IAEA Nuclear Energy Series

No. NF-T-2.7

Basic
Principles

Objectives

Guides

Technical
Reports

Impact of Fuel Density on Performance and Economy of Research Reactors

IAEA
International Atomic Energy Agency

IAEA NUCLEAR ENERGY SERIES PUBLICATIONS

STRUCTURE OF THE IAEA NUCLEAR ENERGY SERIES

Under the terms of Articles III.A.3 and VIII.C of its Statute, the IAEA is authorized to “foster the exchange of scientific and technical information on the peaceful uses of atomic energy”. The publications in the **IAEA Nuclear Energy Series** present good practices and advances in technology, as well as practical examples and experience in the areas of nuclear reactors, the nuclear fuel cycle, radioactive waste management and decommissioning, and on general issues relevant to nuclear energy. The **IAEA Nuclear Energy Series** is structured into four levels:

(1) The **Nuclear Energy Basic Principles** publication describes the rationale and vision for the peaceful uses of nuclear energy.
(2) **Nuclear Energy Series Objectives** publications describe what needs to be considered and the specific goals to be achieved in the subject areas at different stages of implementation.
(3) **Nuclear Energy Series Guides and Methodologies** provide high level guidance or methods on how to achieve the objectives related to the various topics and areas involving the peaceful uses of nuclear energy.
(4) **Nuclear Energy Series Technical Reports** provide additional, more detailed information on activities relating to topics explored in the **IAEA Nuclear Energy Series**.

The IAEA Nuclear Energy Series publications are coded as follows: **NG** – nuclear energy general; **NR** – nuclear reactors (formerly **NP –** nuclear power); **NF** – nuclear fuel cycle; **NW** – radioactive waste management and decommissioning. In addition, the publications are available in English on the IAEA web site:

www.iaea.org/publications

For further information, please contact the IAEA at Vienna International Centre, PO Box 100, 1400 Vienna, Austria.

All users of the IAEA Nuclear Energy Series publications are invited to inform the IAEA of their experience for the purpose of ensuring that they continue to meet user needs. Information may be provided via the IAEA web site, by post, or by email to Official.Mail@iaea.org.

IMPACT OF FUEL DENSITY ON PERFORMANCE AND ECONOMY OF RESEARCH REACTORS

The following States are Members of the International Atomic Energy Agency:

AFGHANISTAN
ALBANIA
ALGERIA
ANGOLA
ANTIGUA AND BARBUDA
ARGENTINA
ARMENIA
AUSTRALIA
AUSTRIA
AZERBAIJAN
BAHAMAS
BAHRAIN
BANGLADESH
BARBADOS
BELARUS
BELGIUM
BELIZE
BENIN
BOLIVIA, PLURINATIONAL STATE OF
BOSNIA AND HERZEGOVINA
BOTSWANA
BRAZIL
BRUNEI DARUSSALAM
BULGARIA
BURKINA FASO
BURUNDI
CAMBODIA
CAMEROON
CANADA
CENTRAL AFRICAN REPUBLIC
CHAD
CHILE
CHINA
COLOMBIA
COMOROS
CONGO
COSTA RICA
CÔTE D'IVOIRE
CROATIA
CUBA
CYPRUS
CZECH REPUBLIC
DEMOCRATIC REPUBLIC OF THE CONGO
DENMARK
DJIBOUTI
DOMINICA
DOMINICAN REPUBLIC
ECUADOR
EGYPT
EL SALVADOR
ERITREA
ESTONIA
ESWATINI
ETHIOPIA
FIJI
FINLAND
FRANCE
GABON
GEORGIA
GERMANY
GHANA
GREECE
GRENADA
GUATEMALA
GUYANA
HAITI
HOLY SEE
HONDURAS
HUNGARY
ICELAND
INDIA
INDONESIA
IRAN, ISLAMIC REPUBLIC OF
IRAQ
IRELAND
ISRAEL
ITALY
JAMAICA
JAPAN
JORDAN
KAZAKHSTAN
KENYA
KOREA, REPUBLIC OF
KUWAIT
KYRGYZSTAN
LAO PEOPLE'S DEMOCRATIC REPUBLIC
LATVIA
LEBANON
LESOTHO
LIBERIA
LIBYA
LIECHTENSTEIN
LITHUANIA
LUXEMBOURG
MADAGASCAR
MALAWI
MALAYSIA
MALI
MALTA
MARSHALL ISLANDS
MAURITANIA
MAURITIUS
MEXICO
MONACO
MONGOLIA
MONTENEGRO
MOROCCO
MOZAMBIQUE
MYANMAR
NAMIBIA
NEPAL
NETHERLANDS
NEW ZEALAND
NICARAGUA
NIGER
NIGERIA
NORTH MACEDONIA
NORWAY
OMAN
PAKISTAN
PALAU
PANAMA
PAPUA NEW GUINEA
PARAGUAY
PERU
PHILIPPINES
POLAND
PORTUGAL
QATAR
REPUBLIC OF MOLDOVA
ROMANIA
RUSSIAN FEDERATION
RWANDA
SAINT LUCIA
SAINT VINCENT AND THE GRENADINES
SAN MARINO
SAUDI ARABIA
SENEGAL
SERBIA
SEYCHELLES
SIERRA LEONE
SINGAPORE
SLOVAKIA
SLOVENIA
SOUTH AFRICA
SPAIN
SRI LANKA
SUDAN
SWEDEN
SWITZERLAND
SYRIAN ARAB REPUBLIC
TAJIKISTAN
THAILAND
TOGO
TRINIDAD AND TOBAGO
TUNISIA
TURKEY
TURKMENISTAN
UGANDA
UKRAINE
UNITED ARAB EMIRATES
UNITED KINGDOM OF GREAT BRITAIN AND NORTHERN IRELAND
UNITED REPUBLIC OF TANZANIA
UNITED STATES OF AMERICA
URUGUAY
UZBEKISTAN
VANUATU
VENEZUELA, BOLIVARIAN REPUBLIC OF
VIET NAM
YEMEN
ZAMBIA
ZIMBABWE

The Agency's Statute was approved on 23 October 1956 by the Conference on the Statute of the IAEA held at United Nations Headquarters, New York; it entered into force on 29 July 1957. The Headquarters of the Agency are situated in Vienna. Its principal objective is "to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world''.

IAEA NUCLEAR ENERGY SERIES No. NF-T-2.7

IMPACT OF FUEL DENSITY ON PERFORMANCE AND ECONOMY OF RESEARCH REACTORS

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 2021

Marketing and Sales Unit, Publishing Section
International Atomic Energy Agency
Vienna International Centre
PO Box 100
1400 Vienna, Austria
fax: +43 1 26007 22529
tel.: +43 1 2600 22417
email: sales.publications@iaea.org
www.iaea.org/publications

Printed by the IAEA in Austria
April 2021
STI/PUB/1917

IAEA Library Cataloguing in Publication Data

Names: International Atomic Energy Agency.
Title: Impact of fuel density on performance and economy of research reactors / International Atomic Energy Agency.
Description: Vienna : International Atomic Energy Agency, 2021. | Series: IAEA nuclear energy series, ISSN 1995–7807 ; no. NF-T-2.7 | Includes bibliographical references.
Identifiers: IAEAL 20-01379 | ISBN 978–92–0–120320–5 (paperback : alk. paper) | ISBN 978–92–0–120420–2 (pdf) | ISBN 978–92–0–120520–9 (epub) | ISBN 978–92–0–120620–6 (mobipocket)
Subjects: LCSH: Uranium as fuel — Quality control. | Nuclear fuels. | Density. | Nuclear reactors.
Classification: UDC 621.039.543.4 | STI/PUB/1917

FOREWORD

The IAEA's statutory role is to "seek to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world". Among other functions, the IAEA is authorized to "foster the exchange of scientific and technical information on peaceful uses of atomic energy". One way this is achieved is through a range of technical publications including the IAEA Nuclear Energy Series.

The IAEA Nuclear Energy Series comprises publications designed to further the use of nuclear technologies in support of sustainable development, to advance nuclear science and technology, catalyse innovation and build capacity to support the existing and expanded use of nuclear power and nuclear science applications. The publications include information covering all policy, technological and management aspects of the definition and implementation of activities involving the peaceful use of nuclear technology.

The IAEA safety standards establish fundamental principles, requirements and recommendations to ensure nuclear safety and serve as a global reference for protecting people and the environment from harmful effects of ionizing radiation.

When IAEA Nuclear Energy Series publications address safety, it is ensured that the IAEA safety standards are referred to as the current boundary conditions for the application of nuclear technology.

This publication considers the major impacts of using higher density uranium fuel on research reactor performance and economy. Owing to fuel cycle and fuel and reactor performance complexities, it is difficult for potential users of high density fuel to clearly understand its possible effects in research reactors, for example when high density uranium–molybdenum (U–Mo) fuel is used instead of U_3Si_2 fuel, which has a typical density of 4.8 gU/cm^3. Several studies of the impacts of changing to high density fuel are available for generic or specific existing research reactors, but an overview is needed of the potential benefits and limitations. Thus, this publication discusses the implications of using such fuel for research reactor irradiation performance and safety, as well as the economic impacts of changes in annual fuel consumption. A preliminary evaluation of the potential impact on fuel cost includes possible changes in the cost of fuel manufacturing and spent fuel management for high density fuel.

In view of the potential advantages of using higher density fuels in research reactors, in April 2013 the IAEA hosted a consultancy meeting to discuss the possible impacts. The participants studied the implications of fuel density for research reactor performance and parts of the fuel cycle. Several major considerations and case studies related to economy and performance were then presented and discussed in detail during a second consultancy meeting in December 2013. The present publication provides an overview of the topic based on those studies.

The IAEA wishes to thank all those who participated in the preparation and review of this publication, in particular Ho Jin Ryu (Republic of Korea) and K. Shropshire (United States of America). The IAEA officers responsible for this publication were F. Marshall and P. Adelfang of the Division of Nuclear Fuel Cycle and Waste Technology.

EDITORIAL NOTE

CONTENTS

1. INTRODUCTION 1

1.1. Background 1
1.2. Objective 2
1.3. Scope 2
1.4. Structure 3

2. PERFORMANCE IMPACT OF INCREASING FUEL DENSITY 6

2.1. Performance of irradiation facilities 6
2.2. Reactivity. 7
2.3. Potential limitations and engineering measures 8
2.4. Safety parameters 9

3. ECONOMIC IMPACT OF INCREASING FUEL DENSITY 11

3.1. Fuel consumption 11
3.2. Preliminary cost evaluation 13

4. CONCLUSIONS 19

REFERENCES 19

ANNEX I: PRELIMINARY ANALYTICAL INVESTIGATION OF HIGH DENSITY U–Mo FUEL: INTRODUCTION TO THE JMTR 21

ANNEX II: COMPARATIVE STUDY OF FUEL MATERIALS FOR THE KIJANG RESEARCH REACTOR 35

ANNEX III: CONVERSION OF OPAL FROM U_3Si_2 TO HIGH DENSITY U–Mo FUEL 44

ANNEX IV: RP-10 RESEARCH REACTOR CONVERSION ANALYSIS. 54

ANNEX V: CONVERSION OF GENERIC REACTORS FROM U_3Si_2 TO URANIUM–MOLYBDENUM (U–Mo) 61

ANNEX VI: COMPARATIVE STUDY ON FUEL DENSITY FOR THE AHR 73

ABBREVIATIONS 81
CONTRIBUTORS TO DRAFTING AND REVIEW 83
STRUCTURE OF THE IAEA NUCLEAR ENERGY SERIES 84

1. INTRODUCTION

This publication summarizes the major impacts of using high density uranium fuel on research reactor performance and economy. Changing to higher density fuel can lead to extended reactor fuel cycle length and greater discharge burnup. This publication provides an overview of the potential benefits and limitations for research reactor performance and the economic impacts in terms of the estimated changes in fuel consumption. The proposed methodology for such estimations may be applied to a specific reactor and fuel to obtain numeric results.

The annexes accompanying this publication are case studies of the impact of high density fuels on generic and existing research reactors; they serve as the basis of the discussion in the main body of the publication. The results of such studies are strongly dependent on the specific reactor design; however, several common attributes can be identified in order to estimate the impacts of high density fuel on reactor performance and economics. Accordingly, each case study is assigned to one of four scenarios representing reactor modifications to accommodate and optimize performance with the new fuel. The four representative scenarios do not cover all possible situations; the design characteristics, performance goals and constraints differ from one research reactor design to another. Any specific reactor must be investigated individually to evaluate fuel consumption, irradiation facility performance and safety parameters when a change to a higher density fuel is proposed, but the annexes may be used as examples to undertake such an investigation.

1.1. BACKGROUND

Uranium–molybdenum (U–Mo) alloy fuel exhibits excellent irradiation stability and offers a higher uranium density than the widely used low enriched uranium (LEU) fuel U_3Si_2. Therefore, U–Mo is considered to be a promising candidate for conversion of highly enriched uranium (HEU) cores to LEU cores in research and test reactors. In addition, high density U–Mo fuels may also offer economic benefits to research reactors already operating with currently qualified and commercially available LEU fuel. U–Mo fuels offer the prospect of higher burnup, potentially reducing the number of fuel assemblies (FAs) to be procured.

High density U–Mo fuels are still under development and will be available following the completion of qualification testing and regulatory review. Development of U–Mo dispersion fuel began in 1996 and is still ongoing because qualification of plate type U–Mo fuel has not yet been completed under the high power conditions required for high performance research and test reactors. Numerous high power irradiation tests of U–Mo dispersion fuel have been performed, primarily in the United States of America (USA) and Europe: the Reduced Enrichment for Research and Test Reactors (RERTR) programme and the Advanced Test Reactor (ATR) full size plate in centre flux trap position (AFIP) tests at the ATR in the USA; FUTURE, E-FUTURE and SELENIUM tests at the Belgian Reactor 2 (BR2) in Belgium; internal rotary inspection system (IRIS) tests at the Osiris reactor in France; and UMUS tests at the High Flux Reactor (HFR) in the Netherlands. Some of these tests resulted in fuel failure at high burnup or high fission rates. Despite this, fuels operating in the safe zone below the burnup and fission rates of the observed failures are expected to be qualified without significant problems.[1]

Owing to the complex characteristics of various fuel cycle factors, potential users of the high density fuel find it difficult to clearly understand what benefits they can expect, for example, when 8.0 gU/cm^3 U–Mo fuel is used instead of 4.8 gU/cm^3 U_3Si_2 fuel. Therefore, the IAEA formed a consultancy group

[1] Of interest is the Jules Horowitz research reactor currently under construction in France and projected to achieve its first criticality in 2021. Its fuel design basis was originally high density LEU U–Mo fuel, but this was replaced with neutronically equivalent (27% enriched) U_3Si_2 until the U–Mo fuel is qualified. The same fuel and plate thickness and hydraulic gap geometry were maintained to allow direct fuel meat replacement in the design.

to study the implications of fuel density for research reactor performance and portions of the fuel cycle. The consultancy group collected studies of uranium density impacts on the economy and performance of research reactor LEU fuel. The following general discussion and comparisons are case studies of existing and generic reactors, including the Open Pool Australian Lightwater (OPAL) reactor in Australia, the Japan Materials Testing Reactor (JMTR) in Japan, the Kijang Research Reactor (KJRR) and the Advanced HANARO Reactor (AHR) based on the High-Flux Advanced Neutron Application Reactor (HANARO) in the Republic of Korea, and the RP-10 research reactor in Peru.

1.2. OBJECTIVE

This publication offers a preliminary evaluation of the performance and cost impacts of using high density fuel in a research reactor. The objective is not to make detailed design modification and economic assessments, but to estimate the general impact of increasing fuel density on the performance and cost of the reactor fuel cycle. Cost is estimated based on FA consumption, and only for replacing fuel meat with higher density fuel, not for fuel qualification, potential FA or core redesign or implementation of power upgrades. Costs are also not estimated for interaction with other reactor areas or with regulatory bodies.

This publication compiles the results of the case studies provided in the annexes, summarizes the major considerations of the impact of increased fuel density on reactor performance and fuel consumption, and offers a simple evaluation of cost savings, based on comparative fuel consumption, from using high density fuel.

1.3. SCOPE

To help decide whether to use high density fuel in a research reactor, potential benefits and drawbacks need to be clearly identified. Of course, performance and economic benefits are sought, but with a desired minimum amount of related redesign and work to execute the fuel change. The following issues are worth considering:

- Compliance with all design criteria related to safety must be maintained or improved. Thus, the main safety constraints of the original reactor have to be clearly understood in order to determine whether the change to a high density fuel is viable.
- Ideally, the fuel change is executed with minimal or no modifications to related reactor systems (e.g. detectors, pumps, piping). This will minimize required procedure changes, safety re-evaluations and interactions with regulatory bodies.
- Changes in operating modes and use of the reactor and irradiation facilities need to be planned and understood. The reactor core, experiments, targets and FAs might require redesign.
- The proposed high density fuel must have qualified performance for burnup and average fission rate, such that failure-free operation is assured within the required reactor operating range.
- Potential constraints related to the fuel should be identified. For example, a tight limit in maximum allowable fuel swelling or oxide layer growth on the cladding might diminish the potential benefits.
- A stable fuel source needs to be identified and established.
- The fuel cycle back end (i.e. spent fuel storage and disposition) needs to be understood and planned.

This publication focuses on the possible use of high density fuel in research reactors, without preferentially promoting the use of any single high density fuel. Evaluations involving monolithic fuel versus dispersion fuel, or HEU–LEU conversion, are not included in this study so that the emphasis is placed solely on the impact of changes in fuel density. Guidance provided here, describing good practices, represents expert opinion but does not constitute recommendations made on the basis of a consensus of Member States.

1.4. STRUCTURE

Because the potential benefits of changing a reactor's fuel from standard LEU fuel to high density LEU fuel are not always clear, several analyses need to be conducted to estimate the impact of the fuel change on areas related to reactor performance (including safety) and operation economy. These are addressed in this publication:

(a) Performance. The impact of the fuel change on the most important irradiation facilities of the reactor (both in-core and ex-core) is analysed to recognize whether irradiation performance will be decreased, maintained or improved. In addition, the effect on safety parameters — such as the power peaking factor (PPF), shutdown margins, reactivity coefficients and kinetic parameters — must be evaluated to ensure that all design and safety criteria are satisfied.
(b) Economy. The overall cost aspects of the research reactor fuel cycle are discussed in Ref. [1], whereas this publication focuses on the impacts of changing to high density fuel. Given that uranium fuel costs vary over time, cost estimates are not made in terms of absolute fuel cost, but as comparative fuel consumption. Fuel consumption before and after the change is compared, in terms of both the number of FAs and the uranium consumption per year, to assess whether there is potential economic benefit in the change.

The results of these evaluations are strongly dependent on the research reactor design. Nevertheless, for a given type of research reactor that will be changed to high density fuel, several significant common attributes can be identified to estimate the effect in the above areas.

Accordingly, this publication considers the key potential benefits and constraints associated with a change to a high density fuel for several research reactor designs, including case studies as both annexes and references, where the impacts on reactor performance and economy are investigated. Each case study is intended to represent a unique combination of research reactor main attributes, design criteria, design constraints and performance goals. The underlying intent of using these case studies is that the primary potential impacts, benefits and constraints of changing to a higher density fuel can be identified for a wide range of research reactor designs. This publication assigns the case studies to one of four representative scenarios:

(1) Direct replacement of fuel meat;
(2) Fuel assembly redesign;
(3) Changes to core;
(4) Change in reactor power.

Even though irradiation facility performance typically only improves for scenarios 3 and 4, FA consumption can decrease for all scenarios. Each new FA will have more uranium because of the higher density fuel, but if the number of FAs is reduced sufficiently, total uranium consumption could also be reduced, therefore potentially reducing overall fuel cost.

1.4.1. Scenario 1: Direct replacement of fuel meat

For a research reactor with a fixed number of FAs, where FA redesign and power upgrade are not required, a change of the original fuel zone, or fuel meat, within the central zone in the FA plate (or rod for some designs) can be proposed. FA design and core layout may not be optimal for the density level of the fuel meat, but redesigns of FAs and core are not expected in this scenario. Increased uranium loading will allow higher discharge burnup or lower FA consumption, but irradiation facility performance usually cannot be improved. Furthermore, even though the original FA design might not be optimal for the higher uranium loading, the impact on other areas — thermohydraulic, safety analysis, instrumentation and control, and so on — is minimized when only a direct replacement of the fuel meat is performed.

1.4.2. Scenario 2: Fuel assembly redesign

To optimize the performance of the high density fuel, an FA redesign may be proposed, while still maintaining a fixed core size and design with no power upgrade. The FA redesign might be proposed in order to improve the moderation ratio (i.e. the ratio between fuel and moderator) of the high density fuel, and thus take advantage of the potential economic or performance benefits of higher uranium loading. FA redesign can also be required in cases where a decrease in parameters such as the PPF is needed. This scenario will lead to greater interaction with other design areas, particularly the thermohydraulic and safety analysis areas; irradiation facility performance usually cannot be improved.

1.4.3. Scenario 3: Changes to core

If the fuel meat is replaced with higher density uranium, which might also involve FA redesign, the total number of FAs inside the core might be reduced. For a research reactor with a fixed core, some FAs can be replaced with in-core experiment irradiation facilities, while for a configurable core, a more compact configuration can be proposed. For this scenario, irradiation facility performance can usually be improved.

1.4.4. Scenario 4: Change in reactor power

Any of the above scenarios might be accompanied with an upgrade for higher reactor power to take greater advantage of the high density fuel.

1.4.5. Annexes

Note that these representative scenarios do not cover all possible situations; design characteristics, performance goals and constraints differ from one research reactor design to another, so any specific reactor needs to be investigated individually to evaluate fuel consumption, irradiation facility performance and safety parameters when a change to a higher density fuel is proposed. The case studies in the annexes present an interesting range of circumstances encountered when changing to high density fuel in research reactors.

Annex I considers the JMTR reactor, which has already been designed for optimal performance with U_3Si_2 fuel, and with a tightly defined limit on maximum discharge burnup. Direct replacement (scenario 1) of the fuel meat with a higher density U–Mo fuel only offers benefits if the U–Mo fuels can offer even higher burnup than the set limit. Otherwise, the fuel change does not provide an advantage regarding irradiation performance or fuel consumption, and it does not satisfy the reactor performance or fuel design criteria.

Annex II presents the evaluation of a backup fuel for the new KJRR. This reactor has been designed for optimal performance with a U–Mo fuel that combines 6.5 and 8.0 gU/cm^3 densities. However, this case study investigates the use of U_3Si_2 fuel as a backup in case U–Mo fuel is not available when the reactor starts up. The circumstances presented in this case study correspond to scenario 1 or scenario 4, and show the clear economic advantages that design basis U–Mo fuel offers.

Annex III studies the possible replacement of U_3Si_2 fuel in the OPAL reactor by U–Mo fuel meat (scenario 1) with densities ranging from 6 to 8 gU/cm^3. Irradiation facility fluxes and reactor safety parameters are adversely affected, but remain within acceptable limits. The significant reduction in fuel consumption is a clear benefit.

Annex IV evaluates replacement of the current U_3O_8 fuel with higher density U_3Si_2 fuel in the Peruvian RP-10 research reactor via FA redesign and a configurable core (scenario 3). Irradiation performance and fuel consumption are both significantly improved; the latter is illustrated with a simple relative cost comparison.

Annex V provides a generic evaluation, assuming a heavy water reflected reactor based on the OPAL core and fuel geometry. It considers four cases in which U_3Si_2 fuel (reference fuel) is replaced with U–Mo fuel of either 7 gU/cm^3 or 16.4 gU/cm^3 density. For each density, a case with direct replacement of the fuel meat (scenario 1) and a case with FA redesign (scenario 2) are considered, and the thicknesses of fuel meat, cladding, plate and water channel are varied to maintain a constant total thickness dimension of the plate–water channel; all other FA dimensions are held constant. In these four cases, fuel consumption is reduced, and irradiation facility performance is acceptable for all except for the highest density in scenario 1. A simple relative cost comparison is offered.

Annex VI presents the new AHR research reactor currently under development using U_3Si_2 fuel of 4.0 gU/cm^3. This study investigates cases that replace the fuel meat (scenario 1) with U–Mo, with a density of either 4.5 gU/cm^3, 6 gU/cm^3 or a combination of these two. Of these, the case using the highest density U–Mo does not meet some performance criteria and so might require an FA redesign. All cases decrease thermal flux levels to some extent, but also significantly reduce fuel consumption due to longer cycle lengths.

Table 1 summarizes the general expected impacts of changing to higher density fuel in the above scenarios and identifies the associated annex case studies.

TABLE 1. EXPECTED PERFORMANCE IMPACTS OF CHANGING TO HIGH DENSITY FUEL IN A RESEARCH REACTOR

		Scenario 1	Scenario 2	Scenario 3	Scenario 4
Description		Direct replacement of fuel meat	Fuel assembly redesign	Changes to core	Change in reactor power
Expected impacts	Interaction with other areas	Low	High	High	High
	Interaction with regulatory bodies	Low	High	High	High
	Optimal FA design	Possibly no	Yes	Possibly yes	Possibly yes
Potential benefits[a]	Minimize FA consumption	Yes	Yes	Yes	Yes
	Minimize U consumption	Yes	Yes	Yes	Yes
	Irradiation facilities performance	Maintained	Maintained	Improved	Improved
Annex case studies		I (JMTR) II (KJRR) III (OPAL) V (generic reactor) VI (AHR)	V (generic reactor)	IV (RP-10)	II (KJRR)

[a] General benefits based on evaluation of irradiation performance and economy of fuel or U consumption.

2. PERFORMANCE IMPACT OF INCREASING FUEL DENSITY

Although high density U–Mo fuel may offer significant economic benefits, it is not always clear what the impact on reactor performance might be. For example, effects on irradiation facility performance can vary depending on fuel and core configurations. On the other hand, high density fuels — and the potential FA and core redesigns necessary to accommodate them — might lead to increases in available reactivity and power density, and so the compliance with design criteria and the impacts on safety parameters must be evaluated. These potential impacts are discussed below.

Regarding the latter example, using a high density fuel might cause potential problems related to discharge burnup, average irradiation time, fuel swelling, oxide layer growth, shutdown margins, the PPF, reactivity coefficients or kinetic parameters. To manage these problems, the designer should include (or modify) burnable poisons and develop suitable fuel management strategies.

2.1. PERFORMANCE OF IRRADIATION FACILITIES

This section of the publication discusses how a change to high density fuel affects neutron flux in a research reactor's main irradiation facilities. If higher power densities are achievable with high density fuels, several performance parameters of the research reactor can be improved, for example, flux level and radioisotope production. Or, if the power density is not raised and the number of FAs in the core is fixed, a higher discharge burnup can be achieved while maintaining the original neutron flux performance in the main irradiation facilities and consequently lowering the fuel consumption. Accordingly, the four scenarios can be used to frame evaluation of irradiation facilities' performance.

2.1.1. With direct replacement of fuel meat

If only the fuel meat needs to be changed, with no required FA or core redesigns and no power upgrade, irradiation facility performance can be largely maintained along with improved fuel consumption. Examples are given in Annexes II, III, V and VI, where for a fixed core, the performances of the irradiation facilities can be maintained or only slightly reduced while fuel consumption is significantly improved.

The case described in Annex II is that of a reactor core specifically designed to successfully optimize the use of U–Mo fuel. Compared with its backup U_3Si_2 fuel option, the U–Mo fuel provides about 8% lower flux at the core region but significantly lower fuel and uranium consumption.

Annexes I and V offer case studies in which fuel consumption cannot be improved, and as a result, either FA or core redesign is necessary.

2.1.2. With fuel assembly redesign

Other reactors require FA redesign but no core redesign and no power upgrade. This scenario can maintain irradiation facility performance with improved fuel consumption, as illustrated in Annex V.

2.1.3. With changes to core

For configurable core reactors, the impact of high density fuels on irradiation facility performance depends on the core design.

In this scenario of changed core — with or without FA redesign — but no power upgrade, the use of high density fuel allows a reduction of the number of FAs in the core in order to (a) develop a more compact core, or (b) add an in-core irradiation facility by replacing an FA in the core. Thus, irradiation facility performance can be improved along with fuel consumption.

For example, Refs [2, 3] present some cases where an improvement is obtained in the flux performances and other cases where there is deterioration in some irradiation positions. Annex IV presents a case where fuel and uranium consumption is reduced and irradiation performance is clearly improved with the change to higher density fuel (from U_3O_8 to U_3Si_2), primarily because a more compact core configuration has been obtained.

2.1.4. With a change in the reactor power

If reactor power is also increased with any of the above scenarios, irradiation facility performance can be improved. This can be achieved for cores with both configurable and fixed numbers of FAs because higher fuel densities offer the possibility, if needed, of reducing fuel plate thickness and increasing the number of fuel plates. The overall power increase occurring in this scenario will lead to higher fluxes in both in-core and ex-core facilities. To illustrate, Annex II and Refs [2, 3] present cases where the overall reactor power was optimized or changed. Several effects were observed with increases in irradiation facility fluxes in some cases and decreases in others.

To summarize, changing to high density fuel can potentially lead to irradiation facility improvements that are dependent on the research reactor considered. The following outcomes are possible:

— Scenarios 1 and 2 maintain irradiation facility performance and improve fuel consumption.
— Scenario 3 can improve irradiation facility performance either by replacing FAs with new in-core facilities or by reducing reactor core size in order to improve ex-core fluxes.
— Scenario 4 changes reactor power to improve performance of both in-core and ex-core irradiation facilities.

For all scenarios, it is necessary to consider the in-core irradiation facilities for neutron irradiation under thermal spectra, because higher density uranium fuels typically lead to hardening[2] of neutron spectra.

2.2. REACTIVITY

When a reactor is changed to high density uranium fuel, the desired effect is to increase the available reactivity. However, there is a corresponding decrease in the moderation ratio (i.e. the ratio between the fuel and moderator), which needs to be mitigated. So, depending on the research reactor design, benefits might not occur without a modification in FA design that helps to achieve acceptable moderation ratios [2].

As an example, Annex V illustrates this effect for very high density fuels, where analysis predicted a decrease in the available core reactivity if this moderation ratio was not corrected through FA redesign. Such redesign is typically achieved by increasing the number and decreasing the thickness of plates in the FA.

In addition, a competing effect occurs when high density fuels use compounds that include a component (such as molybdenum) with a higher capture cross-section than the components of the original fuel (usually an oxide or silicide compound). Because of this, a decrease in reactivity can be observed if the uranium loading is maintained as constant. This effect was analysed using a theoretical approach in Ref. [4], and is also observed in Annex I.

---

[2] Neutron spectral ‘hardening’ is an increase in the average energy of neutrons due to preferential loss at lower energies by absorption, leakage or scattering. The presence of more uranium in the high density fuel causes more loss of low energy neutrons.

2.3. POTENTIAL LIMITATIONS AND ENGINEERING MEASURES

Several potential limitations arise when a higher density fuel is introduced into a reactor with a core that already operates with LEU. For example, as previously discussed, higher density fuels might lead to increases in the available reactivity [5, 6]. Also, cores redesigned to be more compact can have higher power densities [3, 7]. Depending on the specific design of the research reactor being analysed, several constraints might arise regarding safety and technical issues. These constraints include the following possibilities:

- Higher initial uranium loads might reduce the capacity of control rods to shut down the reactor with appropriate margins.
- Changing to high density fuel can modify the average neutronic behaviour, with potential adverse impact on parameters relating to temporal response to transients, for example, kinetic parameters and feedback coefficients.
- Potential limitations might arise in cooling system capability to remove heat generated by the new power distribution from higher density fuel. Such limitations might be caused by an increased PPF in the fresh fuel or decreased thermohydraulic margins.
- When high density fuels operate to increased levels of discharge burnup, other related phenomena are impacted, such as fuel swelling or oxide layer growth[3]. These phenomena can decrease the hydraulic gap between fuel plates and might inhibit convective and conductive fuel cooling and affect reactor fluid flow dynamics. This illustrates that changes in fuel design or burnup levels can impact several parameters in different ways, since the resulting fuel swelling and oxide layer growth both have complex dependences on other design and reactor operation aspects.
- When a reactor changes to higher density fuel, the fuel's average fission rate and burnup must be shown to be within the range at which it has been demonstrated that fuel failures do not occur, proven by experience over past decades.
- For several research reactors that already operate using LEU, additional constraints can be imposed by facility operators or regulatory bodies regarding the maximum allowable burnup. These constraints might impose a strong limitation for changing to high density fuel in these reactors.

The impact of a reactor performance issue depends on the reactor design. Nevertheless, the resulting potential constraints can be addressed through several methods, including, but not limited to the following ones:

- Fuel management modifications;
- Introduction or redesign of neutron absorbing poisons[4];
- FA redesign (primarily modification of fuel plate design);
- Core redesign, including possible modification of control rod positions in the core (for configurable reactors).

For most of the cases analysed, a combination of these methods enables the change to high density fuels while satisfying all design and safety criteria. Particular attention should then be given to cases where additional constraints arise, such as the maximum allowed fuel swelling, oxide layer growth or discharge burnup.

---

[3] Methods to limit or accommodate fuel swelling or oxide growth should be considered as part of changing to any new fuel type. Discussions of such methods and these phenomena are outside the scope of this publication.

[4] Various options are available for burnable and non-burnable neutron poison material, including boron, cadmium, gadolinium and hafnium. Such might be included in reactor coolant solution or in FAs, control rods or other reactor components. Details of options, comparisons, selection criteria and designs are beyond the scope of this publication.

For example, Annex I presents a case where the maximum discharge burnup has a tight limit that has already been optimized for the original core operating with LEU. As a result, the change to higher density fuel does not satisfy the design criteria, so in this case the higher excess reactivity available from the high density U–Mo fuel cannot be used to increase discharge burnup, and consequently, it does not provide an advantage with regard to irradiation performance or fuel consumption.

2.4. SAFETY PARAMETERS

Regardless of the reactor design investigated, the key safety parameters that must be considered when a high density fuelled core is analysed are shutdown margin, PPF, feedback coefficients and kinetic parameters. Each of these parameters is discussed separately below. The impact on each safety parameter should be considered, not only with regard to the original reactor design, but also the main design criteria constraints.

2.4.1. Shutdown margins

The shutdown margin is the instantaneous amount of reactivity by which the reactor is subcritical from its present condition when all full length shutdown and control rods are fully inserted except for the single rod cluster assembly of highest reactivity worth, which is assumed to be fully withdrawn.

The expected effect of changing to high density fuel is to increase the overall uranium load and, by doing so, to increase excess reactivity in the core. If other design parameters are not altered, an expected additional impact is to diminish control rod reactivity worth[5] due to the lower neutron importance[6] of control rods in the core. The combination of these two effects will reduce the shutdown margin and might preclude conformance with design criteria related to shutdown margins. The fixed size core case presented in Annex I illustrates an example of this effect, while an in depth analysis regarding several FA uranium loads can be found in Ref. [6] as a theoretical benchmark case.

The main potential benefit of changing to high density fuel arises from optimization of the fuel management strategy and reactor fuel cycle length. In fact, the adverse effect of high density fuel on shutdown margins can be diminished if higher cycle lengths are successfully attained — thus diminishing excess reactivity from the one present at the beginning of the cycle — and fresh FAs are introduced in specific zones — thus enhancing the neutron importance of the control rods. With these considerations, the impact on control rod worth can be managed, as in the cases presented in Annexes II, III and VI. Furthermore, if the increase in the excess reactivity is desired and control rod redesign is not intended, the inclusion of burnable poisons (such as cadmium wires) in the FAs is an interesting alternative.

By contrast, when a configurable reactor (per scenario 3) is analysed, control rod position in the core grid can be altered; thus, control rod neutron importance and reactivity worth are altered. Such a case is illustrated in Annex IV.

To summarize, when the impact of higher density fuel on reactor shutdown margins is analysed, the following can be anticipated:

(a) Excess reactivity is increased, which deteriorates the shutdown margin (this effect is usually avoided by increasing the reactor fuel cycle length and thus diminishing the excess reactivity from that present at the beginning of the cycle, as well as strategically locating FAs in the reactor to enhance the neutron importance of the control rods).

---

[5] Control rod ‘reactivity worth’ is the amount that reactivity is changed through the control rod’s presence in a reactor core. Control rod insertion into the core decreases reactivity, and control rod removal allows the reactivity to then increase by that same amount.

[6] ‘Neutron importance’ indicates the change in neutron population of a critical reactor due to the introduction or removal of one neutron with specific location, energy and angle of momentum.

(b) Control rod neutron importance is changed (usually increased when the redesigned core is more compact or decreased when the number of FAs is fixed).
(c) Inclusion of burnable poisons can improve shutdown margins without control rod redesign.

2.4.2. Power peaking factor

The PPF is the ratio *a*/*b* of: *a*, the highest local power density (LPD) (i.e. the LPD at the hottest part of a fuel rod or plate in a reactor); and *b*, the average power density in the reactor core.

Generally, the PPF will increase when a reactor core is changed to high density fuel. This is clearly observed when the FA number in the core is maintained constant, as in the cases presented in Annexes II and III, where the total PPF increased by up to 20%. This adverse impact can be managed through implementing a suitable fuel management strategy and including (or modifying) burnable poisons in the FA.

But for cases where the reactor core configuration can be changed, such as in Refs [2, 3] and the case analysed in Annex IV, the PPF can be kept below the maximum allowed limits by using a combination of a suitable fuel management strategy and an appropriate reflector distribution. For example, the case presented in Annex VI controls the PPF by not placing FAs with high density fuel meat at the outer core region.

Finally, for cases where the inclusion of higher density fuel leads to unacceptable increases in the PPF, an FA or core redesign might be compulsory.

2.4.3. Reactivity coefficients

Reactivity coefficients, also called feedback coefficients, represent the effect of reactor operating conditions on reactivity in the core. Of multiple reactivity coefficients, the principal one is the temperature coefficient. This is the partial derivative of reactivity with respect to temperature, integrated across various temperatures which are complexly calculated for multiple components, materials, location and time effects (prompt or delayed); simplifying assumptions are usually made. Other feedback coefficients — which are also affected by temperature with complex calculations — include the power coefficient, the density coefficient and the Doppler (resonant absorption) coefficient.

If a direct replacement of LEU with a higher density fuel meat compound is made without an FA or core redesign, the neutron flux in the reactor hardens due to the change in the moderation ratio. This neutron flux hardening usually causes the water temperature and water density feedback coefficients to decrease, even though the Doppler feedback coefficient increases. This impact can be observed from a theoretical perspective in Ref. [8] for cases of different uranium loadings, and in a preliminary analysis at the cell (i.e. single FA) level in the high density scenario 1 fuel cases presented in Annex V. The latter analysis demonstrates significant changes in the feedback coefficients when the FA geometry is not modified, but only minor changes when the FA is properly redesigned (scenario 2). The results in Annex V also suggest that fuel management strategies and fuel burnup are important determinants of feedback coefficient values.

The cases presented in Annexes III and V demonstrate that for core level analyses, the change to a higher density fuel can be achieved without problems for the feedback coefficients, although FA redesign might be needed. Particular attention should be paid when the core geometry is changed, as in the case presented in Annex IV, because the effect on the feedback coefficients is not always straightforward.

2.4.4. Kinetic parameters

Kinetic parameters indicate the time related behaviour of neutrons. These parameters reflect how quickly reactivity changes, for example, during reactor startup or shutdown, and their values are important for reactor safety.

The kinetic parameters are significant in terms of the temporal behaviour of the reactor, and they have a key function in reactivity insertion accidents. When a reactor is changed to high density fuel, the following impacts on kinetic parameters are possible:

— If direct replacement of fuel meat with a higher density compound without FA or core redesign is considered (scenario 1), neutron flux in the reactor hardens due to the changed moderation ratio. In that case, both the effective delayed neutron fraction[7] and the prompt neutron generation time[8] can be significantly reduced [6].
— If the replacement leads to higher fuel burnup in any scenario, the increased plutonium content in the core can lead to reduction of the effective delayed neutron fraction.
— If the replacement leads to a more compact core and the reflector is changed as in scenario 3, the prompt neutron generation time can be reduced.

The two first impacts listed above can be observed from a theoretical perspective in Ref. [8], whereas the last impact depends on the core and reflector design.

In considering these possible impacts, it should be noted that the actual impact on kinetic parameters depends on FA and core designs. The cases presented in Annexes II, III, IV and V demonstrate that for reactor level analyses, the change to higher density fuel can be achieved with a low impact on the kinetic parameters and little or no impact on reactor transients. Although these parameters are important to consider, the actual effects are expected to be minor and will be evaluated for each proposed FA or core design modification.

3. ECONOMIC IMPACT OF INCREASING FUEL DENSITY

The cost aspects of the research reactor fuel cycle can be evaluated using the information and framework offered by Ref. [1]. This publication advances that discussion by specifically focusing on the impacts of changing to high density fuel.

This section describes the potential impacts a higher density fuel might have on fuel consumption. The change to a higher density fuel can lead to higher uranium loading in the core, which can improve both the reactor fuel cycle length and the burnup discharge, therefore obtaining potential economic benefits. The methodology presented here estimates cost savings, based on comparative fuel consumption, rather than absolute costs.

3.1. FUEL CONSUMPTION

A major potential economic benefit of increasing fuel density in research reactors is a favourable impact on fuel consumption. Since the 1990s, several studies have conceptually analysed fuel consumption in specific research reactors when higher density fuels were considered, using both theoretical approaches [4, 5] and practical analyses [2, 3, 7, 9]. These studies show that the use of high

[7] The ‘effective delayed neutron fraction’ is the ratio of the number of fissions caused by delayed neutrons to the total number of fissions caused by all neutrons (delayed plus prompt): the higher this parameter, the more stable reactor operation is.

[8] The ‘prompt neutron generation time’ is the average time between two generations of prompt neutrons in a critical reactor: the higher this parameter, the more stable reactor operation is.

density fuel might reduce the number of FAs required for reactor operation over a given period of time (e.g. a calendar year) and therefore reduce the number of fresh FAs to be procured and handled.

The direct effect of increasing uranium density in a given FA is increased uranium loading, which can lead to a more reactive configuration (see Ref. [6] and Annex I). Three beneficial options can then be implemented regarding reactor fuel consumption:

(1) Increasing the reactor fuel cycle length;
(2) Decreasing the number of FAs required per cycle;
(3) Combining these two options, where the cycle length can be increased, and the number of FAs required per cycle can be reduced compared with the original lower density fuel core.

For any of these options, the final potential economic savings will also depend on several fuel cycle aspects, including FA fabrication costs, uranium cost, fuel storage, transport and back end associated costs.

The performance impact of changing to high density fuel and the best option for economic benefit will depend on the specific reactor design. The approaches to incorporating high density fuel are broadly defined by the four scenarios, and fuel consumption benefit options can be implemented within each scenario.

3.1.1. With fixed core

Scenarios 1 and 2, where the number of FAs in the core is fixed (and FA design either remains unchanged or does not), can both yield fuel consumption benefit because the higher mass of uranium will lead to a higher excess reactivity, which can be used to increase the reactor fuel cycle length or the discharge burnup. Typically, in these scenarios a savings in the total mass of uranium consumed each year can be obtained, as described in Ref. [4] from a theoretical perspective, and as presented in case studies in Annexes II, III and V.

For these fixed core scenarios, benefit option (1), mentioned above, manifests because the cycle length and uranium density exhibit a semi-proportional behaviour[9] (see Ref. [4]).

Alternatively, option 2 can manifest in these scenarios if the cycle length is maintained as constant and fewer fresh FAs are required in each refuelling, which in turn is accomplished by incrementing the discharge burnup. In this case, uranium saving is proportional to the increase in the discharge burnup.

Option 3 might also manifest, where cycle length can be increased and simultaneously the number of required FAs per cycle can be reduced.

3.1.2. With configurable core or power

In scenarios 3 and 4, where the core configuration or the total power in the reactor is not maintained as constant, the impact of higher density fuel on fuel consumption will depend on the core design and other constraints that are not easy to evaluate.

For example, in Annex II the benefit is illustrated as significant reductions in FA and uranium consumption (option 2), along with a slight increase in reactor power. Annex IV presents a case where the benefit previously described in option 3 was found when the changed reactor design yielded improved average discharge burnup, with fewer FAs required in each cycle, less uranium consumed in each calendar year and increased cycle length. These improvements clearly demonstrate the potential economic benefit.

References [2, 3] present some cases where fuel consumption savings were successfully achieved, but also other cases, where FA or core redesigns were not allowed, where both uranium and FA consumption increased.

[9] Of interest is that for any specific value of uranium density, the longest cycle length can be obtained for the fuel with the least absorbent dispersion binding agent.

Increasing uranium density in the fuel meat will require, in some cases, FA or core redesign to achieve acceptable moderation ratios. However, if the number of FAs or the reactor power is changed, the potential economic benefit will vary depending on the core design and several safety related constraints.

3.2. PRELIMINARY COST EVALUATION

This section gives a preliminary evaluation of the cost impact of using high density fuel in a research reactor. The objective is not to make a detailed economic assessment but to provide a preliminary estimate of the potential reactor fuel cycle cost (FCC) savings. The focus is on replacing fuel meat with higher density fuel; excluded are U–Mo fuel qualification, potential FA or core redesign and implementation of power upgrades. Additionally, costs are not estimated for interaction with either other reactor areas (e.g. experiment facilities) or regulatory bodies.

The FCC includes the purchase of LEU, FA manufacture and spent fuel management. For a given year of operation, these expenses do not happen at the same time: the purchase of LEU and FA manufacturing expenses occur several years before the fuel is used in the reactor, while back end expenses occur several years later. An accurate economic assessment would take these different dates into account by doing a time valued estimate such as the one described in Ref. [1]. However, such an estimate will not be done here, as the purpose of this section is to present preliminary evaluation considerations.

3.2.1. Breakdown of research reactor operating cost

Depending on the reactor performance, experiment capacity, available irradiation facilities, power, flux level and, to a lesser extent, utilization rate (annual number of operating days), the total operating cost of a research reactor is tens of millions of euros per year [10]. Table 2 indicates an approximate breakdown of this cost.

This evaluation focuses on the FCC. The contribution of this factor to the reactor's total operating cost is 20–25%, depending on the level of the LEU market price.[10]

TABLE 2. RESEARCH REACTOR OPERATING COST BREAKDOWN [10]

Research reactor operating cost factors	Cost fraction
Workforce (direct and indirect costs)	~40%
Fuel cycle (LEU supply, manufacturing, back end disposition)	~20%
Other external expenditures (excluding waste treatment)	~20%
Waste and dismantling financial provisions	~20%

[10] LEU market price has fluctuated significantly in recent decades, since the prices of milling, conversion and enrichment have changed abruptly as well. For example, yellow cake price per pound went from around $10 in the late 1990s to a historical maximum of nearly $140 in 2007, to around $50 in 2012, $35 in 2014, $19 in late 2016 and $28 in October 2018, according to Refs [11–13].

3.2.2. Breakdown of fuel cycle cost

From the point of view of a research reactor operator, the FCC includes the following component expenses:

— Purchase of LEU base material (P_{LEU});
— Expenses for FA manufacturing (MF_{FA});
— Cost of the back end (BE_{FA}), as storage or reprocessing;
— Transport and shipping (TR_{FA}).

Even though these expenses are not accrued at the same time, only a preliminary evaluation of the impact of increased fuel density on the cycle cost is performed. Thus, in a simplified way, FCC can be written as:

$$\mathrm{FCC} = \mathrm{P}_{\mathrm{LEU}} + \mathrm{MF}_{\mathrm{FA}} + \mathrm{BE}_{\mathrm{FA}} + \mathrm{TR}_{\mathrm{FA}} \tag{1}$$

A rough breakdown of this cost is 20–33% for P_{LEU}, 33–40% for MF_{FA}, 33–40% for BE_{FA} and the remaining small percentage for TR_{FA} .

3.2.3. Analytical expressions of fuel cycle cost

The annual reactor FCC is defined as the fuel costs for one year of reactor operation. These costs depend on the number n of FAs used to operate the reactor for one year (i.e. the annual consumption of FAs). As previously explained, the FCC comprises several components.

The purchasing cost P_{LEU}, can be calculated as the product of n multiplied by: m_{LEU}, the mass of LEU required for the manufacture (losses included) of a single FA; and p_{LEU}, the price of one unit mass of LEU, say:

$$\mathrm{P}_{\mathrm{LEU}} = n \times m_{\mathrm{LEU}} \times p_{\mathrm{LEU}} \tag{2}$$

Similarly, the other addends of FCC can be expressed by using the costs of manufacturing (mf_{FA}), back end (be_{FA}) and transport (tr_{FA}) per FA, each multiplied by n :

$$\mathrm{MF}_{\mathrm{FA}} = n \times \mathrm{mf}_{\mathrm{FA}} \tag{3}$$

$$\mathrm{BE}_{\mathrm{FA}} = n \times \mathrm{be}_{\mathrm{FA}} \tag{4}$$

$$\mathrm{TR}_{\mathrm{FA}} = n \times \mathrm{tr}_{\mathrm{FA}} \tag{5}$$

Then, the annual FCC can be expressed as:

$$\mathrm{FCC} = n \times \left[\left(m_{\mathrm{LEU}} \times p_{\mathrm{LEU}} \right) + \mathrm{mf}_{\mathrm{FA}} + \mathrm{be}_{\mathrm{FA}} + \mathrm{tr}_{\mathrm{FA}} \right] \tag{6}$$

3.2.4. Impact of increased uranium density on reactor fuel cycle cost

Increased uranium density can increase reactor cycle length, thereby reducing annual FA consumption. Some studies on replacement of U_3Si_2 fuel with U–Mo fuel while maintaining FA geometry are presented in Annexes II, III and VI.[11] Table 3 collects the results of these studies in terms of cycle length and annual consumption.

These results indicate that replacing U_3Si_2 fuel (density 4.8 gU/cm^3) with U–Mo fuel (density 8 gU/cm^3) without changing FA geometry (scenario 1) might increase the cycle length by approximately a third and therefore reduce annual FA consumption by approximately half. Consequently, because there is higher LEU mass in each U–Mo FA (a density of 8 gU/cm^3 provides 1.67 times as much U as a 4.8 gU/cm^3 density) but fewer FAs are being consumed per year (~50% less), annual uranium consumption is reduced to 83% of the uranium annual consumption with U_3Si_2 fuel.

By increasing the cycle length, the increase of uranium density reduces the number of FAs to be manufactured, transported, reprocessed or stored following end of useful life, but the cost reduction is not fully proportional for several reasons:

— Increasing uranium density increases the uranium mass per FA, which affects the annual uranium consumption, as already observed. By replacing U_3Si_2 (4.8 gU/cm^3) with U–Mo (8 gU/cm^3), while maintaining FA geometry, the increase of mass in each FA is proportional to the ratio 8/4.8 (i.e. 1.67).
— The manufacturing cost per FA will probably increase due to the increased amount of fissile material per FA, atomized powder production, matrix modification, increased scrap and enhanced quality control. The manufacturing cost for high density fuel may be significantly affected if additional steps, such as coating, have to be added to an existing fabrication process.
— It is also expected that the change to high density fuel could affect back end costs in the circumstances of storage or reprocessing — especially when the density change is associated with a change of fuel type.[12] A cost evaluation with an acceptable uncertainty for spent fuel management of U_3Si_2 fuel and U–Mo fuel is not currently available.

TABLE 3. CASE STUDIES OF ANNUAL FUEL CONSUMPTION FOR SCENARIO 1

Fuel type	Annex II (KJRR)		Annex III (OPAL) [a]		Annex VI (AHR) [b]	
	Backup	Original	Original	Replacement	Original	Replacement
	U_3Si_2	U–Mo	U_3Si_2	U–Mo	U_3Si_2	U–Mo
Density (g/cm^3)	4.8	8.0 and 6.5	4.8	8.0	4.0	6.0 and 4.5
Cycle length (equivalent full power days)	37.5	50	30.0	40.5	31	50
Annual FA consumption	24	12	34.2	17.2	—[c]	—
Annual U consumption (kg)	45.9	37.6	—	—	—	—

[a] Annex III, case Mo807.
[b] Annex VI, case Core B no annual consumption data were reported, but savings are implied by the increased cycle length.
[c] —: data not available.

[11] Studies from Annex I are not mentioned here as none of those cases offer reduced FA consumption due to constraints on maximum fuel burnup. Also, Annex V studies are not included because they report no cycle length or consumption data except for redesigned FAs (scenario 2).

[12] For example, fuels with higher uranium loading might introduce additional sustainability issues into back end storage options; on the other hand, reprocessing might be easier without the silicon component of U_3Si_2 fuel. Discussion of these issues is beyond the scope of this publication.

To account for these possible cost increases from using high density fuel, α and β are defined as the percentage of increase in the cost of manufacturing an FA ($\mathrm{mf}_{\mathrm{FA}}$), and the cost of back end per FA ($\mathrm{be}_{\mathrm{FA}}$), respectively, but it is assumed that the change in the transport cost per FA is insignificant. New FCC components can thus be written as follows:

$$\mathrm{mf}_{\mathrm{FA}}{}' = (1 + \alpha) \times \mathrm{mf}_{\mathrm{FA}} \tag{7a}$$

$$\mathrm{be}_{\mathrm{FA}}{}' = (1 + \beta) \times \mathrm{be}_{\mathrm{FA}} \tag{7b}$$

and, finally:

$$\begin{aligned} \mathrm{FCC}' &= N' \times \left[\left(m_{\mathrm{LEU}}{}' \times p_{\mathrm{LEU}} \right) + \mathrm{mf}_{\mathrm{FA}}{}' + \mathrm{be}_{\mathrm{FA}}{}' + \mathrm{tr}_{\mathrm{FA}} \right] \\ &= N' \times \left[\left(m_{\mathrm{LEU}}{}' \times p_{\mathrm{LEU}} \right) + (1 + \alpha) \times \mathrm{mf}_{\mathrm{FA}} + (1 + \beta) \times \mathrm{be}_{\mathrm{FA}} + \mathrm{tr}_{\mathrm{FA}} \right] \end{aligned} \tag{8}$$

where

FCC'	is the new annual FCC;
N'	is the annual consumption of new FAs;
$m_{\mathrm{LEU}}{}'$	is the LEU mass needed to manufacture one new FA;
p_{LEU}	is the per mass unit price of LEU (unchanged);
$\mathrm{mf}_{\mathrm{FA}}{}'$	is the cost of manufacturing one new FA;
$\mathrm{be}_{\mathrm{FA}}{}'$	is the cost of back end processing of one new FA;

and $\mathrm{tr}_{\mathrm{FA}}$ is the cost of transport of one FA, which is assumed to be largely unchanged by fuel density.

3.2.5. Quantitative analysis

A quantitative evaluation of the impact of density on FCC uses the above equations to analyse the impact of replacing U_3Si_2 (4.8 gU/cm^3) fuel meat with U–Mo (8 gU/cm^3). Using Eqs (6) and (8), the ratio of FCCs can be expressed as:

$$\frac{\mathrm{FCC}_{\mathrm{U\text{–}Mo}}}{\mathrm{FCC}_{\mathrm{U3Si2}}} = \frac{\mathrm{FCC}'}{\mathrm{FCC}} = \frac{N' \times [\left(m_{\mathrm{LEU}}{}' \times p_{\mathrm{LEU}} \right) + (1 + \alpha) \times \mathrm{mf}_{\mathrm{FA}} + (1 + \beta) \times \mathrm{be}_{\mathrm{FA}} + \mathrm{tr}_{\mathrm{FA}}]}{N \times [\left(m_{\mathrm{LEU}} \times p_{\mathrm{LEU}} \right) + \mathrm{mf}_{\mathrm{FA}} + \mathrm{be}_{\mathrm{FA}}]} \tag{9}$$

As previously noted, high density fuel replacement allows the reduction of annual FA consumption by a factor of around two ($N'/N \approx$ ½), but increases the LEU mass for one FA by a factor of 1.67 ($m'_{\mathrm{LEU}}/m_{\mathrm{LEU}} = 8\mathrm{g}/4.8\mathrm{g} = 1.67$) for unchanged FA geometry. With insertion of these values into Eq. (9), the ratio of FCCs is then calculated as:

$$\frac{\mathrm{FCC}_{\mathrm{U\text{–}Mo}}}{\mathrm{FCC}_{\mathrm{U3Si2}}} = \frac{\left(\frac{1}{2} \times \left(\left(1.67 m_{\mathrm{LEU}} \times p_{\mathrm{LEU}} \right) + (1 + \alpha) \times \mathrm{mf}_{\mathrm{FA}} + (1 + \beta) \times \mathrm{be}_{\mathrm{FA}} + \mathrm{tr}_{\mathrm{FA}} \right) \right)}{\left(\left(m_{\mathrm{LEU}} \times p_{\mathrm{LEU}} \right) + \mathrm{mf}_{\mathrm{FA}} + \mathrm{be}_{\mathrm{FA}} \right)} \tag{10}$$

Two cases of bounding assumptions for possible allocation of FCCs are defined and are expressed in Table 4. The transport component is neglected here as it represents only a few per cent of the total FCC and is assumed to not be very sensitive to fuel density. Other cost components are allocated in the assumption cases as follows.

TABLE 4. GENERAL IMPACTS OF FUEL DENSITY ON FUEL CYCLE COSTS

Cost component	Assumption case 1	Assumption case 2	FCC[a]
Purchase LEU	~20%	~33%	↘ FA number per year ↗ U mass per FA
Manufacture FAs	~40%	~33%	↘ FA number per year ↗ ? Price per FA
Back end (spent fuel storage and disposition)	~40%	~33%	↘ FA number per year ? Price per FA
Transport	Remainder	Remainder	↘ FA number per year
Total	100%	100%	↘ FA number per year ↗ Price per FA

[a] Fuel cycle costs go up (↗) or down (↘), as a general impact of increasing LEU fuel density; an uncertainty in the price per FA (?) is indicated since applicable numerical values are required in the previous equation to determine the outcome for a given research reactor.

(a) Assumption case 1:

LEU purchasing cost, P_{LEU}, assumed to be 20% of FCC:

$$0.2 = \frac{P_{LEU}}{FCC} = \frac{N \times (m_{LEU} \times p_{LEU})}{N \times [(m_{LEU} \times p_{LEU}) + mf_{FA} + be_{FA}]} = \frac{(m_{LEU} \times p_{LEU})}{(m_{LEU} \times p_{LEU}) + mf_{FA} + be_{FA}} \tag{11}$$

Manufacturing costs, MF_{FA}, assumed to be 40% of FCC:

$$0.4 = \frac{MF_{FA}}{FCC} = \frac{N \times m_{FA}}{N \times [(m_{LEU} \times p_{LEU}) + mf_{FA} + be_{FA}]} = \frac{mf_{FA}}{(m_{LEU} \times p_{LEU}) + mf_{FA} + be_{FA}} \tag{12}$$

Back end costs, BE_{FA}, assumed to be 40% of FCC:

$$0.4 = \frac{BE_{FA}}{FCC} = \frac{N \times be_{FA}}{N \times [(m_{LEU} \times p_{LEU}) + mf_{FA} + be_{FA}]} = \frac{be_{FA}}{(m_{LEU} \times p_{LEU}) + mf_{FA} + be_{FA}} \tag{13}$$

(b) Assumption case 2:

One third (33%) for each of the same three cost components. Inserting these assumed percentages into Eq. (10), the ratio of the FCCs due to high density fuel can be expressed as:

(a) Assumption case 1:

$$\frac{FCC_{U-Mo}}{FCC_{U3Si2}} = \frac{1}{2} \times \{(1.67 \times 0.2) + [(1 + \alpha) \times 0.4] + [(1 + \beta) \times 0.4]\} \tag{14}$$

(b) Assumption case 2:

$$\frac{FCC_{U-Mo}}{FCC_{U3Si2}} = \frac{1}{2} \times \left\{ (1.67 \times 0.33) + \left[(1+\alpha) \times 0.33 \right] + \left[(1+\beta) \times 0.33 \right] \right\} \tag{15}$$

The relative FCC change due to using high density U–Mo fuel can be calculated as:

$$\frac{FCC_{U3Si2} - FCC_{U-Mo}}{FCC_{U3Si2}} = 1 - \frac{FCC_{U-Mo}}{FCC_{U3Si2}} \tag{16}$$

Table 5 gives the values of relative FCC changes, calculated based on allocation of FCC components — per assumption cases 1 and 2, as expressed in Eqs (14) and (15) — and assuming α and β values ranged between 0.0 and 0.2.

By replacing U_3Si_2 (4.8 gU/cm^3) with U–Mo (8 gU/cm^3), the savings in FCC are roughly between 35 and 40% of its pre-change cost, mainly depending on the actual cost breakdown of FCC components and the change or not of per-FA cost of manufacturing and back end processes:

- For manufacturing and back end per-FA costs unchanged ($\alpha = \beta = 0$), the savings on the FCC vary between 39 and 43%;
- For a 20% increase in either manufacturing or back end costs ($\alpha = 0.2$ and $\beta = 0$; or $\alpha = 0$ and $\beta = 0.2$), the relative FCC savings are between 36 and 39%;
- For a 20% increase in both manufacturing and back end costs ($\alpha = \beta = 0.2$), the relative FCC savings range between 32 and 35%.

TABLE 5. RELATIVE FUEL CYCLE COST SAVINGS DUE TO USING HIGH DENSITY FUEL

	(a) Assumption case 1[a]				(b) Assumption case 2[b]		
	$\beta = 0.0$	$\beta = 0.1$	$\beta = 0.2$		$\beta = 0.0$	$\beta = 0.1$	$\beta = 0.2$
$\alpha = 0.0$	43%	41%	39%	$\alpha = 0.0$	39%	37%	36%
$\alpha = 0.1$	41%	39%	37%	$\alpha = 0.1$	37%	36%	34%
$\alpha = 0.2$	39%	37%	35%	$\alpha = 0.2$	36%	34%	32%

Note: α indicates increased percentage of FA manufacturing cost (excluding cost of uranium); β indicates increased percentage of FA back end cost (e.g. fuel storage and disposition).

[a] Assumption case 1 uses FCC component allocation per Eq. (14).

[b] Assumption case 2 uses FCC component allocation per Eq. (15).

4. CONCLUSIONS

A variety of conclusions can be drawn from the discussion in the previous sections about the impacts of using higher density fuel in research reactors on both reactor performance and fuel economics. These conclusions are summarized below:

(a) Improved reactor irradiation performance is an additional potential benefit of using high density fuel, but typically only for scenarios when either the reactor core is changed to reduce the number of FAs or reactor power is upgraded.

(b) To help decide whether to use high density fuel in a research reactor, potential benefits and drawbacks should be clearly identified. Performance or economic benefits are sought, but with a desired minimum amount of related redesign and work to execute the fuel change.

(c) Design characteristics, goals and constraints differ from one research reactor design to another, so any specific reactor needs to be investigated individually when a change to a higher density fuel is proposed.

(d) Increasing fuel density may increase a reactor's available reactivity, impact moderation ratios and adversely affect a number of performance parameters, including thermal flux spectra, control rod worth, shutdown margin, PPFs, reactivity coefficients and kinetic parameters.

(e) For most reactor cases analysed, potential issues resulting from a change to high density fuel can be successfully addressed, and design and safety criteria can be satisfied through a combination of several methods, including:
   (i) Fuel management modifications;
   (ii) Introduction or redesign of burnable poisons;
   (iii) FA redesign (primarily modification of fuel plate design);
   (iv) Modification of FA and control rod positions in a configurable core.

(f) Use of higher density fuel in a research reactor is expected to reduce annual FA consumption owing to increased fuel cycle length and burnup, when compared with the commercial research reactor fuels currently used and available. This is shown for all the scenarios considered here.

(g) A preliminary assessment of potential cost savings shows that replacement of U_3Si_2 fuel (4.8 gU/cm^3) with U−Mo fuel (8 gU/cm^3), with no change to FA geometry, increases the cycle length by a third and reduces FA annual consumptions by half, and therefore reduces uranium annual consumption by about 17%. This results in FCC savings of 35−40%, mainly depending on the breakdown of cost components and the possible cost change of manufacturing and back end processes with the high density fuel.

REFERENCES

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Cost Aspects of the Research Reactor Fuel Cycle, IAEA Nuclear Energy Series No. NG-T-4.3, IAEA, Vienna (2010).

[2] AHMED, R., ASLAM, AHMAD, N., Burnup study for Pakistan Research Reactor-1 utilizing high density low enriched uranium fuel, Ann. Nucl. Energy **32** 10 (2005) 1100–1121.

[3] AHMED, R., ASLAM, AHMAD, N., KHAN, M.J., MANAN, A., A study of reduction in fuel requirement and enhancement of neutron flux in typical MTR type research reactors, Prog. Nucl. Energy **49** 2 (2007) 171–179.

[4] BRETSCHER, M.M., MATOS, J.E., SNELGROVE, J.L., "Relative neutronic performance of proposed high-density fuels in water-moderated and D_2O-reflected research reactors", Proc. 19th Int. Mtg on Reduced Enrichment for Research and Test Reactors, Seoul, 1996, Argonne Natl Lab., IL (1996).

[5] BRETSCHER, M.M., MATOS, J.E., Neutronic performance of high-density LEU fuels in water-moderated and water-reflected research reactors, Rep. ANL–TD/RP-91011, Argonne Natl Lab., IL (1996).
[6] MUHAMMAD, F., Effects of high density dispersion fuel loading on the control rod worth of a low enriched uranium fueled material test research reactor, Ann. Nucl. Energy **58** (2013) 19–24.
[7] VILLARINO, E., HERGENREDER, D., "Neutronic performance of the U–Mo fuel type in the replacement research reactor", Proc. 24th Int. Mtg on Reduced Enrichment for Research and Test Reactors, San Carlos de Bariloche, 2002, Argonne Natl Lab., IL (2002).
[8] MUHAMMAD, F., MAJID, A., Effects of high density dispersion fuel loading on the kinetic parameters of a low enriched uranium fueled material test research reactor, Ann. Nucl. Energy **35** 9 (2008) 1720–1731.
[9] VILLARINO, E., "Core performance improvements using high density fuel in research reactors", Trans. Topical Mtg on Research Reactor Fuel Management, St. Petersburg, 2013, European Nuclear Society, Brussels (2013) 93–100.
[10] ECONOMICS LEGISLATION COMMITTEE, Answers to Questions on Notice, Industry Portfolio, Budget Estimates Hearing 2014–15, Question No. BI-2, Parliament of Australia, Canberra (2014).
[11] UX CONSULTING, Ux Weekly **26** 42 (15 Oct. 2012).
[12] UX CONSULTING, Ux Weekly **28** 14 (7 Apr. 2014).
[13] UX CONSULTING, Ux Weekly **32** 44 (29 Oct. 2018).

Annex I

PRELIMINARY ANALYTICAL INVESTIGATION OF HIGH DENSITY U–Mo FUEL: INTRODUCTION TO THE JMTR

N. Takemoto, T. Imaizumi, H. Ide, M. Naka, T. Kusunoki, M. Ishihara

Neutron Irradiation and Testing Reactor Center, Oarai Research and Development Center, Japan Atomic Energy Agency, Tokai-mura, Japan

I–1. INTRODUCTION

The Japan Materials Testing Reactor (JMTR) [I–1] of the Japan Atomic Energy Agency (JAEA) is a light water cooled tank type reactor with a thermal power of 50 MW. The purpose of JMTR was to perform irradiation tests for light water reactor fuel and materials, to establish domestic technology for developing a nuclear power plant, to produce radioisotopes and to conduct education and training. A cutaway view and cross-section of the core are shown in Fig. I–1 [I–2]. The main specifications of the JMTR are listed in Table I–1.

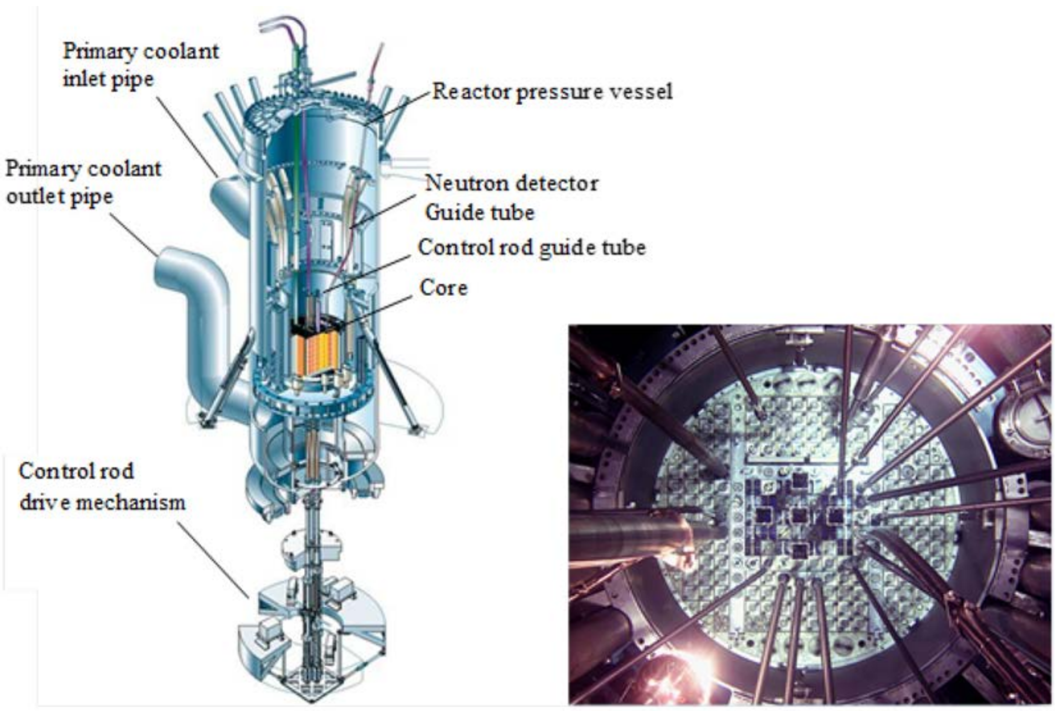


FIG. I–1. The JMTR core: (a) cutaway view; (b) cross-section. Reproduced courtesy of JAEA [I–2].

As part of the Reduced Enrichment for Research and Test Reactors (RERTR) programme [I–3], the JMTR core was converted in 1994 from highly enriched uranium (HEU) fuel — 93%, alloy type — to medium enriched uranium fuel — 45%, aluminide type — in 1986, and then to low enriched uranium (LEU) fuel — 19.75%, silicide type (i.e. U_3Si_2 fuel) — without significant change to the irradiation performance. The specifications of the U_3Si_2 fuel are listed in Table I–2. The JMTR is operated using U_3Si_2 standard fuel assemblies (FAs) (LEU) and U_3Si_2 fuel followers (follower LEUs) of the control rods. Each structure is shown in Fig. I–2.

TABLE I–1. MAIN SPECIFICATIONS OF THE JMTR

Parameter		Value
Reactor type		Light water moderated and cooled tank
Thermal power		50 MW
Neutron flux	Fast (max.)	4×10^{18} n/m^{2}s
	Thermal (max.)	4×10^{18} n/m^{2}s
Fuel assembly	U-235 enrichment	19.5wt%
	Fuel meat	U_3Si_2–Al
	Cladding material	Aluminium
Reflector		Beryllium
Power density		425 MW/m^3
Primary coolant	Core inlet temperature (max.)	49°C
	Core outlet temperature	56°C
	Flow rate	6000 m^3/h
	Operating pressure	1.5 MPa
Irradiation facility	Capsule (irradiation positions)	200
	Shroud irradiation facility	1
	Hydraulic rabbit irradiation facility	1

TABLE I–2. SPECIFICATIONS OF THE JMTR FUEL

		U_3Si_2 fuel	
		Standard fuel assembly	Fuel follower
Fuel type		Plate	Plate
Dimensions (mm)		76 × 76 × 1200	64 × 64 × 890
Fuel plate	Number	19 per assembly	16 per assembly
	U density (g/cm^3)	4.8	4.8
	Thickness (mm)	1.27	1.27
	Length (mm)	780	770
	Width (mm)	71	60
	Cladding thickness (mm)	0.38	0.38
	Cladding material	Aluminium alloy	Aluminium alloy
Fuel meat	Material	U_3Si_2/Al	U_3Si_2/Al
	Thickness (mm)	0.51	0.51
	Length (mm)	760	750
	Width (mm)	62	50
Burnable absorber	Material	Cadmium	Cadmium
	Diameter (mm)	0.3	0.3
	Length (mm)	760	750
	Cladding material	Aluminium alloy	Aluminium alloy
	Cladding thickness (mm)	0.25	0.25
	Number	18 per assembly	16 per assembly
U-235 content (g)	Per assembly	410	275
Burnup		Max. 60%	Max. 60%

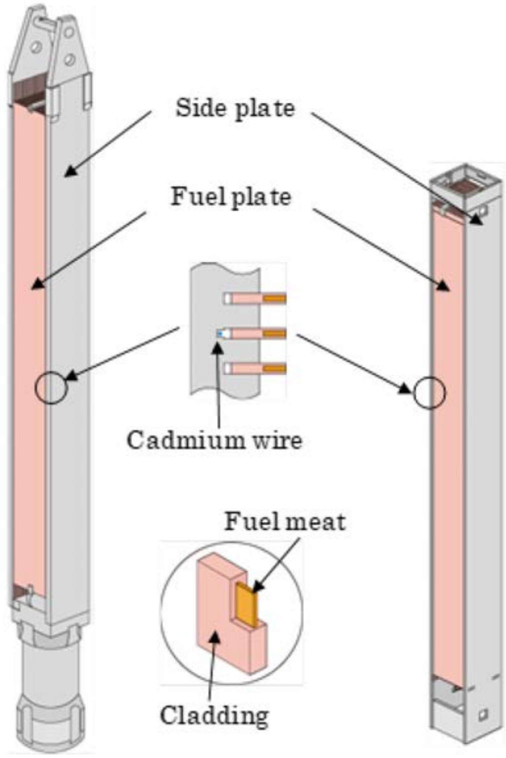


FIG. I–2. Structure of the standard fuel assembly and fuel follower of the JMTR. Reproduced courtesy of JAEA [I–2].

Two kinds of core configurations, called the LEU core and the improved LEU core, are approved for the JMTR, and are listed in Table I–3. The LEU core consists of 22 LEU FAs (used for 2 cycles) and 5 LEU fuel followers (used for 1 cycle) and is operated for about 25 days per cycle (1240 MWd) with maximum burnup of 50%. The improved LEU core consists of 24 LEU FAs (used for 3 cycles) and 5 LEU followers (used for 2 cycles) and is operated for about 30 days per cycle (1490 MWd) with maximum burnup of 60%.

The design criteria for the JMTR are as follows:

— Excess reactivity (%$\Delta k/k$): 15.0% (max.);
— Shutdown margin (k_{eff}): <1.0;
— One stuck rod margin (k_{eff}): <0.9;
— ^{235}U quantity in the core: <11 kg;
— Temperature, void and Doppler coefficients are negative under all operating conditions in the JMTR.

Currently, an international effort to develop uranium–molybdenum (U–Mo) fuel is ongoing, focusing on the HEU to LEU fuel conversion for high power research reactors. U–Mo fuel has the advantages of high density, high burnup and easy reprocessing properties.

Therefore, from the viewpoint of improvement of fuel economy without significant changes to irradiation performance, an analytical investigation was performed for the use of U–Mo fuel in the JMTR [I–3].

TABLE I–3. LEU CORE AND IMPROVED LEU CORE FOR THE JMTR

	LEU core	Improved LEU core
Total number of fuel assemblies	27	29
Standard fuel assemblies	22	24
Cycles used	2	3
Fuel followers	5	5
Cycles used	1	2
Burnup	Max. 50%	Max. 60%
Operation (days per cycle)	25	30
Cumulative power (MW·d)	~1240	~1490
Fuel consumption	64 fuel assemblies for 100 day operation	63 fuel assemblies for 180 day operation

I–2. NEUTRONIC ANALYSIS

I–2.1. Calculating model

A typical core configuration for the improved LEU core [I–4] is used in the calculation, and is shown in Fig. I–3. In the analytical investigation, U_3Si_2 fuels in the JMTR core are converted into U–Mo fuels for comparison. The preconditions are:

— The shape of the U–Mo fuel is the same as that of the U_3Si_2 fuel;
— The neutronic design criteria are also the same as those for the JMTR core.

I–2.2. Specification of U–Mo fuel for analytical investigation

I–2.2.1. Content of Mo in U–Mo alloy

The amount of Mo in U–Mo alloy has an influence on the impact of changing to U–Mo fuel. Increasing the concentration of Mo can decrease irradiation performance, such as neutron flux, because of the large neutron absorption cross-section of Mo. In post-irradiation examination results, good irradiation behaviour is observed when the concentration of Mo is between 6 and 10wt%. Given that irradiation tests of fuel mini-plates of U–Mo with 7wt% Mo (U–7Mo) have been performed internationally, the concentration of Mo was assumed to be 7wt% for this investigation.

I–2.2.2. Density and enrichment of U

This investigation studies three fuels provided by three different combinations of U density and U enrichment. These fuels are named U–Mo fuel 1, U–Mo fuel 2 and U–Mo fuel 3.

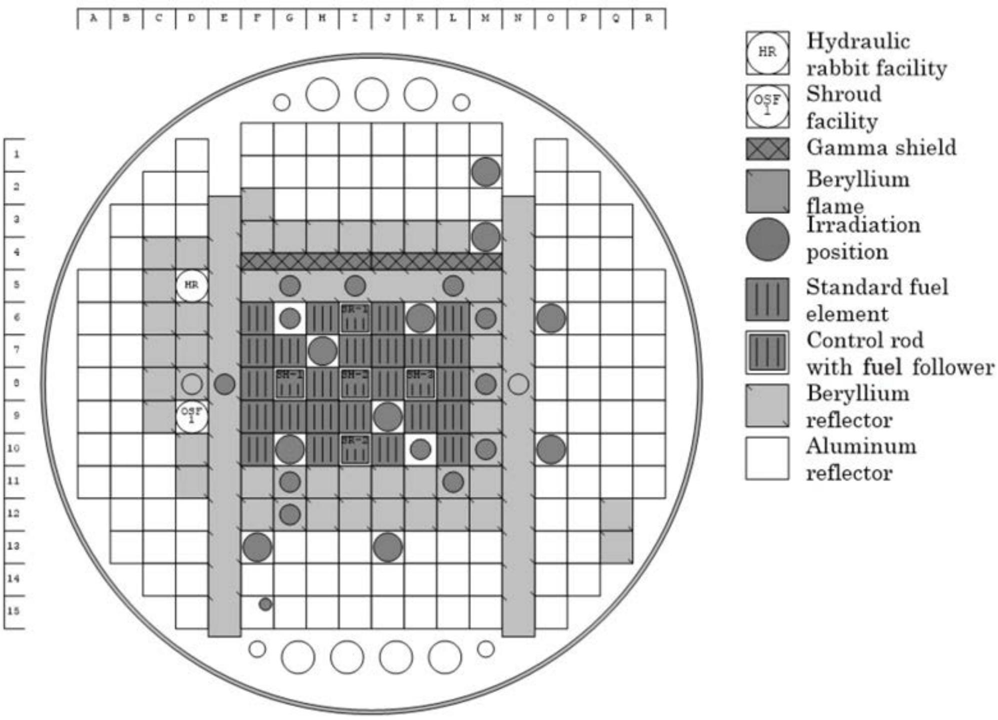


FIG. I–3. Typical improved LEU core configuration of the JMTR. Reproduced courtesy of JAEA.

The uranium density for these combinations is set at either 4.8 or 8.0 gU/cm^3. The first density is that of the current U_3Si_2 fuel, while the other density is based on the results of U–Mo fuel development in other countries.[1]

The uranium enrichment for the combinations is set at either 19.75 or 11.85%. The first enrichment is that of the current U_3Si_2 fuel; the second enrichment is obtained by considering a U–Mo fuel with a density of 8.0 gU/cm^3 that would provide the same mass of ^{235}U in the core as the current fuel.

Table I–4 provides details of the combinations considered for U–Mo fuels 1, 2 and 3.

I–2.3. Calculating code

The neutronic calculations were performed by using the Standard Reactor Analysis Code (SRAC) system (see Refs [I–5, I–6]), which includes codes and subprograms such as PIJ, Citation, CoreBN and ENDFB/IV. Cell calculations for four-group macroscopic cross-sections of each material were performed using the PIJ code, which is based on the collision probability method. The three dimensional whole core calculation was performed using Citation (the diffusion code). The fuel burnup was calculated using the CoreBN. Nuclear cross-section library ENDF-B/IV was used. The SRAC code system has been used for core management of the JMTR (see Refs [I–7, I–8]).

[1] Using an atomized method, U–Mo fuel has been produced with a density of up to 8.4 gU/cm^3 in the United States of America, and of up to 8.5 gU/cm^3 in France.

TABLE I–4. SPECIFICATION OF U–Mo FUEL[a] FOR ANALYTICAL INVESTIGATION

Name	U-235 amount in the core (g)	U density (gU/cm^3)	U-235 enrichment (%)	Remarks
U_3Si_2 fuel	9.629	4.8	19.75	Current fuel
U–Mo fuel 1	9.629	4.8	19.75	Without change of density and enrichment
U–Mo fuel 2	9.629	8.0	11.85	Without change of U-235 quantity in the core
U–Mo fuel 3	16.048	8.0	19.75	Without change of enrichment

[a] Content of Mo in U–Mo alloy: 7wt%.

I–2.4. Calculation

I–2.4.1. Preliminary core burnup calculation using U–Mo fuel

Before the analytical investigation, a preliminary core burnup calculation was performed for the existing JMTR fuel to compare the impact of the new LEU core configuration of the JMTR with all new U–Mo fuel 1s and with all new U_3Si_2 fuels.

I–2.4.2. Burnup calculation for U–Mo fuel plate

Cell burnup calculations for U–Mo fuel 1, 2 and 3 and U_3Si_2 fuel without cadmium wire, coolant, side plate and so on were carried out to compare burnup days, infinite multiplication factor (k_∞)[2] and number density of nuclei in the fuel plate.

I–2.4.3. Core calculation using U–Mo fuel

Four groups of macroscopic cross-sections for each material of U–Mo fuels 1, 2 and 3 and the U_3Si_2 fuel were evaluated using a cell calculation with the number densities of nuclei obtained in the calculations described in Section I–2.4.2. Then, the core calculation for the typical improved LEU core configuration of the JMTR was carried out using those evaluated cross-sections. The following reactor parameters were calculated to evaluate whether the JMTR neutronic design criteria are satisfied:

— Shutdown margin, one stuck rod margin and excess reactivity at the beginning of cycle (BOC) in the cold clean state;
— Excess reactivity after 15% burnup.

[2] The ‘multiplication factor’, k, is the ratio of the total number of neutrons produced during a time interval to the total number of neutrons lost by absorption and leakage during the same interval. The ‘infinite multiplication factor’, k_∞, is the multiplication factor evaluated for an infinite medium or for an infinite repeating lattice.

At the BOC, the burnup for new fuels, single-cycle used fuels and two-cycle used fuels is assumed to be 0, 15 and 30%, respectively. After 15% burnup — which is equivalent to one cycle operation — each burnup is changed to 15, 30 and 45%, respectively.

Moreover, in order to investigate the influence of U–Mo fuel on irradiation performance, cell averaged fast (>1 MeV) and thermal (<0.683 eV) neutron fluxes at several irradiation positions were calculated at the BOC for the U_3Si_2 fuel and U–Mo fuels 1, 2 and 3. Four positions were selected as irradiation positions, as shown in Fig. I–4: fuel region, reflector layer 1, reflector layer 2 and reflector layer 3.

I–3. RESULTS OF NEUTRONIC ANALYSIS AND DISCUSSION

I–3.1. Preliminary core burnup calculation using U–Mo fuel

As a result of the calculation, the excess reactivity in the U–Mo fuel core decreased by roughly 0.5%$\Delta k/k$ compared with the U_3Si_2 fuel core, as shown in Fig. I–5. This is thought to be because the neutron absorption cross-section of molybdenum is larger than that of Si. On the other hand, there is a similar tendency in reactivity change with cumulative power.

I–3.2. Burnup calculation of U–Mo fuel plate

The change to k_∞ as a function of burnup days for U–Mo fuels 1, 2 and 3 is shown in Figs I–6, I–7 and I–8, respectively, with each figure also showing the values for U_3Si_2 fuel, for comparison.

For U–Mo fuel 1, k_∞ decreased a little when compared to the U_3Si_2 fuel, while for U–Mo fuel 2, k_∞ decreased more significantly at the initial stage of burnup, and only a small diminution of the factor was observed at the end state, compared with the U_3Si_2 fuel. By contrast, for U–Mo fuel 3, the factor increased because the amount of ^{235}U was 1.67 times larger than that of the other fuels.

In these figures it is possible to observe the cumulative power per FA by looking at the relation between the burnup and burnup days. For example, the number of burnup days needed to reach 60%

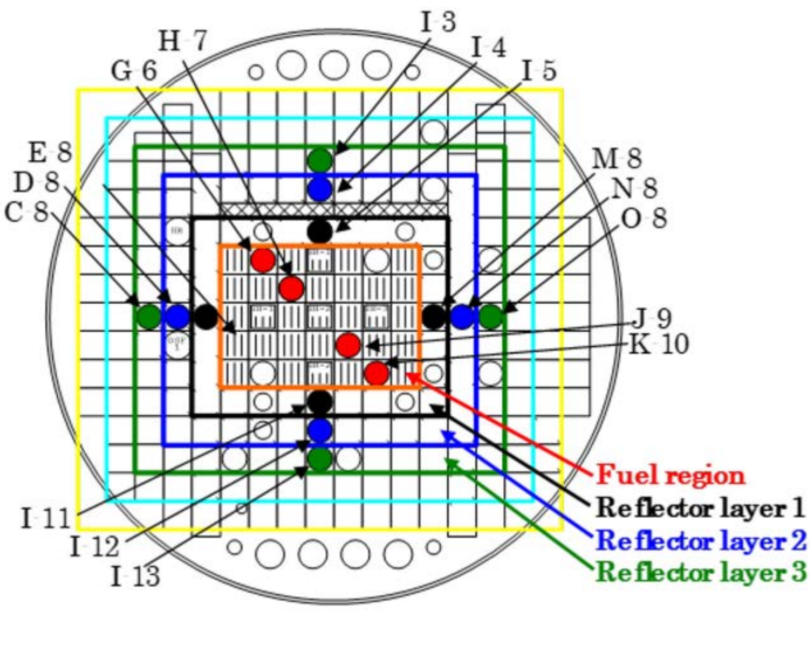


FIG. I–4. Irradiation positions for neutronic calculation. Reproduced courtesy of JAEA.

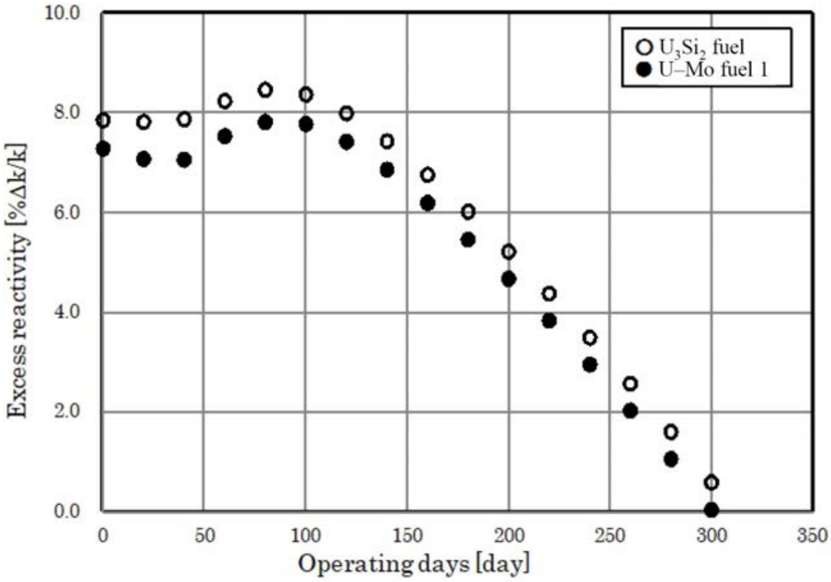


FIG. I–5. Preliminary excess reactivity calculation using U–Mo fuel. Reproduced courtesy of JAEA.

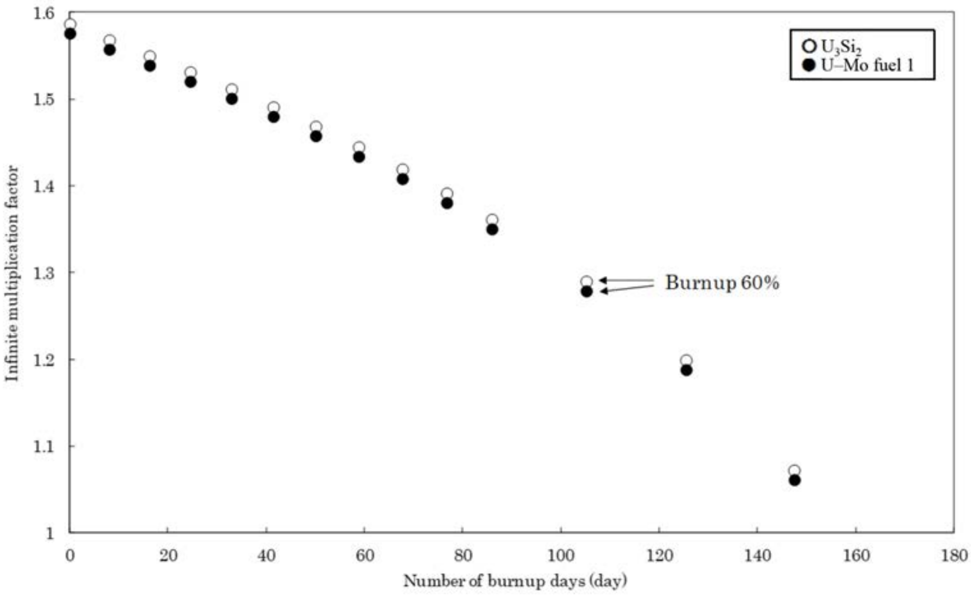


FIG. I–6. Comparison of k_∞ between U–Mo fuel 1 and U_3Si_2 fuel. Reproduced courtesy of JAEA (adapted from Ref. [I–3]; translated from the original).

burnup of the U–Mo fuel is larger than that for the U_3Si_2 fuel (i.e. the cumulative power per FA of the U–Mo fuels is larger than that of the U_3Si_2 fuel).

The cumulative power per FA is almost the same for the U_3Si_2 fuel and U–Mo fuel 1. For U–Mo fuel 2, the cumulative power per FA increases with burnup because of a large resonance absorption by Mo and ^{238}U. For U–Mo fuel 3, the cumulative power per fuel increases dramatically because there is 1.67 times more ^{235}U than in the other fuels.

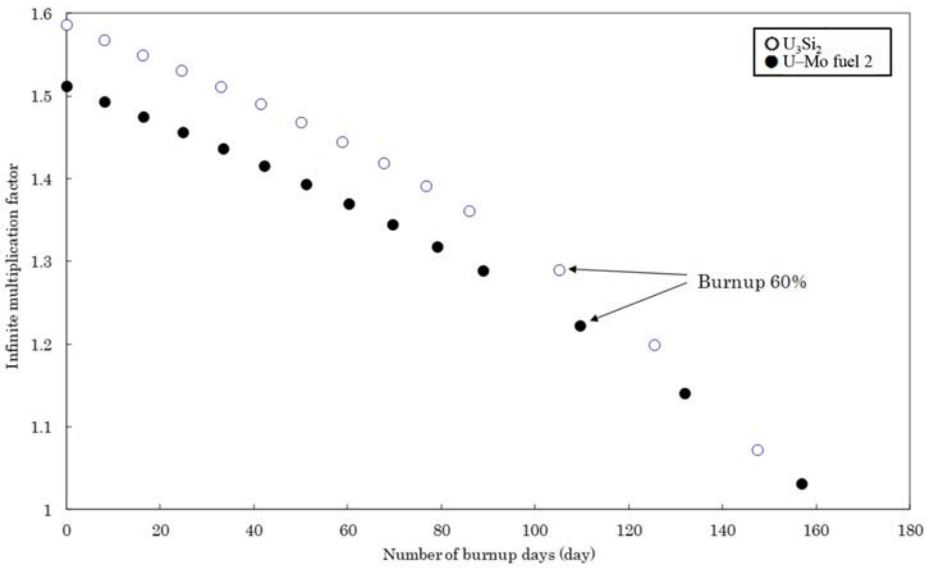


FIG. I–7. Comparison of k_∞ between U–Mo fuel 2 and U_3Si_2 fuel. Reproduced courtesy of JAEA (adapted from Ref. [I–3]; translated from the original).

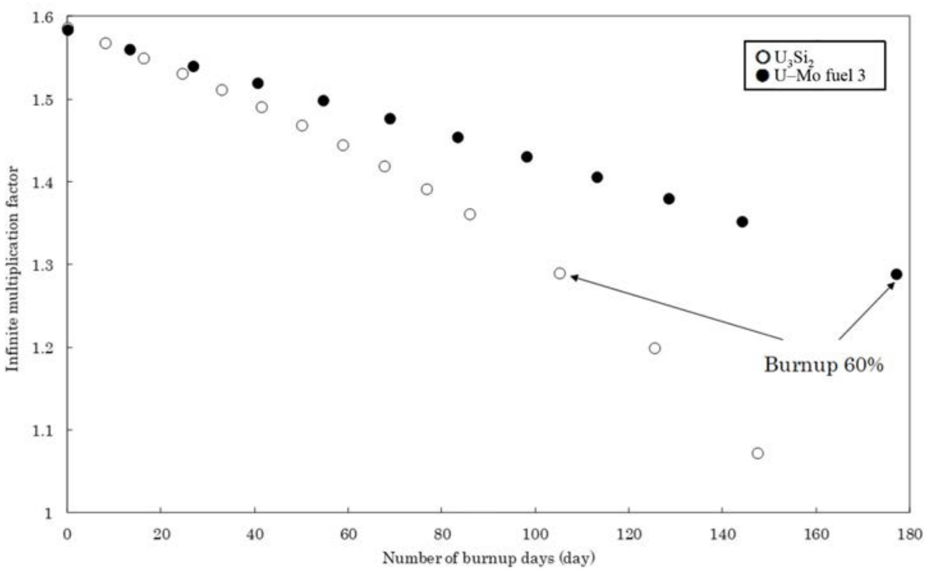


FIG. I–8. Comparison of k_∞ between U–Mo fuel 3 and U_3Si_2 fuel. Reproduced courtesy of JAEA (adapted from [I–3]; translated from the original).

I–3.3. Core calculation using U–Mo fuel

The core calculation results for the U–Mo fuels are summarized in Table I–5. The calculated fast and thermal neutron fluxes of the U–Mo fuel cores are normalized to those of the U_3Si_2 fuel core and plotted in Figs I–9 and I–10, respectively.

TABLE I–5. CALCULATION RESULTS FOR REACTOR PARAMETERS USING U–Mo FUEL IN THE JMTR

	BOC at cold clean state				After 15% burnup	
Fuel type	U-235 amount in the core (g)	Shutdown margin	One stuck rod margin	Excess reactivity (%$\Delta k/k$)	Excess reactivity (%$\Delta k/k$)	Cumulative power (MWd)
U_3Si_2 fuel	9.629	0.893	0.983	12.32	3.78	1378
U–Mo fuel 1	9.629	0.888	0.977	11.85	3.29	1378
U–Mo fuel 2	9.629	0.867	0.954	9.47	0.78	1420
U–Mo fuel 3	16.048	0.952	1.041	16.61	7.77	2305
Design criteria	<11	<0.9	<1.0	<15.0	n.a.[a]	n.a.

[a] n.a.: not applicable.

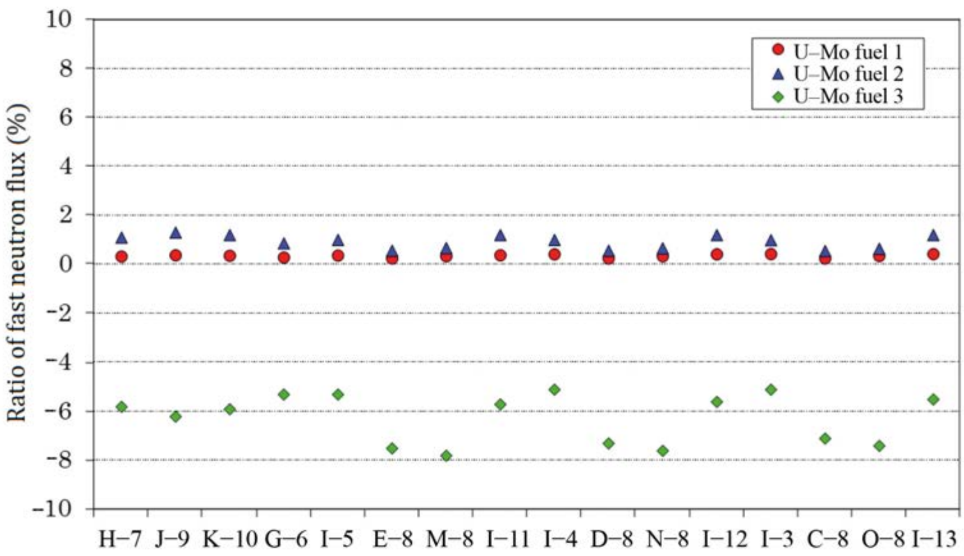


FIG. I–9. Comparison of fast neutron flux between U–Mo fuel and U_3Si_2 fuel core. Reproduced courtesy of JAEA.

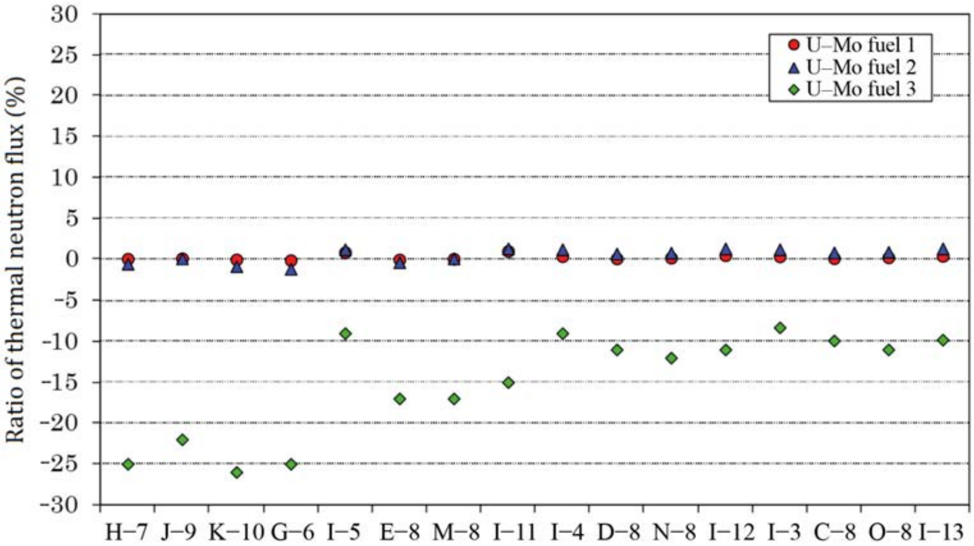


FIG. I–10. Comparison of thermal neutron flux between U–Mo fuel and U_3Si_2 fuel core. Reproduced courtesy of JAEA.

The following observations can be made concerning the comparisons of the U–Mo 1, 2 and 3 fuels to the U_3Si_2 fuel:

— The U–Mo fuel 1 core meets all of the design criteria for the JMTR. The excess reactivity decreases, and the neutron spectrum is hardened a little compared with the U_3Si_2 fuel core because of the larger neutron absorption cross-section of Mo. It is found that it is impossible to increase the cumulative power per fuel by using U–Mo fuel 1.

— The U–Mo fuel 2 core also meets all of the design criteria. The excess reactivity decreases, and the neutron spectrum is hardened compared to that of the U–Mo fuel 1 core because of an increase in Mo and ^{238}U. The cumulative power per FA increases a little after 15% burnup, as already shown in Table I–5, due to the effect of plutonium, which is created by the increased amount of ^{238}U. However, the effect is not sufficient to improve the fuel economy. Based on the calculation results for the excess reactivity and the cumulative power per fuel, as shown in Table I–5, the possible operating days per cycle is about nine days shorter than that of the U_3Si_2 fuel core.

— The U–Mo fuel 3 core substantially exceeds all of the design criteria. In particular, the excess reactivity increases dramatically, and the number of operating days per cycle is also expected to increase. However, fast and thermal neutron fluxes decrease because of increased ^{235}U in the core, resulting in additional neutron absorption without fission, and the irradiation performance decreases drastically. From the calculation results, it is found that it is possible to improve the cumulative power per FA by introducing high density U–Mo and increasing the U amount per core.

In order to meet the design criteria for the JMTR core without significant degradation of the irradiation performance, it is necessary to investigate not only the change of the fuel, but also change of the core configuration and the FA, control rod, cooling system and so on.

I–3.4. Possibility of U–Mo fuel for introduction to the JMTR

In the analytical investigation described above, despite the application of high density fuel, it is impossible to improve the cumulative power per FA within the neutronic design criteria for the JMTR with the previously proposed fuels. Therefore, based on U–Mo fuel 2, which satisfies the design criteria for the JMTR, U–Mo fuel 2′ was considered.

To meet the design criteria for the quantity of ^{235}U (11 kg) in the improved LEU core, for U–Mo fuel 2′ the enrichment of U, the concentration of Mo and the density of U are roughly estimated at 13.50%, 7wt% and 8.0 gU/cm^3, respectively. U–Mo fuel 2′ is listed in Table I–6 with U–Mo fuel 1 and U–Mo fuel 2. The excess reactivity at the BOC when using U–Mo fuel 2′ is roughly estimated to be ~10.96%$\Delta k/k$. Even though this is a little lower than the excess reactivity of U_3Si_2 fuel, Table I–5 shows that there is an increase in the number of operating days for 15% burnup, even with an excess reactivity of as low as 9.47%$\Delta k/k$, because of the increased quantity of U in the core. With this rough estimation, the fuel economy is thought to be the same as for the U_3Si_2 fuel core.

In order to improve fuel economy, it has already been found that operation for about 190 days per year (an increase of approximately 10 days) will be possible when using U_3Si_2 fuels by changing the existing three cycle core to a six cycle core with 50 MW without increasing fuel consumption [I–9]. Moreover, if U–Mo fuel is developed that has higher burnup than the U_3Si_2 fuel, the fuel economy will be expected to improve. Therefore, it is concluded that the high density with higher burnup U–Mo fuel will be applicable to the JMTR core together with the refuelling technique without significant irradiation performance change, and that the U–Mo fuel will increase the cumulative power per fuel and improve the fuel economy while meeting the JMTR design criteria.

I–4. CONCLUSIONS

A U_3Si_2 fuel with LEU has been used in the JMTR since 1994. Given the performance advantages of a high density U–Mo fuel with regard to high burnup and easy reprocessing, an analytical investigation of the introduction of U–Mo fuel to the JMTR was carried out. The investigation results indicate that U–Mo fuel will be applicable to the JMTR, showing almost the same irradiation performance as the U_3Si_2 fuel, while satisfying the neutronic design criteria (i.e. the fuel economy of the U–Mo fuel is almost the same as the that of the U_3Si_2 fuel).

In this situation, it is thought that two effective methods will improve fuel economy for the JMTR:

(1) The adoption of U–Mo fuel showing higher density and higher burnup than U_3Si_2 fuel with an increase in fuel cycles for each FA and shuffling to increase the cumulative power per FA;
(2) The adoption of U–Mo fuel if it is cheaper than U_3Si_2 fuel.

TABLE I–6. POSSIBLE U–Mo FUEL[a] INTRODUCTION TO THE JMTR

Name	U-235 amount in the core (g)	U density (gU/cm^3)	U-235 enrichment (%)	Remarks
U–Mo fuel 1	9.629	4.8	19.75	Without change of density and enrichment
U–Mo fuel 2	9.629	8.0	11.85	Without change of U-235 amount in the core
U–Mo fuel 2’	10.965	8.0	13.5	Rough estimation

[a] Content of Mo in U–Mo alloy: 7wt%.

REFERENCES

[I–1] JAPAN ATOMIC ENERGY RESEARCH INSTITUTE, Conceptual Design of the Japan Materials Testing Reactor, Rep. JAERI 1056, JAERI, Tokai-muri (1964).

[I–2] TAKEMOTO, N., et al., Simulator for Materials Testing Reactors, Rep. JAEA-Technology 2013-013, JAEA, Tokai-muri (2013).

[I–3] U–MO FUEL RESEARCH WORKING GROUP, U–Mo Fuel Research Working Group Report (FY2008), Rep. JAEA-Technology 2009-062, JAEA, Tokai-muri (2009).

[I–4] DEPARTMENT OF JMTR OPERATION, Annual Report of JMTR, FY2006, 1 April 2006–31 March 2007, Rep. JAEA-Review 2008–006, JAEA, Tokai-muri (2008).

[I–5] TSUCHIHASHI, K., ISHIGURO, Y., KANEKO, K., IDO, M. (Eds), Revised SRAC Code System, Rep. JAERI 1302, JAERI, Tokai-muri (1986).

[I–6] OKUMURA, K., KANEKO, K., TSUCHIHASHI, K., SRAC95; General Purpose Neutronics Code System, Rep. JAERI-Data/Code 96-015, JAERI, Tokai-muri (1996).

[I–7] KOMUKAI, B., et al., Neutronics Analysis of the Improved LEU Core in JMTR, Rep. JAERI-Tech 2002-067, JAERI, Tokai-muri (2002).

[I–8] NAGAOKA, Y., et al., Neutronics Design of JMTR–LEU Core, Rep. JAERI-M 92-098, JAERI, Tokai-muri (1992).

[I–9] TAKEMOTO, N., NAKA, M., NAGAO, Y., “Investigation of optimization on in-core fuel management with high reactor-availability-factor of JMTR”, paper presented at 12th Mtg International Group on Research Reactors, Beijing, 2009, IGORR (2009).

Annex II

COMPARATIVE STUDY OF FUEL MATERIALS FOR THE KIJANG RESEARCH REACTOR

C.G. Seo

Reactor Core Design Division, Korea Atomic Energy Research Institute, Deajeon, Republic of Korea

II–1. INTRODUCTION

The Korea Atomic Energy Research Institute (KAERI) has been developing a new research reactor, the Kijang Research Reactor (KJRR) [II–1]. The KJRR will mainly be used for isotope production, neutron transmutation doping (NTD) production and related research activities. The KJRR is a medium flux reactor of 15 MW using materials testing reactor (MTR) type fuel assemblies (FAs), which use uranium–molybdenum (U–Mo) with 7wt% Mo (U–7Mo) dispersion fuel, with a uranium density of 8.0 gU/cm^3, as a reference fuel. Its fuel has not yet been fully qualified, but the KJRR adopts high density fuel for higher fuel economy. A silicide uranium fuelled KJRR core was prepared in order to understand the U–7Mo core well and to obtain a fallback option [II–2]. This report presents a nuclear comparative study on the differently fuelled cores.

II–2. NUCLEAR ANALYSIS

II–2.1. Core configuration

The core configuration was optimized according to its purpose. The core is located within a core box, which will prevent core uncovering in any emergency state. The core design is strongly dependent on the number of in-core irradiation positions and control absorber rods (CARs).

A core model with three in-core irradiation sites fully surrounded with FAs was selected, as shown in Fig. II–1 [II–3]. This core is composed of a 7 × 9 lattice with an active length of 60 cm. The nominal core consists of 22 FAs, in which 16 standard and 6 follower FAs are loaded. There are six detachable CARs to control and shut down the reactor. The reactor regulating system shares four CARs with the reactor protection system, and these are driven by stepping motors. The independent secondary shutdown system uses two CARs, which are fully withdrawn by hydraulic force at a normal operation state. The arrangement of the CARs is studied carefully to minimize the flux perturbation and maximize the reactivity worth.

Figure II–2 shows that six fission molybdenum targets are inserted at the lateral positions. A hydraulic transfer system (HTS) is located within the core box. Two pneumatic transfer systems and five NTD positions are located outside the core box. One position is prepared for the fast neutron irradiation facility, which can easily be used for NTD. The outside of the core box is surrounded with aluminum, beryllium and graphite; its materials are chosen depending on their accessibility and the fast flux level.

The standard fuel and follower fuel have the same box size, as shown in Fig. II–3. When a follower fuel is loaded into the core, an Hf absorber is attached to the end of the fuel. As the FA and Hf absorber are moving together, a larger control rod worth is available to control the KJRR core with large uranium loading. Each KJRR FA has two types of fuel plates, one with a fuel meat density of 8.0 gU/cm^3 and the

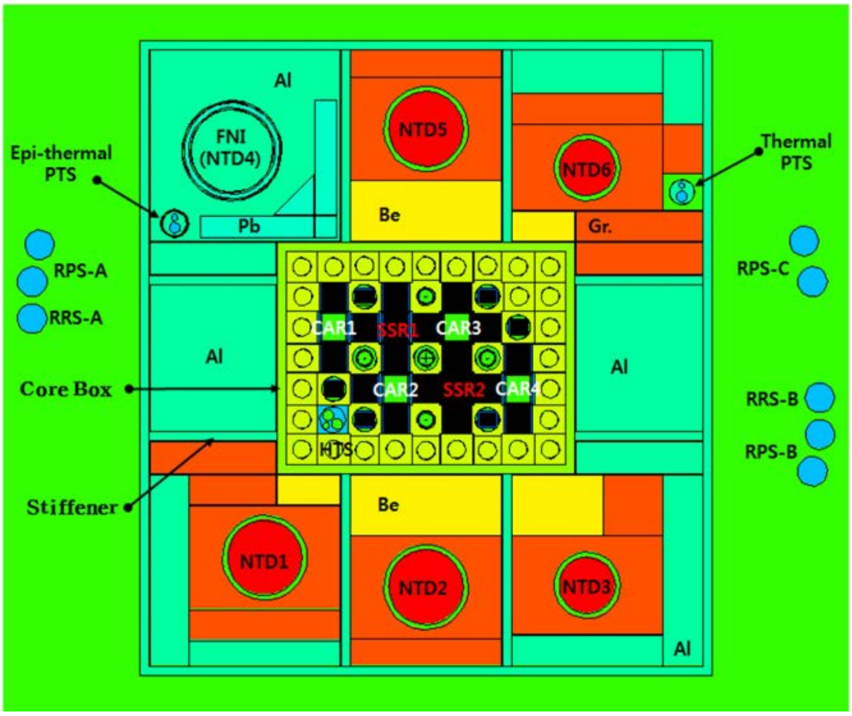


FIG. II–1. Plan view of the KJRR core. Reproduced courtesy of KAERI [II–3].

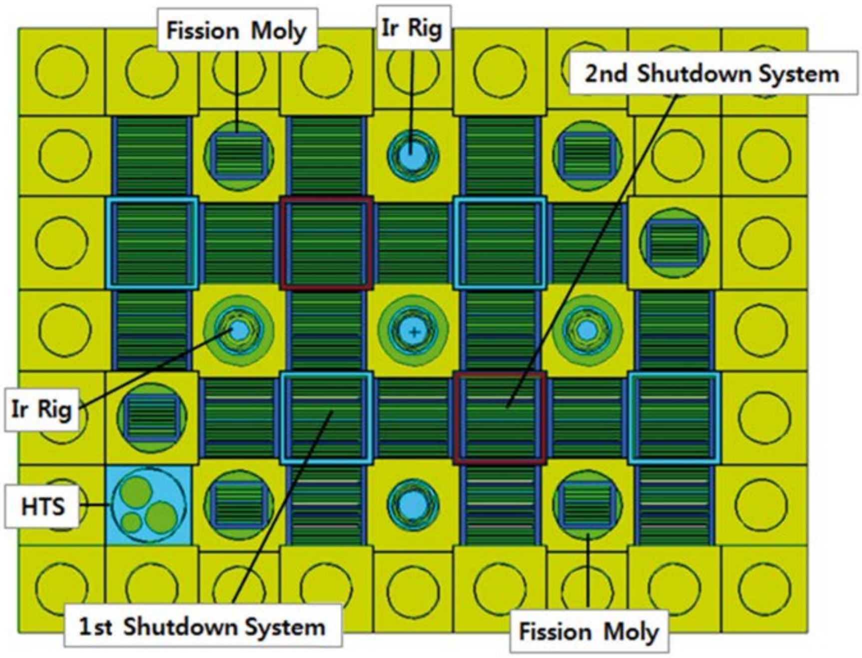


FIG. II–2. Core configuration of the KJRR. Reproduced courtesy of KAERI [II–1].

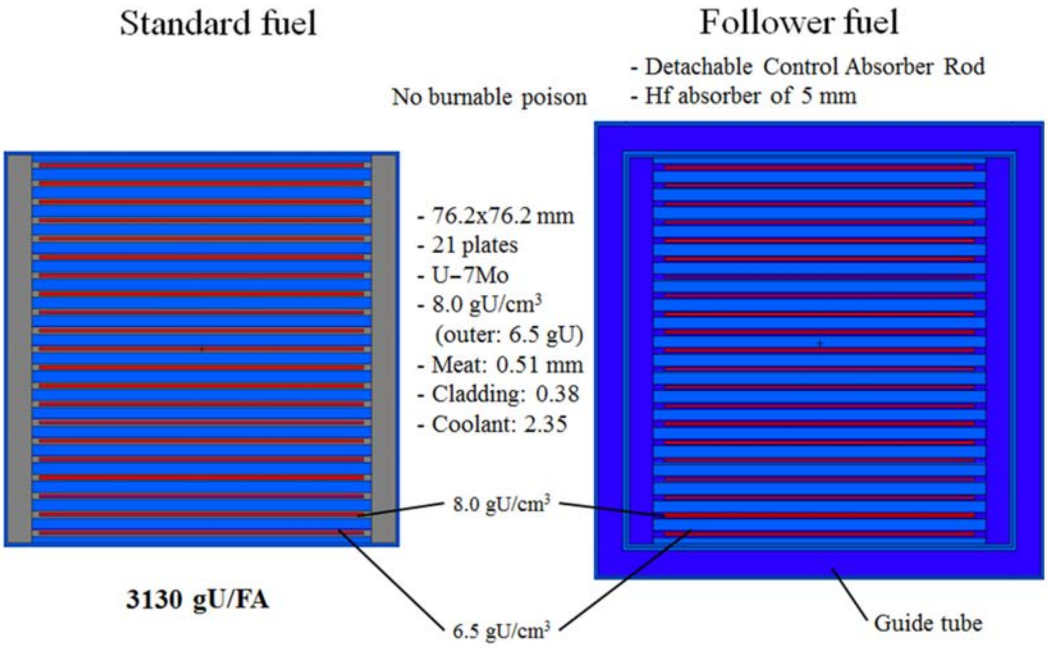


FIG. II–3. Cross-sectional view of the KJRR FAs. Reproduced courtesy of KAERI.

other with one of 6.5 gU/cm^3. The FA is filled with 19 interior fuel plates of 8.0 gU/cm^3 and two exterior fuel plates of 6.5 gU/cm^3. The total uranium inventory of each FA is 3130 g.

II–2.2. Analyses and results

In this analysis, the McCARD code, a Monte Carlo (MC) neutron–photon transport simulation code designed for neutronic analyses of various nuclear reactor and fuel systems, was used to obtain detailed information for the burnt cores. McCARD is capable of burnup analysis using the built-in depletion equation solver module. Unlike with many existing MC burnup analysis codes, it is not necessary to couple the MC neutronic analysis modules with an external depletion code [II–4]. The McCARD code uses a continuous energy library based on ENDF/B-VII.

At the current design stage of KJRR, nuclear analyses are mainly performed for an equilibrium cycle of a reference core. An equilibrium core is dependent on an operation strategy, so there may be various equilibrium cores according to reactor operating strategies and reactor utilization needs. Two fresh FAs are loaded for one cycle operation considering discharge burnup, cycle length and excess reactivity at beginning of cycle (BOC) and end of cycle (EOC). As there are many loading patterns, a sophisticated study is required. A loading pattern is selected to satisfy all design requirements at the same time. As a loading pattern is determined, a fresh core converges to an equilibrium core by repeated core calculations. For the selected equilibrium core, the cycle length was estimated to be 50 days.

Even though the KJRR is designed to work optimally with U–7Mo fuel, it is of interest to consider and study an alternative fuel, since this analysis will enable a better understanding of U–7Mo fuelling. The U_3Si_2 fuel, with 4.8 gU/cm^3 density, is already being used in many research reactors, which is why it has been selected as the alternative fuel for this study.

Lower uranium loading in the FAs reduces its cycle length. The silicide FA only uses fuel plates of U_3Si_2 (4.8 gU/cm^3). So, the total uranium loading of the U_3Si_2 fuelled core is 42.1 kgU.

When two fresh U_3Si_2 FAs are loaded at every cycle, the cycle length is estimated to be 25.5 days. A cycle length with four fresh FAs is 50 days — larger batch size helps the core use the FAs more efficiently. Figure II–4 shows how cycle lengths are dependent on uranium loading and batch size.

The design requirement for cycle length is 37.5 days for the KJRR. For the U_3Si_2 fuelled core, three or four fresh FAs are required to fulfil the minimum cycle length. It was determined that the appropriate number of FAs is three and also that the reactor power can be reduced to 14.5 MW for the required cycle length and the same neutron flux. Its reactivity swing is compared to that of the U–Mo core in Fig. II–5.

The reactivity swing is almost the same, but the U–7Mo core is better from the viewpoint of utilization. The basic reactor physics parameters were generated using the different FA loading patterns, but the new pattern for the U_3Si_2 fuelled core is based on that of the U–7Mo core. The compared data are summarized in Table II–1.

In Fig. II–6, the thermal fluxes at the U_3Si_2 fuelled core are compared with those of the original U–7Mo fuel. In the U_3Si_2 fuelled core, maximum thermal flux is increased by ~9.4%, but thermal flux at the reflector region is reduced by a power decrease of the U_3Si_2 core, with a difference of core power of approximately −3.3%. The changed flux differences at each site are caused by the different loading patterns. And because thermal fluxes at the in-core region are higher with U_3Si_2 fuel, the power of the molybdenum targets increases by about 9.2%.

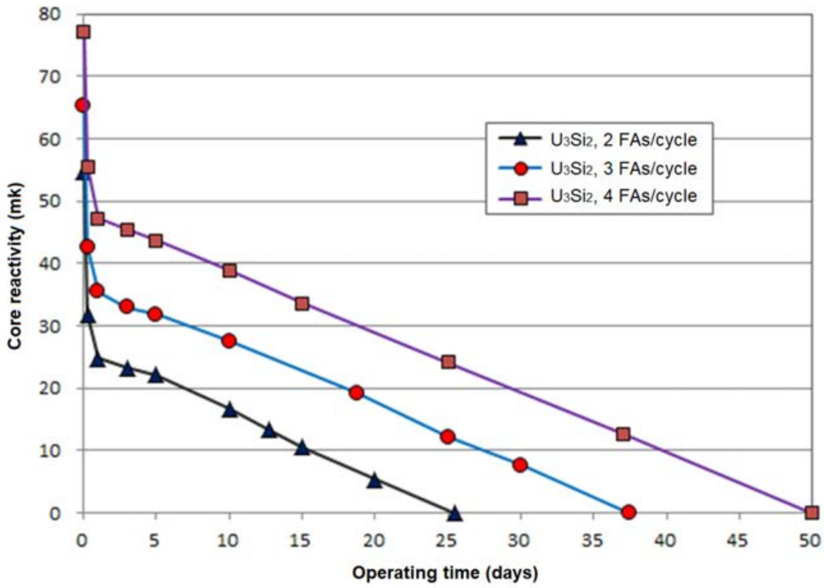


FIG. II–4. Fuel loading and cycle length. Reproduced courtesy of KAERI.

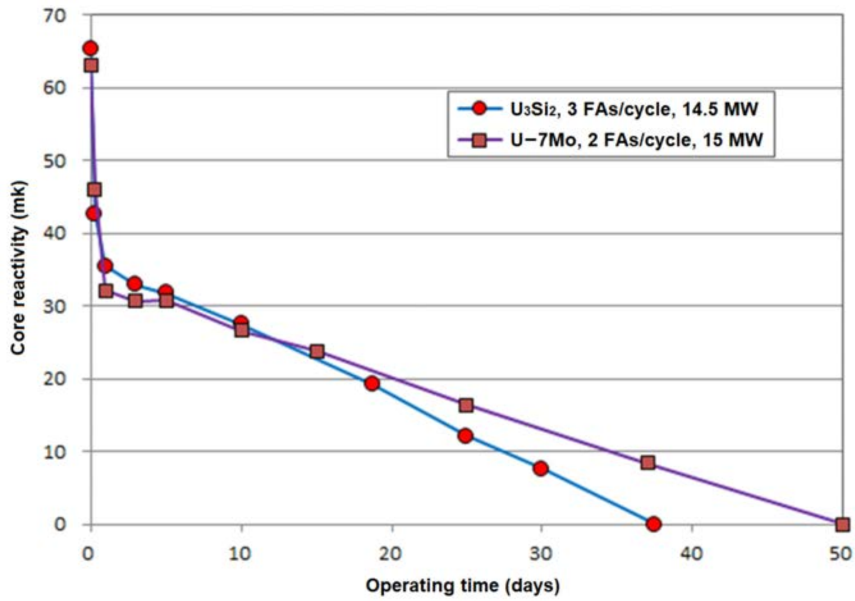


FIG. II–5. Comparison of reactivity swing. Reproduced courtesy of KAERI.

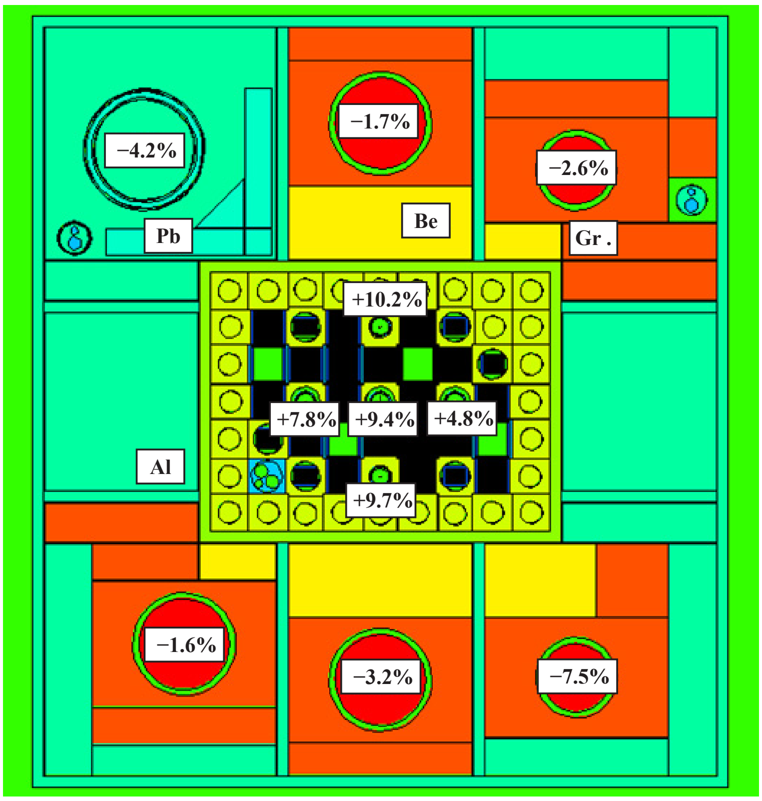


FIG. II–6. Comparison of thermal flux. $\left[\% = \left(\frac{U_3Si_2}{U-7Mo} - 1\right) \times 100\right]$. *Reproduced courtesy of KAERI.*

TABLE II–1. COMPARISON OF MAJOR REACTOR PHYSICS PARAMETERS

Parameters		U–7Mo core	U_3Si_2 core
Reactor power (MW)		15	14.5
Max. thermal neutron flux (n/cm^{2}s)		3.2×10^{14}	3.2×10^{14}
Cycle length (days)		50	37.5
FA consumption per year		12 FA (37.56 kgU)	24 FA (45.89 kgU)
Local peak burnup (%U-235)		86.0	78.0
Reactivity swing (mk)		63	65
Shutdown margin (mk)		32	49
Max. power peaking factor		2.58	2.38
Power defect (mk)		−1.23	−1.11
Beta effective ($\times 10^{-3}$)	BOC	6.72	6.84
	EOC	6.72	6.77
Prompt neutron lifetime ($\times 10^{-4}$ s)	BOC	1.55	1.68
	EOC	1.59	1.75

The cycle length of the U_3Si_2 core is shorter than that of the U–7Mo core, and it requires 50% more FAs per batch. As a result, its FA consumption per year is doubled and its uranium consumption per year increases by about 22%. These increases make the benefit of the U–7Mo core clear in terms of fuel economy.

Table II–1 shows that the shutdown margin of the U–7Mo core is smaller than that of the U_3Si_2 core, but different cycle lengths are being considered. On the other hand, Fig. II–7 shows a comparison of the reactivity swing for both fuels with the same 50 day cycle length being considered. The uranium loading of the U_3Si_2 core is 7.648 kgU per cycle, which is larger than the 6.26 kgU of the U–7Mo core. Comparison of the two cores adjusted to have the same 50 day cycle length shows that the reactivity swing of the U_3Si_2 core is larger by about 17 mK and the shutdown margin of the U_3Si_2 core is smaller by 5 mK (i.e. the increase of uranium loading for the same cycle length reduces the shutdown margin).

Power peaking factors (PPFs) are strongly dependent on the CAR position. The PPFs in Table II–1 were compared under almost the same conditions. High density fuel adversely increases the PPF. The U–7Mo core uses lower density fuel of 6.5 gU/cm^3 at two exterior plates to reduce its high peaking factor, but the transverse peaking factor is not fully suppressed. Figure II–8 compares the transverse relative power factors at each of the hottest fuel plates.

The power defects were evaluated under the same temperature conditions. The power defect comprises the fuel, coolant and moderator temperature defects. It is assumed that the difference of the power defects is caused by the fuel temperature defect. Figure II–9 confirms that this assumption is valid.

The kinetic parameters were changed by the core burnup and the neutron spectrum. Figure II–10 shows that the neutron spectrum of the U–7Mo core is slightly hardened.

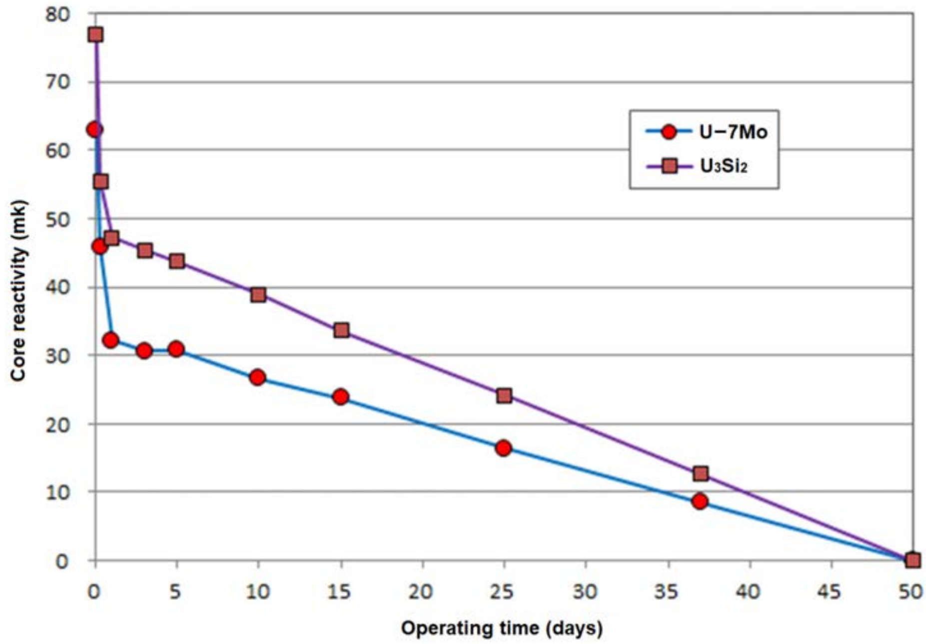


FIG. II–7. Comparison of reactivity swing at the same cycle length. Reproduced courtesy of KAERI.

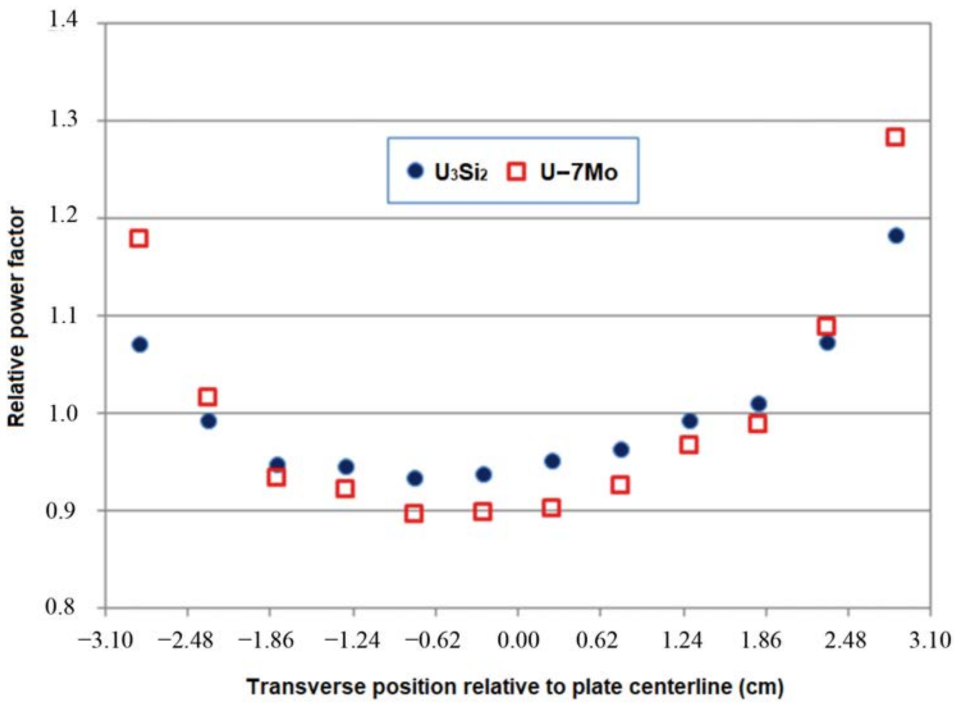


FIG. II–8. Comparison of transverse relative power distribution. Reproduced courtesy of KAERI.

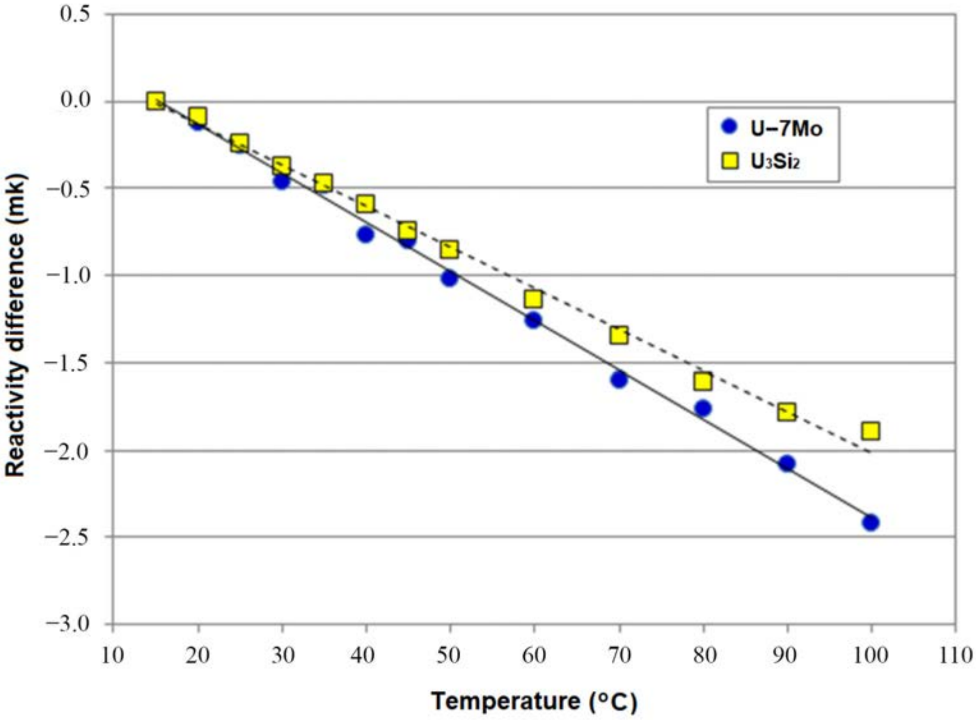


FIG. II–9. Comparison of fuel temperature defects. Reproduced courtesy of KAERI.

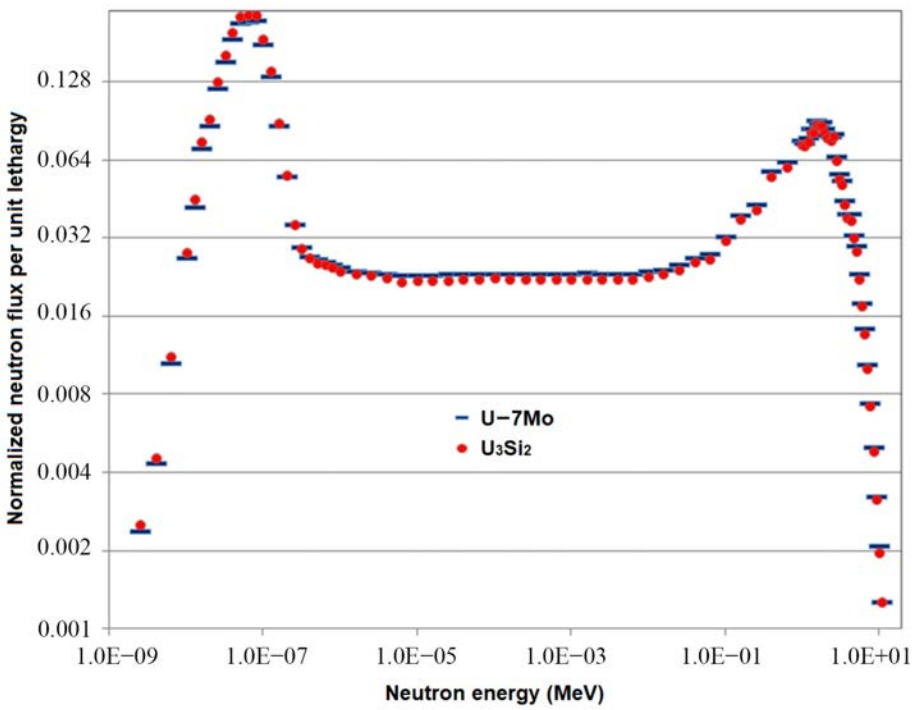


FIG. II–10. Comparison of neutron flux spectra. Reproduced courtesy of KAERI.

II–3. CONCLUSIONS

When comparing the U_3Si_2 fuel (fallback) to the U–7Mo fuel (reference) in the KJRR core, the following differences are noted:

- The total reactivity worth of the CARs is smaller with U–7Mo fuel, but the difference in shutdown margins is negligible when compared at the same cycle length.
- The reactivity effects on irradiation targets are reduced by about 6–8% for the U–7Mo core.
- The higher burnup and hardened neutron spectrum of the U–7Mo core change the kinetic parameters. In particular, the hardened spectrum makes the thermal flux in the core region lower.
- The U–7Mo core has a more negative power defect, which might be caused by a more negative fuel temperature defect.

In addition, two severe disadvantages of U–7Mo fuel are identified in comparison with the U_3Si_2 fuel:

(1) High power peaking at the U–7Mo core leads to the use of two kinds of fuel plates in the KJRR FA, but the peaking factor is still higher than that of the U_3Si_2 core.
(2) The change in thermal neutron flux at the reflector region is negligible between U–7Mo and U_3Si_2 cores. However, at the core region, the thermal flux is significantly lower for the U–7Mo core.

Overall, however, the use of high density fuel satisfies all of the defined design requirements and provides a significant economic benefit:

- The U–7Mo fuel loading results in a smaller reactivity swing, which in turn prevents the need for any burnable poison.
- The batch size of the U–7Mo core increases and thus the uranium resource can be used more efficiently.

REFERENCES

[II–1] SEO, C.G., CHAE, H.T., LEE, B.C., JUN, B.J., LIM, I.C., "Conceptual nuclear design of the Kijang research reactor", Trans. Topical Mtg on Research Reactor Fuel Management, St. Petersburg, 2013, ENS, Brussels (2013) 491–496.

[II–2] SEO, C.G., CHAE, H.T., SONG, J.S., WU, S.I., "Silicide fueled core for the Kijang research reactor", Proc. of the Korean Nuclear Society Fall Meeting, Kyungju, 2013, KNS, Daejeon (2013).

[II–3] PARK, C., KIM, Y.K., LEE, B.C., RYU, J.S., KWON, Y.S., "Overview of KJRR design features", 11th Natl Conf. on Nuclear Science and Technology, Da Nang City, 2015, VAEI, Hanoi (2015).

[II–4] SHIM, H.J., HAN, B.-S., JUNG, J.-S., PARK, H.J., KIM, C.-H., McCARD: Monte Carlo code for advanced reactor design and analysis, Nucl. Eng. Technol. **44** 2 (2012) 161–176.

Annex III

CONVERSION OF OPAL FROM U_3Si_2 TO HIGH DENSITY U–Mo FUEL

G. Braoudakis

Nuclear Analysis Section, Australian Nuclear Science and Technology Organisation (ANSTO), Lucas Heights, Australia

E. Villarino

Nuclear Engineering Department, INVAP S.E., San Carlos de Bariloche, Argentina

III–1. INTRODUCTION

The ongoing development of high density low enriched uranium (LEU) fuels to replace the current application of highly enriched uranium (HEU) fuels, in accordance with the non-proliferation and security objectives of the Global Threat Reduction Initiative (GTRI), offers an opportunity to assess the impact of such fuels on performance and reactor core parameters. The performance of the current 4.8 gU/cm^3 U_3Si_2 fuel in the Open Pool Australian Lightwater (OPAL) reactor is compared against that estimated for a variety of potential U–Mo high density fuels that range in density from 6.0 to 8.0 gU/cm^3. The comparison includes cycle length, fuel usage, shutdown system performance, neutron fluxes in irradiation positions, kinetic parameters and reactivity feedback coefficients.

High performance research reactors depend on a compact core design to maximize the neutron flux available for irradiation and beam facilities. The most readily available technology to achieve this goal is the use of HEU fuels. The extensive experience and demonstrable reliability and performance of such fuels makes them a clear choice for most of the high performance research reactors around the world. In addition, there are gains in fuel economy through maximization of fuel burnup and minimization of parasitic neutron absorption, which is present in some of the LEU fuels. In the spirit of GTRI [III–1], new high density LEU fuels are being developed with the intention of providing a non-proliferation option while maintaining much of the performance expected from HEU fuels. One of the most promising high density LEU fuels is a uranium–molybdenum (U–Mo) alloy dispersion type fuel [III–2] in an aluminium matrix. The addition of Mo stabilizes the uranium during irradiation and several different densities of uranium and alloy compositions are considered in this study, as no qualified fuel exists at this time.

The current core design is optimized for the use of 4.8 gU/cm^3 U_3Si_2 fuel. This study was performed while maintaining the core and fuel dimensions (specifically the fuel meat, fuel plate and coolant channel thicknesses) and only the fuel meat composition was modified. The estimated performance parameters were calculated using the same codes and methods that were used for the U_3Si_2 calculations. In this way a direct comparison can be made to assess the impact of using high density U–Mo fuel.

III–2. OPAL RESEARCH REACTOR DESCRIPTION

The OPAL reactor is a multipurpose open pool type research reactor. The nominal power is 20 MW. A compact core is situated inside a chimney surrounded by heavy water contained inside a reflector vessel. The whole assembly is located at the bottom of the reactor pool, which is filled with light water that acts as the coolant, moderator and biological shielding.

The core is a 4 × 4 array of 16 flat plate type fuel assemblies (FAs) with five control rods that comprise the first shutdown system and reactivity control during operation. Each FA is square in cross-section with 21 fuel plates and 20 cadmium wires as burnable poison for reactivity compensation and power peaking trimming. The coolant flow is upwards through the core. A second diverse and independent shutdown system is provided by partial drainage of the heavy water in the reflector vessel. There are no irradiation positions within the core.

The compact core is designed to optimize the flux within the reflector vessel, where numerous irradiation facilities are situated for radioisotope production, neutron activation analysis (NAA) and neutron transmutation doping (NTD). Five tangential neutron beams are also located within the reflector, including two coupled to a cold neutron source to provide thermal and cold neutron beams for research. A schematic of the reactor, including the core and facilities, is presented in Fig. III–1, with a more detailed view of the core in Fig. III–2.

III–3. SCOPE OF STUDY

This study considers the impact of changing the composition of the fuel meat from U_3Si_2 at 4.8 gU/cm^3 to high density U–Mo dispersion fuel. A range of uranium densities and fuel compositions were considered for the U–Mo fuel. The parameters assessed include FA usage, core reactivity, shutdown margin, power peaking factor (PPF), performance of the second shutdown system, neutron fluxes in the irradiation and beam facilities, reactivity feedback coefficients and kinetic parameters. For the last two sets of parameters only one U–Mo composition was evaluated, and this is considered to be sufficient to assess the impact of high density fuel within the scope of this study.

III–4. CALCULATION METHODOLOGY

The calculation methodology relied on the same codes, tools and models as those used for the current reactor design and existing calculations. The methodology is based on the INVAP MTR-PC [III–3] suite of codes. The calculation line [III–4] follows the usual method for deterministic calculations with the basic steps being multigroup cross-section library (group constants) generation followed by global reactor calculations. The various regions of the reactor, including FA, control rods, core structures, heavy water reflector and the various reflector facilities (irradiation and beam) were modelled individually using the code CONDOR in two dimensional geometry; the collision probability method was used to solve for flux. The calculations were carried out in 69 groups using the WIMS [III–5] library with some isotopes from the HELIOS library. The solution is condensed to a three group structure for global reactor calculations. The details for the FA in the cell calculations are provided in Fig. III–3, which shows a one quarter model of the FA with individual fuel meat, clad, coolant channel, side plate and burnable cadmium wires in the side plate.

The global reactor calculations were performed using the CITVAP code, which is a modified version of CITATION-II [III–6] that solves one, two and three dimensional multigroup diffusion problems. The models used were three dimensional in rectangular geometry. The FAs were represented using distinct fuel plate regions and non-fuelled side plate regions. The heavy water was represented by three different materials, depending on the distance from the core, to account for spectral changes in the neutron flux. Burnup calculations were performed along a cycle with control rods moved to search for critical configurations.

III–5. FUEL COMPOSITION

The fuel currently licensed and used in OPAL is U_3Si_2 dispersion fuel with a density of 4.8 gU/cm^3. Results are presented for this fuel along with several different densities and compositions of U–Mo dispersant fuel. The relevant characteristics of the fuels are presented in Table III–1. A range of uranium densities was considered to evaluate the dependence of the various reactor performance characteristics on density. The composition has also been selected to reflect what could reasonably be expected for qualified fuels.

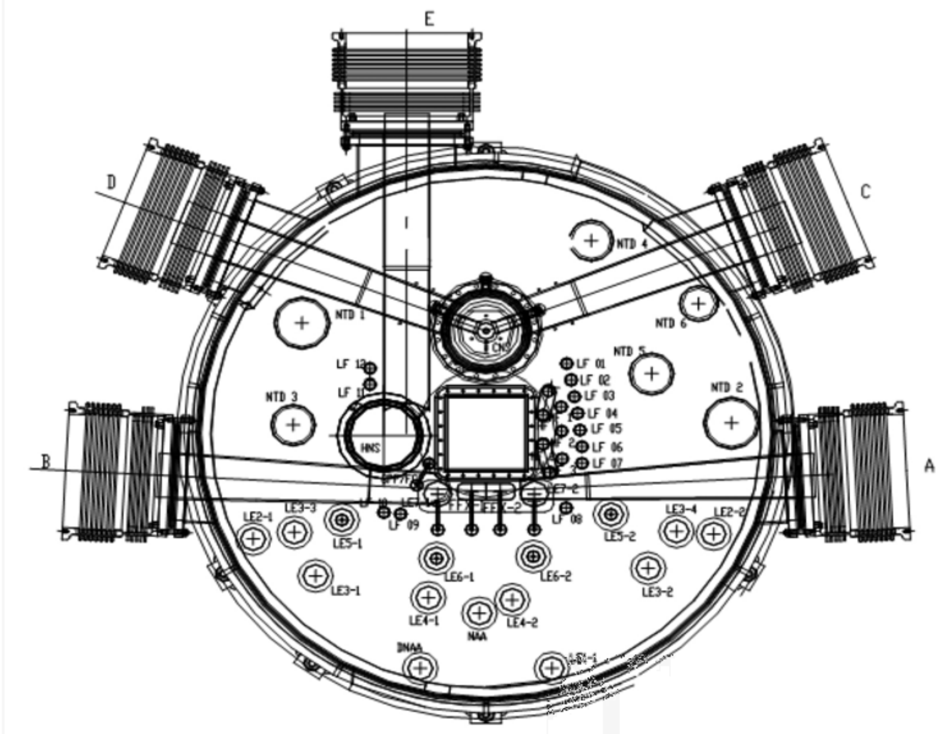


FIG. III–1. Schematic of the OPAL research reactor showing the various irradiation and beam facilities. Reproduced courtesy of INVAP.

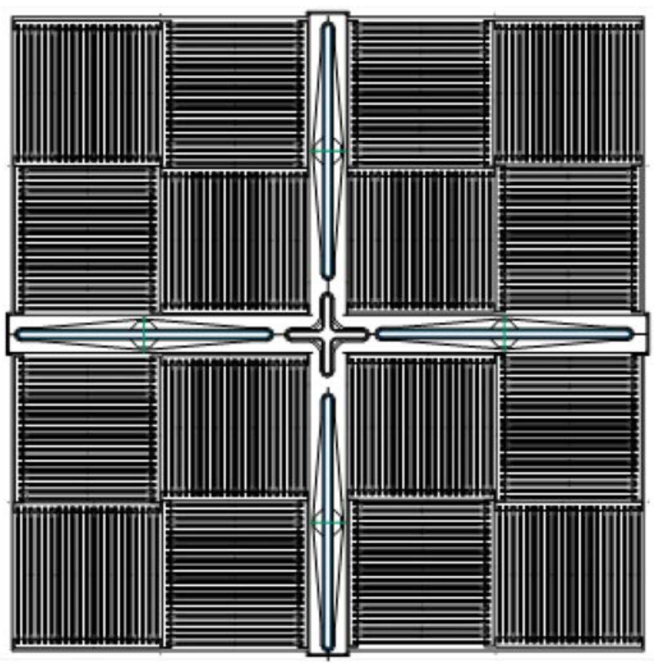


FIG. III–2. Schematic of the OPAL reactor core showing the 16 FAs and 5 control rods. Reproduced courtesy of INVAP.

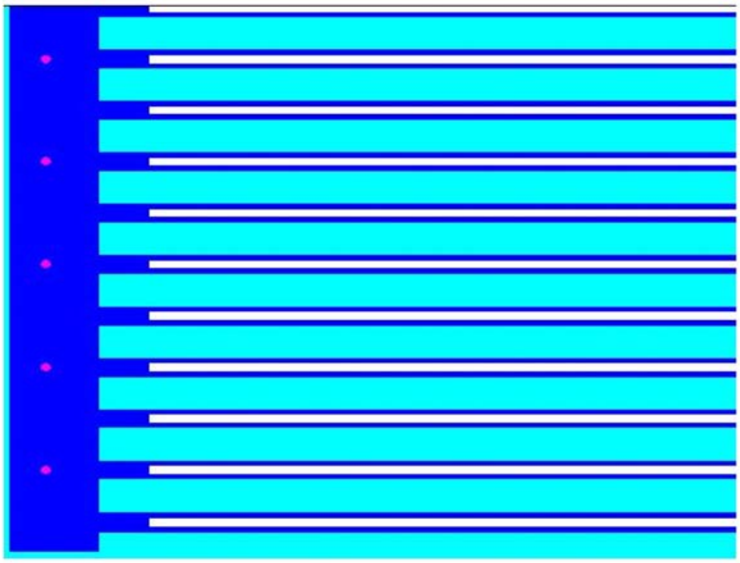


FIG. III–3. Details for the FA cell calculation. Reproduced courtesy of INVAP.

TABLE III–1. MAIN CHARACTERISTICS OF LEU HIGH DENSITY FUELS

FA type	Label	U density (gU/cm^3)	U-235 (g)
U_3Si_2	Si480	4.8	~484
U–6%Mo	Mo606	6.0	~605
U–9%Mo	Mo709	7.0	~706
U–6%Mo	Mo706	7.0	~706
U–7%Mo	Mo807	8.0	~809

III–6. RESULTS AND DISCUSSIONS

Results are presented for the calculations performed to assess the effect of using high density U–Mo fuels in the OPAL reactor.

III–6.1. General core parameters

The current fuel management strategy at OPAL consists of replacing three Si480 FAs at the end of each cycle. The cycle length is nominally 30 days. The increased uranium mass density of the high density U–Mo fuel and the constant dimensions and geometry of the FAs will result in a corresponding increase in the uranium loading of the FAs. This needs to be considered in the management of core reactivity and to ensure the reactor can be safely shut down. The aim is to use the higher fuel loading to either increase the fuel cycle length or decrease the number of FAs replaced every cycle, or a combination of both. The result will be a reduction in the amount of fuel (either FAs or mass of ^{235}U) used. The criterion adopted for this study to determine the length of the cycle is that the end of cycle (EOC) reactivity is greater than 1200 pcm and the cycle length is greater than 26 days. These constraints determined the minimum number of FAs changed every cycle and the upper limit was determined by the shutdown margin required with single failure (SDM-1) (highest worth control rod fails to drop).

The analysed fuel management strategy implemented for the high density fuel cases was the same for all high density fuel types, as this resulted in the same number of new FAs every cycle, but different from that of the Si480 case. The basic strategy was to load fresh FAs in the outer positions and gradually move them to the central core positions. Core reactivity was calculated with all control rods fully extracted. The shutdown margin was calculated with all control rods fully inserted and a single failure with the highest control rod fully extracted. The PPF was calculated as the ratio of the highest power density anywhere in the core to the average core power density. This presents the most challenging condition in the core from a thermohydraulic perspective.

The results for the various core performance parameters are presented in Table III–2. In all high density fuel cases, the required minimum cycle length can be met by replacing only two FAs as opposed to the current three FAs for Si480. Along with the increased cycle length permitted by the denser fuels, this reduces the number of FAs to about half for the densest fuel considered. These benefits need to be balanced against some of the penalties, such as increased PPF, decreased SDM-1 and decreased worth of the second shutdown system. These are all safety significant, as they decrease existing safety margins, although assessing the real impact requires review and possibly revision of the current safety analyses. Such work falls outside the scope of the present study.

III–6.2. Reactivity coefficients

The reactivity coefficients are significant in terms of reactor dynamics and response to transients, in particular, reactivity insertion accidents. Ideally, reactors are designed with negative reactivity coefficients so that transients are self-limiting. The reactivity feedback coefficients were calculated for the relevant fuel, coolant and reflector contributions for one type of high density U–Mo fuel, Mo709, to assess the expected impact.

TABLE III–2. GENERAL REACTOR CORE PARAMETERS FOR U_3Si_2 AND HIGH DENSITY U–Mo FUELS

	Cycle length (days)	FA per cycle	FA annual consumption	Hot with Xe EOC reactivity (pcm)	PPF	SDM (pcm)	SDM-1 (pcm)	SSS (pcm)
Limit	>26.0	n.a.	n.a.	>1 200	>3.0	>3 000	>1 000	>3 000
Si480	30.0	3	34.2	1 510	2.14	10 760	6 130	6 370
Mo606	27.8	2	24.5	1 420	2.38	10 160	5 790	6 000
Mo709	33.4	2	20.6	1 430	2.51	9 230	5 110	5 910
Mo706	34.5	2	20.0	1 450	2.53	8 980	5 180	6 300
Mo807	40.5	2	17.2	1 400	2.65	8 350	5 170	5 740

Note: n.a.: not applicable; EOC — end of cycle; FA — fuel assembly; PPF — power peaking factor; SDM-1 — shutdown margin with single failure; SSS — second shutdown system.

The various temperature coefficients were calculated using differences in core reactivity due to a 50°C increase in temperature of that material. Coolant and reflector coefficients were calculated for temperature increases at constant density and with the associated density change. The coolant void corresponds to a 10% decrease in density and the reflector void to a 5% decrease in density.

Results for hot (fuel and coolant temperatures at nominal full power and equilibrium Xe) and cold (fuel and coolant temperatures at shutdown and no Xe) states are presented in Table III–3. The higher density fuel increases the moderation ratio (i.e. the ratio between fuel and moderator) and hardens the neutron spectrum. These changes appear to increase the absolute value of the fuel temperature feedback coefficient but decrease the coolant and reflector temperature coefficients at constant density. As the densities of the coolant and reflector decrease (increasing void), the spectrum effects become important and the absolute value of the void feedback coefficient increases for the high density fuel. The changes in reactivity feedback coefficients are relatively small, with the largest difference being in the case where the coolant temperature feedback is 13.6% for the hot state and 11.3% for the cold state. This complex dependence of reactivity coefficients on the fuel density means that the ultimate impact during transients can only be determined by detailed transient analyses for the scenarios of interest. Such analyses are beyond the scope of this report, but most of these changes are within the uncertainties for these parameters. The impact on the outcome of the transient analyses is, therefore, expected to be minor.

III–6.3. Kinetic parameters

The kinetic parameters are significant in terms of reactor dynamics and response to transients, in particular, reactivity insertion accidents. High density fuels change the neutron spectrum and uranium–plutonium (U–Pu) ratios that change the prompt neutron lifetime and delayed neutron fraction. The kinetic parameters were calculated for one type of high density U–Mo fuel, Mo709, to indicate the expected impact. These values were calculated directly by CITVAP from the flux and adjoint weighted flux solutions. The results are presented in Table III–4 for hot and cold beginning of cycle (BOC) and end of cycle (EOC) cores. The changes in effective delayed neutron fraction are about 2% and for prompt neutron lifetime about 4%. Such changes are minor and will likely have little impact on the outcome of transients.

TABLE III–3. THE IMPACT OF HIGH DENSITY U–Mo FUEL ON REACTIVITY FEEDBACK COEFFICIENTS

Feedback coefficient	Hot		Cold	
	Si480	Mo709	Si480	Mo709
a_{ft} — fuel temperature (pcm/°C)	−2.05	−2.2	−2.05	−2.2
a_{ct} — coolant temperature density fixed (pcm/°C)	−7.95	−5.9	−7.35	−5.45
a_{ct} — coolant temperature (pcm/°C)	−17.65	−15.25	−13.7	−12.15
a_{cv} — coolant void (pcm/%)	−196.35	−199.05	−191.4	−195.8
a_{rt} — reflector temperature density fixed (pcm/°C)	−1.35	−1.25	−2.05	−1.85
a_{rt} — reflector temperature (pcm/°C)	−8.7	−8.4	−6.4	−6.1
a_{rv} — reflector void (pcm/%)	−133.05	−130.75	−126.85	−124.45

TABLE III–4. IMPACT OF HIGH DENSITY U–Mo FUEL ON KINETIC PARAMETERS

Core state	β_{eff} (pcm)		ι_p (μs)	
	Si480	Mo709	Si480	Mo709
Hot BOC	730	716	176	169
Cold BOC	731	716	180	174
Hot EOC	718	705	181	174
Cold EOC	719	706	186	179

Note: β_{eff} — effective delayed neutron fraction; l_p — prompt neutron lifetime.

III–6.4. Fluxes in irradiation facilities

A key outcome of operating the reactor is to provide neutrons for the various irradiation facilities. There are three main classes of irradiation facilities for OPAL: (i) the bulk irradiation facilities, (ii) the pneumatic facilities, and (iii) the silicon NTD facilities. The main performance parameter for all these is the thermal neutron flux, although for some, flux uniformity and thermal to fast ratio are also of interest. In this study, only the neutron fluxes have been assessed. The thermal flux represents the integrated flux below 0.6 eV and the fast flux represents the integrated flux above 1 MeV.

III–6.4.1. Bulk irradiation facilities

The thermal neutron fluxes for the three groups of bulk irradiation facilities are presented in Table III–5 for Si480 fuel. The relative changes to this caused by the high density U–Mo fuels are also presented. The differences in thermal neutron fluxes for these facilities appear to be small and not of any significance. There is a gradual decrease in flux as the uranium loading increases and there is also a decrease associated with the increased fraction of Mo in the fuel dispersant. Both results are as expected, but these detailed calculations enable the magnitude of this effect to be estimated.

III–6.4.2. Pneumatic irradiation facilities

The pneumatic facilities comprise a series of general facilities that provide seven different levels of thermal neutron fluxes, a neutron activation analysis (NAA) facility, a delayed neutron activation analysis (DNAA) facility and a fast flux (FFX) facility optimized for fast neutron flux. Results are presented in Table III–6 for the various facilities with the absolute flux for the Si480 case and the relative change for

TABLE III–5. THE RELATIVE EFFECT OF HIGH DENSITY U–Mo FUEL ON FLUXES IN THE BULK IRRADIATION FACILITIES

Facility	Reference flux Si480 (n/cm^{2}s)	Change relative to reference flux (%)			
		Mo606	Mo709	Mo706	Mo807
High flux	2.4×10^{14}	0.5	−0.7	0.4	−2.1
Medium flux	1.5×10^{14}	1.4	0.4	0.7	−0.4
Low flux	8.9×10^{13}	−1.0	−2.1	−1.4	−2.6

the high density U–Mo fuels. The fast neutron flux above 1 MeV is indicated for the fast flux facility. The changes are somewhat larger than for the bulk facilities, but the general trends are the same, with a gradual decrease in flux as the uranium loading increases, and there is also a decrease associated with the increased fraction of Mo in the fuel dispersant.

III–6.4.3. Silicon irradiation facilities

There are six large volume irradiation facilities dedicated to silicon NTD. An average flux over the six facilities is reported. The average thermal neutron flux for the Si480 fuel and the effect of higher density fuels is presented in Table III–7. As for the bulk facilities, the changes are small and the general trends are the same, with a gradual decrease in flux as the uranium loading increases; there is also a decrease associated with the increased fraction of Mo in the fuel dispersant.

TABLE III–6. THE RELATIVE EFFECT OF HIGH DENSITY U–Mo FUEL ON FLUXES IN THE PNEUMATIC IRRADIATION FACILITIES

Facility	Reference flux Si480 (n/cm^2s)	Change relative to reference flux (%)			
		Mo606	Mo709	Mo706	Mo807
LE-1	3.6×10^{12}	−1.5	−3.0	−3.0	−3.0
LE-2	8.7×10^{12}	−1.5	−3.0	−3.0	−3.0
LE-3	1.6×10^{13}	−1.5	−3.0	−3.0	−3.0
LE-4	3.2×10^{13}	−1.5	−3.0	−3.0	−3.0
LE-5	5.2×10^{13}	−1.5	−3.0	−2.5	−3.4
LE-6	7.2×10^{13}	−1.4	−3.9	−2.5	−3.4
LE-7	1.2×10^{14}	−3.2	−3.0	−4.3	−5.3
NAA	2.5×10^{13}	−1.5	−3.0	−3.0	−3.0
DNAA	5.9×10^{12}	−1.5	−3.0	−3.0	−3.0
FFX	7.6×10^{12}	0.3	−0.3	1.1	3.0

TABLE III–7. THE RELATIVE EFFECT OF HIGH DENSITY U–Mo FUEL ON FLUXES IN THE SILICON IRRADIATION FACILITIES

Facility	Reference flux Si480 (n/cm^2s)	Change relative to reference flux (%)			
		Mo606	Mo709	Mo706	Mo807
Si NTD	8.9×10^{12}	−1.5	−3.0	−2.5	−3.4

TABLE III–8. THE RELATIVE EFFECT OF HIGH DENSITY U–Mo FUEL ON FLUXES IN THE BEAM FACILITIES

Facility	Reference flux Si480 (n/cm^{2}s)	Change relative to reference flux (%)			
		Mo606	Mo709	Mo706	Mo807
Cold source	7.5×10^{13}	−2.9	−4.2	−2.5	−5.2
Hot source	1.5×10^{14}	−2.9	−6.2	−5.2	−7.5

III–6.5. Fluxes in beam facilities

The performance of the OPAL beam facilities is determined by the neutron flux in the region near the entrance of the beam tubes. For the cold beams, this is best defined by the cold neutron source and the flux of interest is that below 0.01 eV. In the case of the thermal beams, it is defined as a volume of heavy water near the entrance of the thermal beam tubes. The neutron flux of interest in this case is that below 0.1 eV. The relevant beam fluxes for the case of Si480 fuel and the relative effect due to high density fuels are indicated in Table III–8. The changes are the largest seen so far for fluxes, although they may still be considered minor. The general trends are again the same, with a gradual decrease in flux as the uranium loading increases, and there is also a decrease associated with the increased fraction of Mo in the fuel dispersant.

III–7. CONCLUSIONS

The impact of the use of high density U–Mo fuel in the OPAL research reactor was assessed. The present fuel management strategy and use of U_3Si_2 (4.8 gU/cm^3) dispersion fuel was compared to the use of U–Mo dispersion fuel with densities ranging from 6.0 to 8.0 gU/cm^3. The impact varied depending on the parameters and reactor characteristics of interest. A clear benefit was the increased cycle length and reduced fuel usage possible with increasing uranium density. This benefit, however, came at a penalty (but still within limits) of increased PPF and reduced shutdown margins. There were also impacts on the reactivity feedback coefficients and kinetic parameters that should be further evaluated by performing detailed transient calculations, although it is expected that such changes would make minor differences to the ultimate outcome of these analyses. Finally, the effect on the fluxes of the irradiation and beam facilities was evaluated. As expected, the neutron flux in most facilities decreased with increasing uranium density, although the effect was small for the irradiation facilities and somewhat larger for the beam facilities.

Fuel conversion of an existing reactor such as OPAL will be an involved and complex process and such a significant change will require more specific details on available high density fuel, the cost of such fuels and the priorities for the reactor into the future. Commitment to any such conversion could then be made with complete awareness of the benefit and penalties to be expected. Such choices offered by new fuels are always an opportunity to further optimize the performance and outcome of existing and new research reactors.

REFERENCES

[III–1] INTERNATIONAL ATOMIC ENERGY AGENCY, Global Threat Reduction Initiative International Partners' Conference: Summary of the Proceedings and Findings of the Conference, INIS-XA-738, IAEA, Vienna (2004).

[III–2] JAROUSSE, Ch., LAVASTRE, Y., GRASSE, M., "CERCA's experience in UMo fuel manufacturing", Proc. 25th Int. Mtg on Reduced Enrichment for Research and Test Reactors, Chicago, 2003, Argonne Natl Lab., IL (2003).

[III–3] VILLARINO, E., MTR-PC v3.0 System: User Manual, INVAP, Bariloche (2001).

[III–4] MOCHI, I., INVAP's nuclear calculation system, Sci. Technol. Nucl. Install. **2011** (2011) 215363.

[III–5] ASKEW, J.R., FAYERS, F.J., KEMSHELL, P.B., A general description of the lattice code WIMS, J. Br. Nucl. Energy Soc. **5** (1966) 564–585.

[III–6] FOWLER, T.B., VONDY, D.R., CUNNINGHAM, G.W., Nuclear Reactor Core Analysis Code: CITATION, Rep. ORNL-TM-2496 (Rev. 2), Oak Ridge Natl Lab., TN (1971).

Annex IV

RP-10 RESEARCH REACTOR CONVERSION ANALYSIS

E. Villarino, D. Ferraro

Nuclear Engineering Department, INVAP S.E., San Carlos de Bariloche, Argentina

IV–1. INTRODUCTION

The ongoing development of high and very high density fuels is mainly driven by the desire to enable the conversion of high performance research reactors from high enriched uranium (HEU) to low enriched uranium (LEU) fuels to meet the current non-proliferation and security goals. Furthermore, for research reactors of variable configuration that are already fuelled by LEU, conversion to higher density fuels provides a very promising opportunity to obtain compact cores that can reduce the number of fresh assemblies needed annually in the core and improve reactor performance in irradiation facilities. Hence, the potential economic benefits due to an operational cost reduction and increased income from irradiation device performance will be analysed.

The conversion of the Peruvian 10 MW research reactor RP-10 [IV–1] to high density fuel is analysed with regard to neutronic performance, safety features and potential economic benefits. Several aspects of this work are based on Ref. [IV–2], which was supported by the IAEA Technical Cooperation Programme project PER/4/023, Modernizing and Improving the Utilization of the RP-10 Reactor.

IV–2. RP-10: MAIN DESCRIPTION

The RP-10 [IV–1] is a materials testing reactor (MTR) pool type reactor, with a thermal power of 10 MW and downwards coolant flow. It became critical in 1989 and is used for research in reactor physics and radioisotope production.

The fuel assembly (FA) has aluminium cladding, with LEU (19.75% enrichment) in the form of U_3O_8 dispersed in an aluminium matrix. With regard to shutdown systems, the reactor has five fork type control rods, made of a Cd–Ag–In alloy. The core, surrounded by graphite and beryllium reflectors, rests at the bottom of a cylindrical tank 4 m in diameter and 11 m deep.

The reactor core is configurable, whereas the current U_3O_8 core design has 24 standard fuel assemblies (SFAs) and five fork type control fuel assemblies (CFAs), one in-core irradiation position and eight ex-core irradiation positions. In addition, five beam tubes and a thermal column are placed outside the reactor core, where one of the radial beams is used for neutron radiography.

IV–3. CONVERSION ANALYSIS: MAIN CONSTRAINTS

One of the main objectives of this analysis is to obtain a more compact core in order to both improve the irradiation fluxes and decrease the operational cost by reducing the number of FAs consumed. This FA consumption reduction is to be achieved through the utilization of FAs with higher uranium density and higher discharge burnup.

To do this, a well known U_3Si_2 dispersed in Al plate type FA is proposed. The use of such an FA allows the uranium load to be increased using a technology that has already been proven in several reactors around the world [IV–2].

Furthermore, to avoid an impact on other reactor systems, the design of the new FA is guided by several constraints from the current reactor design:

— There will be no changes in the nuclear safety design criteria.
— Higher fluxes are expected.
— The new FA will have the same external dimensions as the current FA.
— No changes in the absorber area of the CFA are expected.
— The same absorber rods are to be used, with the same control and safety functions.
— The coolant channels will be similar to minimize hydraulic perturbations in the core.

In addition, using the original RP-10 FA design as a basis, a change to U_3Si_2 fuel is proposed, with a minor modification in the fuel plate being considered to allow an increase of the power density. The main characteristics of the U_3Si_2 FA and its comparison with the current U_3O_8 FA are presented in Table IV–1. Additionally, the main core characteristics are also listed for the proposed compact core with the U_3Si_2 FA.

Schematic views of the core configurations of the original design and the proposed U_3Si_2 FA compact core are provided in Fig. IV–1.

IV–4. ANALYSIS OF REACTOR PERFORMANCE AND SAFETY FEATURES OF THE CONVERTED RP-10 CORE

All of the main performance and safety features of the proposed compact core [IV–2] have been analysed, ensuring that all design criteria are fulfilled and no impact on the outcome of the transients and

TABLE IV–1. RP-10 U_3O_8 AND U_3Si_2 FUEL ASSEMBLY PARAMETERS

Parameter		U_3O_8	U_3Si_2
Meat thickness (cm)		0.1	0.074
Internal cladding (cm)		0.038	0.037
External cladding (cm)		0.045	0.045
Internal channel (cm)		0.33	0.33
External channel (cm)		0.165	0.148
SFA U mass (g)		1418	2330
SFA U-235 mass (g)		280	460
Fuel plates per FA		16	17
Fuel plates per CFA		12	13
Core characteristics	SFA in the core	24	17
	CFA in the core	5	5
	Total number of fuel plates	444	388

Note: CFA — control fuel assembly; FA — fuel assembly; SFA — standard fuel assembly.

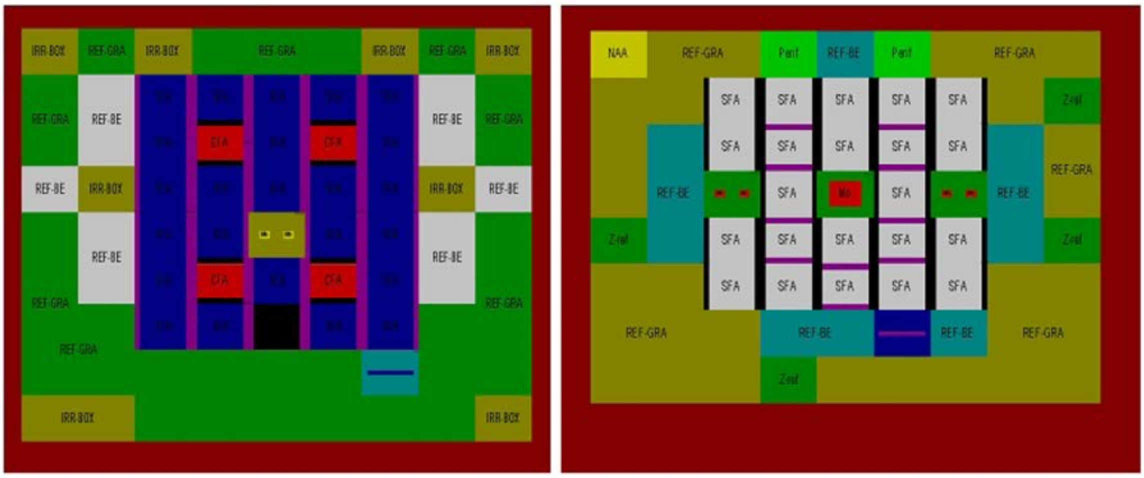


FIG. IV–1. U_3O_8 (left) and U_3Si_2 (right) RP-10 core configurations. Reproduced courtesy of INVAP [IV–2].

safety features of the reactor is observed (due to kinetic parameters and reactivity coefficients changes related to a more compact core).

IV–4.1. Calculation scheme and tools

The calculation tools used for the numerical analysis of the U_3Si_2–Al equilibrium core were the WIMS [IV–3] cell level code and the CITVAP [IV–4] core level code. Furthermore, several of INVAP's proprietary administrative tools were also used for cross-section handling, input geometry modelling and output processing [IV–4].

IV–4.2. Main core parameters

The proposed compact core fulfils all of the neutronic design criteria related to both nuclear safety and operational requirements. The main results are listed in Table IV–2 (see Ref. [IV–2] for detailed information).

IV–4.3. Reactivity coefficients

Reactivity coefficients were calculated for beginning of cycle (BOC) with Xe cases to determine the impact of the use of higher density fuels on the safety features, reactor dynamics and response to transients. The results are presented in Table IV–3; the reference case information has been obtained for similar reactors from the literature.

Analysing the results from Table IV–3, the observed changes for reactivity coefficients are minor and are expected to have little impact on the transients and safety features of the reactor.

IV–4.4. Kinetic parameters

In addition to reactivity coefficients, kinetic parameters were also calculated, and the results are presented in Table IV–4. A review of the results displayed in Table IV–4 indicates that the observed changes for kinetic parameters are minor and are not expected to affect the transients and safety features of the reactor.

TABLE IV–2. MAIN DESIGN CRITERIA FOR VERIFICATION OF PROPOSED CORE

Criteria	Limit	Current value
Number of FA	≥21	22
Max. PPF	2.80 (102.3 W/cm^2)	2.57 (93.9 W/cm^2)
Min. EOC reactivity (pcm)	≥1 000	1 599 (reserve for experiments)
CRW/max. reactivity excess	≥150%	294%
Min. control rod worth (pcm)	≥780	2 356
Max. control rod worth (pcm)	≤6 000	4 074
Shutdown margin (pcm)	≥3 000	14 754
Shutdown margin with single failure (pcm)	≥1 000	5 881
Regulating rod worth (pcm)	≤780	713
Av. in-core thermal flux (E4) (n/cm^{2}s)	$\geq 2.0 \times 10^{14}$	2.67×10^{14}
Av. in-core thermal flux (C4, G4) (n/cm^{2}s)	$\geq 1.5 \times 10^{14}$	1.72×10^{14}
Ex-core facility flux (level 1) (n/cm^{2}s)	$\sim 8.0 \times 10^{13}$	8.48×10^{13}
Ex-core facility flux (level 2) (n/cm^{2}s)	$\sim 4.0 \times 10^{13}$	5.69×10^{13}
NAA facility flux (n/cm^{2}s)	$\sim 2.0 \times 10^{13}$	3.16×10^{13}

Note: CRW — control rod worth; EOC — end of cycle; NAA — neutron activation analysis; PPF — power peaking factor.

TABLE IV–3. REACTIVITY COEFFICIENTS OF ANALYSED FUELS

Reactivity coefficients	U_3O_8 (estimated)	U_3Si_2 (calculated)
Fuel temperature (pcm/°C)	−1.8	−2.13
Coolant temperature (pcm/°C)	−17	−18.89
Void (pcm/%)	−195	−227

TABLE IV–4. KINETIC PARAMETERS OF ANALYSED FUELS

Kinetic parameter	U_3O_8 (estimated)	U_3Si_2 (proposal)
β_{eff} (pcm)	730	801
Λ (μs)	59	60.1

Note: β_{eff} — effective delayed neutron fraction; Λ — prompt neutron generation time.

TABLE IV–5. RP-10 U_3O_8 AND U_3Si_2 FUEL ASSEMBLIES NEUTRONIC CORE PARAMETERS

Parameter	U_3O_8	U_3Si_2
Cycle length (days)	14	21
SFA discharge burnup (%)	51.8	54.1
CFA discharge burnup (%)	46.9	66.3
Central in-core thermal flux (n/cm^{2}s)	1.16×10^{14}	2.67×10^{14} (+130%)
Lateral in-core thermal flux (n/cm^{2}s)	1.02×10^{14}	1.74×10^{14} (+71%)
Ex-core facility flux (level 1) (n/cm^{2}s)	8.56×10^{13}	9.51×10^{13} (+11%)
Ex-core facility flux (level 2) (n/cm^{2}s)	5.45×10^{13}	5.73×10^{13} (+5%)
Annual consumption of U-235 (300 FPDs) (g)	7125	6265
Annual consumption of FAs (SFA + CFA)	26.8 (21.4 + 5.4)	14.3 (11.4 + 2.9)
Annual saving U (%)	n.a.	12.1
Annual saving FAs (%)	n.a.	46.6

Note: Adapted from Ref. [IV–2]; n.a.: not applicable; CFA — control fuel assembly; FA — fuel assembly; FPD — full power day; SFA — standard fuel assembly.

IV–4.5. Summary of conversion effect in core parameters

The overall impact on the main operational performance parameters is presented in Table IV–5 (adapted from Ref. [IV–2]), where the FA consumption and relevant irradiation facility fluxes are compared with those of the original RP-10 design. The results in Table IV–5 show that a 12% savings in uranium consumption is obtained. In addition, more significant savings are obtained for the number of FAs used. Furthermore, as the proposed U_3Si_2 core is more compact, the improvement to flux performance is also significant; the in-core fluxes are 130 and 70% higher in the central and lateral zones, respectively.

IV–5. SIMPLIFIED ECONOMIC ANAYLSIS OF THE CONVERSION

The results described in the previous sections show that a reduction in FA consumption can be achieved by using higher density fuel, obtaining a compact core that fulfils all of the neutronic design criteria related to nuclear safety and operational requirements, along with an increase in the neutron fluxes. Nevertheless, the potential economic benefits of such modification need to be considered.

With regard to FA costs, the economic benefits will be strongly dependent on the fabrication cost of the new FA; simplified estimates can be obtained for the cost of each FA using the following options:

(a) The cost of the FA is 50% manufacturing cost and 50% uranium cost for the uranium loading in the current FA design.
(b) The cost of the FA is 100% manufacturing cost (neglecting uranium loading cost).
(c) The cost of the FA is 100% uranium cost (neglecting manufacturing cost).

Combining this case analysis with the results from Table IV–5, a simplified cost analysis is performed in order to identify potential economic benefits. The results are presented in Table IV–6.

Regarding results in Table IV–6, a potential economic benefit can be identified for both uranium consumption and reduced use of FAs. Potential conservative savings of around 30% can be expected, whereas the final cost reduction will depend on the final relative manufacturing and uranium costs. Additionally, the potential economic impact is expected to be higher due to the reduction of costs associated with the storage, manoeuvring and back end of FAs.

Furthermore, the final economic impact of the improvement will also include the higher performance achieved in irradiation facilities.

IV–6. CONCLUSIONS

The analysis for the RP-10 reactor showed that the proposed conversion to higher density fuels (from U_3O_8 to U_3Si_2) for LEU fuelled research reactors of variable configuration provides a very promising opportunity to obtain a general improvement to reactor performance and operating costs.

The performed analysis showed that a more compact core with higher density fuels can be obtained, fulfilling all of the neutronic design criteria concerning both nuclear safety and operational requirements, with little impact on the safety features of the reactor related to reactivity coefficients and kinetic parameters.

Finally, the potential economic benefits of such a conversion are very promising because the uranium consumption is lowered, the number of FAs required per year is reduced and the irradiation facilities are improved.

TABLE IV–6. SIMPLIFIED COST COMPARISON FOR DIFFERENT FAs

Parameter	U_3O_8	U_3Si_2 [a]		
		(a)	(b)	(c)
FA cost[b]	1.0	1.32	1.0	1.64
Annual consumption cost[b]	26.8	18.90	14.3	23.49
Annual saving (%)	0.0	29.5	46.6	12.3

[a] For simplified cost estimations described above as options (a), (b) and (c).
[b] Relative to the original U_3O_8 fuel assembly cost.

REFERENCES

[IV–1] MONTOYA, M., CARPIO, R., AVILA, J., El Núcleo del Reactor RP-10, Informe Nuclear **5** (1989) 15–23.

[IV–2] VILLARINO, E., GALLARDO PADILLA, A., "Core performance improvement using U3Si2-Al fuel in the RP-10 modernization", Proc. Int. Mtg on Reduced Enrichment for Research and Test Reactors, Santiago, 2011, Argonne Natl Lab., IL (2011).

[IV–3] ASKEW, J.R., FAYERS, F.J., KEMSHELL, P.B., A general description of the lattice code WIMS, J. Br. Nucl. Energy Soc. **5** (1966) 564–585.

[IV–4] MOCHI, I., INVAP's nuclear calculation system, Sci. Technol. Nucl. Install. **2011** (2011) 215363.

Annex V

CONVERSION OF GENERIC REACTORS FROM U_3Si_2 TO URANIUM–MOLYBDENUM (U–Mo)

E. Villarino, D. Ferraro, C. Mazufri

Nuclear Engineering Department, INVAP S.E., San Carlos de Bariloche, Argentina

P. Bouza

Balseiro Institute, National University of Cuyo, Mendoza, Argentina

V–1. INTRODUCTION

The ongoing development of high and very high density fuels is mainly driven by the desire to enable the conversion of high performance research reactors from highly enriched uranium (HEU) to low enriched uranium (LEU) fuels in accordance with global non-proliferation and security goals. Furthermore, for research reactors that are already fuelled with LEU, conversion to high and very high density fuels is considered to be a very promising opportunity to reduce the number of fresh assemblies needed to be procured and later handled in spent fuel management; however, this conversion must be accomplished without negative reactor performance impact for fixed core reactors. Hence, some analysis is required to fully understand the potential economic benefits coming from operational cost reduction, as well as the impact on income from irradiation device performance.

Unfortunately, the conversion to high and very high density fuels is not straightforward for an already operational research reactor because of design and licensing constraints. Furthermore, beyond the performance improvements that may be observed, the economic benefits will be strongly dependent on the fabrication cost of the new fuel assemblies (FAs). Regarding these aspects, it is difficult for potential users to understand clearly what economic or performance benefits can be expected from this conversion.

The main problem for the conversion analysis for a given high performance research reactor comes from the design and licensing constraints. Typically, a direct swap of the original fuel meat in the assembly plates with a high density fuel is not a viable solution, and a FA redesign is necessary to fulfil the given design criteria.

Therefore, an integrated approach is proposed based on Ref. [V–1], selecting the fuel from a thermohydraulic point of view with the evaluation of the main neutronic parameters. For this analysis a given heavy water reflected reactor of fixed core size is considered, using U_3Si_2 dispersion fuels as a reference and high density molybdenum fuels, namely U–10Mo/Al (dispersion fuel) and U–10Mo (monolithic fuel) as fuel alternatives.

Finally, to identify potential economic benefits, a simplified analysis is performed.

V–2. MAIN ASPECTS OF DESIGN ANALYSIS

The main idea of a high performance research reactor is to maximize the thermal and fast flux to be used. Both fluxes are proportional to the power (density) of the reactor; thus, for a compact core the following approximations can be made:

— Thermal flux is proportional to power divided by the mass of uranium of the core.

— Fast flux is proportional to power divided by the volume of the core.

The in-core facilities will mainly be defined by the first aspect, whereas fast flux will be the driver flux for the thermal ex-core facilities in a heavy water reflected core.

To have an integrated approach, the current analysis includes the following steps:

— Preliminary neutronic analysis at cell level to identify the main aspects of conversion;
— Thermohydraulic analysis in which the meat, cladding and coolant channel thicknesses are defined considering a flat plate MTR type FA;
— Core level analysis for main safety and operational features;
— Simplified economic analysis to identify potential benefits.

Additionally, in order to simplify the analysis, the Open Pool Australian Lightwater (OPAL) reactor geometry [V–2] was used as a reference.

V–2.1. Design criteria

The developed analysis is guided by the following design criteria [V–1]:

(a) General criteria:
    (i) Upwards coolant flow (light water);
    (ii) Inlet temperature at 37°C;
    (iii) 10 m of water column over the core;
    (iv) Heavy water reflector.
(b) Neutronic:
    (i) Negative power feedback coefficient. In order to satisfy this point, all feedback coefficients will be required to be negative.
    (ii) Shutdown margin with single failure (SDM-1) >1000 pcm.
    (iii) Minimum end of cycle (EOC) reactivity excess >1000 pcm.
    (iv) Minimum discharge burnup (~50% ^{235}U).
    (v) Cycle length (>26 full power days).
(c) Fuel plate mechanical stability:
    (i) Coolant velocity <2/3 critical velocity.
(d) Steady state core thermohydraulic parameters:
    (i) Departure from nucleate boiling ratio >2.0;
    (ii) Redistribution ratio >2.0;
    (iii) Onset of nucleate boiling ratio >1.3.
(e) Combined neutronic and thermohydraulic criteria:
    (i) Power peaking factor (PPF) < 3.

V–2.2. Fuel materials

To analyse the high density conversion, three different fuels are considered:

(1) Dispersion fuel U_3Si_2 at 4.8 gU/cm^3, used as a reference;
(2) U–Mo dispersion fuel with 10wt% Mo (U–10Mo/Al) at 7.0 gU/cm^3;
(3) U–Mo monolithic fuel with 10wt% Mo (U–10Mo) at 16.4 gU/cm^3.

Furthermore, in all cases the structural material is Al 6061 for frames and plates. Note that, despite the well known failures for high burnup for U–Mo fuels with Al clad at high power densities [V–3], the main results of this analysis are fully valid as a reference.

V–3. RESULTS AND ANALYSIS

V–3.1. Neutronic analysis at fuel assembly level

In order to identify the main neutronic aspects of a generic conversion to high density fuel, a preliminary cell level calculation was performed using INVAP's CONDOR cell code [V–4]. A simplified two dimensional cell model was developed (shown in Fig. V–1), where the fuel meat was changed for each case analysed. The dimensions were set using OPAL fuel as a reference for U_3Si_2 fuel material, as listed in Table V–1.

To determine how increasing the density of uranium changes the moderation ratio of the fuel, the comparison was performed considering five different cases:

(a) Maintaining the geometry of the OPAL reference fuel:
  - (i) U_3Si_2 reference fuel, labelled U_3Si_2;
  - (ii) Dispersion U–Mo fuel, labelled UMO_D_G;
  - (iii) Monolithic U–Mo fuel, labelled UMO_M_G.

(b) Maintaining the same moderation ratio as the U_3Si_2 fuel:
  - (i) Dispersion U–Mo fuel, labelled UMO_D_M;
  - (ii) Monolithic U–Mo fuel, labelled UMO_M_M.

Figure V–2 shows the obtained infinite multiplication factor (k_∞)[1] for the five cases listed above, while Fig. V–3 shows the effective multiplication factor (k_{eff}) considering the geometrical buckling of the OPAL reactor with 8 cm and 16 cm of axial and radial reflector savings, respectively.

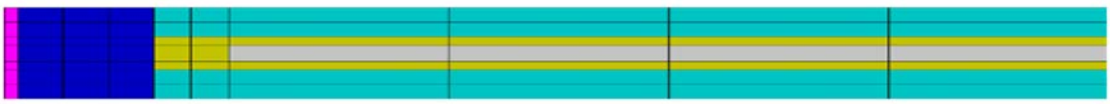


FIG. V–1. Simplified cell model. Reproduced courtesy of INVAP.

TABLE V–1. REFERENCE GEOMETRIC FUEL ASSEMBLY DATA

Geometric characteristic	Value (cm)
Meat thickness	0.061
Cladding thickness	0.037
Coolant channel	0.245

[1] The 'multiplication factor', k, is the ratio of the total number of neutrons produced during a time interval to the total number of neutrons lost by absorption and leakage during the same interval. The 'infinite multiplication factor', k_∞, is the multiplication factor evaluated for an infinite medium or for an infinite repeating lattice. The 'effective multiplication factor', k_{eff}, is the multiplication factor evaluated for a finite medium.

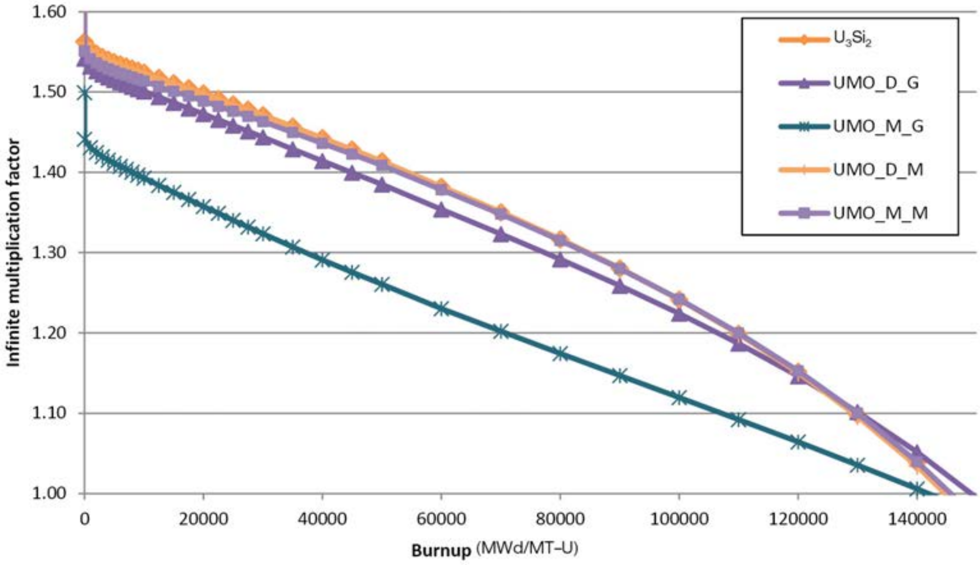


FIG. V–2. Infinite multiplication factor. Reproduced courtesy of INVAP.

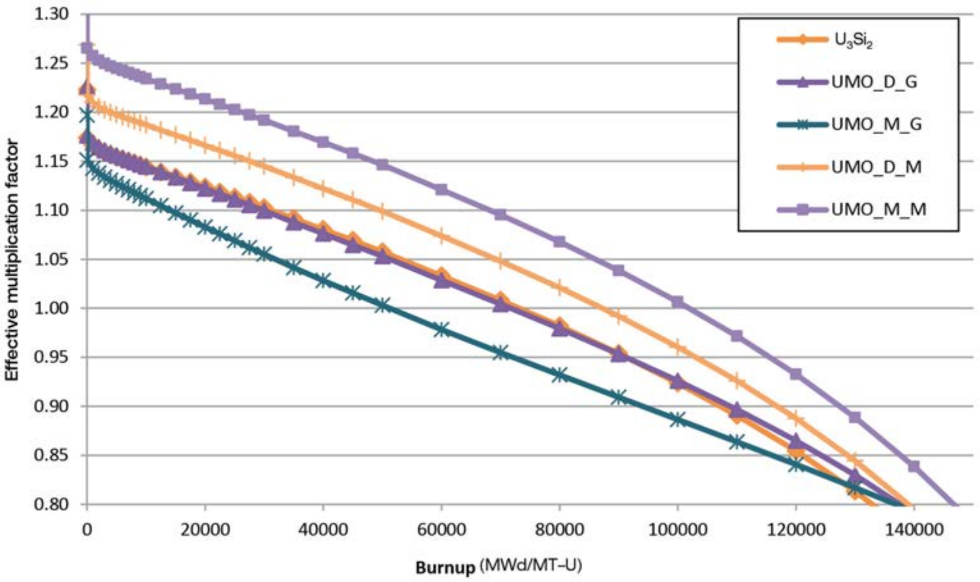


FIG. V–3. Effective multiplication factor. Reproduced courtesy of INVAP.

Figures V–2 and V–3 show that a direct swap of U_3Si_2 for higher density fuel (represented by UMO_D_G and UMO_M_G) does not offer an improvement because the moderation ratio is significantly altered. Furthermore, as expected, k_∞ shows no significant difference when the moderator ratio is preserved (as by UMO_D_M and UMO_M_M), but there is an important improvement in k_{eff} (evaluated using a core buckling).

In addition, because the fuel density changes, it is crucial to analyse the impact on the FA feedback coefficients, namely, the fuel temperature (T_f), coolant temperature (T_c) and void coefficients. An evaluation at cell level was performed for the cases listed above, for both infinite and effective conditions, and the results are shown in Table V–2 for several FA burnups.

Note that all the effective feedback coefficients shown in Table V–2 are negative for the high density fuels. The results from this cell level analysis indicate that the geometry (i.e. the coolant channel thickness) should be changed for a higher density fuel, whereas it is expected that reactivity feedback coefficients remain negative for the modified geometry.

V–3.2. Thermohydraulic analysis

To ensure both the thermohydraulic and mechanical design criteria (i.e. limits in coolant velocity), a simplified analysis was performed to define the meat, cladding and coolant channel thicknesses.

A parametric analysis was performed [V–1] to determine the maximum allowable coolant velocity while avoiding mechanical collapse. The results are shown in Fig. V–4, where the increment of the plate thickness (i.e. thickness of the meat and its two claddings) shows an increment of the coolant velocity (due to a higher margin to flow induced instability), in turn allowing an increment of the power to be removed.

TABLE V–2. INFINITE AND EFFECTIVE FEEDBACK COEFFICIENTS CALCULATED AT CELL LEVEL

Burnup (MW·d/MTU):		0	40 000	80 000	120 000	0	40 000	80 000	120 000
Fuel	Type of coefficient	Coefficient values in infinite condition				Coefficient values in effective condition			
U_3Si_2	T_f	−2.1	−2.2	−2.5	−3.0	−2.7	−2.9	−3.3	−3.9
	T_c	−1.8	−3.3	−4.2	−3.8	−24.2	−28.6	−33.0	−38.1
	Void	−9.1	−16.3	−20.7	−13	−416.5	−466.3	−524.8	−600.6
UMO_D_G	T_f	−2.2	−2.4	−2.8	−3.3	−2.8	−3.1	−3.6	−4.2
	T_c	−2.8	−4.7	−6.2	−6.7	−23.4	−28.1	−32.8	−37.9
	Void	−34.0	−50.1	−65.8	−74.8	−417	−478.6	−548.7	−634.1
UMO_M_G	T_f	−2.5	−2.9	−3.5	−4.1	−3.1	−3.6	−4.3	−5.0
	T_c	−3.8	−5.9	−7.9	−9.7	−19.8	−24.6	−29.4	−34.5
	Void	−62.5	−89.6	−121.5	−157.7	−368.7	−443.6	−527.6	−627.3
UMO_D_M	T_f	−1.8	−2.0	−2.3	−2.7	−2.3	−2.5	−2.8	−3.4
	T_c	−1.8	−3.3	−4.2	−3.8	−21.1	−25.0	−28.9	−33.2
	Void	−12.1	−19.1	−23.1	−14.4	−365.6	−409.1	−459.1	−522.4
UMO_M_M	T_f	−1.3	−1.4	−1.6	−1.8	−1.5	−1.7	−1.9	−2.2
	T_c	−0.4	−1.6	−2.2	−1.4	−16.0	−19.1	−22.0	−24.8
	Void	−0.6	−4.9	−5.7	−6.6	−290.9	−323.9	−360.5	−404.3

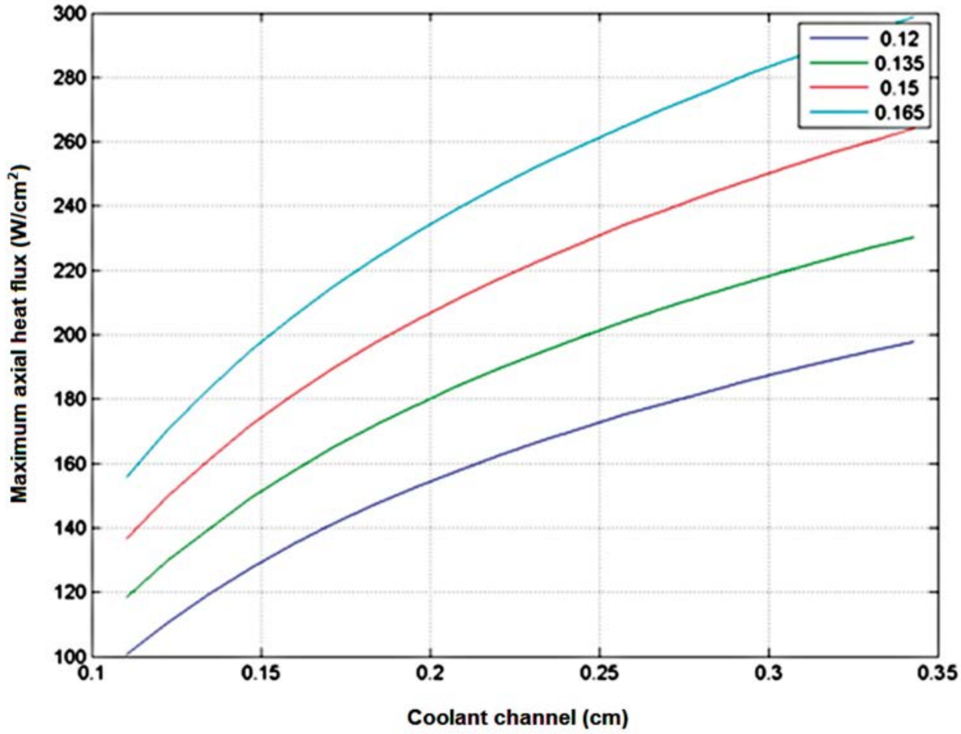


FIG. V–4. Maximum heat flux dependence with coolant channel for several plate thicknesses. Reproduced courtesy of INVAP.

Unfortunately, even though the increment in plate thickness allows higher maximum heat fluxes (due to a higher margin to flow induced instability), for a coolant channel thickness above ~0.2 cm, it reduces the power density of the reactor, thus reducing the flux performance. The combined effects lead to a maximum that depends on the combination of both channel and plate thickness. Accordingly, Fig. V–5 shows the maximum power density that can be removed for different plate thicknesses as a function of coolant channel thickness.

To finish defining the FA geometry, analysis of the redistribution margin also has to be considered. Figure V–6 shows the results for several plate thicknesses as a function of coolant channel thickness. It shows that a coolant channel lower than ~0.2 cm does not fulfil the design criteria for a redistribution ratio greater than 2.0.

In summary, the following criteria were used to define the geometry of the fuel assemblies:

— The OPAL meat cladding thickness was used as a reference for the U_3Si_2 fuel.
— The coolant channel was slightly reduced in the U_3Si_2 fuel with respect to the OPAL FA.
— To have the same fuel assembly geometry for all the fuel material it was decided to preserve the total thickness, 0.37 cm, of the plate plus the water channel. This criterion gives the same power density for all the fuel materials.
— The reduction of the coolant channel compensated for increased plate thickness, allowing a higher coolant velocity.
— The remaining geometry data for the FA were the same for the entire analysis (i.e. active height, side plate thickness).

According to the criteria listed above, and considering the results presented in Figs V–5 and V–6, a selection for the meat thickness, cladding thickness and coolant channel was made, which is listed in Table V–3.

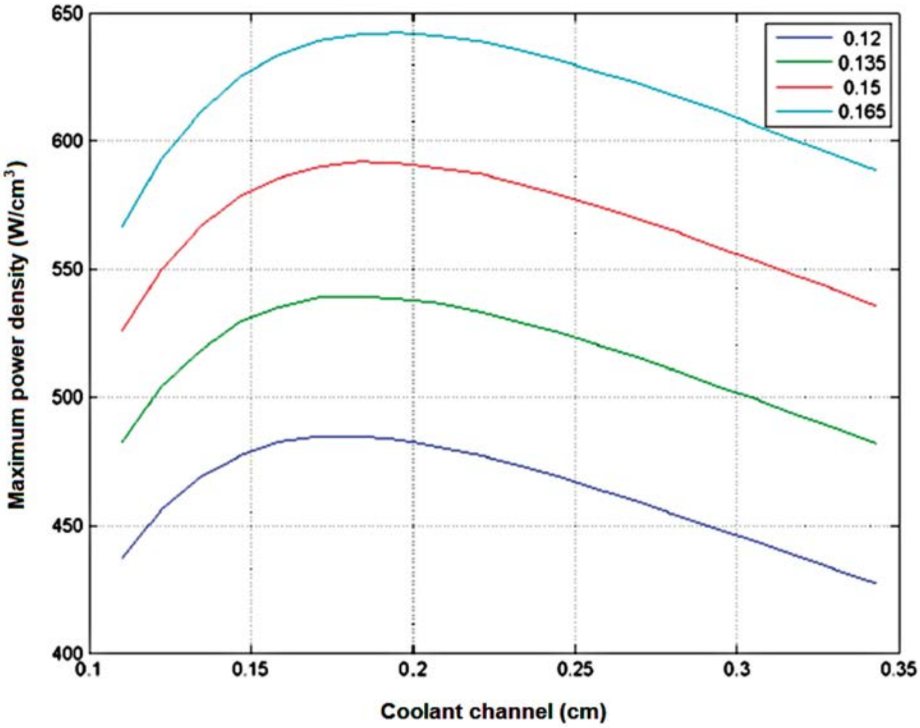


FIG. V–5. Maximum power density dependence with coolant channel for several plate thicknesses. Reproduced courtesy of INVAP.

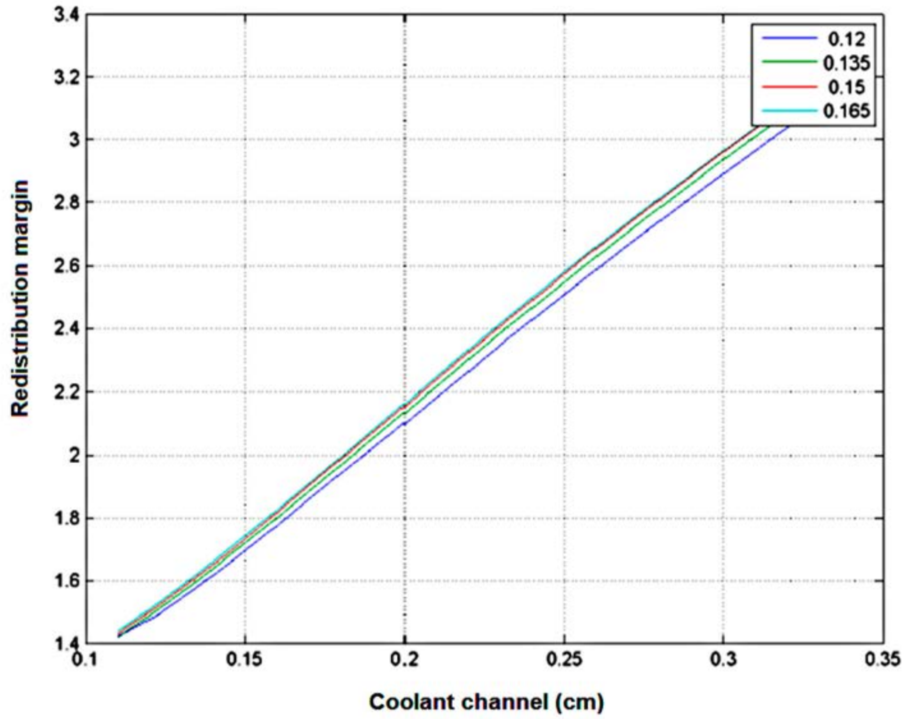


FIG. V–6. Redistribution margin dependence with coolant channel for several plate thicknesses. Reproduced courtesy of INVAP.

As indicated in Table V–3, the cladding of the U–Mo monolithic fuel can be reduced using a different design in the FA (e.g. a slightly curved plate or additional combs in the FA), but in this conceptual analysis a similar FA design was preferred.

V–3.3. Core level calculations

In order to study the final impact on core performance, core level analyses with INVAP's proprietary calculation line [V–4] were performed using FAs with the geometric characteristics listed in Table V–3. Figure V–7 shows the core model used for the core analysis; it is similar to OPAL [V–5].

TABLE V–3. SELECTED FUEL ASSEMBLY GEOMETRY

Zone	Thickness (cm)		
	U_3Si_2	Dispersion U–Mo	Monolithic U–Mo
Meat	0.061	0.06	0.03
Cladding	0.037	0.045	0.06
Channel	0.235	0.220	0.220
Plate	0.135	0.150	0.150
Total	0.370	0.370	0.370

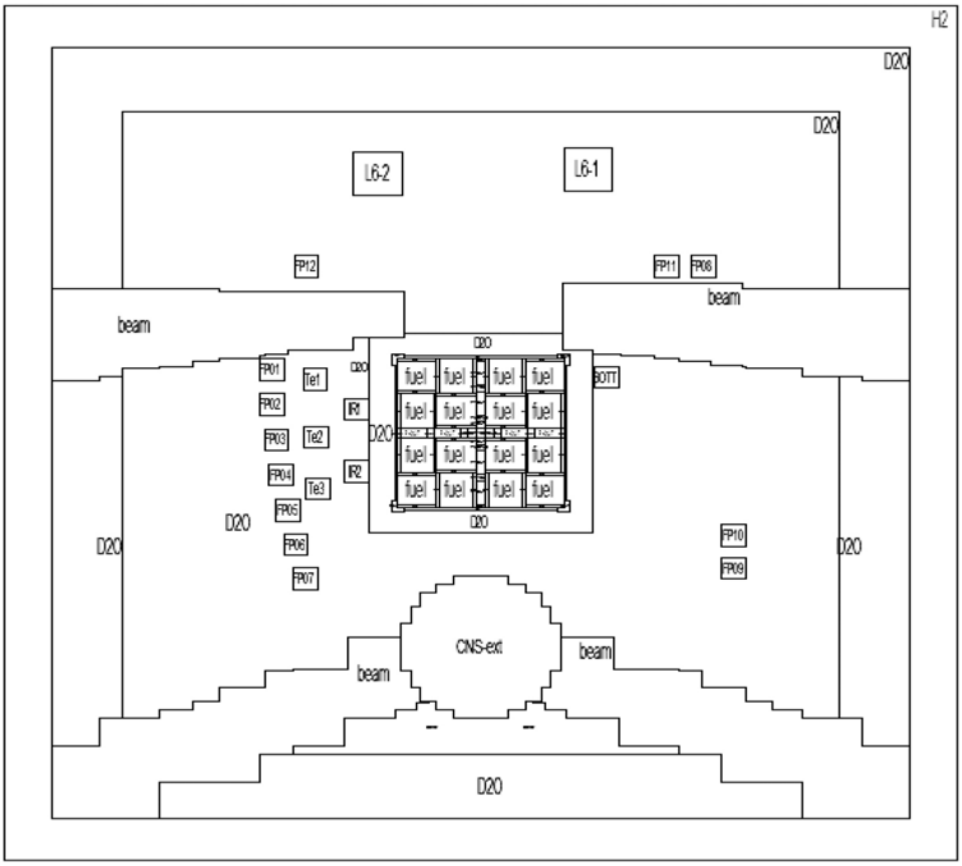


FIG. V–7. Core model. Reproduced courtesy of INVAP.

TABLE V–4. MAIN PERFORMANCE AND SAFETY PARAMETERS FOR ANALYSED FUELS

Parameter	U_3Si_2	Dispersion U–Mo	Monolithic U–Mo
Discharge burnup (MW·d/MTU)	87 600	94 600	97 000
PPF	−2.43	−2.51	−2.84
Full power days	33	35	41
BOC reactivity (pcm)	5 060	3 670	4 050
EOC reactivity (pcm)	1 270	1 000	1 290
SDM-1 (pcm)	14 000	13 800	13 300
Very high thermal flux facilities (n/cm^{2}s)	1.50×10^{14}	1.60×10^{14}	1.50×10^{14}
High thermal flux facilities (n/cm^{2}s)	1.10×10^{14}	1.10×10^{14}	1.00×10^{14}
Cold neutron source thermal flux facilities (n/cm^{2}s)	1.60×10^{14}	1.60×10^{14}	1.60×10^{14}
U-235 mass per FA (g)	485.5	696.4	815.8
Number of FAs per cycle	3	2	2
Number of FAs per year	31.3	19.7	17.0
U-235 mass per year (kg)	15.19	13.74	13.85

V–3.3.1. Main core performance

Main core performance was analysed, during which, due to the difference in the uranium loading, different fuel management strategies were applied for each FA. A similar EOC reactivity was sought for each FA. The results for the proposed FA are presented in Table V–4, in which the average FA consumption per year is proved to be greatly reduced for both U–Mo alternatives without a relevant impact on reactor performance.

V–3.3.2. Reactivity coefficients

Regarding the impact of high density fuels on the safety features, reactor dynamics and response to transients, reactivity coefficients were calculated for cases of beginning of cycle (BOC) with Xe. The results are presented in Table V–5, where it can be seen that the U_3Si_2 results are slightly different from OPAL's due to the fact that the FA used is not exactly the same.

Analysing the results from Table V–5, the observed changes for reactivity coefficients are minor and are expected to have little impact on the transients and safety features of the reactor.

V–3.3.3. Kinetic parameters

In addition to reactivity coefficients, kinetic parameters were also calculated; the results are presented in Table V–6.

TABLE V–5. REACTIVITY COEFFICIENTS OF ANALYSED FUELS

Reactivity coefficient	U_3Si_2	Dispersion U–Mo	Monolithic U–Mo
Fuel temperature (pcm/°C)	−2.3	−2.6	−2.6
Coolant temperature (pcm/°C)	−18.1	−17.6	−16.6
Void (pcm/%)	−204.8	−229.5	−221.8

TABLE V–6. KINETIC PARAMETERS OF ANALYSED FUELS

Kinetic parameter	U_3Si_2	Dispersion U–Mo	Monolithic U–Mo
β_{eff} (pcm)	708	699	690
Λ (μs)	197	192	196

Note: β_{eff} — effective delayed neutron fraction; Λ — prompt neutron generation time.

Similar to the reactivity coefficients, when the results from Table V–6 are analysed, the observed changes for kinetic parameters are minor and are expected to have little impact on the transients and safety features of the reactor.

V–4. SIMPLIFIED ECONOMIC ANALYSIS

The results from the previous sections show that a reduction of FA consumption due to high density fuel conversion can be achieved with little or no impact on the performance and safety features of the reactor. Nevertheless, the potential economic benefits of such modifications need to be studied.

Although the proper economic benefits will be strongly dependent on the fabrication cost of the new FA, simplified estimates can be obtained for the relative cost of each FA using the following case options:

(a) For the U_3Si_2 reference case:
   (i) The cost of the FA is assumed to be 50% manufacturing cost and 50% uranium cost based on 4.8 gU/cm^3 density for the U_3Si_2 fuel.

(a) For the U–Mo cases:
   (i) The cost of the FA is assumed to be 50% manufacturing cost and 50% uranium cost based on 7.0 gU/cm^3 density in the current U–Mo dispersion FA design, and 16.4 gU/cm^3 density in the current U–Mo monolithic FA design (i.e. per Table V–3);
   (ii) The cost of the FA is 100% manufacturing cost (neglecting uranium loading cost);
   (iii) The cost of the FA is 100% uranium cost (neglecting manufacturing cost);
   (iv) The cost of the FA is 200% of the manufacturing cost of the U_3Si_2 FA with the same uranium cost as case (a).

The expected relative economic benefits between the reference case and the proposed U–Mo cases are shown in Table V–7. A potential economic benefit is expected for cases (i), (ii) and (iii). On the other hand, if the manufacturing costs for U–Mo are too high, as in case (iv), the reduction in FA consumption does not compensate for the increasing costs, and thus, the potential economic benefits can disappear.

TABLE V–7. ESTIMATED POTENTIAL RELATIVE ECONOMIC BENEFITS FROM FUEL ASSEMBLY CONSUMPTION REDUCTION

Parameter	Fuel	Case option			
		(i)	(ii)	(iii)	(iv)
FA cost[a]	U_3Si_2	1.0	1.0	1.0	1.0
	Dispersion U–Mo	1.22	1.0	1.43	1.72
	Monolithic U–Mo	1.34	1.0	1.68	1.84
Annual FA consumption	U_3Si_2	31.3	31.3	31.3	31.3
	Dispersion U–Mo	19.7	19.7	19.7	19.7
	Monolithic U–Mo	17	17	17	17
Annual saving[a] (%)	U_3Si_2	0.0	0.0	0.0	0.0
	Dispersion U–Mo	23.4	37.1	9.7	−7.5
	Monolithic U–Mo	27.2	45.7	8.7	0.1

[a] Relative to the original U_3Si_2 fuel assembly cost.

This simplified analysis does not consider other potential economic benefits, such as reduction of FA manoeuvring, storage and back end costs.

V–5. CONCLUSIONS

This conceptual analysis shows the possibility of converting (or designing) a high performance research reactor core using U–Mo fuel, using an integrated neutronic and thermohydraulic approach, and estimating potential relative economic benefits.

It has been noted that the conversion from U_3Si_2 fuel to U–Mo (dispersion or monolithic) fuel will need a detailed analysis for the specific reactor. The use of the same geometry will change the neutronic characteristics of the core, and even though the same geometry can conceptually be used in U–Mo dispersion fuel, it is not possible in the monolithic case. Thus, for a given reactor to be converted, it is important to perform a neutronic and thermohydraulic analysis, where an FA redesign may arise. In particular, to improve the performance of U–Mo monolithic fuel, an additional redesign of the FA geometry is needed; for example, using thinner curved plates appears to be an attractive option.

Finally, it has been shown that the conversion from U_3Si_2 to U–Mo fuel can be achieved with little or no impact on the performance and safety features of the reactor, and with an important reduction in the number of FAs required for operation. The potential benefit of such a conversion will depend on both fuel manufacturing costs and other associated costs, such as those from storage, manoeuvring and back end, which would need to be included in a detailed consideration of the conversion decision.

REFERENCES

[V–1] BOUZA, P., Conceptual Research Reactor Design Using High Density Fuels, BA Thesis, Balseiro Institute, National University of Cuyo (2012).

[V–2] AUSTRALIAN NUCLEAR SCIENCE AND TECHNOLOGY ORGANISATION, Opal Nuclear Reactor (2018), http://www.ansto.gov.au/AboutANSTO/OPAL/index.htm

[V–3] MEYER, M.K., et al., "U.S. progress in the development of very high density low enrichment research reactor fuels", Trans. Topical Mtg on Research Reactor Fuel Management, Prague, 2012, ENS, Brussels (2012) 10–13.

[V–4] MOCHI, I., INVAP's nuclear calculation system, Sci. Technol. Nucl. Install. **2011** (2011) 215363.

[V–5] VILLARINO, E., HERGENREDER, D., "Neutronic performance of the U–Mo fuel type in the replacement research reactor", Proc. 24th Int Mtg on Reduced Enrichment for Research and Test Reactors, San Carlos de Bariloche, 2002, Argonne Natl Lab., IL (2002).

Annex VI

COMPARATIVE STUDY ON FUEL DENSITY FOR THE AHR

C.G. Seo

Reactor Core Design Division, Korea Atomic Energy Research Institute, Deajeon, Republic of Korea

VI–1. INTRODUCTION

The Korea Atomic Energy Research Institute (KAERI) developed an Advanced HANARO Reactor (AHR) based on the High-Flux Advanced Neutron Application Reactor (HANARO). The AHR is a 20 MW multipurpose reactor that uses HANARO rod type fuel assemblies (FAs), comprising U_3Si_2 dispersion fuel with a uranium density of 4.0 gU/cm^3 as a reference fuel. Higher uranium density fuel is required for better fuel economy and high density uranium–molybdenum (U–Mo) fuel with a uranium density of up to 6.0 gU/cm^3 has been considered. This report compares the neutronic characteristics of 4.0 gU/cm^3 U_3Si_2 fuel and up to 6.0 gU/cm^3 U–Mo fuel in the AHR core. The compared parameters are linear heat generation rate (LHGR), control rod worth, cycle length and neutron fluxes.

VI–2. NUCLEAR ANALYSIS

VI–2.1. Core description

AHR is a conceptual multipurpose 20 MW reactor [VI–1]. The following are the basic design principles:

— Multipurpose medium power research reactor;
— Adaptation of the HANARO concepts;
— High neutron flux;
— Superior safety and economic features;
— Improvement of operability and maintainability;
— Sufficient space and expandability of the facility.

The changed core design will need to satisfy the basic design principles while achieving high neutron flux, which is very important in a research reactor. Although the reactor physics design of AHR using the current HANARO fuel could provide higher neutron flux, the uranium density is too low to obtain high discharge burnup. The AHR requires a higher uranium density fuel for improved performance without economic loss. Two types of FAs, hexagonal and circular, are used in the AHR core, as shown in Fig. VI–1.

The core configuration needs to be optimized according to its purpose. As AHR is a multipurpose research reactor, the flux level should be high at both the core and reflector regions. A multipurpose research reactor in general provides at least one irradiation position at the core region, in which the fast neutron flux can be high. The reactor core should be as compact as possible to obtain high neutron flux at the reflector region. The sites of the control absorber rods (CARs) are restricted by the cooling method and the position of the control driving units. An upward forced convection cooling system was applied to AHR. The CARs are located at the periphery of the core. The number of fuel channels should be

FIG. VI–1. Hexagonal and circular FAs. Reproduced courtesy of KAERI.

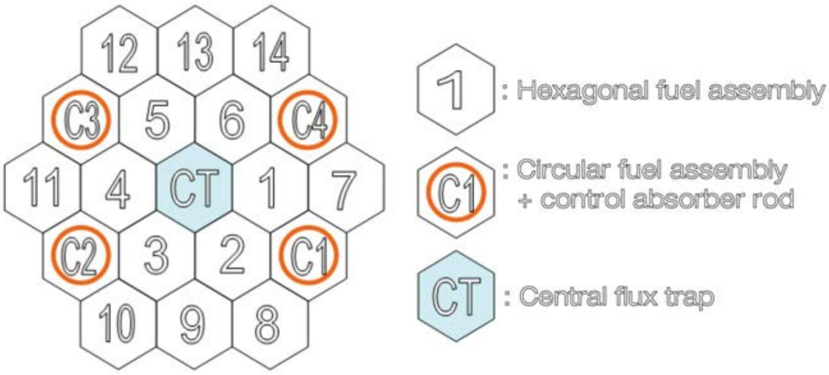


FIG. VI–2. The core layout of AHR. Reproduced courtesy of KAERI [VI–2].

optimized for the reactor power. Various options for the reactor core have been studied and the core model in Fig. VI–2 [VI–2] was selected as the reference.

Fourteen channels contain the hexagonal FAs, four channels contain the circular FAs and one channel is devoted to the central flux trap (CT). The core reactivity is controlled by four CARs made of hafnium, which are used as the first shutdown system. The secondary independent shutdown system will be a heavy water drainage system. The number of vertical irradiation positions and horizontal beam tubes at the reflector region will be determined later. Only the reactivity effect at the irradiation positions is considered at this design stage. The nominal fission power of AHR is 20 MW and the other design characteristics are similar to those of HANARO.

VI–2.2. Comparison of the U_3Si_2 and U–Mo fuel

Most of the parameters were calculated using the MCNP [VI–3] code. HELIOS [VI–4] was used for the burnup calculations, in which the two dimensional full core model was used.

The maximum unperturbed thermal (<0.625 eV) neutron flux levels at the core and reflector regions are estimated to be approximately 5.03×10^{14} and 4.36×10^{14} n · cm^{-2} · s^{-1}, respectively. The fast (>1.0 MeV) neutron flux in the CT is estimated to be about 1.67×10^{14} n · cm^{-2} · s^{-1}. The partially inserted CARs cause the maximum LHGR to become larger. The peak powers were evaluated for all CARs insertion depths.

At the beginning of cycle (BOC), the maximum LHGR was estimated to be 118.6 kW/m and the total peaking factor was 2.54. The flux levels and the peaking factor were estimated for a fresh core. To compensate for the reactivity decrease present in the equilibrium core (due to xenon, temperature, power, xenon override, fuel burnup and irradiation experiment configurations), a core excess reactivity of up to 102 mK will be included in the analysis at the BOC. This excess reactivity is obtained by loading two fresh hexagonal FAs or one hexagonal FA plus two circular FAs. The core excess reactivity at the end of cycle (EOC) will be above 30 mK, in which each 15 mK is reserved for a xenon override and a typical target loading. The total reactivity worth of CARs is about 212 mK and it meets the shutdown margin at the BOC of the equilibrium core. It is estimated that the reactor can operate at 20 MW without refuelling for 31 days. The moving span of the CARs in AHR is shorter than that of HANARO. The average burnup values in the equilibrium core at BOC and EOC were about 28 and 35%, respectively. The average discharge burnup of the FAs is about 58%.

Physics analyses for a comparison of the U_3Si_2 and U–Mo fuel were limited to the fuels considered in HANARO. The analysed fuels will be 4.0 gU/cm^3 U_3Si_2, 4.5 gU/cm^3 U–Mo with 7wt% of Mo (U–7Mo) and 6.0 gU/cm^3 U–7Mo. The compared physics parameters are the distributions of the LHGR, CAR worth, cycle length and neutron flux [VI–2].

A total of 576 fuel rods of a standard type were used in the AHR core, resulting in a core uranium loading of 51.1 kgU. The number of fuel rods is equivalent to the number of fuel rods in HANARO after a core conversion from the thermohydraulic view point. HANARO uses two types of fuel rods to obtain a more uniform power distribution in the FA. If AHR has a large power peak, a reduced rod will be used, as it is in HANARO, or other remedies will be used. As U–Mo fuel has enough density for a high discharge burnup, the combination case of 4.5/cm^3 and 6.0/cm^3 fuel could be an option for flattening the power distribution. Table VI–1 shows the core characteristics of the reference and U–Mo cores, such as the core uranium loading, maximum LHGR and CAR worth.

The cycle length and discharge burnup are estimated at the same refuelling condition as the reference. Core A is similar to the reference core except for the cycle length. The use of Mo instead of Si requires a further uranium loading of 12% for the same excess reactivity. The increased neutron absorption due to use of Mo and the further uranium loading decreases the CAR worth. The change of the CAR worth is less than 10% even when the uranium loading increases by 50% (core C). The increase of the cycle length is proportional to the amount of uranium loading. The maximum LHGRs, except for core C, are below 120 kW/m, which was selected as the physics design limit of the LHGR at the design stage. As a result, a reduction of maximum LHGR is required for core C.

Next, the neutron flux distributions are compared at the fresh core state:

— The fluxes at the bare core without the irradiation facilities are calculated in a 2 cm × 2 cm × 2 cm rectangular mesh size. As the thermal and fast fluxes in the CT are calculated at its centre, the average thermal flux within the CT is lower and the average fast flux is higher. The CARs are located at an estimated average position during the reactor operation. Table VI–2 shows the maximum neutron flux in the CT and the reflector region.
— The flux at the core with the irradiation facilities is calculated at each irradiation position.

Although the design of the irradiation facilities is very important, features such as the vertical positions and the beam tubes were not fixed at the design stage. At a preliminary conceptual design stage, it is important to obtain a core model to provide a high and broad flux at the reflector tank. This requirement is fulfilled by a compact core design, which requires a large reactivity load. The reactivity load of 20 mK in the AHR design is reserved for the irradiation facilities. When the same facilities as HANARO are arranged at the reflector tank, the reactivity effect in the AHR core is 18.5 mK. A comparison of the neutron flux at the important facilities is shown in Table VI–3 for the reference and U–Mo cores. The core model using MCNP is plotted in Fig. VI–3. For the comparison in Table VI–3, the irradiation positions at a forced convection area are loaded with dummy FAs and the other positions are filled with light water. The neutron fluxes are the averaged values from −35.0 to +35.0 cm from the axial core centre. The decrease of neutron fluxes of the U–Mo cores is lower in irradiation facilities than in the CT and reflector positions.

TABLE VI–1. CORE CHARACTERISTICS OF REFERENCE AND U–Mo CORES

Parameter	Core label			
	Reference	Core A	Core B	Core C
Fuel specifications	4.0 gU/cm^3 U_3Si_2	4.5 gU/cm^3 U–7Mo	6.0 and 4.5 gU/cm^3 U–7Mo	4.5 gU/cm^3 U–7Mo
Core U loading (kg)	51.5	57.5	66.6	76.6
Ratio to the reference	1.00	1.12	1.30	1.50
Max. LHGR (kW/m)	118.6	118.8	115.3	124.2
k_{eff} all rods out	1.269 31	1.267 45	1.273 46	1.287 42
CAR worth (mk)	183.5	179.8	174.0	165.4
Cycle length (days)	31	34	40	50
Av. discharge burnup (%)	57.7	56.4	56.9	61.6

Note: CAR — control absorber rod; LHGR — linear heat generation rate.

TABLE VI–2. MAXIMUM NEUTRON FLUX IN CENTRAL FLUX TRAP AND REFLECTOR REGIONS

Region	Neutron flux (n/cm^{2}s)			
	Reference	Core A	Core B	Core C
CT fast flux (>1.0 MeV)	1.67×10^{14}	1.56×10^{14}	1.52×10^{14}	1.50×10^{14}
CT thermal flux (<0.625 eV)	5.03×10^{14}	4.65×10^{14}	4.54×10^{14}	4.27×10^{14}
Reflector thermal flux (<0.625 eV)	4.36×10^{14}	4.40×10^{14}	4.29×10^{14}	4.31×10^{14}

Note: Core models without the irradiation facilities.

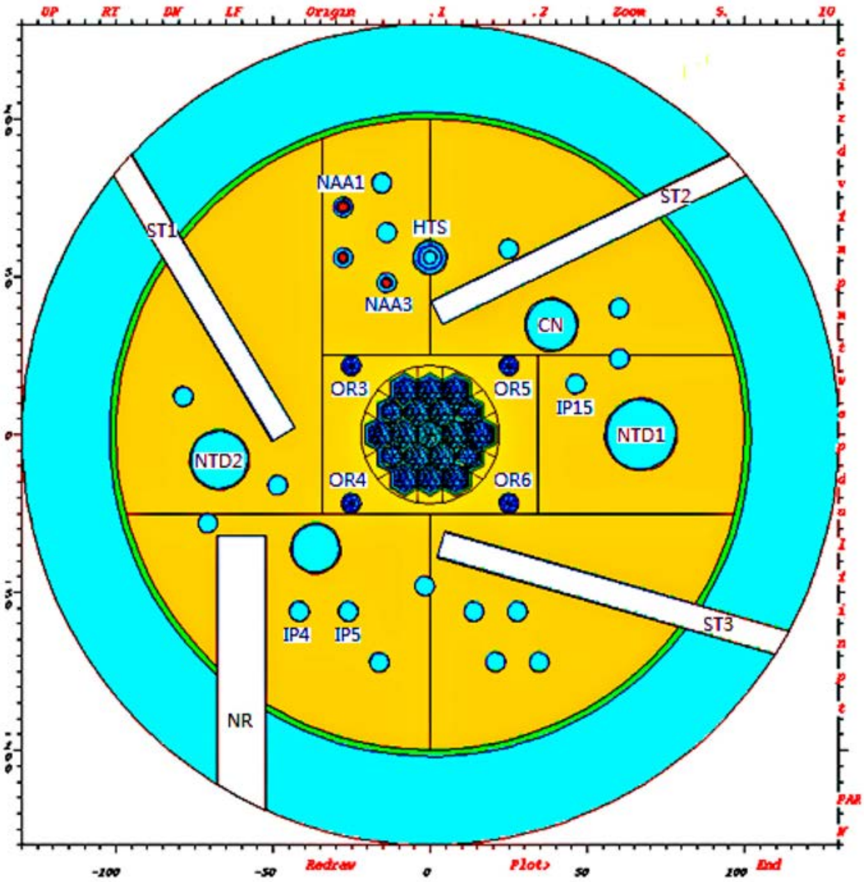


FIG. VI–3. The core model with the HANARO irradiation facilities. Reproduced courtesy of KAERI.

TABLE VI–3. COMPARISON OF THE NEUTRON FLUXES AT THE IRRADIATION FACILITIES

Irradiation facility		Reference neutron flux ($n \cdot cm^{-2} \cdot s^{-1}$)	Difference[a] (%)		
			Core A	Core B	Core C
Vertical positions (forced convection)[b]	OR3 thermal	2.25×10^{14}	−1.4	−2.4	−3.8
	OR4 thermal	2.18×10^{14}	−2.0	−3.8	−4.9
	OR5 thermal	2.15×10^{14}	−1.8	−2.3	−3.6
	OR6 thermal	2.33×10^{14}	−1.3	−2.7	−3.7
	Average	n.a.[c]	−1.6	−2.8	−4.0

TABLE VI–3. COMPARISON OF THE NEUTRON FLUXES AT THE IRRADIATION FACILITIES (cont.)

Irradiation facility		Reference neutron flux ($n \cdot cm^{-2} \cdot s^{-1}$)	Difference[a] (%)		
			Core A	Core B	Core C
Vertical positions (forced convection)[b]	OR3 fast	3.02×10^{12}	+5.9	+2.5	+0.7
	OR4 fast	3.22×10^{12}	+2.1	−4.1	−2.0
	OR5 fast	3.03×10^{12}	+1.6	+7.1	+5.3
	OR6 fast	3.01×10^{12}	+8.7	−0.3	+13.8
	Average	n.a.	+4.6	+1.3	+4.5
Vertical positions (natural convection)	IP4	2.53×10^{13}	−2.7	−3.2	−6
	IP5	5.31×10^{13}	−1.4	−1.7	−4.1
	IP11	8.06×10^{13}	−1.8	−3.3	−5.1
	IP15	9.86×10^{13}	−0.4	−2.8	−3.1
	NTD1	1.87×10^{13}	−1.4	−2	−3.4
	NTD2	2.03×10^{13}	−0.5	−2.4	−3.7
	NAA1	2.53×10^{13}	−2.3	−2.4	−3.5
	NAA3	1.16×10^{14}	−2.8	−2.4	−4.8
	HTS	5.23×10^{13}	−2.3	−2.4	−4.2
	Average	n.a.	−1.7	−2.5	−4.0
Beam tubes	ST1	1.51×10^{14}	−1.4	−2.3	−4.0
	ST2	1.90×10^{14}	−2.3	−1.9	−5.3
	ST3	2.28×10^{14}	−0.9	−2.0	−4.7
	ST4	1.70×10^{14}	−1.1	−1.2	−4.2
	NR	3.09×10^{13}	−0.6	−3.8	−6.5
	IR	2.51×10^{14}	−1.5	−3.4	−4.9

TABLE VI–3. COMPARISON OF THE NEUTRON FLUXES AT THE IRRADIATION FACILITIES (cont.)

Irradiation facility		Reference neutron flux ($n \cdot cm^{-2} \cdot s^{-1}$)	Difference[a] (%)		
			Core A	Core B	Core C
	CN	5.85×10^{13}	−2.4	−3.5	−5.3
	Average	n.a.	−1.5	−2.6	−5.2

[a] Relative to the reference core values.
[b] Thermal energies: less than 0.625 eV; fast energies: more than 1.0 MeV.
[c] n.a.: not applicable.

VI–3. CONCLUSIONS

The AHR core adopts U_3Si_2 dispersion fuel of 4.0/cm^3 as the reference fuel, but needs higher density fuel for higher fuel economy. The U–Mo cores are favourable for a longer cycle core. The reactor physics characteristics of the U_3Si_2 and U–Mo fuelled core are compared and summarized as follows:

- The increase of the cycle length is proportional to the amount of uranium loading in the FAs;
- The change of the CAR worth is less than 10% even with a uranium loading increase of 50%;
- The power peaking factor increases, but could be controlled by reducing fuel density at the outer region;
- The decrease of thermal flux is significant at the core region, but is negligible at the reflector region.

REFERENCES

[VI–1] PARK, C., et al., “Design approach to the development of an advanced HANARO research reactor”, Proc. Int. Symp. on Research Reactor and Neutron Science, Daejeon, 2005, KNS, Daejeon (2005) 45–50.

[VI–2] SEO, C.G., LEE, C.S., LEE, B.C., CHAE, H.T., “Conceptual neutronic design of an advanced HANARO reactor using the U_3Si_2 and U–Mo fuel,” Proc. 28th Int. Mtg on Reduced Enrichment for Research and Test Reactors, Cape Town, 2006, Argonne Natl Lab., IL (2006).

[VI–3] BRIESMEISTER, J.F. (Ed.), MCNP™ — A General Monte Carlo N-Particle Transport Code, Version 4A, Rep. LA-12625-M, Los Alamos Natl Lab., NM (1993).

[VI–4] VILLARINO, E.A., STAMM’LER, R.J.J., FERRI, A.A., CASAL, J.J., HELIOS: Angularly dependent collision probabilities, Nucl. Sci. Eng. **112** (1992) 16–31.

ABBREVIATIONS

AHR	Advanced HANARO Reactor
BOC	beginning of cycle
CAR	control absorber rod
CFA	control fuel assembly
EOC	end of cycle
FA	fuel assembly
FCC	fuel cycle cost
HANARO	High-Flux Advanced Neutron Application Reactor
HEU	highly enriched uranium
JAEA	Japan Atomic Energy Agency
JMTR	Japan Materials Testing Reactor
KAERI	Korea Atomic Energy Research Institute
KJRR	Kijang Research Reactor
LEU	low enriched uranium
LHGR	linear heat generation rate
MC	Monte Carlo
MTR	materials testing reactor
NAA	neutron activation analysis
NTD	neutron transmutation doping
OPAL	Open Pool Australian Lightwater
PPF	power peaking factor
SDM-1	shutdown margin with single failure
SFA	standard fuel assembly

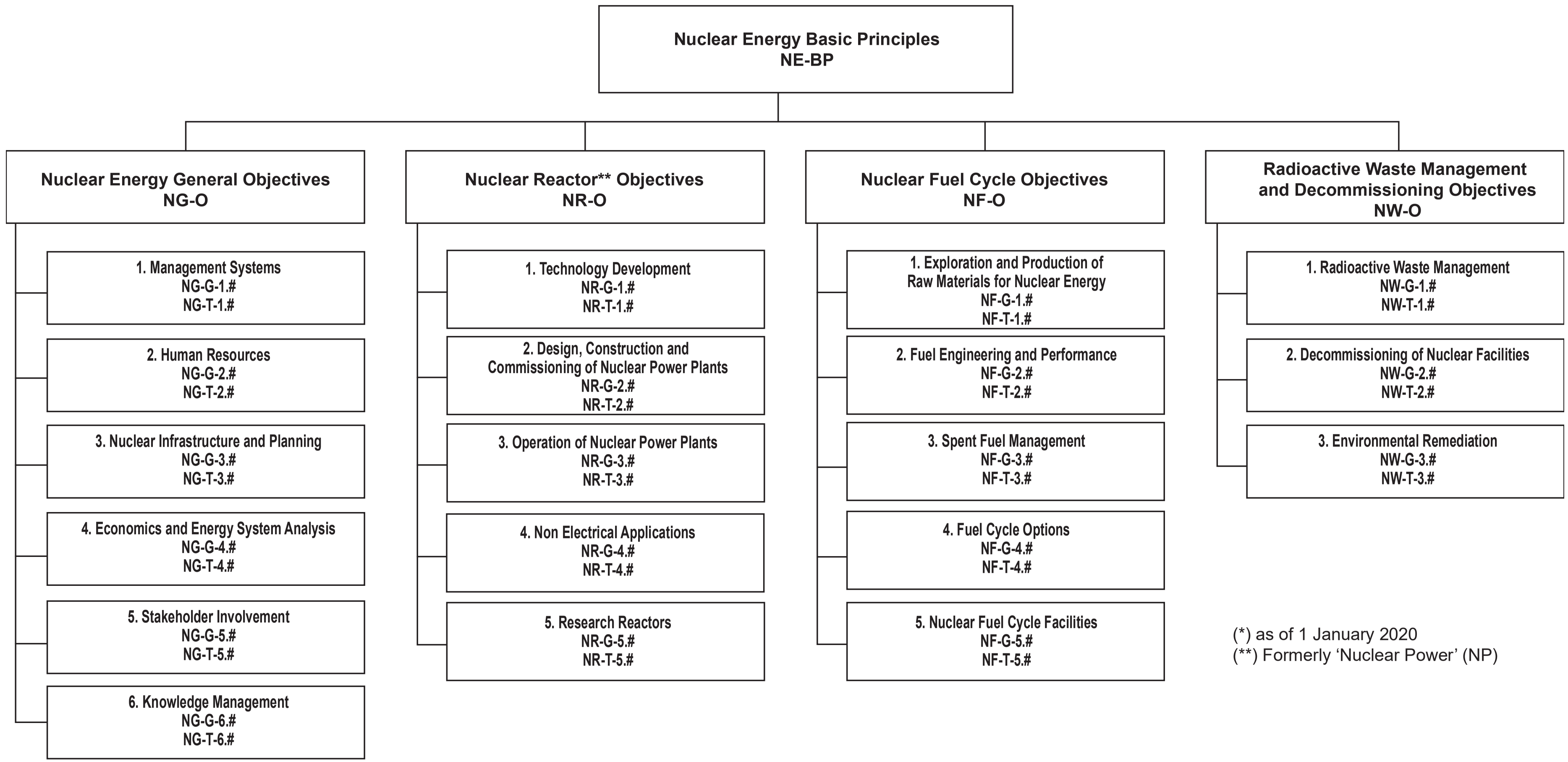
CONTRIBUTORS TO DRAFTING AND REVIEW

Adelfang, P.	International Atomic Energy Agency
Anselmet, M.-C.	French Alternative Energies and Atomic Energy Commission, France
Bouza, P.	Balseiro Institute, National University of Cuyo, Argentina
Braoudakis, G.	Australian Nuclear Science and Technology Organisation, Australia
Dionne, B.	Argonne National Laboratory, United States of America
Ferraro, D.	INVAP, Argentina
Fuentes Solis, N.O.	International Atomic Energy Agency
Gater, R.	International Atomic Energy Agency
Ide, H.	Japan Atomic Energy Agency, Japan
Imaizumi, T.	Japan Atomic Energy Agency, Japan
Ishihara, M.	Japan Atomic Energy Agency, Japan
Klaassen, F.C.	NRG, Netherlands
Koonen, E.	SCK CEN, Belgium
Kusunoki, T.	Japan Atomic Energy Agency, Japan
Lemoine, P.	French Alternative Energies and Atomic Energy Commission, France
Marshall, F.	International Atomic Energy Agency
Mazufri, C.	INVAP, Argentina
Naka, M.	Japan Atomic Energy Agency, Japan
Roglans-Ribas, J.	Argonne National Laboratory, United States of America
Ryu, Ho Jin	Korea Advanced Institute of Science and Technology, Republic of Korea
Seo, Chul Gyo	Korea Atomic Energy Research Institute, Republic of Korea
Shropshire, K.	Consultant, United States of America
Takemoto, N.	Japan Atomic Energy Agency, Japan
Villarino, E.A.	INVAP, Argentina

Consultants Meetings

Vienna, Austria: 17–19 April 2013, 11–13 December 2013

Structure of the IAEA Nuclear Energy Series*



(*) as of 1 January 2020
(**) Formerly 'Nuclear Power' (NP)

Key

BP:	Basic Principles
O:	Objectives
G:	Guides and Methodologies
T:	Technical Reports
Nos 1–6:	Topic designations
#:	Guide or Report number

Examples

NG-G-3.1:	Nuclear Energy General (**NG**), Guides and Methodologies (**G**), Nuclear Infrastructure and Planning (topic **3**), **#1**
NR-T-5.4:	Nuclear Reactors (**NR**)*, Technical Report (**T**), Research Reactors (topic **5**), **#4**
NF-T-3.6:	Nuclear Fuel (**NF**), Technical Report (**T**), Spent Fuel Management (topic **3**), **#6**
NW-G-1.1:	Radioactive Waste Management and Decommissioning (**NW**), Guides and Methodologies (**G**), Radioactive Waste Management (topic **1**) **#1**

IAEA
International Atomic Energy Agency

No. 26

ORDERING LOCALLY

IAEA priced publications may be purchased from the sources listed below or from major local booksellers.

Orders for unpriced publications should be made directly to the IAEA. The contact details are given at the end of this list.

NORTH AMERICA

Bernan / Rowman & Littlefield

15250 NBN Way, Blue Ridge Summit, PA 17214, USA
Telephone: +1 800 462 6420 • Fax: +1 800 338 4550
Email: orders@rowman.com • Web site: www.rowman.com/bernan

REST OF WORLD

Please contact your preferred local supplier, or our lead distributor:

Eurospan Group

Gray's Inn House
127 Clerkenwell Road
London EC1R 5DB
United Kingdom

Trade orders and enquiries:

Telephone: +44 (0)176 760 4972 • Fax: +44 (0)176 760 1640
Email: eurospan@turpin-distribution.com

Individual orders:

www.eurospanbookstore.com/iaea

For further information:

Telephone: +44 (0)207 240 0856 • Fax: +44 (0)207 379 0609
Email: info@eurospangroup.com • Web site: www.eurospangroup.com

Orders for both priced and unpriced publications may be addressed directly to:

Marketing and Sales Unit
International Atomic Energy Agency
Vienna International Centre, PO Box 100, 1400 Vienna, Austria
Telephone: +43 1 2600 22529 or 22530 • Fax: +43 1 26007 22529
Email: sales.publications@iaea.org • Web site: www.iaea.org/publications

20-03469E-T

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA