be the LEU U_3Si 621 g ^{235}U element (No. 13 in Table 1) since this element contains the largest fissile loading. The end-fittings on the fuel elements were also neglected in the calculations.

Table A1. Eigenvalue Calculations for an Infinite Row of Number 13 Fuel Elements (LEU U_3Si 621 g ^{235}U) as a Function of Element Separation in a Row and Element Orientation in a Row. (See Figs. A, B, and C in Appendix C.)

Element Separation in a Row, cm	Water Reflector Thickness, cm	Fuel Element Plates Perpendicular to the Row	Square Representation of the Fuel Element	Fuel Element Plates Parallel to the Row
0.0	10.0	0.7920	0.8038	0.8154
0.25	10.0	0.7937	0.8049	0.8161
0.50	10.0	0.7942	0.8046	0.8153
0.75	10.0	0.7934	0.8031	0.8132
1.00	10.0	0.7914	0.8002	0.8098
1.50	10.0	0.7838	0.7910	0.7994
2.00	10.0	0.7718	0.7775	0.7849
0.75[a]	10.0	0.8250	0.8311	0.8458[c]
0.75[b]	7.0	0.8912	0.8965	0.9141[c]
0.25	7.0	-	-	0.7909
0.25	8.0	-	-	0.8036
0.25	12.0	-	-	0.8206
1.77	8.6	-	-	0.7860[d]

[a]Data for an infinite-by-infinite array with a 20-cm row separation.

[b]Data for an infinite-by-infinite array with a 14-cm row separation.

[c]See Table 2 for parallel-plate data.

[d]k_{eff} is 0.8431 for an infinite-by-infinite array with a 17.2-cm row separation.

Results of these diffusion theory calculations are shown in the first part of Table A1 and in the bottom three curves of Fig. A2. Clearly, the fuel element orientation in a row has a reactivity effect. When the fuel plates are parallel to the row, this results in the largest amount of fuel per unit length of the row and therefore, is the most reactive orientation.

For the three orientations of fuel elements in a row, the eigenvalues differ by up to 2.3% $\delta k/k$ for zero separation between fuel elements and 1.3% $\delta k/k$* for 2 cm separation. As the separation increases, the fuel element orientation reactivity effect is smaller.

It is also evident that the most reactive configuration is not at zero separation between fuel elements; the optimum separation is between 0.25 and 0.5 cm depending upon the fuel element orientation. This suggests that for separations larger than the optimum separation, water effectively begins to isolate a fuel element from its neighbor. Source multiplication experiments[3,7] suggest that about 17 fuel elements, with optimum spacing, are equivalent to an infinite row of fuel elements.

Infinite-by-Infinite Array of Fuel Elements

Fuel element orientation reactivity effects were also examined for configurations with more than just a single infinite row of fuel elements. Calculations were performed for infinite-by-infinite arrays of fuel elements with 14 and 20 cm of water between rows.

These results are shown in the second part of Table A1 and are plotted in Fig. A2. The same general trend of the eigenvalue being a function of the fuel element orientation in a row is evident in these data, as is also, the effectiveness of water in isolating one row from a neighboring row as the separation between rows increase.

*For convenience in this report, the reactivity unit $\delta k/k$ is defined to be δk_{eff}.

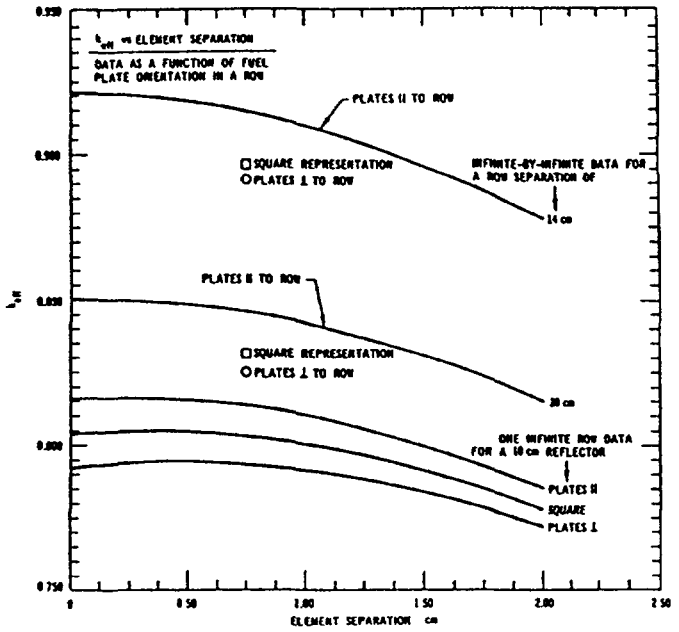


Fig. A2. Eigenvalues for Various Fuel Element Orientations in Infinite Arrays of LEU U_3Si 621 g ^{235}U Fuel Elements as a Function of the Separation Between Elements.

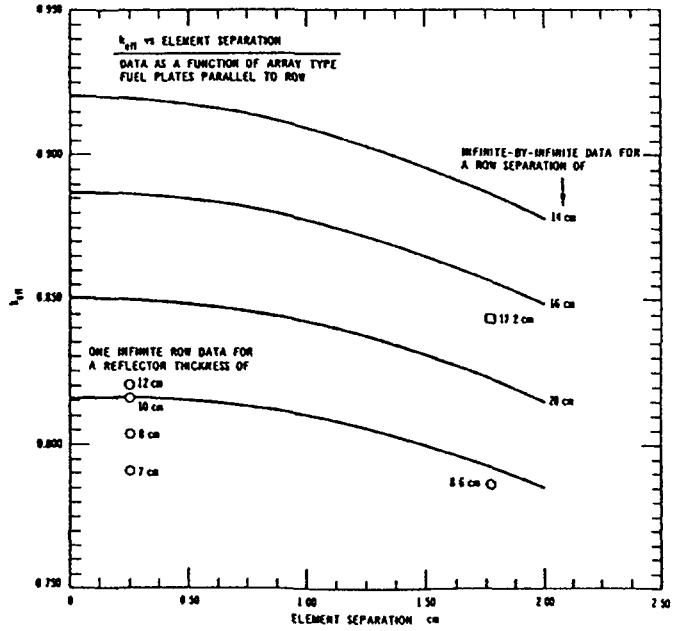


Fig. A3. Eigenvalues for Various Reflector Thicknesses in Infinite Arrays of LEU U_3Si 621 g ^{235}U Fuel Elements as a Function of the Separation Between Elements.

The optimum fuel element orientation in these types of storage configurations is therefore, with the fuel plates of the fuel element parallel to the rows. This orientation gives the maximum eigenvalue and therefore, is the most conservative in judging the acceptable reactivity for a storage rack configuration.

WATER REFLECTOR THICKNESS

Infinite Row of Fuel Elements

The reactivity effect of various thicknesses of water on an infinite row of fuel elements was examined to determine the effective infinite thickness of reflector material. These calculations were performed for a storage configuration with optimum spacing (0.25 cm) of LEU U_3Si 621 g ^{235}U fuel elements and 7, 8, 10, and 12 cm water reflectors. As in the fuel element orientation studies, the storage rack compartments and the fuel element end-fittings were neglected.

These results are listed in the third part of Table A1 and are plotted at the bottom of Fig. A3. These data show that the reactivity increases by just over 1% $\delta k/k$ as the reflector thickness increases from 7 to 8 cm, and increases only another 1% $\delta k/k$ for a 2 cm increase from 8 to 10 cm. The increase in k_{eff} from 10 to 12 cm is substantially smaller which indicates that 12 cm of water is nearly an infinite reflector with respect to an infinite row of fuel elements.

Infinite-by-Infinite Array of Fuel Elements

As an extension of the above data, the reactivity effects of adding additional rows of fuel elements were examined. These data are also plotted in Fig. A3 for 14, 16, and 20 cm thicknesses of water between rows of an infinite-by-infinite array of fuel elements.

These results indicate that the coupling between rows rapidly decreases as the water reflector thickness increases. The reactivity difference between a single infinite row of fuel elements with a 10 cm reflector and an infinite-by-infinite array with 20 cm of water between rows is only 3.4% $\delta k/k$. This reactivity difference increases to 12.9% $\delta k/k$ for the case of 7 and 14 cm water reflector thicknesses.

As a function of the separation between elements in a row, this coupling decreases as the separation increases. For example, with reflector thicknesses of 8 and 16 cm, the coupling is 8.3% $\delta k/k$ at 0.25 cm separation, and 5.7% $\delta k/k$ at 1.77 cm separation with reflector thicknesses of 8.6 and 17.2 cm.

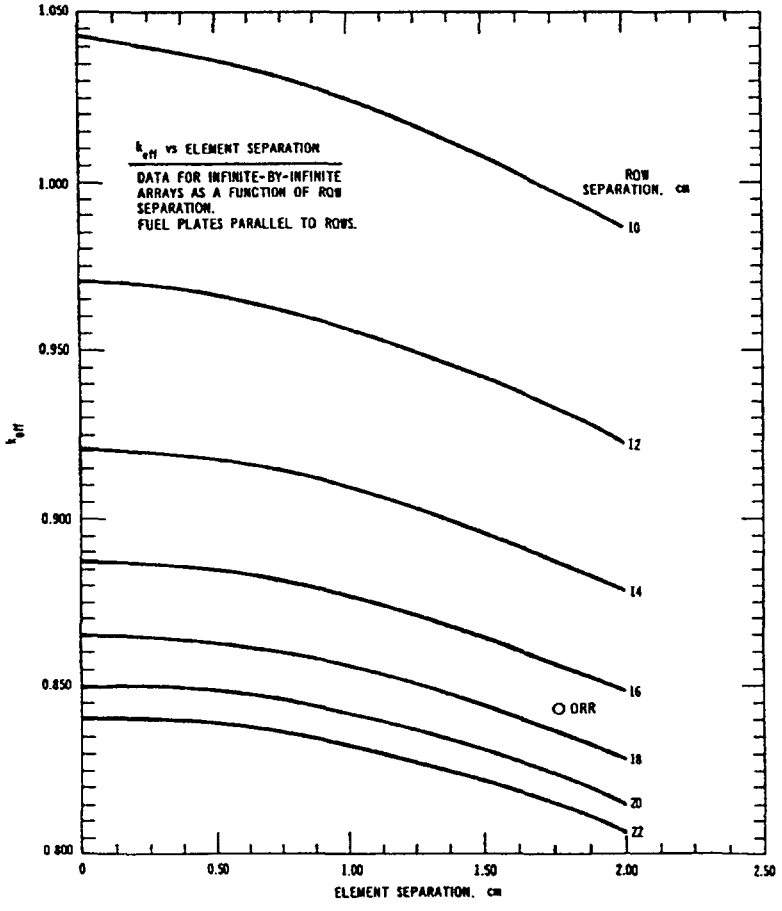


Fig. A4. Eigenvalues for Various Row Separations in Infinite-by-Infinite Arrays of LEU U_3Si 621 g ^{235}U Fuel Elements as a Function of the Separation Between Elements.

FUEL ELEMENT SPACING

As a result of an indicated optimum fuel element spacing and an effective infinite reflector thickness for a single row of fuel elements, an examination was made to determine the corresponding optimum spacing and reflector thickness for an infinite array of fuel elements.

The calculations assumed an infinite-by-infinite array of LEU U_3Si 621 g ^{235}U fuel elements with row separations of from 10 to 22 cm, and element separations of 0.0 to 2.0 cm. In all cases, both the aluminum of the storage rack and the fuel element end-fittings were neglected.

The results of these calculations are shown in Table 2 and are plotted in Figs. A4 and A5. The data are plotted in Fig. A4 as a function of element separation for various row separations, and in Fig. A5 as a function of row separation for various element separations.

Optimum Fuel Element Spacing

As clearly shown in Fig. A5, the optimum fuel element spacing is a function of the row separation. For row separations greater than about 18 cm, the optimum fuel element spacing is about 0.25 cm. This optimum spacing is about the same as was noted previously for an infinite row of fuel elements with a 10 cm reflector.

Effective Infinite Separation

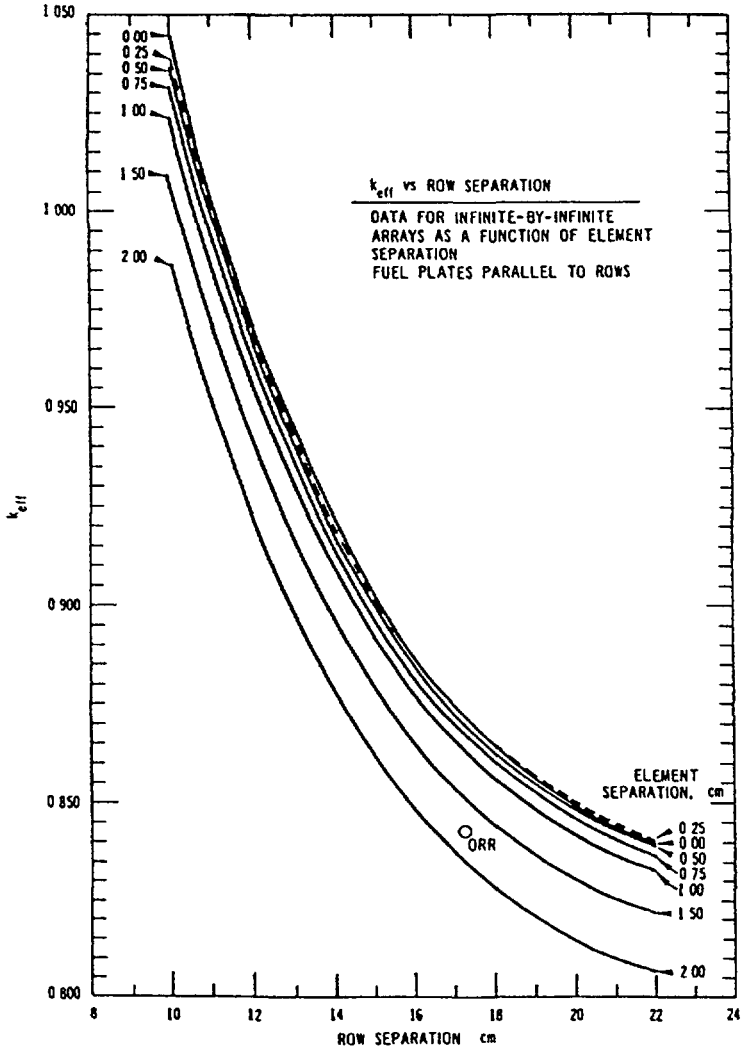


Fig. A5. Eigenvalues for Various Element Separations in Infinite-by-Infinite Arrays of LEU U_3Si 621 g ^{235}U Fuel Elements as a Function of the Separation Between Rows.

In case of coupling between rows, Fig. A5 shows the slopes of the curves are becoming fairly flat when the row separation is about 22 cm. For larger separations, the reactivity effect will be substantially smaller which indicates that 22 cm separation between rows is nearly an infinite separation. This separation is consistent with the 12 cm reflector thickness noted previously for an infinite row of fuel elements.

APPENDIX B

EIGENVALUES FOR VARIOUS FUEL ELEMENT TYPES IN VARIOUS STORAGE RACK CONFIGURATIONS

CHANGING THE STORAGE RACK CONFIGURATION

The storage rack configuration considered here is simular to the ORR storage rack. In this case, the same 1.77 cm spacing between fuel elements is assumed, but the separation between rows is decreased from 17.2 to 14 cm.

Based upon the data of Fig. 2, the reactivity change for this new configuration is about +4.4% $\delta k/k$. When this 0.044 change in eigenvalue is added to the curves of Fig. 3, the estimated eigenvalues for the various LEU and HEU fuel element types are obtained; these parallel curves are shown in Fig. B1.

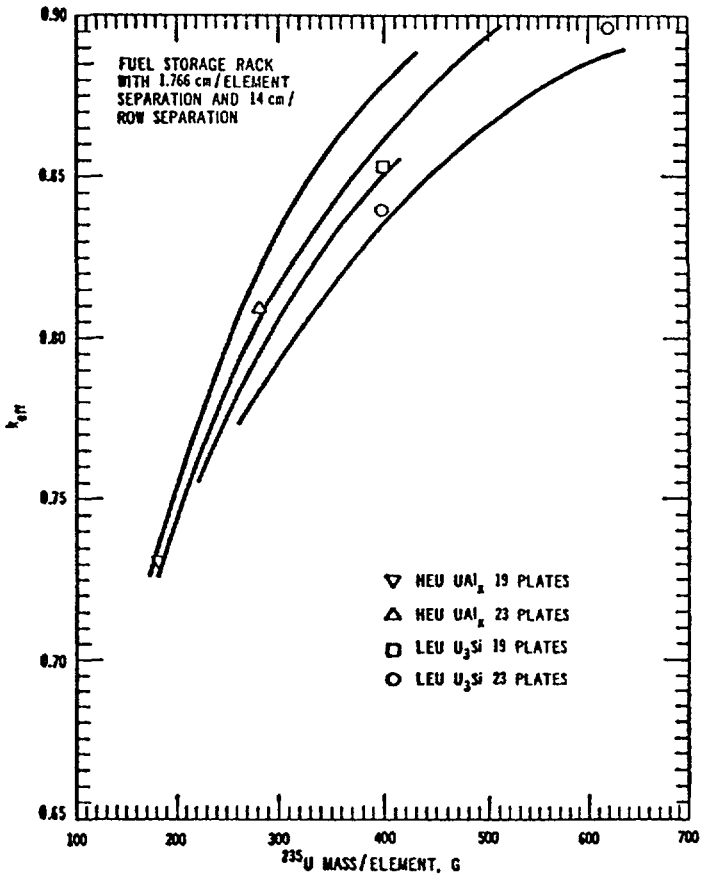


Fig. B1. Eigenvalues for Various LEU and HEU Fuel Elements in Infinite-by-Infinite Arrays With Separations of 1.766 cm Between Elements and 14 cm Between Rows as a Function of the ^{235}U Fuel Element Loading.

As a check of these estimates, eigenvalues were calculated for five of the twenty fuel element types in this new configuration. These data points are listed in Table B1 and plotted in Fig. B1. In these calculations, both the aluminum storage rack compartments and the fuel element end-fittings were included.

The data in Fig. B1 show that the three LEU and the two HEU fuel types very nearly fall on their estimated eigenvalue curves. Based upon this indicated good agreement, an important observation can be made. This is, that eigenvalue scaling using the LEU U_3Si 621 g ^{235}U fuel element data (Fig. 2) works well for other fuel types and ^{235}U loadings. It can, therefore, be expected that reasonable eigenvalue estimates can be made for a wide range of LEU and HEU fuel element types in a variety of storage rack configurations.

Table B1. Eigenvalue Calculations for Five Fuel Element Loadings, Each in an Infinite-by-Infinite Array Assuming an Element Separation of 1.766 cm and a Row Separation of 14 cm. (See Fig. F in Appendix C.)

Fuel Element Loading Number	Fuel Type	Plates/ Element[a]	^{235}U/ Element	k_{eff} Eigenvalue
9[b]	LEU U_3Si	19	395	0.8528
12[c]	LEU U_3Si	23	395	0.8394
13	LEU U_3Si	23	621	0.8962
14	HEU UAl_x	19	180	0.7300
18	HEU UAl_x	23	280	0.8094

[a]End-fitting Al/H_2O volume fractions are: 19-plate element 38.98/61.02 and 23-plate element 44.53/55.47.

[b]Uranium density is 5.391 g/cm^3.

[c]Uranium density is 4.454 g/cm^3.

EXCHANGING HEU AND LEU FUEL ELEMENT TYPES

Based upon the data of Fig. 3, reactivity estimates can be made for fuel element geometry differences and ^{235}U loading changes. For a given ^{235}U fuel element loading, these reactivity estimates are summarized below:

	Fuel Element Configuration Change	Reactivity Change
1.	LEU vs. HEU fuel for the same fuel element geometry	2 to 3% δk/k less reactive
2.	^{235}U loading increment for the same fuel element geometry	0.5% δk/k per 10 grams
3.	Number of plates per element for the same fuel meat thickness	-0.5% δk/k per plate
4.	Fuel meat thickness increment for the same number of plates per element	-2.0% δk/k per 0.25 mm

As an example of how these reactivity effect figures can be used, the estimated eigenvalues for 300 g ^{235}U HEU fuel elements and two geometries of 400 g ^{235}U LEU fuel elements in the ORR storage rack configuration are given below:

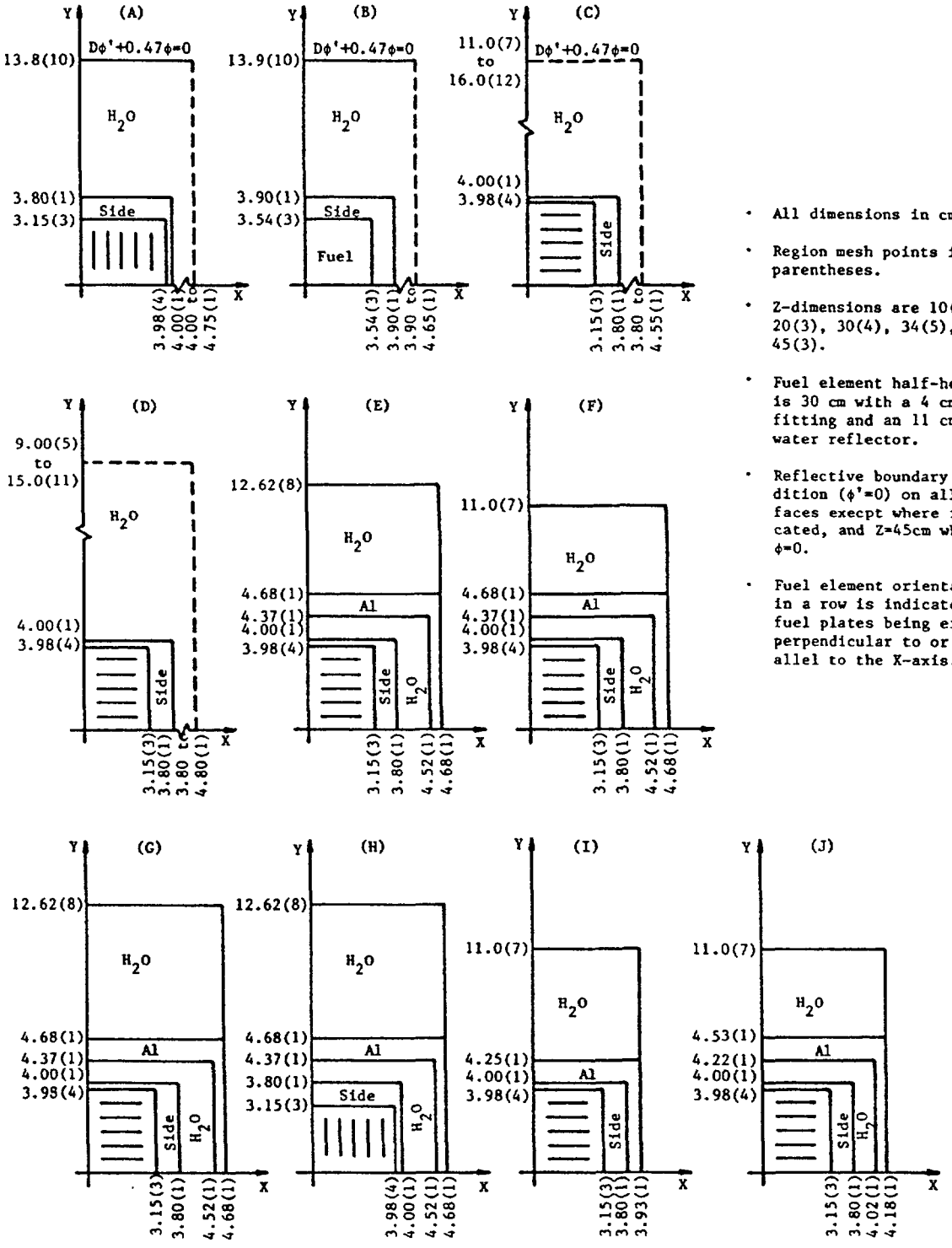
ORR Storage Rack Configuration Loading	k_{eff} Estimated	k_{eff} Figure 3
HEU 300 g ^{235}U, 19 plates, 0.51 mm meat	---	0.791
LEU 400 g ^{235}U, 23 plates, 0.51 mm meat	0.796	0.793
1. LEU vs. HEU: 0.791 - 0.025 = + 0.766		
2. ^{235}U loading: + 10 * 0.005 = + 0.050		
3. Number of plates: - 4 * 0.005 = - 0.020		
4. Fuel meat thickness: no change		
LEU 400 g ^{235}U, 22 plates, 0.76 mm meat	0.781	0.781
1. LEU vs. HEU: 0.791 - 0.025 = + 0.766		
2. ^{235}U loading: + 10 * 0.005 = + 0.050		
3. Number of plates: - 3 * 0.005 = - 0.015		
4. Fuel meat thickness: - 1 * 0.02 = - 0.020		

In this particular example there would be very little change expected in the storage rack reactivity with the three fuel element types.

While the above reactivity coefficients are approximate and applicable over a limited range, they provide bases to estimate what would be the net effect on a storage rack if LEU fuel elements were to replace HEU fuel elements.

APPENDIX C

THREE-DIMENSIONAL DIFFUSION THEORY CALCULATIONAL MODELS



- All dimensions in cm.
- Region mesh points in parentheses.
- Z-dimensions are 10(4), 20(3), 30(4), 34(5), 45(3).
- Fuel element half-height is 30 cm with a 4 cm end-fitting and an 11 cm axial water reflector.
- Reflective boundary condition ($\phi'=0$) on all faces execpt where indicated, and Z=45cm where $\phi=0$.
- Fuel element orientation in a row is indicated by fuel plates being either perpendicular to or parallel to the X-axis.

Appendix N-3.2

FRESH FUEL STORAGE

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Abstract

A criticality analysis on the storage of MTR-type medium-enriched uranium (MEU) fuel elements was performed. It was assumed that the MEU fuel elements were arranged vertically in two rows in a stainless steel box, and that the boxes were placed in parallel infinitely at a certain distance. In both sides of the box, 0.3w/o boron-loaded stainless steel plates were used. Moreover, it was assumed that the inside and outside of the box were filled up with water. K_{eff} was calculated as a function of the distance between the boxes by using ANISN-JR code. The effect of boron on K_{eff} was also examined. A similar analysis on the storage of HEU fuel elements was performed and compared with the results on MEU fuels.

INTRODUCTION

On fuel management, it is important to store fuel elements safely and efficiently in a fuel storage facility. When a great number of fuel elements are stored in the facility, safe and simple handling of fuels, radiation shielding, decay heat removal and criticality safety control as well as an appropriate physical protection are required. Especially, the problem of criticality safety control is peculiar to nuclear material storage.

At the Research Reactor Institute of Kyoto University (KURRI), many MTR-type fuel elements containing highly-enriched uranium (HEU, 93%) have been stored to be kept $K_{eff} \leq 0.95$ in a fresh fuel storage room and a spent fuel pond.

Recently, the Reduced Enrichment for Research and Test Reactors (RERTR) program has been forwarded, and the KURRI is cooperating with the Argonne National Laboratory in the joint study of the RERTR program. Accompanying the reduction of enrichment, it is necessary to confirm the criticality safety for the storage of reduced enrichment uranium fuels.

This paper presents the results of criticality analysis on the storage of medium-enriched uranium (MEU, 45%) fuel elements.

MEU FUEL ELEMENT AND STORAGE METHOD

The MEU fuel element considered in the criticality analysis is shown in Fig. 1. The dimension of the element is the same as that of the fuel element of Kyoto University Reactor (KUR). One fuel element is composed of 18 fuel plates. The clearance (water gap) between the plates is 2.81 mm. Each fuel plate consists of a 45% enriched uranium-aluminide (UAl_x-Al) meat cladded with aluminum. The thickness of the meat is 0.5 mm and that of the clad is 0.51 mm. The average length of the meat is 59.4 cm. The uranium density of the meat is 1.6833 g/cm^3 and the U-235 density is 0.7575 g/cm^3.

The storage method for the MEU fuel elements is also same as that for HEU fuel elements of KUR. That is, the fuel elements are arranged vertically in two rows and stored in a rectangular box made of stainless steel. The fuel storage box is shown in Fig. 2. In both larger sides of the box, stainless steel plates containing natural boron are partially used. The position of the boron-loaded stainless steel plates corresponds to the meat position of the fuel element. The content of natural boron is about 0.3w/o, and the thickness of the side plate is 2 mm. The fuel storage boxes are placed in parallel at a suitable distance in a storage room and fixed to the building floor. K_{eff} in the storage room is controlled less than 0.95 even if the room is filled up with water.

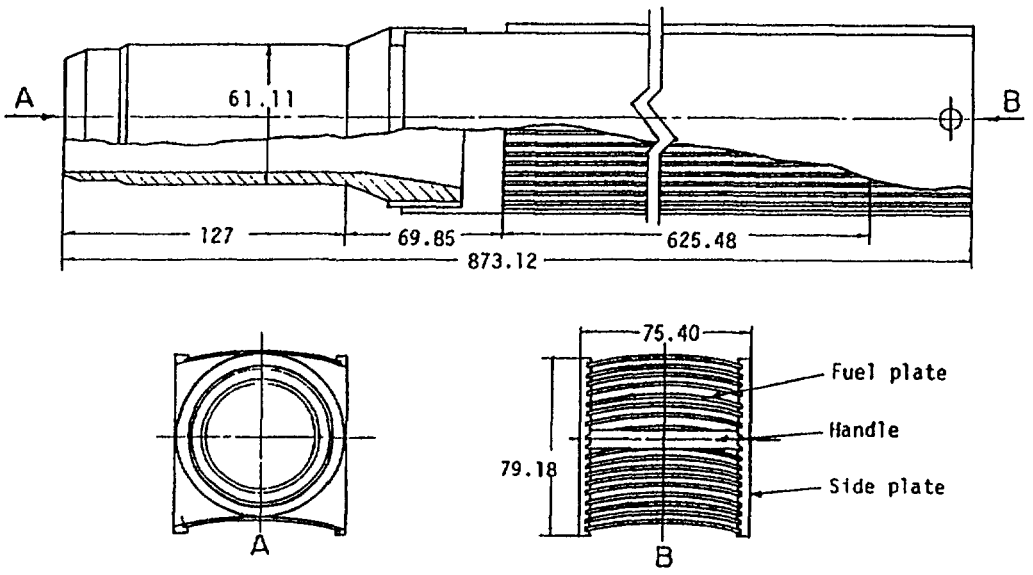

Fig. 1. MTR-type MEU fuel element considered in criticality analysis (scale : mm)

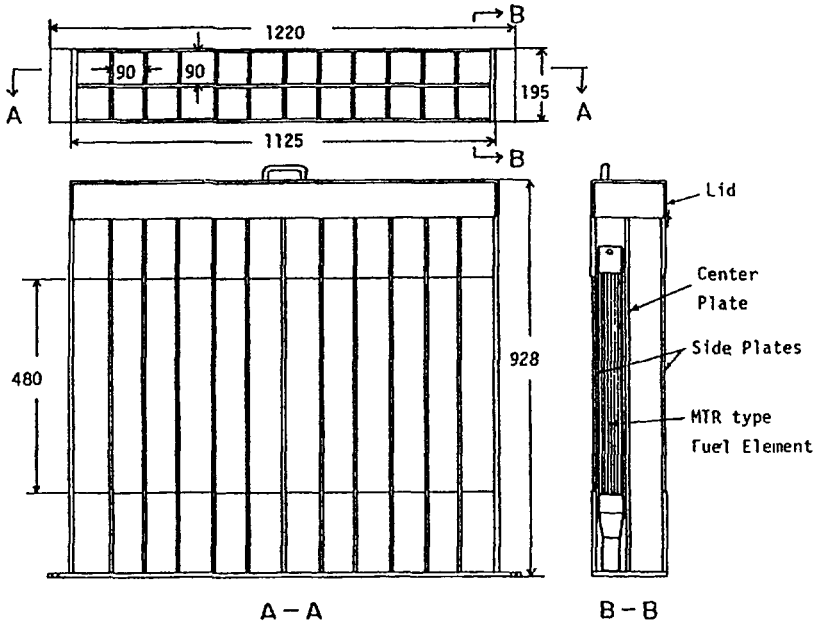

Fig. 2. Fuel storage box (scale : mm)

CRITICALITY ANALYSIS

In order to calculate K_{eff} for the storage of MEU fuel elements, one dimensional transport code ANISN-JR[1] was used. The number of employed energy groups was 26, 15 thermal and 11 fast neutron energy groups, and the cross sections were taken from MGCL 26 library[2].

The calculation was performed on condition that:

(1) the fuel storage box was filled with fresh MTR-type MEU fuel elements mentioned above,
(2) the inside and outside of the box were filled up with water,
(3) for the region where the fuel elements were in line, macro cross sections were obtained by cell calculation so called unit cell - super cell,
(4) the fuel region was considered as a 62.55 cm-high, infinitely long slab fuel, and
(5) two slab fuels were combined with boron-loaded and ordinary stainless steel plates and placed in parallel infinitely at a certain distance.

The above conditions are illustrated in Fig. 3.

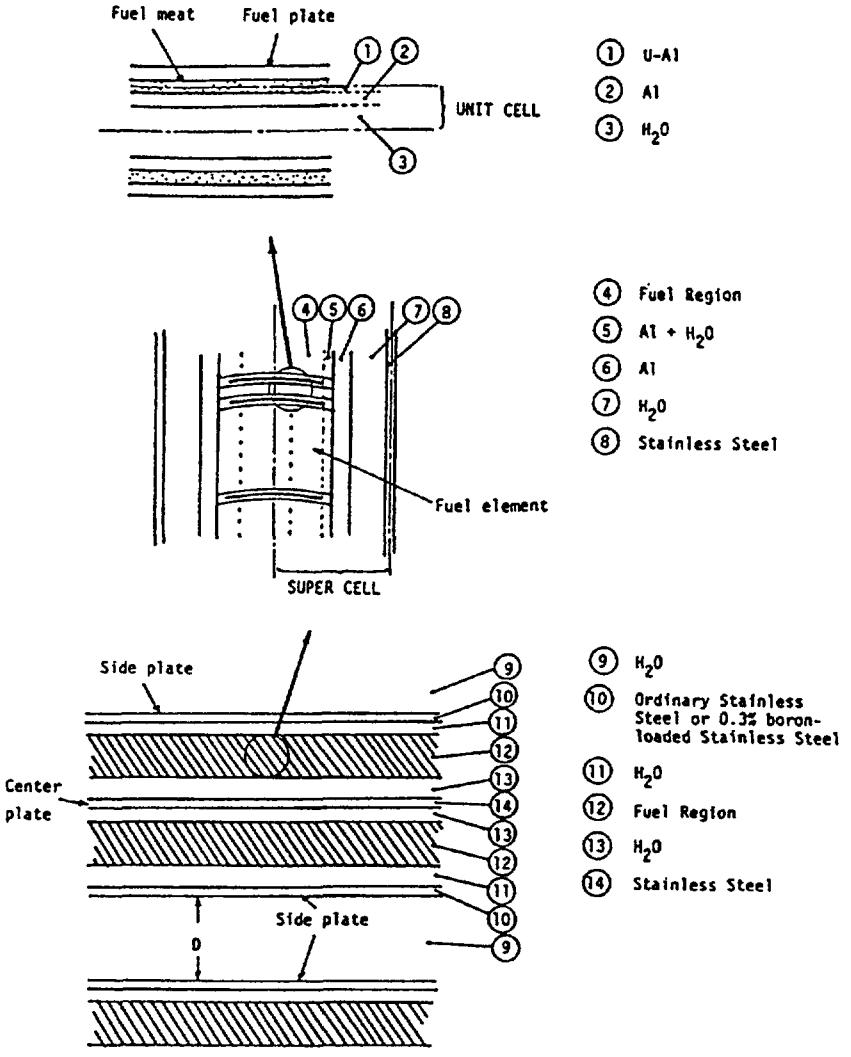

Fig. 3. Outline of criticality analysis.

Prior to the K_{eff} calculation, the validity of using ANISN-JR code was confirmed by comparing the experimental K_{eff}-value with the calculated value. The experimental value was obtained from the critical experiments using MEU fuel in the Kyoto University Critical Assembly (KUCA). As a result the calculated K_{eff}-value turned out to be in agreement with the experimental value within 1%.

K_{eff} was calculated as a function of the distance between the fuel storage boxes. In order to examine the effect of boron containing in the side plate on K_{eff}, we also carried out similar calculation of K_{eff} for the case of using ordinary stainless steel side plates. Moreover, a series calculation of K_{eff} was performed for the storage of MTR-type HEU fuel elements, and the results were compared with the results of MEU fuels. The properties of MEU and HEU fuel meats used in the calculation are listed in Table 1.

Table 1. Properties of MEU and HEU fuels used in criticality analysis.

	MEU	HEU
Fuel material	UAl_x - Al	U-Al
Enrichment (%)	45	93.15
Uranium loading (%)	42	20
Meat density (g/cm^3)	4.0081	3.2598
Uranium density (g/cm^3)	1.6833	0.6520
U-235 density (g/cm^3)	0.7575	0.6073

RESULTS AND DISCUSSION

The calculated results of K_{eff} are shown in Fig. 4. In the figure, 'without-boron' or 'with-boron' means a case of using ordinary or 0.3% boron-loaded stainless steel side plate, respectively. In the case of with-boron for MEU, the calculated K_{eff} is less than unity when the distance (D) between the fuel storage boxes is larger than 6 cm and less than 0.95 at $D \geq 8$ cm. If ordinary stainless steel plates are used in both sides of the box, $D \geq 10$ cm is required to be subcritical and K_{eff} is less than 0.95 at $D \geq 15$ cm. For HEU fuels K_{eff} is less than 0.95 at $D \geq 5$ cm and 10 cm in cases of with-boron and without-boron, respectively. These results are summarized in Table 2.

In this calculation, it has become clear that K_{eff} decreases about 6 to 10% when 2-mm thick 0.3% boron-loaded stainless steel plates are used in both sides of the box instead of ordinary stainless steel plates.

As comparing the results for MEU with HEU, K_{eff} for MEU is about 6% larger than that for HEU in both cases of without- and with-boron. This is mainly caused by the difference of the U-235 density of the meat.

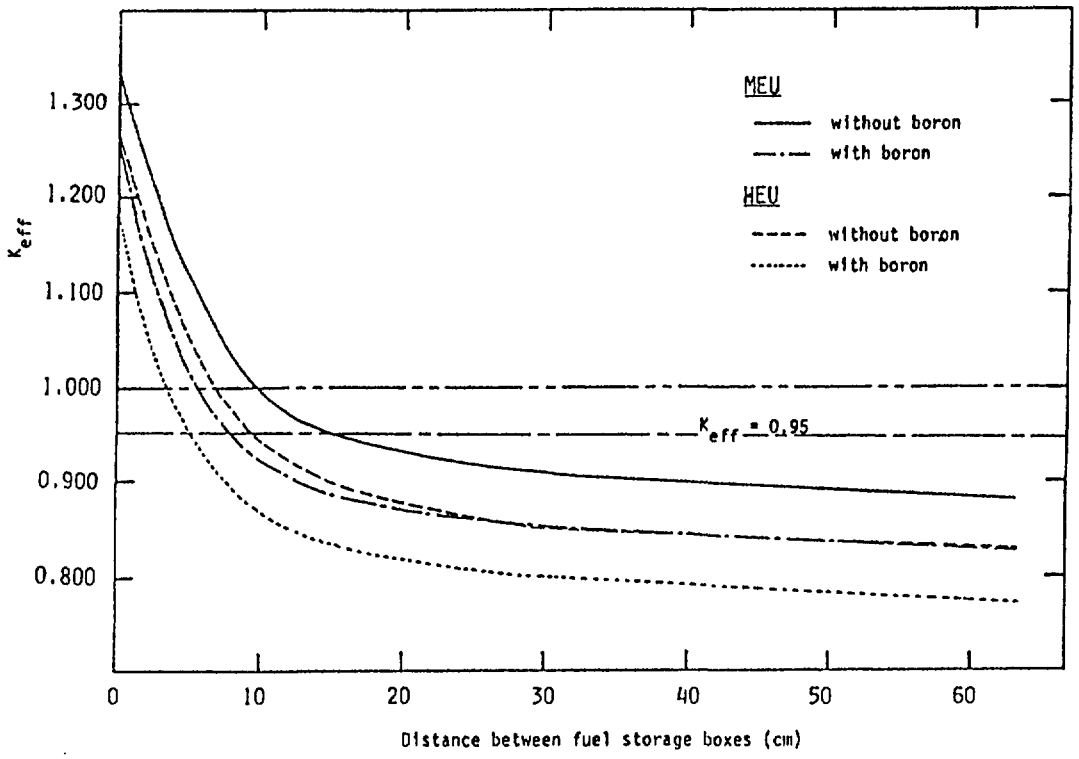

Fig. 4. Calculated K_{eff}-values as a funstion of the distance between fuel storage boxes for MEU and HEU.

Table 2. Minimum required distance between fuel storage boxes.

	MEU	HEU
with-boron	8 cm	5 cm
without-boron	15 cm	10 cm

CONCLUSION

A criticality analysis on the storage of fresh MTR-type MEU fuel elements has been performed. In the calculation of K_{eff}, it is assumed that the array of the fuel elements are considered as an infinitely long slab fuel, the fuel and its surroundings are filled up with water and the slab fuels are placed in parallel infinitely at a certain distance. A similar analysis for HEU fuels has also been performed on the same condition of MEU to be compared with the results of MEU fuels. From these results of the K_{eff} calculation, it has been concluded that:

(1) since the U-235 density of the MEU fuel considered in this analysis is higher than that of the HEU fuel, it is required for the storage of MEU fuel elements to keep the distance between the fuel storage boxes about 1.5 times larger than the case of HEU fuel storage,

(2) by using 2 mm-thick stainless steel plates containing 0.3% natural boron in both sides of the box instead of ordinary stainless steel plates, it is possible to store the amount of MEU or HEU fuel elements by 1.2 times.

As mentioned above, it is valuable to use the boron-loaded stainless steel plate in the fuel storage box when a great number of fuel elements are stored in a limited area of the storage facility.

Since the fuel storage boxes of KURRI are placed in parallel at a distance of larger than 60 cm and, moreover, 0.3% boron-loaded stainless steel plates are used in both sides of the box, our storage method is sufficient for the storage of MEU fuel elements in the criticality safety control with considerable surplus.

These results should also be useful for spent fuel storage in a pond.

ACKNOWLEDGEMENTS

The authors are most grateful to Messrs. M. Hayashi and S. Shiroya of KURRI for their helpful advices and operating computers.

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Appendix N-3.3

SPENT FUEL STORAGE*

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Abstract

A criticality analysis on the storage of highly enriched HFR fuel elements, containing 420 g ^{235}U and 1000 mg ^{10}B in the side plates, was performed. The fuel elements are stored in the pool in specially designed compact racks, consisting of cladded cadmium boxes. For reasons of safety and flexibility the analysis of the (spent) fuel elements was performed for an infinite array of fresh elements with 450 g ^{235}U without ^{10}B in the side plates.
In the framework of the safety and licensing guidebook additional calculations were performed for LEU fuel elements, containing 450, 475, 600, 675 and 1000 g ^{235}U respectively and compared with the results of the HEU fuel elements. Also the reactivity effect of the cadmium boxes and the presence of Be elements in the boxes was examined.

INTRODUCTION

Up till 1982 the licence for the storage of fresh fuel elements in the pool was restricted to 12 elements of 405 g ^{235}U with 1000 mg ^{10}B in the side plates. It was not allowed to store both fuel and beryllium elements in the racks. The old storage racks consisted of two rows of six positions separated by a thick layer of water and at the outer sides provided with a cadmium liner with a height of 30 cm.

In order to store fuel elements more efficiently and to get rid of the above mentioned restrictions new fuel racks were designed for under water storage of (spent) fuel and other elements such as beryllium reflector elements. The new pool storage racks are provided with a 1 mm thick cadmium box around each element position, see Fig. 1. A number of these boxes are placed next to each other in a tank which is open at the upper side.

For reasons of safety and flexibility the analysis of the (spent) fuel elements was performed for an infinite array of fresh fuel elements with 450 g ^{235}U without ^{10}B in the side plates. An analysis with ^{10}B in the side plates is more complicated due to the reactivity increase during the first burn-up steps caused by the fast depletion of ^{10}B with regard to ^{235}U. Also the reactivity effect of replacing fuel elements by Be elements was examined.

In the framework of the safety and licensing guidebook additional calculations were performed for LEU elements, containing 450, 475, 600, 675 and 1000 g ^{235}U respectively. These results were compared with the results of the HEU fuel elements. Also the reactivity effects of the cadmium boxes were examined both for LEU and HEU elements.

* The work described in this report has been carried out under contract to the European Commission and has been financed by the JRC budget.

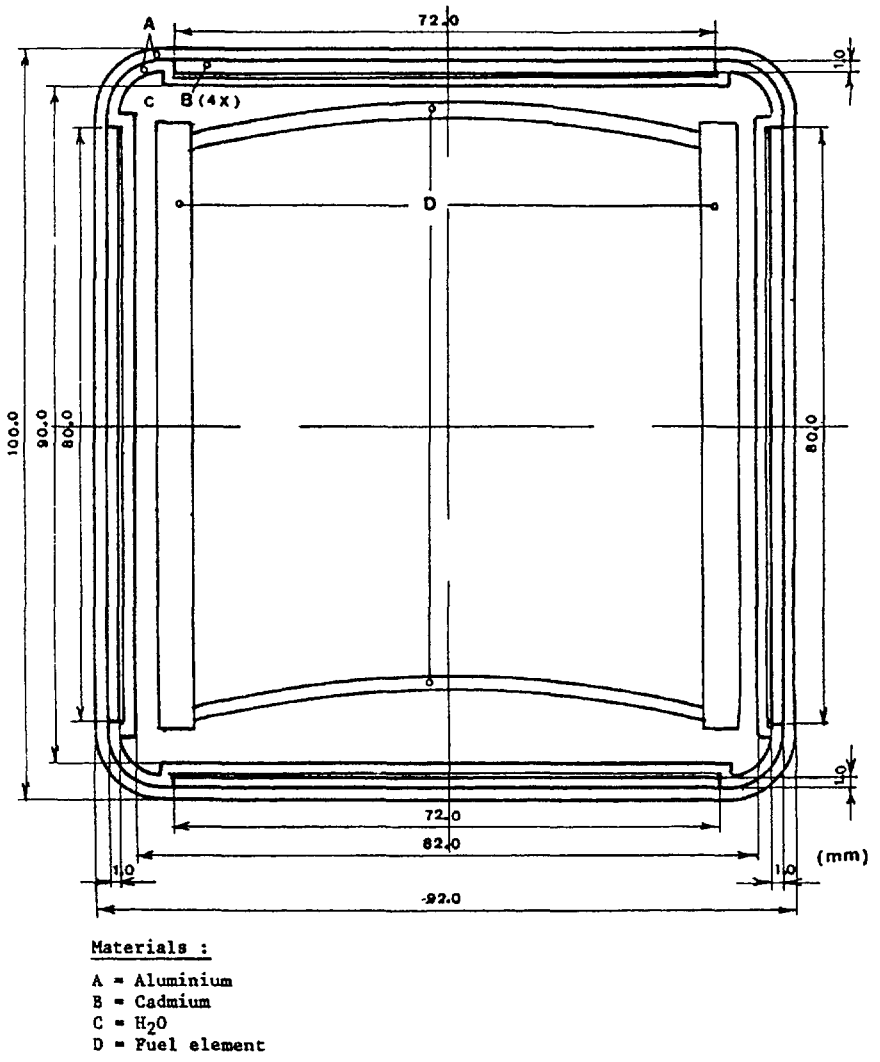


Fig. 1. Horizontal cross section of a 23 plates fuel element in a storage box, provided with cadmium plates in the four aluminum walls.

CALCULATION PROCEDURE

The multiplication factor of the filled storage racks were computed with the aid of the 2 dimensional diffusion code TEDDI-M [3].

The nuclear constants, required in this code are computed for five energy groups. These energy groups were :

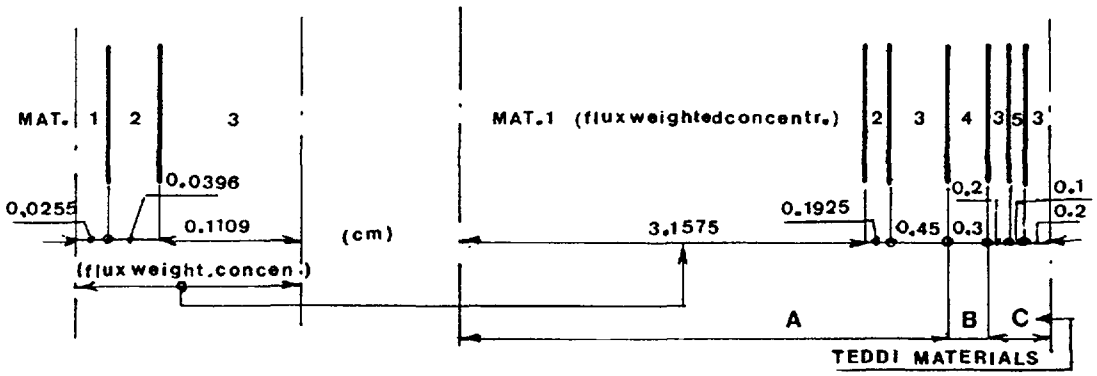
Group 1 = 14 10^6 eV - 1.353 10^6 eV
Group 2 = 1.353 10^6 eV - 0.0674 10^6 eV
Group 3 = 0.0674 10^6 eV - 0.683 eV
Group 4 = 0.683 eV - 0.3 } thermal groups
Group 5 = 0.3 eV - 0 }

The computer programs used for the determination of the nuclear constants were :

a) GGC-IV |2| for the epithermal and fast groups (above 0.683 eV)

b) MICROFLUX-2 |1| for the thermal groups (below 0.683 eV).

The MICROFLUX calculations were carried out in the following way : first determination of the "flux weighed" number densities in the fuel region, consisting of the fuel meat, the cladding and the H_2O cooling channel (see Fig. 2). These "flux weighed" number densities were used in the fuel region (material 1 in Fig. 3) of the second microflux calculation of the whole box.



mat. 1 = fuel
mat. 2 = aluminium
mat. 3 = H_2O

Fig. 2 Geometry used in first microflux calculation.

mat. 1 = flw. conc. of fuel, Al and H_2O
mat. 2 = H_2O + Al
mat. 3 = Al
mat. 4 = H_2O
mat. 5 = cadmium

Fig. 3 Geometry used in second microflux calculation.

At the boundaries of the cadmium material region the so called black boundary condition (no neutrons returning from the cadmium material) had applied for the lowest thermal group number 5.

The multiplication factors were calculated for an infinite two dimensional array of storage racks, placed next to each other. So at the four outer boundaries of a single box the zero next current boundary condition had applied. The buckling factor in the vertical direction, B_z^2, used in the diffusion code was 0.019, corresponding with a fuel height to 60 cm and taking into account a reflector saving of about 8 cm.

CALCULATED CASES AND RESULTS

The criticality of the storage racks were determined for the following cases :

Case 1 : The storage racks only filled with 450 grams ^{235}U HEU fuel elements (94.4% enriched). Two geometrical models for the cadmium box with fuel elements were used (compare Fig. 4 and Fig. 5 case 1). The multiplication factors, k_{eff}, are 0.555 and 0.553 (a difference of 190 pcm) for the models given in Fig. 4 and Fig. 5 (case 1) respectively.

Case 2 : The storage racks partly filled with Be elements in the ratio of three fuel elements against one Be element (see Fig. 5). The k_{eff} for this combination is 0.475.

Case 3 : As case 2, but the numbers of fuel and Be elements are equal (see Fig. 5). The k_{eff} in this case is 0.370.

Case 4 : As case 2, but the number of Be elements is 3 times the number of fuel elements (see Fig. 5). The k_{eff} is 0.239.

Case 5 : As case 1, but for LEU elements (20% enriched). The storage racks are completely filled with fuel elements containing 450, 475, 600, 675 and 1000 g ^{235}U respectively. The results are given in Table 1.

Case 6 : In addition to case 1 and 5, the multiplication factors of the storage racks without cadmium boxes were calculated for the case they were filled completely with HEU or LEU elements. The results are also presented in Table 1.

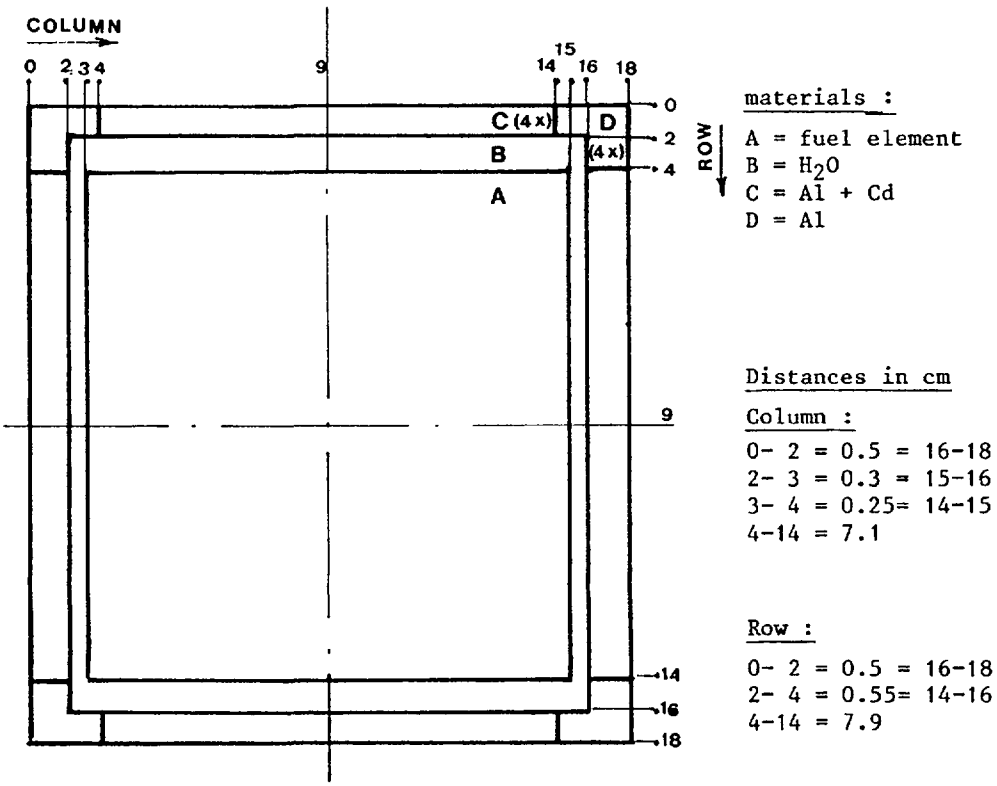
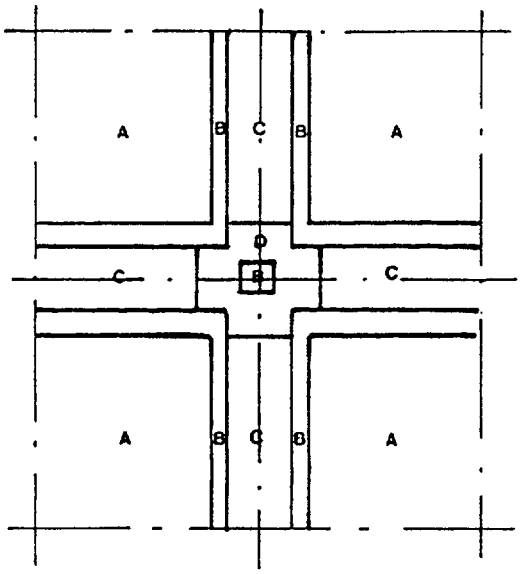
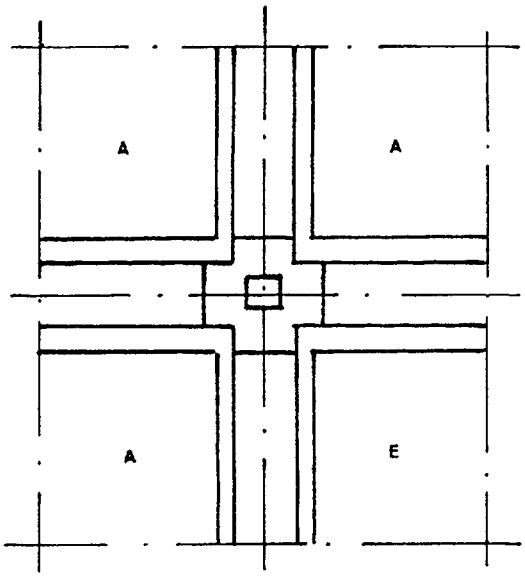


Fig. 4 Geometrical model used in the first TEDDI-M calculation for storage racks, filled with fuel elements only.



Case 1 : 4 fuel elements (A)

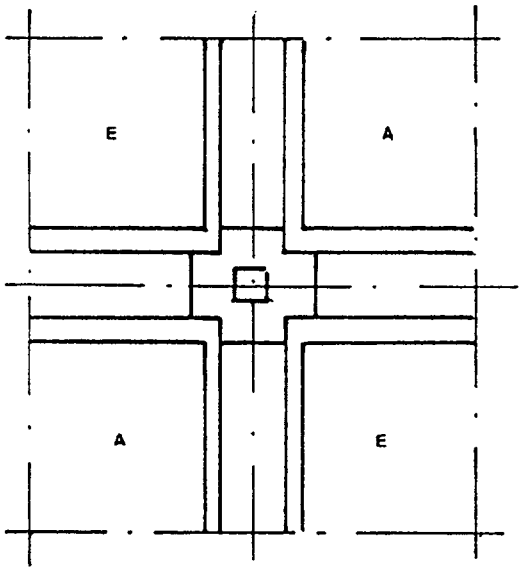
k_{eff} = 0.553



Case 2 : 3 fuel elements (A)
+ 1 Be element (E)

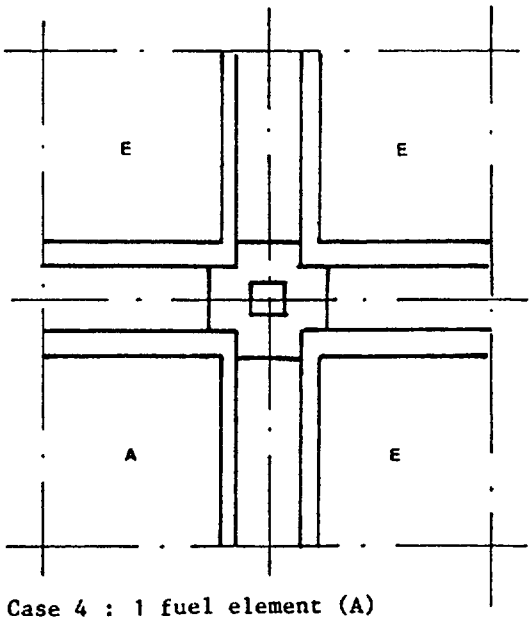
k_{eff} = 0.475

Material : A = fuel element
B = H_2O
C = Al + Cd
D = Al
E = Be element



Case 3 : 2 fuel elements (A)
2 Be elements (E)

k_{eff} = 0.370



Case 4 : 1 fuel element (A)
3 Be elements (E)

k_{eff} = 0.239

Fig. 5. Geometries and material compositions used in the TEDDI-M calculations of the storage racks filled with fuel and Be elements.

Table 1. Multiplication factor of HFR compact storage racks for HEU (90%) and LEU (20%) fuel elements.

HEU/LEU	^{235}U in element gram	k_{eff} with Cd	k_{eff} without Cd
HEU	450	0.56	1.45
LEU	450	0.54	1.33
LEU	475	0.56	1.34
LEU	600	0.64	1.38
LEU	675	0.68	1.39
LEU	1000	0.82	1.41
LEU	> 1500	-	1.45

CONCLUSIONS

According to the analyses, the k_{eff} of the storage racks, fully loaded with highly enriched 450 grams ^{235}U fuel elements without ^{10}B, is lower than required. If these fuel storage racks are partly filled up with Be reflector elements, the multiplication factor will be even lower. The same racks can be used for low enriched fuel with higher ^{235}U contents.

In the compact storage racks, in which each fuel element is placed in a cadmium box, there is only a small difference in reactivity between HEU and LEU fuel elements containing the same amount of ^{235}U. Without the cadmium absorbing plates the difference in reactivity is significant.

REFERENCES

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Users and programmers manual for the GGC-IV multigroup cross section code, G.A. 7157.

|3| J. Raven, G.J. Bakkenes, H.P. Struch
TEDDI-M. Een programma voor de oplossing van de twee dimensionale neutronen-diffusie vergelijking voor veel groepen in x-y en r-z geometrie.

Appendix N-4

RECEIPT AND FINANCIAL SETTLEMENT PROVISIONS FOR NUCLEAR RESEARCH REACTOR FUELS

UNITED STATES DEPARTMENT OF ENERGY
Washington, D.C.,
United States of America

Abstract

U.S. Federal Register notices (as of 30 December 1987) on DOE's "Receipt and Financial Settlement Provisions for Nuclear Research Reactor Fuels" are provided. DOE's current commitment to provide receipt and financial settlement services for qualifying research reactor fuels extends until 31 December 1988 for HEU fuels and until 31 December 1992 for LEU fuels. The need for extension beyond these dates will be evaluated by DOE when appropriate.

Receipt and Financial Settlement Provisions for Nuclear Research Reactor Fuels

AGENCY: Department of Energy.

ACTION: Notice

SUMMARY: The Department of Energy is amending the provisions of its current policy providing for the receipt and financial settlement of U.S.-origin spent research reactor fuels to include certain research reactor fuels in which the uranium-235 content is less than 20 percent of the total uranium weight. Additionally, the Department is providing its financial settlement terms for providing this service.

FOR FURTHER INFORMATION CONTACT: Louis R. Willett, Mail Stop DP-131, Office of Nuclear Materials Production. U.S. Department of Energy, Washington, D.C. 20545, 301/353-3968.

SUPPLEMENTARY INFORMATION: On November 9, 1982, the Department of Energy announced in the **Federal Register** that it was extending until December 31, 1987, its policy for the receipt and financial settlement of U.S.-origin spent research reactor fuels (47 FR 50737). At that time, the provisions of this policy were restricted to: (1) uranium-aluminum research reactor fuels with enrichments; i.e., uranium-235 content as a percentage of total uranium weight, of greater than 20 percent; and (2) uranium-zirconium hydride TRIGA fuel types. In this notice, DOE indicated that it was studying the reprocessing of uranium-aluminum fuel compositions with uranium enrichments of less than 20 percent and would extend the provisions of the notice to include these LEU fuels if a reprocessing capability could be established.

The DOE has determined that reprocessing capabilities for LEU fuels will be available and is, therefore, prepared to extend the provisions of its current policy to include the LEU uranium-aluminum fuel types currently under development. The conditions governing DOE's offer to provide this service remain unchanged; i.e., that commercial fuel processing services must be unavailable at reasonable terms and conditions and that the reactor fuel must be of U.S. origin (composed of nuclear materials produced or enriched in this country). The fuel processing charges used for settlement under this program for LEU fuel receipt are based upon the estimated actual cost of providing this service at a DOE spent fuel processing facility.

In its reviews of this policy extension, DOE has determined: (1) No commercial fuel processing services for LEU fuels are expected to be available to meet anticipated needs; (2) basic beneficial nuclear research would have to be curtailed absent a spent fuel disposal capability; and (3) availability of an LEU fuels disposal capability will encourage the conversion of research reactors currently using highly enriched uranium to LEU. Since it is anticipated that these disposal services for LEU fuels will not be needed until completion of reactor conversions planned for the late 1980's, DOE proposes to extend the LEU fuel receipt provisions from the date of this publication through December 31, 1992. The need for extension beyond this time will be evaluated when appropriate.

To provide for inclusion of LEU fuels in DOE's current policy, the terms and conditions for DOE services described in paragraphs numbered 1 through 11 in the **Federal Register** notice entitled "Receipt and Financial Settlement Provisions for Nuclear Research Reactor Fuels" 47 FR 50737 published November 9, 1982, are hereby deleted and the following substituted in place:

1. This policy applies to irradiated nuclear research reactor fuels and blanket materials (reactor materials). This policy pertains only to reactor materials from research reactors other than those involved in the conduct of research and development activities leading to the demonstration of the practical value of such reactor for industrial or commercial purposes.

2. Commercial fuel processing must be unavailable at reasonable terms and conditions.

3. The fuel must be of U.S. origin - that is, composed of nuclear materials produced or enriched in the United States.

4. This policy applies solely to the following types of reactor fuels:

a. Aluminum-clad reactor fuels where the uranium-235 content is greater than 20 percent, by weight, of the total uranium content. The active fuel region of these fuels may be configured as uranium-aluminum

alloy, uranium oxide or uranium-aluminide. Fuels containing significant quantities of uranium-233 are excluded from receipt.

b. Aluminum-clad reactor fuels where the uranium-235 content is less than or equal to 20 percent by weight of the total uranium content. The active fuel regions of these fuels may be configured as uranium-silicide, uranium-aluminide or uranium oxide. Fuels containing significant quantities of uranium-233 are excluded from receipt.

c. Aluminum or stainless steel clad, uranium-zirconium hydride (other than uranium-233) TRIGA fuel types.

The percentage of uranium-235 of the eligible fuel types shall be that measured or estimated at the time of delivery to DOE.

5. DOE will undertake, under contracts individually negotiated with persons licensed pursuant to sections 53.a.(4), 63.a.(4), 103 or 104 of the Atomic Energy Act of 1954, and persons operating research reactors abroad fueled with materials produced or enriched in the United States, who possess or will possess eligible reactor materials, to receive such reactor materials at DOE-designated facilities, and to make a settlement, therefore, in accordance with this Notice and other established DOE policies. This settlement will take into account the charges for chemical processing of received reactor materials and any conversion of recovered uranium to the standard form, uranium hexafluoride, for which specifications and prices have been established by DOE. Furthermore, DOE may chemically process and convert all such received reactor materials to the extent, in such manner, and at such time, and place as it deems advisable, or otherwise dispose of such materials as it may deem advisable.

6. DOE's commitment to provide fuel receipt and financial settlement services will terminate on the following dates:

a. For research reactor fuels described in 4.a. and 4.c.-December 31, 1987; and

7. Firm charges for DOE services provided under this policy will be part of each contract. These charges will be expressed in terms of a unit weight charge fixed by DOE to the reactor materials in question, to apply over the total number of units of weight.

The charges for chemical processing services provided under this policy will reflect the Government's full cost for providing this service, in accordance with the provisions of 10 CFR Part 1009. The basic charges for processing services will be reviewed periodically and adjusted when necessary.

8. For those research reactor fuels described in 4.a. and 4.c. above, as of January 1, 1983, the following charges will be applied to DOE processing services under this policy:

a. For aluminum-clad research reactor fuels, including alloy, oxide and aluminide composition, $1000 per kilogram of total delivered weight. Of this charge, $375 is capital related and $625 is related to operating costs; and

b. For aluminum and stainless steel-clad uranium-zirconium hydride research reactor fuel, $1050 per kilogram of total delivered weight. Of this charge, $395 is capital related and $655 is related to operating costs.

The capital-related charges for DOE-provided services shall be adjusted to reflect changes in price levels from the base date of June 1982, in accordance with the Official Monthly Construction Cost Indices appearing in "Engineering News Record." The operations-related charges for DOE-provided services shall be adjusted to reflect changes in price levels from the base date of June 1982, in accordance with the Basic Inorganic Chemical Index appearing in "Wholesale Price Indexes," published by the U.S. Bureau of Labor Statistics.

9. For those research reactor fuels described in 4.b. above, as of July 1, 1985, the following charges will be applied to DOE processing services under this policy:

a. For uranium oxide fuel compositions - $660 per kilogram of total delivered weight.

b. For uranium-silicide compositions - $835 per kilogram of total delivered weight; and

c. For uranium-aluminide compositions - $1110 per kilogram of total delivered weight.

DOE will periodically review the charges for processing of these research reactor fuels and revise said charges as appropriate.

10. The charge for conversion to uranium hexafluoride of the purified nitrate salt of uranium that is converted by DOE in its processing of reactor materials is $175 per kilogram of contained uranium.

11. A minimum charge of $44,500 will be applied to each batch of fuel material delivered to DOE under the provisions of this policy. This charge reflects the minimum cost to DOE of providing processing services for small-batched fuel materials. The size of the processing batch to be shipped shall be as specified by the person seeking the processing services. DOE will permit a person to combine its batch with those of other persons in order to avoid the full impact of the minimum charge for handling a small batch size. Persons must notify DOE of their intent to combine batches prior to the delivery of any reactor materials to be included in a proposed batch. Specific arrangements must include a formula for distributing the processing charges and other settlement factors associated with delivery of the reactor materials to DOE.

12. DOE has the option of compensating the reactor operator for enriched uranium recovered in the processing of reactor materials delivered to DOE facilities in accordance with the appropriate DOE-published price schedule for enriched uranium material. Such compensation by DOE will consist of providing materials or services of equivalent value. DOE will, thereby, acquire title to the uranium for which it provides compensation. DOE will also acquire title, without cost, to all waste and other materials contained in the reactor materials.

The enriched uranium recovered in processing reactor materials (or its equivalent) delivered to DOE facilities and not compensated for by DOE, shall be returned to the reactor operator. Enriched uranium will be returned to the reactor operator f.o.b. the DOE processing site, in a reactor-operator furnished cask suitable for shipment offsite.

13. In lieu of processing uranium-zirconium hydride fuel types, DOE will agree to provide disposition services for such fuels. In this case, no compensation for recovered uranium will be made. Research reactor operators may prefer to write off the value of uranium contained in the fuel and accept this service. Additional information concerning DOE's disposition service may be obtained from the Manager, Idaho Operations Office, U.S. Department of Energy, 785 DOE Place, Idaho Falls, Idaho 83402.

Dated: January 3,1986.

Sylvester R. Foley, Jr.,
Assistant Secretary for Defense Programs.

[FR Doc. 86-3452 Filed 2-14-86; 8:45 am]
BILLING CODE 6450-01-M

Receipt and Financial Settlement Provisions for Nuclear Research Reactor Fuels

Correction

In FR Doc. 86-3452 beginning on page 5754 in the issue of Tuesday, February 18, 1986, make the following correction:

On page 5755, in the middle column, in paragraph 6, subparagraph b was omitted. Paragraph 6 is corrected to read as follows:

6. DOE's commitment to provide fuel receipt and financial settlement services will terminate on the following dates:

a. For research reactor fuels described in 4.a. and 4.c. - December 31, 1987; and

b. For research reactor fuels described in 4.b. - December 31, 1992.

DEPARTMENT OF ENERGY

Receipt and Financial Settlement Provisions for Nuclear Research Reactor Fuels

AGENCY: Department of Energy.

ACTION: Notice

SUMMARY: The Department of Energy is amending the provisions of its current policy providing for the receipt and financial settlement of U.S.-origin spent research reactor fuels by extending the date by which it will receive highly enriched uranium (HEU) fuels to December 31, 1988.

FOR FURTHER INFORMATION CONTACT: Louis R. Willett, Office of Nuclear Materials Production, DP-133.2-GTN, U.S. Department of Energy, Washington, D.C. 20545, 301/353-3968.

SUPPLEMENTARY INFORMATION: On November 9, 1982, the Department of Energy announced in the **Federal Register** that it was extending until December 31, 1987, its policy for the receipt and financial settlement of U.S.-origin spent research reactor fuels (47 FR 50737). It was determined at that time that there was a continued need in the research reactor community for a fuel return capability and that the U.S. interests in limiting worldwide inventories of HEU were served by an extension of the policy. This extension was restated, without change, in a February 1986 **Federal Register** notice that expanded DOE's fuel receipt and financial settlement provisions to include low enriched uranium research reactor fuels (51 FR 5754).

DOE has determined that this need still exists and that once again it is in the best interest of the United States to extend the effective date for the receipt and financial settlement for HEU research reactor fuels of U.S. origin. The Department has reviewed the policy extension under the National Environmental Policy Act (NEPA) and has found that the extension itself clearly has no significant impact. Exports of or subsequent arrangements involving nuclear materials are reviewed by DOE on a case-by-case basis in accordance with the Guidelines for Implementing Executive Order 12114, Environmental Effects Abroad of Major Federal Actions, and NEPA. In 1987, DOE initiated studies, including a study of the potential cummulative environmental effects, to determine the impact of a 10-year extension of this policy on DOE programs. These studies are ongoing and have identified a number of important issues that must be resolved prior to extending the provisions of this policy for the long term.

To provide for continuation of beneficial research reactor programs and to permit the additional time required for DOE to complete its review of a 10-year extension of this policy, DOE is amending its fuel receipt and financial settlement provisions by extending the effective date for receipt of U.S.-origin HEU research reactor fuels to December 31, 1988. To provide for this extension, the following amendment to the **Federal Register** notice entitled "Receipt and Financial Settlement Provisions for Nuclear Research Reactor Fuels," 51 FR 5754, published February 18, 1986, and as corrected on March 4, 1986 (51 FR 7487), is made:

1. Delete paragraph 6.a. and substitute in its place:

"a. For research reactor fuels described in 4.a. and 4.c.--December 31, 1988."

Troy E. Wade II,
Acting Assistant Secretary for Defense Programs.

[FR Doc. 87-29918 Filed 12-29-87; 8:45 am]

BILLING CODE 6450-01-M