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Light Water Reactor Fuel Enrichment beyond the Five Per Cent Limit: Perspectives and Challenges

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LIGHT WATER REACTOR FUEL ENRICHMENT BEYOND THE FIVE PER CENT LIMIT: PERSPECTIVES AND CHALLENGES

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LIGHT WATER REACTOR FUEL ENRICHMENT BEYOND THE FIVE PER CENT LIMIT: PERSPECTIVES AND CHALLENGES

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FOREWORD

Fuel high burnup operation has been considered as an option to improve the sustainability of the fuel cycle, since the annual consumption of natural uranium is reduced with extended discharge burnup and the reload fraction of the core is decreased owing to increased flexibility in core management. Long fuel cycles, which are practical with increased fuel burnup, provide economic benefits due to the increased capacity factors resulting from decreased refuelling times.

One important limitation that restricts the further development of fuel high burnup operation and long fuel cycle operation in light water reactors (LWRs) is the current 5% ^{235}U enrichment limit of LWR fuel. Although there is no strong justification for this enrichment limit, it has existed since the establishment of the first requirements for nuclear fuel design and manufacturing and has become the industry standard for LWR fuel operation. The 5% enrichment limit is now being reconsidered in the light of the industrial deployment of LWR fuels that pursue the achievement of economic benefits from high burnup, long fuel cycle operation in the reactor and reduced used fuel inventory. The use of UO_2 fuel with higher enrichment than the 5% limit in LWRs will facilitate the industrial deployment of water cooled small modular reactors.

The development of accident tolerant fuel and innovative fuels requiring higher enriched uranium also challenges the 5% limit. There have been discussions on what the limit for enrichment should be, given that the 5% limit was once seen as providing a large safety margin. This has become a costly limitation and the industry, rather than simply choosing a new limit, needs to consider what real physical and safety limitations to apply in defining a maximum enrichment for the wide variety of fuel designs under consideration. Many fuel cycle experts therefore have suggested that the industry aim to license nuclear fuel cycle facilities for operation up to 20% ^{235}U, which is a limit chosen to be in compliance with the non-proliferation treaty.

Under these circumstances, the limitation of ^{235}U enrichment has become a new concern among Member States, which have requested the IAEA to provide a platform to facilitate a comprehensive review of current status, prospective and challenges associated with the use of fuels having enrichments higher than 5% ^{235}U in LWRs.

In response to these Member State requests, the IAEA organized two Technical Meetings on Light Water Reactor Fuel Enrichment beyond the 5% Limit: Perspectives and Challenges, in 2015 and in 2018. These technical meetings were intended to foster the exchange of information on national and international programmes, focusing on opportunities and implementation issues for the use of high assay low enriched uranium (HALEU) in LWRs. (The enrichment of ^{235}U in HALEU is in the range from 5to 19.75%.) This publication is intended to compile the results and conclusions of these technical meetings, especially in terms of benefits to be obtained from the use of HALEU fuel and consideration of safety issues that arise from its use. The papers presented and presentations made at the two technical meetings are included as an annex and are available on-line as separate supplementary files.

The IAEA wishes to thank all the participants in the two technical meetings for their active involvement and presentations, and the subject matter experts who took part in the preparation of this publication for their valuable contributions and review. The contributions of those members of the Technical Working Group on Fuel Performance and Technology (TWG-FPT) who participated in the peer review of this publication are also appreciated. The IAEA officers responsible for this publication were J. Killeen and K. Sim of the Division of Nuclear Fuel Cycle and Waste Technology.

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1. INTRODUCTION

1.1. BACKGROUND

Typical light water reactor (LWR) fuel is UO_2 with a ^{235}U enrichment in the range of 3% to just below 5% and operates with an average discharge burnup below approximately 60 MW·d/kgU. Indeed, extending fuel discharge burnup has been pursued over the last decades in order to benefit in various ways. Economic benefit can be attributed to maximized fuel utilization, operational benefit to increased operational flexibility with various options on in-core management schemes. Economic benefit is also expected from a reduction of spent fuel as well as from extended cycle lengths that in turn contribute to higher capacity factors.

Previous studies [1–3] have demonstrated such benefits through theoretical case studies of operating UO_2 fuel to extended discharge burnups. For example, one study [2] indicated that, for both boiling water reactors (BWR) and pressurized water reactors (PWR), fuel costs continued to decline with increasing batch average discharge burnups, in this case up to 65.2 MW·d/kgU for BWR with 24-month cycles and 70.3 MW·d/kgU for PWR with 18-month cycles. The economic calculations included a cost penalty caused by the higher enrichment cost of fuel, in excess of the current limit of 5% ^{235}U, which was necessary to achieve the high discharge burnups considered.

The maximum discharge burnup of fuel in a reactor can be determined by taking account of various factors that include but are not limited to:

- 5% ^{235}U enrichment limit;
- Cycle length (12, 18 or 24 months);
- Limits on oxydation thickness and hydrogen pickup of the cladding;
- Radial power distribution across the core;
- Fuel performance in the reactor;
- Economics.

Extended discharge burnup of LWR fuels can be obtained simply by using higher ^{235}U enrichment (Fig.1), although other methods are available to some extent, such as, subdivision of fuel rods in a fuel assembly to reduce the maximum thermal load imposed on the fuel rod, reconstitution of spent fuel assemblies, and reshuffling of spent fuel assemblies. Use of mixed oxide (MOX) fuel (a combination of plutonium and uranium oxides) is also an alternative.

The 5% ^{235}U enrichment limit has existed since the establishment of the first requirements for nuclear fuel design and manufacturing and has become the industry standard for LWR fuel operation. Although no real justification can be found for this 5% ^{235}U enrichment limit, it has remained unchanged. However, LWR fuel enrichments of ^{235}U have gradually increased and in recent designs the ^{235}U enrichment effectively reaches the 5% limit, with 4.95% used to allow for a manufacturing uncertainty to the 5% limit.

The limitation of ^{235}U enrichment to 5% can also affect Accident Tolerant Fuel (ATF) development programmes. New fuel cladding materials[1] developed for ATF may require an increased enrichment of fuel pellets to compensate for their higher thermal neutron absorption than for Zircaloy cladding.

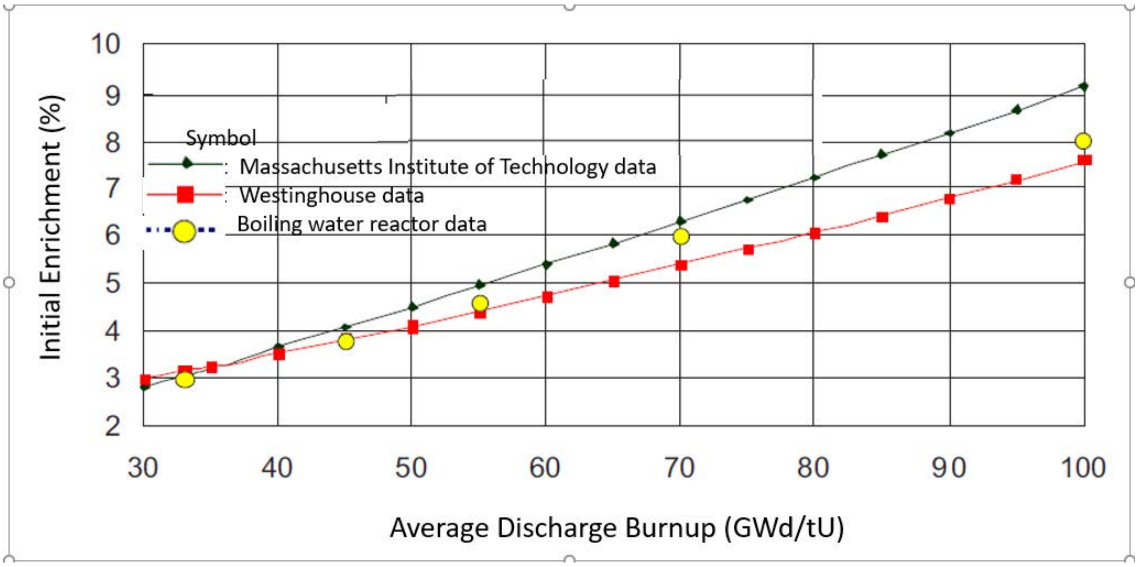

FIG. 1. Initial ^{235}U enrichment versus estimated average discharge burnup in LWRs, Reproduced courtesy of the IAEA [5].

Therefore, the limitation of ^{235}U enrichment to 5% has become a new concern among Member States.

In response to the request of Member States, the IAEA organized two technical meetings on Light Water Reactor Fuel Enrichment beyond the 5% Limit: Perspectives and Challenges in 2015 and in 2018, in order to facilitate a review of national and international programmes focusing on opportunities and implementation issues for the use of High Assay Low Enriched Uranium (HALEU), whose ^{235}U enrichment ranges from 5% up to 19.75%, in LWRs and to facilitate the exchange of relevant knowledge and experience among R&D organizations, utilities, regulatory authorities and governmental agencies.

This publication is intended to compile the results and conclusions of these technical meetings especially in terms of the benefits to be obtained from the use of HALEU fuel and consideration of safety issues that arise.

Similar publications [2,3,6,7] made by other organizations mainly focus on specific aspects, such as economic benefit, domestic fuel cycle infrastructure and licensing practice.

1.2. OBJECTIVES

The objective of this publication is to document perspectives and challenges with the use of HALEU fuel in LWRs, based on the results of the two technical meetings on Light Water

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[1] These include coated cladding, FeCrAl steel cladding, lined Mo alloy cladding, SiC/SiC composite cladding; see [4] for detailed information.

Reactor Fuel Enrichment beyond the 5% Limit: Perspectives and Challenges, held in Vienna, 2015 and in Moscow, 2018. Specifically, this publication is intended to:

- Present various benefits, in qualitative and quantitative manners, attainable through the use of HALEU fuel in LWRs;
- Identify safety concerns that would need to be overcome for the use of HALEU fuel in LWRs; and propose methods to mitigate critical challenges based on practice in Member States;
- Present collected materials (papers and presentations) from the aforementioned two technical meetings.

1.3. SCOPE

The scope of this publication covers:

- Feedback from national and international R&D programmes on the use of HALEU;
- Technological options and corresponding issues regarding fuel and core design, safety analysis and assessments and other aspects related to the use of nuclear fuels for LWRs having enrichments higher than 5% in ^{235}U, such as: manufacturing, handling, transportation, storage, irradiation, and performance in normal and accident conditions;
- Assessment of accident tolerant fuel (ATF) iron-based cladding design options that require higher enrichment and their compatibility with existing reactor designs.

Some design options for the ATF and for innovative fuels to enhance thermal margins in the reactor may need to use HALEU to accommodate for a large absorption of neutrons by the cladding and structural materials. This publication addresses their impact on the ^{235}U enrichment of the fuel and their nuclear compatibility in the existing reactors. Subsequently, the selection of burnable absorber materials and their performance investigation in the material test reactors are of interest. This publication does not address required activities for the design and design verification of the ATF and innovative fuels.

1.4. STRUCTURE

In Section 1, the justification of this project is described.

In Section 2, a worldwide overview on using HALEU in LWRs is described.

In Section 3, benefits that can be achieved from the use of HALEU fuel in power reactors are described from various aspects: high burnup, cycle length, power uprating, economics, adaptability of innovative fuels, waste generation, use of reprocessed uranium and use of ATF.

In Section 4, impacts of ATF iron-based cladding design options on their compatibility with existing reactor designs are described.

In Section 5, additional safety considerations and challenges that come from the use of HALEU fuel are described from viewpoints of international standards, front-end fuel cycle facilities, reactor safety, back-end fuel cycle facilities, accident analysis and validation of computer codes used for safety analysis.

In Section 6, R&D activities on burnable absorbers to facilitate the use of HALEU are described.

In Section 7, conclusions are provided.

Papers presented at the two technical meetings are contained in supplementary material (see Annex I).

2. NATIONAL PERSPECTIVES

Fifteen countries have participated in the technical meetings on LWR Fuel Enrichment beyond the 5% Limit: Perspectives and Challenges, held in Vienna in 2015 and in Moscow in 2018. This Section provides a summary of national perspectives, based on presentations and discussions made at the two technical meetings.

Brazil

A study of PWR fuel is underway, to enable self-reliance and to develop advanced technology, in cooperation with Korea Nuclear Fuel Company and Westinghouse. Spent fuel storage and management is an area of interest with regards to the use of HALEU fuel in power reactors. No difference is expected in the licensing requirements/process between conventional UO_2 and HALEU fuels for power reactors.

Mobile nuclear power plants[2] are being considered to provide electrical energy to isolated regions where the conventional electric network cannot reach, such as: isolated cities, islands, offshore and merchant ships. These reactors would be built on a floating barge that could go where power is needed. An enrichment over 5% is believed to be necessary for such reactors to optimize maintenance and plant availability, reduce costs and improve the competitiveness of this plant type [8].

Some criticality testing has been done at 4.3% enrichment using stainless steel rods, but further testing above 5% enrichment is needed to justify burnable poison use.

China

In China, there is a trend to operate nuclear fuel at extended burnups and extended cycles up to 18 months, taking into account the incentive to achieve economic benefit [9]. For operating reactors there is little incentive to increase current burnup or enrichment limits. For new builds, however, economic evaluation has shown that it would be possible to increase cycle length from 18 months to 24 months utilizing an enrichment of 5.45%.

Specifically, there is a potential need to use fuel enrichment above 5% for LWRs, with the use of the internally and externally cooled annular fuel (IXAF), ATF, or high temperature gas cooled reactor fuel (~8.5% enrichment). High performance fuel is under development which

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[2] The mobile nuclear power plant is the small modular reactor (SMR) that would be built on a barge that could transport it to a required place to operate.

comprises an annular fuel pellet with double cladding (i.e. IXAF). This fuel needs higher enrichment (up to 8%) but will operate at much lower temperatures than standard fuel. Irradiation tests with higher enriched fuel are under way since 2018.

The potential use of ATF is under consideration and research on options has been undertaken.

Transport and storage are issues; high burnup (>50 MW·d/kgU) and ATF need more information in these aspects.

With respect to licensing high burnup HALEU fuel, the computer codes used to model HALEU fuels need qualification for the range of enrichments greater than 5% to 10% or more planned in HALEU.

Czech Republic

There is no plan to use HALEU in power reactors; average initial enrichment of power reactor fuel is around 4.5%. However, there is no limit in applying the current licensing process to HALEU fuel for use in power reactors. ATF fuel with higher enrichments would be considered as long as the fuel supplier can demonstrate meeting all the safety requirements applied to current fuel enrichment (<5%), based on operational experience.

Finland

No specific limits for UO_2 fuel enrichment above 5% for power reactors are used. There is an interest to use nuclear fuel made from reprocessed uranium in Hanhikivi-1 NPP [10]. To compensate for neutron absorption by ^{236}U that exists in the reprocessed uranium, the actual enrichment of ^{235}U in Hanhikivi-1 fuel is anticipated to exceed the 5% limit to achieve equivalent neutronic conditions as the conventional UO_2 fuel with 4.95% enrichment.

Licensee should demonstrate that the same safety criteria used for any nuclear fuel are met. Final disposal needs to be considered in fuel licensing.

France

Currently, fuel enrichment above 5% is not considered in the industry.

However, fuel enrichment is licensed to 6% at the Georges Besse II plant, though transport and packing are limited to 5% enrichment. It is believed that an extension to 6% enrichment would be feasible, but that there would be significant problems at higher enrichments, for example with criticality concerns for single assemblies in water.

It was noted that transport is an international issue and any modification to existing limits or new flasks would be very time consuming, requiring revised codes and validation.

Germany

As the German Government decided to phase out their NPPs by 2022, the utilities are no longer interested in the increased fuel enrichment beyond 5%. However, fuel suppliers (URENCO and FRAMATOME) have an unlimited licence and proceed with their operations. They continue to provide technical innovations for further improvements of safety and fuel performance.

India

In India, stainless steel is being considered as a potential cladding material for use in VVER-1000 type reactor fuels in light of the advantages of insignificant hydrogen generation, possible increase in cycle length, coolant outlet temperature, etc. [11]. In this case, HALEU fuel would need to be used together with appropriate burnable absorbers in LWRs in India.

Japan

Japan was operating a closed fuel cycle before the Fukushima Daiichi nuclear power plant accident and was working on environmentally friendly fuel improvement. The same drivers remain today, to use high burnup fuel to reduce high level waste, to help resource minimization with lower uranium residuals and to implement the efficient burning of uranium.

The need for criticality experiments was indicated due to limited criticality test data [12].

There are no major hurdles for the use of HALEU.

Pakistan

There is no interest at the moment to consider HALEU. The fuel for the Pakistani reactors is supplied from China and there is interim spent fuel storage at the NPP.

Russian Federation

A feasibility study supported the benefit of using HALEU in VVERs in four ways:

- Benefit from the use of reprocessed uranium;
- Benefit from a reduction of fresh fuel assemblies (by 10%) for reloading;
- Benefit from the 2-year fuel cycle length;
- Benefit from ATF.

The current licensing limits remain unchanged. The utility intends to address the economic benefits, considering all factors such as safety cases, equipment replacement, reduction of fuel assemblies and fabrication cost, potential increase of discharge burnup, long term disposal and reprocessing. Extended discharge burnup could be a promising economic option in Russian plants.

Lead test assemblies with 7% enrichment are due to be loaded in a commercial VVER using erbium as the poison.

Russia has the capability to transport enrichments up to 25% based on their fast reactor programme.

Slovakia

Slovakia safety authority would accept the use of increased fuel enrichments based on operational experience in power plants. An initial two-year licensing will allow the use of an increased fuel enrichment within existing operating limits; based on the successful operation for the first licensing period, a second two-year licensing will allow the operation of the increased fuel enrichment with changed operating limits. An environmental impact assessment is required for the increased enriched fuel.

Sweden

Sweden has a nuclear taxation system that does not encourage the use of high burnup fuel, so there is no incentive for the utilities to move to HALEU.

There is a significant fuel manufacturing operation in Sweden, which supplies fuel to other countries and if there is a demand for HALEU it is believed that a safety case could be made to go to 6% (or possibly 7%).

Ukraine

The fuel supplier is TVEL and the Ukrainian utilities will follow the developments in Russian fuel designs. There is no current interest in moving to HALEU.

United Kingdom

With respect to enrichment, a relaxation of the current 5% limit is under discussion. A new limit can be considered from a review of security and operational safety requirements. The industry seems inclined to support the use of HALEU for economic enhancement, the development of ATF for safety enhancement, and next generation fuel/reactor vendors, e.g. for metallic fuel and high temperature gas cooled reactor fuel. It was noted that there is a grey area between the market driven and the criticality safety.

United States of America

Both the safety enhancements of ATF and the longer operating cycles afforded by enrichments beyond 5% are attractive to utilities. Note that ATF has been significantly driven in the U.S. market. Fuel enriched above 5% is considered together with the safety requirements imposed on fuel cycle facilities and NPPs. Therefore, the business decision by the nuclear industry to pursue increased enrichments will be cost related considering all factors. As long as all safety requirements are met and economic benefits are foreseen, there is no reason not to increase enrichments beyond 5%. It is noted that additional data collection and updated analyses for various points within the fuel cycle will be required to assure safety requirements continue to be met.

3. INCENTIVES FOR INCREASING ENRICHMENT FOR LIGHT WATER REACTOR FUEL

3.1. HIGH BURNUP

Japanese investigators [12] have studied the case of using UO_2 fuel with a high enrichment of ^{235}U (less than 10%) to simulate high burnup and longer cycle operation in the next generation LWRs. By using higher enriched fuel, it is anticipated that the number of spent fuel assemblies generated are reduced and thus fuel cycle costs are also reduced whilst power plant availability is improved. As shown in Table 1, the discharge burnup of 70 MW·d/kgU could be achieved by 6% ^{235}U enriched fuel with Zircaloy cladding in a BWR and by 6.5% enriched fuel with Zircaloy cladding in a PWR.

To prepare for higher enriched fuel, fuel cycle equipment such as transportation containers and equipment used in reprocessing facilities will need to be replaced with new ones as appropriate.

The increased cost due to such new equipment has been taken into account in the economic assessment. The results show improved economics with HALEU. The required amount of natural resources was also reduced compared to the conventional UO_2 fuel cases.

TABLE 1. ANALYSIS CASES FOR NEXT GENERATION LWRS IN JAPAN

	PWR-70 MW·d/kgU	PWR-90 MW·d/kgU	BWR-70 MW·d/kgU
Plant Type	PWR	PWR	BWR
Cladding Material	Zirconium alloy	Stainless steel	Zirconium Alloy
Target Discharge Exposure (MW·d/kgU)	70	90	70
Initial Uranium Enrichment (wt.%)	6.5	9.0	6.0
Specific Power (kw/kg)	32.1	32.1	26

- Reproduced courtesy of K. Hiraiwa et al. [12].

An expert from Slovakia [13] used the SCALE code (Version 6.1.2) to analyse the reactivity of fuels in PWR (17×17) and VVER-440 for enrichment up to 6% ^{235}U (without considering burnable absorbers). Figure 2 shows that the discharge burnup of a fuel assembly can be extended to around 80 MW·d/kgU for VVER-440 and to around 97 MW·d/kgU for PWR (17×17) with HALEU (6% enrichment, without burnable poisons).

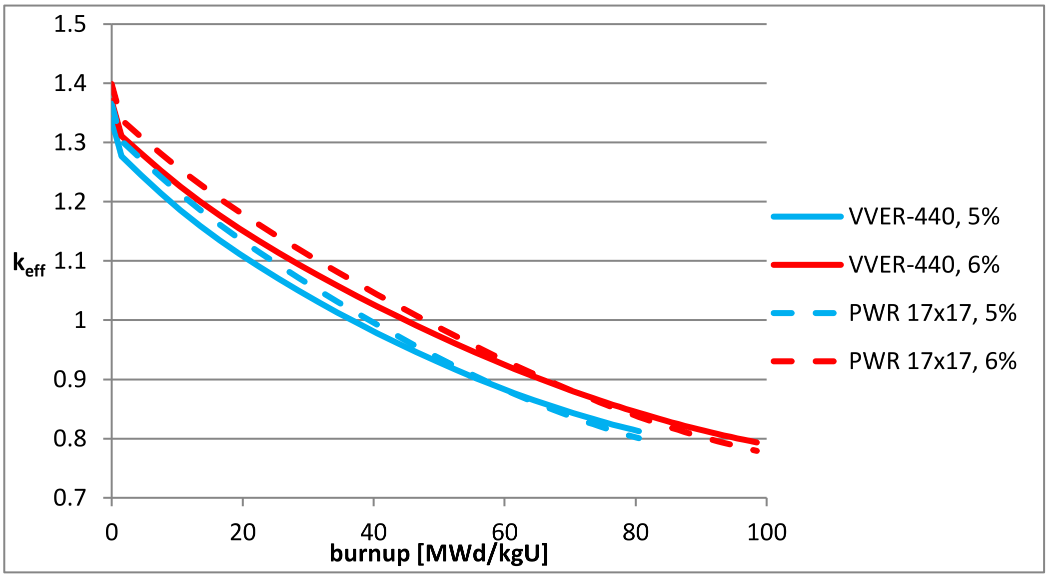


FIG. 2. Criticality of assembly during irradiation (infinite array, average boron, full power), Reproduced courtesy of V. Chrapciak and B. Hatala [13].

Based on the experience of VVER operation and considering the general tendency in LWRs, the following changes [14] were proposed to improve the reactor efficiency in terms of operating flexibility and economics:

- To increase power generation for energy unit through increased heat capacity and/or extended operating period between fuel reloadings (e.g. 18-month and 24-month cycles);

- To enhance the utilization of nuclear fuel for a predetermined level of power generation.

Further, an increase in the energy potential of the fuel loading, which is required to increase the unit power generation, can be realized through three options (see Fig. 3; in this figure, the number of fresh fuel assemblies are between 36 and 78, and the initial ^{235}U enrichment is between 4.4 and 6.5):

- Increasing the quantity of fresh loaded fuel assemblies (reducing the fuel efficiency);

- Increasing the uranium mass in the fuel assemblies without changing the enrichment (increasing the energy potential by not more than 8%);

- Increasing the fuel enrichment in fuel assemblies (dischage burnup can be increased which improves the fuel efficiency).

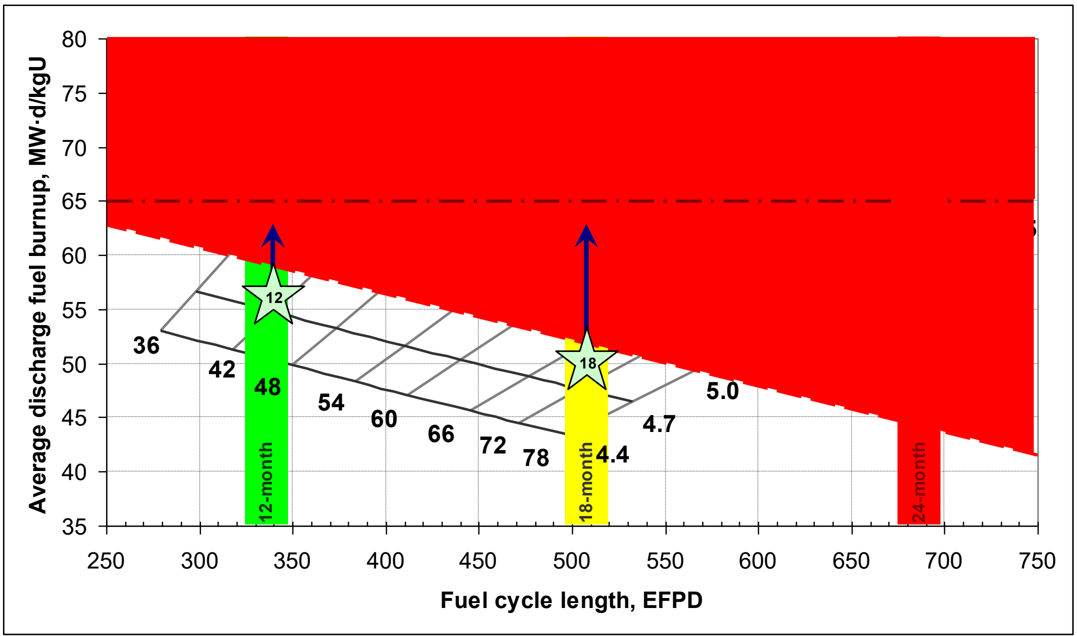


FIG. 3. Dependency of the discharge burnup and campaign duration on the quantity of loading fresh fuel assemblies and their initial ^{235}U enrichment in VVER-1000, Reproduced courtesy of A. Pavlovichev, E. Kosourov et al. [15].

Today, VVER-440/1000/1200 reactors are operated for 12-month or 18-month fuel cycles. Almost all of them are operated at increased power levels, up to 104% of rated power. In all cycles the maximum fuel enrichment is 4.95% and the average enrichment of fresh loaded fuel is about 4.7–4.8%. The current licensed limit of fuel assembly burnup is about 65–70 MW·d/kgU, whilst the average burnup of discharged fuel in 12-month cycle operation is about 55–60 MW·d/kgU, and for 18-month cycles, the average burnup of discharged fuel does not exceed 50 MW·d/kgU.

By using higher enriched fuel (above 5%), the average fuel burnup can be increased (Fig. 3). For 12-month fuel cycles, burnup is close to the licensed limit. An increase in the fuel burnup licensed limit would require serious calculation and experimental studies of fuel performance.

Fuel assemblies operating in 18-month fuel cycles have a significant margin to the licensed limit of fuel burnup. For this reason, an increase in fuel enrichment up to 5.5–5.7 % (Fig. 3) would allow an increase of the average burnup of discharged fuel and reduce the specific fuel cost. Further increases in fuel enrichment would be reasonable after operational experience and carrying out calculational studies and experimental research to increase the licensed limit of fuel burnup.

There is no operational experience of 24-month fuel cycles in VVER reactor types. These cycles are of interest in terms of operating cost. However, the average burnup of discharged fuel would only be around 40 MW·d/kgU for fuel enriched up to 5%. This low fuel efficiency reduces any interest for the utility. Implementation of 24-month fuel cycles, with an average burnup of discharged fuel that is close to the licensed limit, would require an increase in fuel enrichment up to 5.7–6.0 %.

Researchers agree that an increase in fuel enrichment for VVER reactors should be divided into two stages. At the initial stage, the enrichment can increase up to about 6%, this will allow a decrease in the specific fuel cost for 18-month fuel cycles and permit the implementation of economically efficient 24-month fuel cycles. The second stage of enrichment increase (above 6%) can be considered after the completion of design basis and experimental justification of the possibility of fuel operation with burnups above 70 MW·d/kgU.

3.2. CYCLE LENGTH AND ECONOMIC EVALUATION

The 24-month cycles provide more electricity generation than 18-month cycles. The 24-month fuel management strategies for the Chinese pressurized reactor (CPR)-1000 and European pressurized reactor (EPR) were studied [16]. In the study, equilibrium cycle loading patterns were generated with the SCIENCE code package developed by AREVA. Detailed results are described below:

- The fuel management results for CPR-1000 24-month cycle with both enrichment cases (4.45%/4.95% and 5.45%, respectively) showed that a 24-month cycle can be achieved in both cases, meeting all design constraints (Table 2, Table 3). The boron concentrations and enthalpy rise hot channel factor values are relatively high compared to the 18-month refuelling of the typical CPR-1000 reactor at present. The high core power density for the reference CPR-1000, combined with the very long cycle length, resulted in the need to replace more than 50% of the assemblies each cycle. The 24-month cycle for the CPR-1000 with fresh assemblies of 5.45% enrichment saved 24 fuel assemblies, compared to the fuel cycle with fresh assemblies of 4.95% enrichment;

- The fuel management results for an EPR 24-month cycle for both enrichment cases (4.95% and 5.45%, respectively) showed that the 24-month cycle can be achieved in both cases, meeting all design constraints. The boron concentrations and enthalpy rise hot channel factor values are relatively high compared to the 18-month refuelling of the typical EPR-1000 reactor at present. The 24-month cycle for the EPR with fresh assemblies of 5.45% enrichment saved 16 fuel assemblies, compared to the fuel cycle with fresh assemblies of 4.95% enrichment;

- The economic evaluation was performed based on the fuel management results. The economic evaluation method considers power generation revenue and cost including fuel cycle cost, refuelling and overhaul cost and spent fuel storage costs. Preliminary results showed that the CPR-1000 with the 24-month cycles and an enrichment beyond the 5% limit (e.g. 5.45%) had a similar net revenue as the 18-month cycle with an enrichment below the 5% limit (e.g. 4.45%), which indicates that various aspects of economic and operational benefits can be more realized with the fuel option of a higher enrichment than conventional fuel option for a similar revenue condition. The same results were obtained from the economic evaluation exercises for EPR. The economic evaluation was sensitive to assumptions made for natural uranium price, electricity price and overhaul cost.

A study from Czech Republic [17] also showed that an increase in the initial enrichment of nuclear fuel resulted in a possible reduction in fuel cycle cost due to lengthening the reactor campaign from 12 to 18 or even 24 months or due to a reduction of fuel batch reloaded every year, e.g. 1/3, 1/4 or 1/5 part of the core. In [17], an extension of a fuel cycle at the VVER-440 reactors by construction changes and increasing fuel enrichment beyond 5% is feasible. Three different cycle lengths were considered: 6x12 months, 4x18 months and 3x24 months (Table 4). Prepared fuel batches were examined by ANDREA code with regard to selected safety and operation limits of NPP with VVER-440 reactors.

TABLE 2. COMPARISON OF TWO FUEL MANAGEMENT CASES (WITH INITIAL ENRICHMENT OF 4.45/4.95% AND 5.45%, RESPECTIVELY) FOR CPR-1000 24-MONTH CYCLE

	Design constraints for CPR-1000	24-month cycle with 4.45/4.95% U-235	24-month cycle with 5.45% U-235
Number of fresh assemblies	157	100	76
- 8 Gd rods		(28)	(28)
- 20 Gd rods		(24)	(20)
- 24 Gd rods		(24)	(28)
- 20 Gd rods (4.45% U-235)		(24)	(nil)
Average enrichment (wt.%)	n.a.	4.68	5.26
Cycle length (EFPD – Effective Full Power Days)	620 (at 85% capacity factor) minimum	639	622
Maximum HZP MTC (pcm/ºC)	Zero	-3.32	-0.78
Boron concentration (ppm)			
BOL, HZP, ARO	n.a.	2195	2634
BLX, HFP, ARO	n.a.	1536	1937
Max. FDH (Nuclear enthalpy rise hot channel factor)	1.48	1.45	1.55
Average discharged burnup of fuel assemblies (MW·d/kgU)	n.a.	42.8	51.9
Maximum discharge burnup of fuel assemblies (MW·d/kgU)	n.a.	55.0	58.3
Maximum discharge burnup of fuel rod per assembly (MW·d/kgU)	62	60.3	61.5

- Reproduced courtesy of J. Wei et al. [16].

TABLE 3. COMPARISON OF TWO FUEL MANAGEMENT CASES (WITH INITIAL ENRICHMENT OF 4.95% AND 5.45%, RESPECTIVELY) FOR EPR 24-MONTH CYCLE

	Design constraints for EPR	24-month cycle with 4.95% U-235	24-month cycle with 5.45% U-235
Number of fresh assemblies	241	116	100
- 8 Gd rods		(nil)	(nil)
- 20 Gd rods		(44)	(52)
- 24 Gd rods		(72)	(48)
- 20 Gd rods (4.45% U-235)		(nil)	(nil)
Average enrichment (wt.%)	n.a.	4.82	5.29
Cycle length (Effective Full Power Days)	620 (at 85% capacity factor) minimum	671	666
Maximum HZP MTC (pcm/ºC)	Zero	-4.36	-1.89
Boron concentration (ppm)			
BOL, HZP, ARO	n.a.	1877	2191
BLX, HFP, ARO	n.a.	1439	1732
Max. FDH (Nuclear enthalpy rise hot channel factor)	1.58	1.54	1.53
Average discharged burnup of fuel assemblies (MW·d/kgU)	n.a.	51.2	58.8
Maximum discharge burnup of fuel assemblies (MW·d/kgU)	62	60.3	61.8
Maximum discharge burnup of fuel rod per assembly (MW·d/kgU)	68	66.7	67.1

- Reproduced courtesy of J. Wei et al. [16].

TABLE 4. THREE DIFFERENT STRATEGIES OF OPERATION 6×12 MONTHS, 4×18 MONTHS AND 3×24 MONTHS

Notation	Er_2O_3 (%)	Gd_2O_3 (%)	Average enrichment (%)	Batch length (d)	Burnup (MW·d/tU)
er_12_1	0.3	n.a.	4.995	337.2	38 979.1
er_12_2	0.4	n.a.	4.683	305.2	38 185.0
gd_12_1	n.a.	3.35	4.833	331.2	38 768.4
er_18_1	0.7	n.a.	5.819	482.0	43 464.5
er_18_2	0.9	n.a.	5.643	482.7	43 666.8
gd_18_1	0.1	5	5.543	480.3	43 419.7

Notation	Er_2O_3 (%)	Gd_2O_3 (%)	Average enrichment (%)	Batch length (d)	Burnup (MW·d/tU)
er_24_1	1.0	n.a.	6.414	681.2	46 069.4
er_24_2	1.2	n.a.	6.238	698.9	46 804.5
gd_24_1	0.5	8	6.167	654.7	45 054.8

NOTE 1. Reached of six-year fuel cycle with prolonged batches is possible in VVER-440 reactors;
NOTE 2. Modified design and two burnable absorbers (Er, Gd) were considered;
NOTE 3. Increase of fuel cycle length leads to increase of burnup by 16%, 31% and 39% for 6x12, 4x18 and 3x24, respectively.

- Reproduced courtesy of L. Heraltova [17].

3.3. POWER UPRATING

The economic enhancements in LWRs have been primarily investigated through increasing the nuclear core power density in the form of power uprates given existing designs or future plant constructions [18]. Moderate power-uprates (e.g. by 5–8%) have been realized with UO_2 fuel enrichment levels below the 5% limit in existing LWRs. Significant power-uprates (e.g. > 20%) are also archivable but with UO_2 fuel enrichment levels beyond the 5% limit.

The increase in power also needs to come with the ability to maintain or improve the current LWRs' safety standards and margins, in line with international safety standards. The changes to the fuel geometry, materials and/or operating conditions have been previously investigated. This work focuses on changing the fuel geometry while maintaining the operating conditions similar to current LWRs, in order to achieve a significant power uprate. By focusing on improving the nuclear fuel geometry, the improved fuel is still compatible with other fuel types such as high-density fuels (e.g. uranium nitride), high temperature fuels (e.g. tri-structure isotropic or TRISO fuel) and/or metallic and ceramic composite claddings with improved corrosion resistance and high burnup performance.

Given the current LWR fuel material, the cylindrical pin (fuel rod) geometry is limited in providing enough surface area for heat transfer while maintaining desirable structural integrity. Previous parametric studies have shown that cylindrical pin geometry cannot result in beyond 20% power uprate in LWRs and other geometries need to be explored. The internally and externally cooled annular fuel (IXAF) is one such geometry that increases the heat transfer area of the fuel rod significantly and has shown to be able to increase the power output of existing PWRs by 50% and existing BWRs by 25%. The IXAF is cooled on both the external and internal surfaces, resulting in significantly lower average fuel temperatures, even at 50% higher power rating, compared to equivalent solid pin design. The larger heat transfer area and power density of IXAF, comes at the cost of reduced total fuel loading and an increase in enrichment, beyond 5%.

The Helical Cruciform-shaped Fuel (HCF) geometry uses the strategy of fins to increase the heat transfer area and the twisted tapes approach to increase the swirl and intra-bundle mixing of the flow, to increase margin to critical heat flux compared to the traditional cylindrical fuel rod bundle geometries. The HCF design in [18] is a four-petal design which originates from similarly twisted three petal metallic fuel in Russian ice breaker nuclear reactors. The HCF fuel has potential to increase the power density of PWRs and BWRs by up to 50% and 25%, respectively. The HCF concept also eliminates the need for spacer grids as the fuel rods are supported by resting on each other. Like IXAF concept, HCF reduces the average fuel temperature considerably compared to the cylindrical pin geometry. The HCF shape is not ideal to maintain optimum neutron economy. Furthermore, the higher cladding volume along with significant power uprate requires fuel to be enriched above 5%.

3.4. ADVANCES IN LWR FUEL DESIGN

The aim of using HALEU fuel in LWRs is to obtain economic benefits through increasing the discharge burnup and prolongation of the length of fuel cycles to 24-months. Features of nuclear fuel operation at VVER-1000 and VVER-1200 differ from western PWRs, due to higher linear heat generation rate, smaller size of reactor vessels and spent fuel pools located inside the containment. Because of this, heightened requirements should be established for the state of the fuel matrix, fuel cladding, design of fuel assemblies and operating conditions under the necessity of boron regulation in the primary coolant and spent fuel pools. At the same time, the technical and economic analysis should be carried out for the entire life cycle of nuclear fuel, including the fabrication, performance, spent fuel management at NPPs and subsequent storage and processing.

Assessments [19] have shown that the stated objectives for VVERs could be achieved by solving the following tasks:

- **The use of uranium dioxide pellets without a central hole with grain size greater than 25μm and homogenous admixture of erbium to nuclear fuel as a burnable absorber** – The technology for fabrication of similar UO_2 fuel pellets with a dopant of erbium has already existed for Russian RBMK (High Power Channel Type Reactor) uranium fuel. Erbium reduces the neutron-multiplying properties of fuel with enrichment more than 5% to a level equivalent of less than 5%. Consequently, nuclear safety requirements will be fulfilled without the need to change existing technologies related to fuel handling and storage at all stages of nuclear fuel cycles. Retention of fission products in the fuel matrix due to the large grain size will prevent fuel rod damage due to swelling and fission gas release;

- **Ensuring the integrity of the fuel under increased linear heat generation rate by using mixing vane grids in fuel assemblies to intensify its heat transfer** – Strict requirements should be established for primary coolant chemistry and properties of zirconium fuel cladding to prevent corrosion by reducing the thickness of laminar sublayer and the increase of corrosion products deposits on fuel rods due to turbulent coolant flow;

- **Use of MOX fuel in a closed fuel cycle with the aim of nuclear fuel economy –** Special attention should be paid to ensuring radiation safety at all stages of nuclear fuel cycles, including fresh fuel storages and spent fuel pools, equipped with heat removal systems and water treatment systems that remove soluble and insoluble fission and corrosion products, as well as sealed casks for storage of all fuel assemblies with failed fuel rods without the mixing of high-level radioactive water from casks with pond water.

3.5. WASTE GENERATION

It is important to reduce an amount of nuclear waste in the fuel cycle to keep nuclear energy sustainable and environmentally friendly. Higher enrichment fuels reduce the number of neutrons captured by actinides, resulting in the reduction of the production of trans-uranium nuclides. In addition, high-moderated fuel bundle (or fuel assembly) (where the amount of moderator in a fuel assembly increases) also contributes to the reduced production of trans-uranium nuclides by increasing the fission rates of the actinides. Therefore, the high-enriched fuel and the high-moderated fuel bundle are an ideal combination to decrease the production of trans-uranium nuclides in LWRs. In Fig.4, a design concept to reduce the production of trans-uranium nuclides using surplus enriched uranium in the high-moderated fuel bundle is proposed [20].

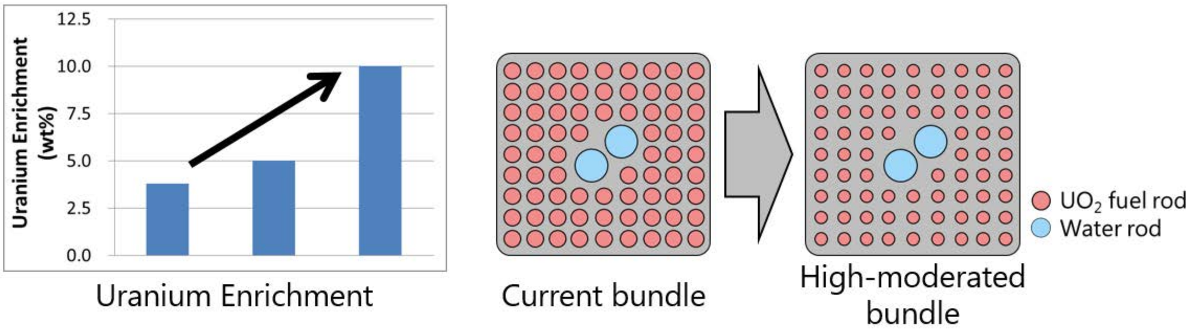


FIG. 4. Design concept of high-moderated assembly to reduce the production of trans-uranium nuclides in higher enriched UO_2 fuel rods, Reproduced courtesy of S. Wada et al. [20].

3.6. REPROCESSED URANIUM FUEL

The Hanhikivi-1 NPP is a VVER-1200 type reactor currently in the process of applying for a construction licence in Finland. It is planned that Hanhikivi-1 nuclear fuel will be made from reprocessed uranium. The main limitation of reprocessed uranium is the presence of even uranium isotopes in the fresh fuel, in particular, ^{236}U, a significant neutron absorber. This limits the effective enrichment of Hanhikivi-1 fuel and leads to an increased fresh assembly batch size and a rather low discharge burnup. Enriching the fuel beyond the 5% limit would allow the batch size to be decreased, with the benefit of a reduced amount of fuel going for final disposal and possibly better optimization of core loadings.

The designed equilibrium cycle with 5.2 % enriched reprocessed uranium (RepU) fuel fulfils the most essential core design parameters related to power distribution, burnup, etc. (Note that here not all core design parameters have been fully checked.) Specifically, the following calculation results [10] were obtained:

- It would be possible to operate Hanhikivi-1 VVER-1200 with RepU fuel enriched to about 5.20 %
  - 5.20 % RepU with 0.7 wt% ^{236}U is equivalent to 4.95 % UO_2 in terms of nutronic characteristics,
  - Core design margins are reduced compared to the basic equilibrium cycle;
- Number of fresh assemblies would be reduced from 48 to 45 assemblies
  - Some savings exist in final disposal costs,

    - Discharge burnup increases but still remains below the planned limit 60 MW·d/kgU;

- Additional analysis on criticality safety may not be needed due to higher enrichment
    - Most likely no problems would occur in terms of criticality safety,
    - Higher than 5.2% enrichment could be challenging without changes in fuel pools etc.

Also note that:

- Core design margins to operational limits are reduced due to some challenges with power distribution;
- Further optimization of the core and/or fuel would likely improve the margins to a certain degree;
- Currently Fennovoima doesn't have confirmative plans to use fuel enriched beyond the 5 % limit but it is interested to follow the progress of this topic.

4. ACCIDENT TOLERANT FUEL IRON-BASED CLADDING

4.1. STAINLESS STEEL CLADDING: LATTICE CHARACTERISTICS

Light water reactors have been extremely successful from considerations of economic advantage, ease of operation, as well as plant system reliability. The earliest LWRs used stainless steel as the fuel cladding material, however, the excellent behaviour of zirconium alloys, as well as their significant advantage of neutron economy over stainless steel, has resulted in the use of zirconium alloys as cladding material.

However, extensive metal-water reaction in the event of severe accidents and subsequent hydrogen generation can be considered as a disadvantage for zirconium alloys. This disadvantage has also resulted in grave consequences both during Chernobyl and Fukushima accidents. Many attempts have been initiated worldwide to find an alternative cladding material that would be more resilient under accident conditions. One of the candidate materials is the reconsideration of stainless steel as fuel cladding [21]. The technology and material science of stainless steel have been studied extensively and at present, understanding of the material is also superior. Some of the nickel alloys may also be considered.

In addition to changing the cladding material, changing the fuel from UO_2 pellets to cermet is also a feasible solution. However, this change amounts to increasing enrichment to significantly high levels and hence is considered having relatively less potential for acceptance.

A study has been made to understand the reactor physics implications of using stainless steel as the cladding material [21]. The candidate system considered for the study is similar to VVER-1000, two units of which are installed at Kudankulam in India.

Due to the considerations of neutron economy, such a change would require an increased fissile inventory and subsequently increased enrichment in the core. However, it is felt that the fuel cost has relatively less influence on the unit energy cost vis-à-vis the operation and maintenance cost for the reactor. The disadvantage in the fuel cost can be compensated, to some extent, by increasing the discharge burnup of the fuel resulting in better fuel utilization. Hence, the slight cost disadvantage due to use of stainless steel as cladding material may be seen as conferring additional advantage of insignificant hydrogen generation, possible increase in cycle length, coolant outlet temperature, etc.

For the present, only preliminary reactor physics studies have been carried out to estimate the extent of increase in enrichment if cladding material is changed from zirconium alloy to stainless steel. The WIMS code with ENDF/B-VII.0 library was used. The study also includes a parametric study for difference in cladding thickness and increased cycle length. In addition, this study identifies aspects of special design provisions to be made for reprocessing of discharged fuel with high initial fissile content. The parametric studies at lattice level have been carried out on a VVER like hexagonal LWR fuel assembly where the clad material has been changed from Zr-1% Nb (E110) to stainless steel.

Figure 5 represents the analysis results of UO_2 fuel cladded with stainless steel tube in comparison with the UO_2 fuel with Zircaloy E110 cladding. The consideration of stainless-steel cladding has resulted in an increase of the inventory of heavy metal per batch from 23.7 tonne (4.5% enrichment: reference) to ~26 tonne (6% enrichment) for 0.6 mm clad thickness.

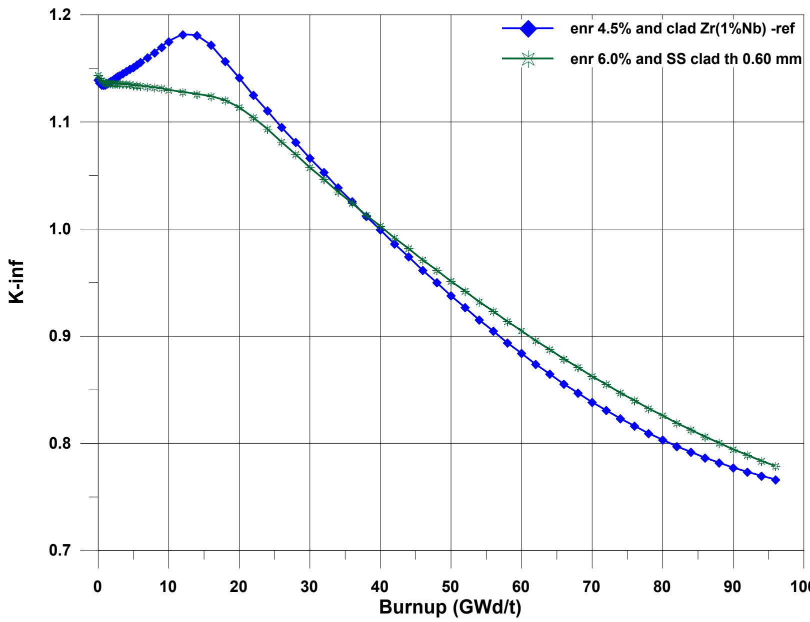


FIG. 5. Characteristics of the reference and modified lattices, Reproduced courtesy of K.K. Yadav et al. [21].

Other important results are: there is a reduced soluble boron worth and a reduced xenon load; fuel temperature load is almost similar to that with reference lattice; requirement of fuel inventory is increased, which can be compensated, to some extent, by increasing the discharge burnup.

The rate constant of hydrogen production is significantly less for stainless steel compared to Zr-alloy at high temperature [21]. Thus, the rate of release of hydrogen into containment is expected to have less severity in comparison to Zr based cladding.

Subsequently, the increase of ^{235}U enrichment would increase the fuel cost per batch by about 50%. The fuel cost has relatively less weight in the unit energy cost compare to the operation and maintenance cost for the reactor. The disadvantage in the fuel cost due to the increased enrichment can be compensated, to some extent, by increasing the discharge burnup.

4.2. Fe-Cr-Al CLADDING: REACTIVITY LOSS

A preliminary neutronic assessment [22] has been done, using MCNP code, for the iron-based alloy cladding to investigate possible reactivity penalties due to the increase of neutron absorption in the different iron-based alloy cladding materials. The trade-off analysis in terms of fuel enrichment and clad thickness has also been evaluated.

Table 5 shows the calculated reactivities for different cladding materials under AP-1000 conditions. The reactivity is affected by the cladding material options as well as fuel material options. With use of FeCrAl cladding, the reactivity could be reduced by around 7% compared with the reactivity of the standard fuel materials (i.e. UO_2 in ZIRLO cladding). The loss of reactivity can be compensated with the thin wall thickness of the cladding, which is described in Section 4.3.

TABLE 5. REACTIVITIES FOR DIFFERENT CLADDING MATERIALS ENVISIONED FOR USE IN AP-1000

Fuel Material	Cladding Material		
	k_∞ (ZIRLO)	k_∞ (FeCrAl)	k_∞ (SS-348)
UO_2	1.46454 ± 0.00026	1.36927 ± 0.00025	1.31416 ± 0.00024
UN	1.32759 ± 0.00024	1.27253 ± 0.00023	1.25515 ± 0.00023
UMo	1.38279 ± 0.00026	1.33076 ± 0.00024	1.31404 ± 0.00024
U_3Si_2	1.45244 ± 0.00024	1.37274 ± 0.00025	1.34904 ± 0.00026
UO_2-Beo	1.42981 ± 0.00025	1.36443 ± 0.00023	1.34405 ± 0.00024

(Note: Reproduced courtesy of Abe et al. [22])

4.3. Fe-Cr-Al CLADDING: TUBE WALL THICKNESSES AND INFLUENCE ON FUEL ENRICHMENT

The cladding materials based on iron alloys have the highest neutronic penalty, mainly due to the presence of Fe, that is already well known as the main neutron absorber.

Three separate approaches can be considered to overcome the neutronic penalty for iron alloys:

- Increase the fuel enrichment;
- Minimize the clad thickness; or
- Increase the fuel mass inside the core.

The first approach implies that the current geometry of fuel is conserved while ^{235}U enrichment is increased. The second and third approaches are coupled in that as the clad thickness is reduced; for a given gap, the additional fuel can be loaded while fixing ^{235}U enrichment at a constant value.

In order to enhance the safety of nuclear power plants, ATF is getting more attention throughout the international nuclear industry. From the standpoint of utility and industry, the development and application of ATF will require not only high safety features, but also acceptable economical costs. Regarding the cladding or fuel candidates with higher neutron absorption, or lower fissile loading, fuel enrichment increase could be an important solution to balance the cost for these ATF concepts. Some preliminary considerations on the issue of enrichment increase for ATF were introduced in [23].

From Fig. 6, the following advantages for FeCrAl cladding material are identified:

- High strength at high temperature;
- Excellent oxidation resistance against high temperature steam;
- Wide application in other fields.

On the other hand, a challenge is also identified:

- High thermal neutron absorption (about 12 times of Zircaloy).

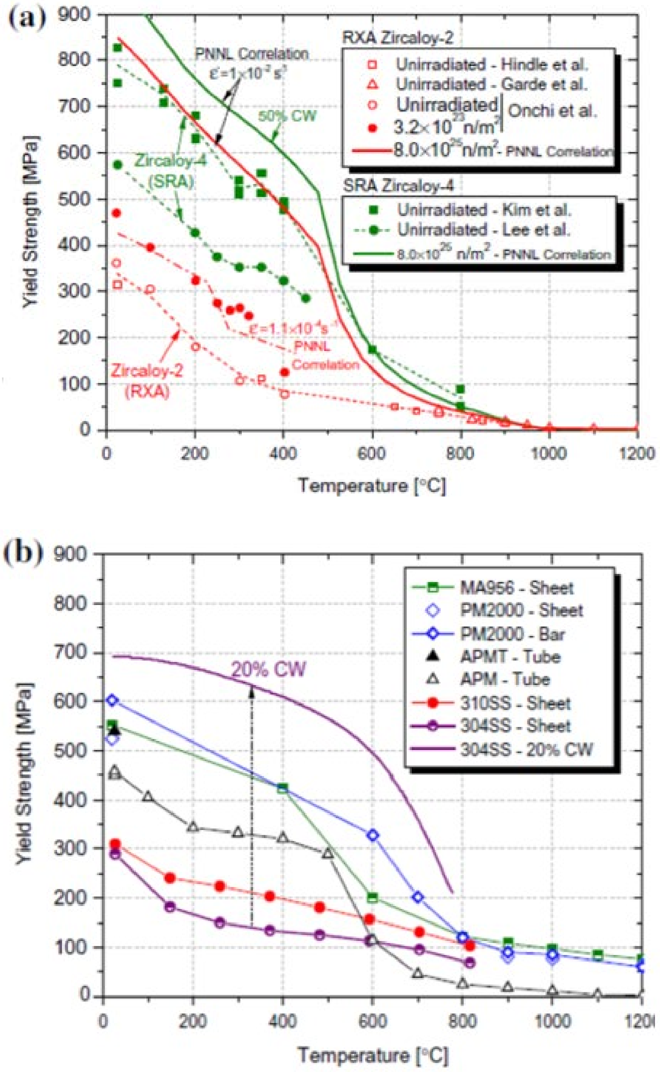
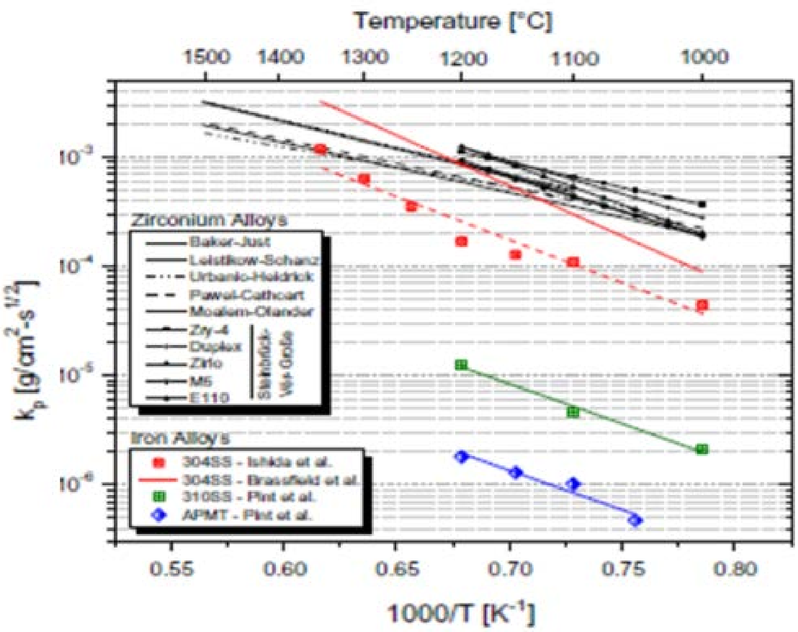


FIG.6. Properties changes of various cladding materials, Reproduced courtesy of Q. Ren et al. [23].



Material	Density (g/cm^3)	Microscopic thermal neutron absorption cross section (barns)
Zircaloy	6.56	0.20
304SS	7.9	2.86
310SS	8.03	3.21
FeCrAl	7.1	2.43
APMT	7.3	2.47
SiC	2.58	0.086

FIG.6 (Continued) Properties changes of various cladding materials, Reproduced courtesy of Q. Ren et al. [23].

In order to achieve equal cycle length with current UO_2-Zr system (cladding thickness is 0.57mm with enrichment of 4.9%), the cladding thickness should be reduced to 0.3 mm or the enrichment increased by 0.96%, when using FeCrAl as cladding material, mainly because of its high thermal neutron absorption.From the point of view of cladding mechanical design (Fig.7):

- FeCrAl has almost the same yield strength with Zircaloy under normal operation temperature (about 360°C) and roughly twice the Young's modulus;
- In order to match the margin in mechanical design, FeCrAl cladding should have the same thickness with Zircaloy;
- For FeCrAl cladding, increasing enrichment or using high uranium density fuel should be taken into consideration to increase fissile material loading.

In order to match the margin in mechanical design, FeCrAl cladding should have almost same thickness with Zircaloy of conventional PWR fuel. In that case the enrichment should be increased by 0.96% to match the cycle length of reactor core.

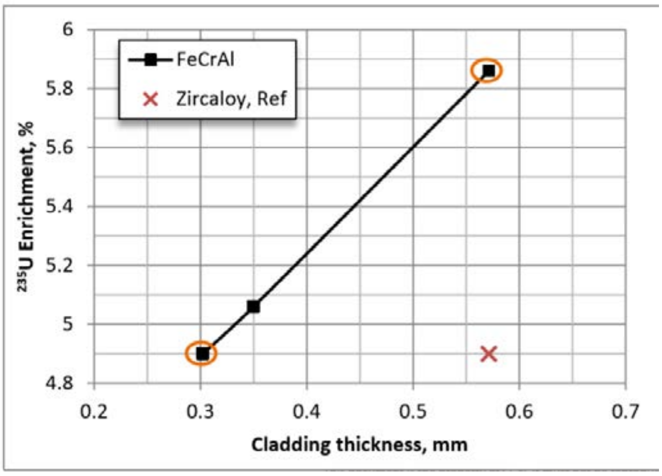


FIG. 7. Required ^{235}U enrichment vs Fe-Cr-Al cladding wall thickness, Reproduced courtesy of Q. Ren et al. [23].

5. SAFETY CONSIDERATIONS AND CHALLENGES

5.1. GENERAL

A significant factor considered for the handling and storage of HALEU at various areas for nuclear fuel cycle is to maintain subcriticality margins. These nuclear fuel cycle areas include: enrichment facilities, fuel fabrication facilities, NPPs, fuel transport, fuel storage facilities, spent fuel reprocessing facilities and facilities for radioactive waste processing and disposal. In the reactor, legacy fuel safety issues that stem mainly from high burnup operation, e.g. fuel fragmentation, relocation and dispersal phenomenon are still important.

5.2. APPLICABLE INTERNATIONAL STANDARDS

5.2.1. IAEA safety standards

The IAEA Safety Standards reflect an international consensus on what constitutes a high level of safety for protecting people and the environment from the harmful effects of ionizing radiation. The IAEA Safety Standards consists of three sets of publications as described in the web site https://www.iaea.org/resources/safety-standards. These are the Safety Fundamentals; the Safety Requirements; and the Safety Guides. While the first of these establishes the fundamental safety objective and principles of protection and safety, the second level documents set out the requirements that must be met to ensure the protection of people and the environment, both now and in the future. The Safety Guides provide recommendations and guidance on how to comply with the requirements.

Figure 8 shows the IAEA Safety Standards applicable to nuclear fuel cycle facilities:

- The IAEA Safety Standard, Specific Safety Requirements publication SSR-4 [24] specifies applicable requirements for safety at all stages in the lifetime of nuclear fuel cycle facilities, including facilities for conversion, enrichment, nuclear fuel production, storage of fresh and spent fuels, reprocessing, preparation for disposal and associated research and development facilities;

- The SSR-4 is supported by five Specific Safety Guide publications including SSG-5 [25] on conversion and uranium enrichment facilities, SSG-6 [26] on uranium fuel fabrication facilities, SSG-7 [27] on uranium and plutonium mixed oxide fuel fabrication facilities, SSG-42 [28] on reprocessing facilities and SSG-43 [29] on nuclear fuel cycle R&D facilities.

- The IAEA Safety Standards, General Safety Requirements publication GSR Part 4 (Rev.1) [30] specifies applicable requirements to be fulfilled in safety assessments for these facilities. The IAEA Safety Standard, Specific Safety Guide publication SSG-27 [31] provides recommendation on criticality safety in these fuel cycle facilities.

- The IAEA Safety Standards, Specific Safety Guide publication SSG-15 [32] provides guidance and recommendations on the design, safe operation and assessment of safety for the different types of spent nuclear fuel storage facility (wet and dry), by considering different types of spent nuclear fuel from nuclear reactors, including research reactors.

In principle, most of these safety standards can be applied above the 5% enrichment limit, whilst the safety standard SSG-5 is currently applicable to enrichment up to 6%. IAEA safety standards should have an authoritative basis at all uranium enrichments in common use.

IAEA safety standards applicable to fuel storage/handling and fuel behaviour in nuclear power plants includes:

- The IAEA Safety Standards, Specific Safety Guide publication SSG-52 [33] on design of the reactor core and Safety Guide publication SSG-63 [34] on design of fuel handling and storage systems in NPPs.

The main issue from transportation aspects concerns package design, which is specified in the following safety standards:

- The IAEA Safety Standard, Specific Safety Requirements publication SSR-6 (Rev.1) [36] specifies "safety requirements for the safe transport of radioactive material. The SSR-6 Regulations apply to the transport of radioactive material by all modes on land, water, or in the air, including transport that is incidental to the use of the radioactive material;

- The IAEA Safety Standard, Safety Guide publication TS-G-1.3 [37] provides guidance on meeting requirements for the establishment of radiation protection programmes for the transport of radioactive material.

Matters relating to nuclear security or to the State system of accounting for, and control of, nuclear material are out of the scope of this publication. From the viewpoint of physical protection measures, which is considered as an interface between safety and security, there is no information to suggest that nuclear security would become more restrictive up to 10% enrichment (i.e. Category III, [35]).

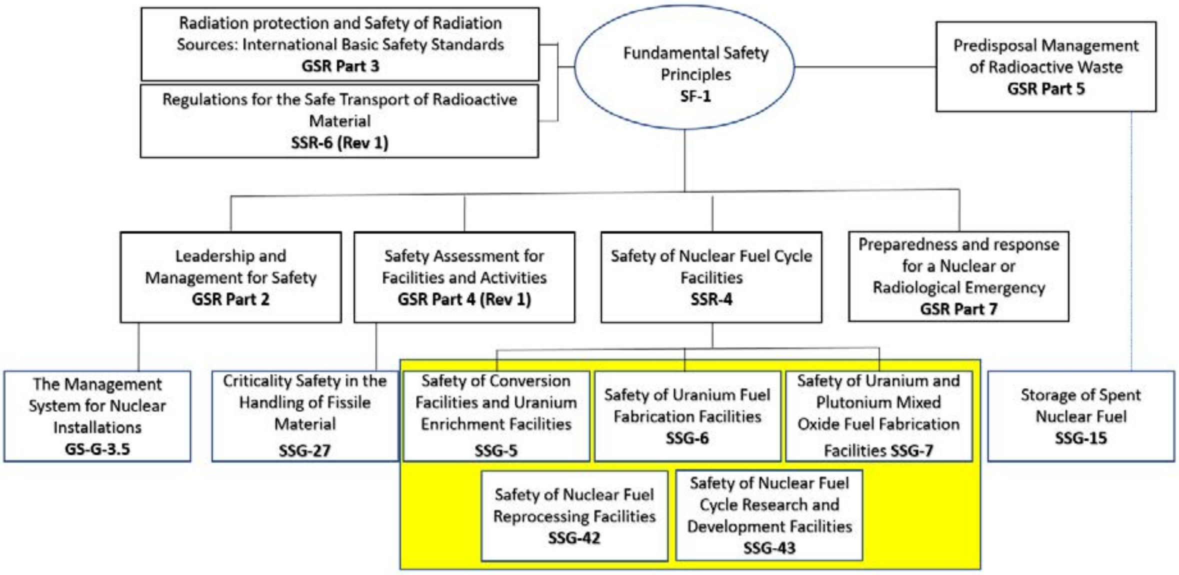


FIG. 8. IAEA Safety Standards publications related to nuclear fuel cycle facilities.

5.2.2. International standards

Several international standards that apply to the nuclear industry may be related to the determination of the maximum level of fuel enrichment:

- ISO 7195 standard for 12 and 30 inch UF_6 cylinders [38];
- ANS-N14.1 standard for 30 inch UF_6 cylinders [39];
- ASTM C-996-15 for the purity of uranium [40].

The first two international standards require a 5% enrichment limit for 30B cylinder for UF_6 transport. The third standard provide standards for purity of UF_6 up to 5% enrichment but does not limit UF_6 enrichment to 5%. Purity standards can be agreed between supplier and purchaser above 5%.

5.3. FRONT-END FUEL CYCLE FACILITIES

5.3.1. Enrichment facilities

In a uranium enrichment facility, feed and withdrawal area, cascades area and product blending devices, including transport and storage cylinders, may be affected due to higher enrichment. Current centrifuge plants may accommodate some degree of higher enrichment in terms of criticality safety. For example, a new enrichment facility in France has been designed for an enrichment of 6% of ^{235}U. Support facilities have been designed for 5%, but the transition to 6% is under progress. Since many facilities for uranium enrichment are licensed for operation up to 5% enrichment, operating licence amendments to modify existing facilities to accommodate HALEU may be challenging.

5.3.2. Transportation to fuel fabrication facilities

As regards transportation of enriched UF_6 from the fuel enrichment facilities to the fuel fabrication facilities, current authorization is up to 5% enrichment. It is noted, however, that preliminary assessments, by Russian experts [41], for transport cylinders with UF_6 enriched up to 7% indicated:

- k_{eff} were less than 0.95 for below 0.1 wt.% of hydrogen in UF_6. (Water bearing concrete bottom has weak impact on k_{eff} owing to design and convex form of cylinder.);

- Flooding of single transport cylinder or group of cylinders by water of different density without water entry does not increase k_{eff} higher than 0.95 for UF_6 enriched up to 7%.

These preliminary results conceptually support that the present transport cylinder for UF_6 can be used to enrichments up to 7% pending on detailed safety assessments.

For UF_6 enriched higher than 7% or for non-acceptable results from the detailed safety assessments for UF_6 enriched not higher than 7%, the current Class A (type 30B) transportation package may need to be redesigned with Class B or relicensed in order to ship fresh enriched UF_6 enriched to greater than 5%.

Alternative solutions to ensure that there is a margin to the criticality limit may include: a reduction in the actual payload and new package design that incorporates physical fixed neutron absorbers inside the package.

5.3.3. Fuel fabrication facilities

As regards fabrication facilities, novel fuel matrices and cladding may require new processes (e.g. high-temperature or pressure). Additional criticality control may be required for the receipt and storage of HALEU at fuel fabrication facilities.

As an example, a Russian fuel fabrication plant, PJSC NCCP has preliminarily evaluated the availability of technology for manufacturing VVER and PWR nuclear fuels with ^{235}U enrichment up to 7% [42]. All process stages required for fuel fabrication and transportation have been considered. These included:

- Area of uranium dioxide production by reducing pyrohydrolysis method;
- Area of nuclear ceramic fuel pellets production;
- Area of fuel rod fabrication;
- Area of fuel assembly fabrication;
- Alarm system for signaling of self-sustained chain reaction;
- Finished-products storage area;
- Shipping container for fuel assembly transportation.

The evaluation results indicated that overall the fuel fabrication plant is capable of producing uranium dioxide powder, pellets, fuel rods and fuel assemblies with U-235 enrichment up to 7%, if some modification for equipment are done. The modification should include: the quantity of the nuclear-hazardous equipment used for fuel assembly manufacturing with up to 5% enrichment and of the equipment that would become nuclear-hazardous when used for fabrication of fuel with enrichment exceeding 5%.

A similar preliminary safety assessment on other Russian fuel fabrication plant, PJSC MSZ [43] represented the results in quantitative manner. That is,

(1) Based on the optimized analysis,
- 90% of the equipment (421 items) may be used without replacements or modifications;
- 10% of the equipment may need lowering of the loading (accumulation) rate, which may result in up to 20 % output capacity loss;
- 3% of the equipment may need installation of deflector shields (baffles) at 140mm distance;
- 7% of the equipment may need replacement with safe-type equipment.

(2) Based on the conservative analysis,
- 88% of the equipment (412 items) may be used without replacements or modifications;
- 10% of the equipment may need lowering of the loading (accumulation) rate, which may result in up to 20 % output capacity loss;
- 12% of the equipment may need replacement with safe-type equipment.

The fabrication of the burnable absorber is an additional consideration. A separate production line for the powder, pellets and fuel rods of burnable absorbers, as well as their inspection methods, should be available.

5.3.4. Transportation to nuclear power plants

As implied, current transportation package may need to be redesigned or relicensed in order to ship fresh enriched fuel assemblies to NPPs.

For fuel assembly transportation packages, the criticality analysis of the packages does take into account burnable absorbers in the assembly. This may allow a slight increase of ^{235}U enrichment above 5%. For example, with use of a small amount of erbium (less than 0.2%) in the fuel assembly with 6% enriched UO_2 fuel, the effective multiplication factor in fresh fuel transport package and fresh fuel shipping cask could remain at a similar level with 5% enriched UO_2 fuel without burnable absorber [41]. The same result was obtained with an amount of erbium less than 0.4% in the fuel assembly with 7% enriched UO_2 fuel [41].

In case of transporting fuel assemblies that use gadolinium as burnable absorber or do not contain burnable absorber, a reduction of the payload could be considered; however, some packages do not have such option because only one assembly is loaded in a package. The new package design may include fixed burnable absorbers in the packaging walls and in the internal structures of the package.

5.4. REACTOR

The safety requirements applicable to the fuel and its operation in the reactor are not changed in any way due to an increase in enrichment, nor to the implementation of ATF design. Any such change is justified, and the fuel and reactor remain within the bounds of the safety requirements.

Legacy fuel safety issues that mainly stem from high burnup operation, e.g. fuel fragmentation, relocation and dispersal phenomenon are still important. It may be necessary to modify plants or amend procedures to ensure that the safety requirements are met even with a conventional HALEU fuel.

There will be a need to carry out criticality assessments to cover normal and accident conditions at the reactor site and it is important to be sure that the codes used have proper validation.

5.4.1. Fresh fuel handling and storage at reactor site

The arrival of fresh fuel at a reactor storage facility will require a demonstration that there will be no safety concern, particularly a criticality concern, during unloading operations and storage in the reactor pond. Such safety cases will need to demonstrate that under both normal conditions and accident conditions, such as a dropped assembly or a misloading in the pond a criticality incident will not occur, nor operational exposures exceeded.

Increasing the enrichment above the 5% limit that is currently widely used, will require a full reassessment of the safety case to cover the higher reactivity of the new assemblies. In many cases, this may result in a need to amend operating procedures, though plant modifications, such as additional, fixed absorbers in the spent fuel pool may also be required.

HALEU fuel will generally be supplied with some form of burnable absorber, which can be discrete or integral within the fuel rods. Criticality assessments will need to account for this, particularly if it is possible to misload a discrete absorber. For example, with use of a small amount of erbium burnable absorber (i.e. less than 0.2% and 0.4%, respectively) in the UO_2 fuel enriched to 6% and 7%, the effective multiplication factor was estimated to be similar with 5% enriched UO_2 fuel without burnable poison for fresh fuel storage rack [41].

The higher reactivity of HALEU fuel will require careful consideration for sequencing the fuel load into the reactor.

5.4.2. Reactor physics design and fuel management in the reactor

The main drivers for increasing fuel enrichment are either to gain commercial benefit (e.g. power uprate, cycle length extension, burnup extension) or to recover lost reactivity due to a feature of an ATF design.

There are many examples of calculations to increase cycle length or to achieve higher burnup, which lead to a requirement for HALEU fuel, particularly in VVER fuel. Sections 3.1 and 3.2 of this report describe some of these in more detail. A Chinese study [16] has shown fuel cycle improvement with a small increase in maximum enrichment to 5.45%

Calculation for the use of any ATF will need to be carried out in detail, even if HALEU is not needed. Such calculations will need to cover mixed core issues [44] as well as equilibrium core issues.

Stainless steel cladding has been used in early PWRs with low burnup designs and is known to be thermally and mechanically compatible with PWR conditions. This cladding material was generally replaced by zirconium alloys for better neutronics. However, the current interest in ATF designs has renewed interest in stainless steel cladding, and several countries have prepared initial calculations showing the need for HALEU with stainless steel cladding to match the performance of standard fuel designs [11,21,22]. For example, in [45], various cladding material options (from various alloys) have been investigated, as part of ATF study, for a possible replacement of zirconium alloy cladding of a UO_2 fuel rod from neutronic aspects. For given cladding material options together with thin wall thickness, the UO_2 fuel needs to be enriched more than 5% for use in VVER-1200.

There are also examples of fuel designs to allow for significant whole core power increases which alter the basic pin geometry to include a coolant path through the centre of the pellet (co-axial coolant flow paths). Another option is to design cooling fins on the outer surface of the cladding [18]. These fuel designs require increased enrichment to compensate for the reduction in fuel volume associated with such designs.

If HALEU fuel is used in reactor cores, it will be necessary to demonstrate the validation of any codes used for the new materials and enrichments.

5.4.3. Fuel behaviour in the reactor

Longer reload fuel cycle and power uprates may result in increased discharge burnup. Challenges of high burnup and extended in-reactor service include:

- Changes in pellet microstructures (e.g. grain growth, rim structure, cracking);
- Decrease in pellet thermal conductivity;
- Higher fission gas release and rod internal pressure;
- Higher decay heat loads;
- Changes in fuel cladding microstructure and properties due to irradiation damage and hydrogen pickup;
- Water side corrosion, hydrogen pickup and crud deposit;
- Dimensional changes in fuel rod and assembly components due to irradiation-induced growth, creep and corrosion.

5.4.4. Spent fuel handling and storage at reactor site

Spent HALEU fuel will generally require longer storage in a spent fuel pond due to its higher burnup and higher neutron emission on discharge. This could put pressure on storage capacity, even with smaller numbers of fuel assemblies to be stored.

5.5. BACK-END FUEL CYCLE FACILITIES

Spent nuclear fuel produced by operation of NPP represents jointly highly radioactive waste and nuclear materials. Therefore, it is important to take into consideration all characteristics of spent nuclear fuel when designing facilities and equipment for storage, transport and handling of it. There are three main tasks to be solved when handling spent nuclear fuel:

- Maintaining subcriticality;
- Residual heat removal;
- Protection against ionizing radiation.

At the same time, it is necessary to consider both the physical and chemical attributes of spent fuel and its construction materials and design.

Existing back end fuel cycle facilities were designed for the handling of spent fuel with initial enrichment less than 5 % of ^{235}U. Even though there are some safety and design margins, the handling of spent fuel with initial enrichment above 5 % of ^{235}U will require additional measures in order to meet the requirements for handling this spent fuel.

5.5.1. Spent fuel pool

Existing spent fuel pools (SFP) – At Reactor SFP and Away from Reactor SFP – may be not able to store spent fuel with enrichment above 5 % of ^{235}U, since hotter and more reactive fuel may squeeze pool capacity. Therefore, additional calculations, design changes and other measures may be necessary. Necessary changes resulting from subcriticality calculations (possibly using burnup credit), residual heat production calculations and radiation load calculation may apply:

- Changing of the spent fuel storage grid (distance between spent fuel assemblies, increasing of boron content in cooling water, insertion of solid neutron absorbers into storage racks);
- Increasing of cooling performance of heat removal cooling system;
- Additional shielding due to increased neutron and gamma radiation level;
- Time needed for cooling the spent fuel assemblies before transport to long term storage or to final disposal.

5.5.2. Spent fuel transport

For existing transport containers, the increase of initial enrichment may lead to increased burnup, reactivity and radiation level. Neutron dose rates may go up exponentially due to curium-242 and -244. These changes may apply:

- Changing of the inside transport cask (distance between spent fuel assemblies, insertion of solid neutron absorbers into the storage racks);
- Requirement for increased shielding on package design;
- Requirement for increased storage time before transportation;
- The limitation in residual heat removal capacity of the transport container may lead to necessity not to use the whole capacity of the transport container.

Uranium-erbium fuel cycles based on HALEU have been assessed from spent fuel transport aspects by Russian experts in [41]. The average residual heat and the average power of neutron source for discharged uranium-erbium fuel cycles based on HALEU are compared with those for the standard 12-month fuel cycle in Fig.9[3] and Fig.10[3]. These values for uranium-erbium fuel cycles based on HALEU remain not exceeded those values for the standard 12-month fuel cycle in VVER-1200, except for two cases: (1) 18-month fuel cycle with a reduced number of fresh fuel assemblies (from 72/73 to 54) and (2) 24-month fuel cycle with 72/73 fresh fuel assemblies. These results conceptually indicate that if the fuel discharge burnup is restricted

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[3] In Figs 9 and 10, X-axis represents various fuel types in VVER-1000/1200. Specifically, ‘A’ indicates VVER-1000 and ‘B’ indicates VVER-1200. ‘12’, ’18’ and ‘24’ indicate 12-month, 18-month and 24-month cycle length, respectively. ‘G’ indicates Gd_2O_3 as burnable absorber and ‘E’ indicates Er_2O_3 as burnable absorber. ‘42’, ‘54’, ‘61’, ‘73’, ‘82’ and ‘108’ represent the number of fresh fuel assemblies in the reactor, that is, 42 fuel assemblies, 54 fuel assemblies, 60/61 fuel assemblies, 72/73 fuel assemblies, 81/82 fuel assemblies and 108 fuel assemblies, respectively. For example, ‘A18G73’ indicates 72 or 73 fresh fuel assemblies with 18-month cycle length, Gd_2O_3 burnable absorber in the VVER-1000 reactor.

(for example, by means of the number of fresh fuel assemblies), the transport package used for the standard 12-month fuel cycle could also be used for HALEU.

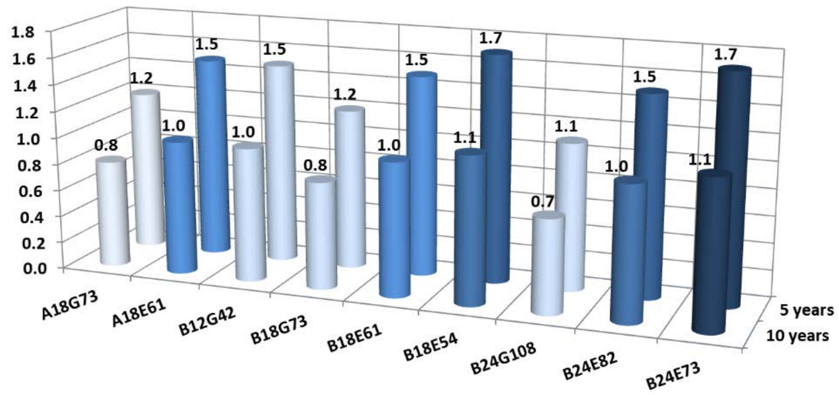

FIG. 9. Average residual heat for discharged fuel as function of residence time (kW/FA), Reproduced courtesy of K.Y. Vergazov, A.V. Ugryumov et al. [41].

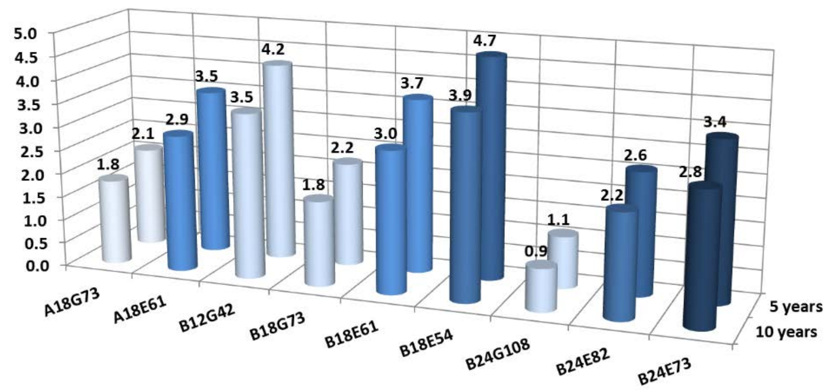

FIG. 10. Average power of neutron source for discharged fuel as function of residence time (10^8•neutron/FA), Reproduced courtesy of K.Y. Vergazov, A.V. Ugryumov et al. [41].

5.5.3. Spent fuel dry storage

For existing dry storage installation, the same provisions may apply as for wet storage:

- Changing of the spent fuel storage grid (distance between spent fuel assemblies, insertion of solid neutron absorbers into the storage grid, etc.);
- Control calculation of cooling performance of heat removal passive cooling system;
- Additional shielding due to increased neutron and gamma radiation level;
- Time needed for cooling the spent fuel assemblies before transport to final disposal.

Casks designed for storage of MOX fuel have improved heat removal capability, although the capacity of the storage system may be reduced. There is a potential trade-off between cooling time prior to dry storage and dry storage capacity.

5.5.4. Reprocessing

Reprocessing involves operations to recover the uranium and plutonium from waste products (e.g. fission products, minor actinides in fuel assemblies) after the fuel has been irradiated. In the wet spent fuel reprocessing process, criticality safety of the equipment is mainly control by volume and the moderator, especially in the chemical separation, purification and the conversion areas. Handling of reprocessed uranium may be hazardous.

5.5.5. Other concerns

5.5.5.1. Long term storage

Re-criticality may be a concern.

5.5.5.2. Subcriticality margin

Subcriticality margin remains all the same; for normal operational conditions k_{eff} should be less than 0.95, for accident condition less than 0.98.

5.5.5.3. Heat loads from decay heat

Spent fuel with enrichment above 5% of ^{235}U will have higher burnup. This will result in higher residual heat production. Such fuel will need longer cooling time before to be suitable for transport. On the other hand, the limitation of fuel discharge burnup will lead to a decrease of the residual heat production and thus of cooling time.

5.6. ACCIDENT CONDITIONS

5.6.1. Accident conditions at the fuel handling and storage area

Fuel handling and storage systems at all stages in the nuclear fuel cycle needs to be designed to fulfil the main safety functions in all facility states of the nuclear fuel cycle facility: that is, (a) confinement and cooling of radioactive material and associated harmful materials; (b) protection against radiation exposure; (c) maintaining subcriticality of fissile material (Requirement 7 of SSR-4 [24]). Therefore, postulated initiating events taken into account for the design of fuel handling and storage systems include those events that potentially lead to a reduction in subcriticality margin, a reduction in decay heat removal capability (if applicable), a significant release of radioactive material, or a significant direct radiation exposure of operating personnel.

Increased fuel enrichment together with increased burnup needs to be considered in accident cases, e.g., fuel handling accidents and seismic events.

The IAEA Safety Guide SSG-63 [34] is under development, which describes accident conditions and internal/external hazards that should be considered in the design and safety analysis of fuel handling and storage systems at the reactor site. For the fuel handling and storage in other areas except for the reactor, the Specific Safety Guide SSG-15 (Rev.1) [32] describes the same contents.

5.6.2. Accident conditions at the reactor

To ensure the reactor safety, the combination of the increased enrichment and changed behaviour of the fuel in the core needs to be considered in the safety analysis.

All safety requirements considered need to be met. Safety analysis methods and conditions are described in IAEA Specific Safety Guide SSG-2 (Rev.1) [46].

5.7. CODE VALIDATION

Increasing the uranium enrichment of LWR fuel effects every aspect of the fuel cycle, from enrichment and fuel fabrication to storage or reprocessing of spent fuel. Correspondingly, the codes and methods currently used to model the performance of contemporary fuel designs with conventional enrichments may not be sufficient to model the performance of HALEU fuels and assure safety and design requirements are met unless modifications are made. This is also true for the modelling of innovative fuel designs and ATF fuel designs.

The increased concentration of ^{235}U that is characteristics of HALEU, along with the presence, type and concentration of burnable neutron absorbers, alters the isotopic composition of the fuel matrix in comparison to conventional enrichments. This affects both the neutron spectrum and the isotopic concentrations of fission products as a function of burnup. For core physics codes, correctly modelling these phenomena is vital for the accurate prediction of fuel pin and assembly powers throughout the life of the core, and the subsequent determination of available margin to safety and design requirements.

Fuel performance analysis codes are also affected by the change in fuel matrix composition. In the case of fuel thermal mechanical codes, the change in fuel pellet porosity influences fuel pellet thermal expansion and fission gas release rates, which require accurate modelling to assure the potential for pellet-cladding interaction and higher rod internal pressures do not challenge cladding integrity. If increased burnups are considered, fuel pellet fragmentation and relocation may be issues that require a higher fidelity modelling. If innovative fuel designs and ATF fuel designs are considered, the cladding materials may require new models for cladding stress, creep, growth and oxidation.

Thus, increasing enrichment will impact phenomena in ways that, depending on the code, may not have been previously considered, and attempting to use these codes to model HALEU will cause them to behave in ways not previously examined. To assure safety and design requirements are met, codes need to be validated for use with HALEU fuel, innovative fuel designs and ATF fuel designs.

Assessment of code applicability should ideally begin with the identification and ranking of importance of the phenomena impacted by increasing uranium enrichments or new fuel designs, followed by the verification and validation of the code to model the impacted phenomena, especially if modifications to the code are needed to achieve the necessary modelling capabilities. Uncertainty quantification is also of prime importance for assuring enough margin exists to safety and design requirements. While aspects of this approach are generically applicable (e.g., impacted phenomena), the unique nature of each analysis code requires application of the approach on an individual basis (e.g., for demonstrated results and quantified uncertainties).

Validation of a code involves the comparison of code results to real world data. At present, there is little real-world data available that characterizes the thermal and neutronic performance of fuels with enrichments beyond 5%. Thus, there is a need to expand existing databases (e.g., criticality) so that code validations can be performed. While comparisons of code results to those of higher-order codes and methods do provide support of the correct functioning of a code, there is no substitute for real-world data.

6. R&D PROGRAMMES FOR INCREASING ENRICHMENT FOR NUCLEAR FUEL: BURNABLE ABSORBER RODS

From the standpoint of the reactor core safety, the main issue of 18-month and 24-month fuel cycles is assurance of the negative coolant temperature feedbacks. This issue can be addressed by using burnable absorbers. Burnable absorbers are also required for decreasing the power distribution irregularity especially in the case of low neutron leakage configuration when the most burnt fuel assemblies are distributed in the peripheral cells of the core. At the present time, gadolinium burnable absorber is successfully used in Russian type of light water reactor VVER.

If fuel enrichment is beyond 5 % in 18-month and 24-month fuel cycles, the number of gadolinium burnable absorber rods and the content of uranium oxide should be increased to provide negative coolant temperature feedbacks. It is known that the more gadolinium burnable absorber rods in fuel assemblies the higher irregularity of power distribution within a fuel assembly. This also leads to substantial power redistribution between fuel assemblies and along the length of each fuel assembly during operation.

Besides gadolinium, investigators have also considered boron, erbium, samarium, hafnium etc. for possible use as burnable poisons in water cooled reactors.

Russian investigators have considered the following types of burnable absorbers and their distribution within fuel assembly, besides well-known gadolinium oxide distributed in the limited number of fuel rods:

- Erbium oxide distributed in all fuel rods;
- Dysprosium oxide distributed in the limited number of fuel rods and homogeneously distributed in in all fuel rods;
- Zirconium diboride applied in a thin layer on the side surface of fuel pellet;
- Hafnium containing in fuel rod claddings;
- Samarium oxide distributed in the limited number of fuel rods and homogeneously distributed in all fuel rods.

The content of burnable absorber has been chosen in such a way to provide the same k_{eff} of fuel assembly as for fuel assembly enriched to 5% with 6 U-Gd fuel rods with gadolinium oxide content of 5 %. The comparison with fuel assembly, enriched to 5 % that contains 27 U-Gd fuel rods with gadolinium oxide content of 8 %, was also considered.

Based on the calculational results of k_{eff} versus burnup, Gd_2O_3, Er_2O_3 and ZrB_2 are selected to be attractive burnable absorbers in the viewpoint of reactivity penalties caused by the incomplete burn of absorbers.

For Russia with its existing experience of industrial-scale manufacture of uranium-erbium oxide fuel for RBMK reactors, the use of erbium oxide is of special interest.

From the point of erbium oxide impact on thermal conductivity and thermal creep of uranium oxide fuel [47] (refer to Fig. 11 and Fig.12) the following conclusions have been done:

- There is a tendency to increase the rate of stationary creep with increasing the content of oxide of erbium;

- In the temperature range from 650–1650 °K the thermal conductivity of the samples decreases with increasing content of erbium in solid solution $(U,Er)O_2$;

- It is supposed to use uranium-erbium fuel with erbium content about 1 %mass Er_2O_3.

The technology of uranium-erbium fuel was tested at an RBMK reactor, however, in order to create a technical design project, it is required to conduct reactor tests in accordance with federal norms and regulations in the field of nuclear energy use.

The first stage of technical design is testing in a research reactor for obtaining experimental data on:

- Thermal mechanical behavior;
- Gas release and internal pressure in the fuel rod;
- Structural changes, radiation stimulated densification and swelling of fuel.

The results of the reactor tests verify the design codes and expand the data bank on the properties of uranium-erbium fuel with erbium content up to 1%.

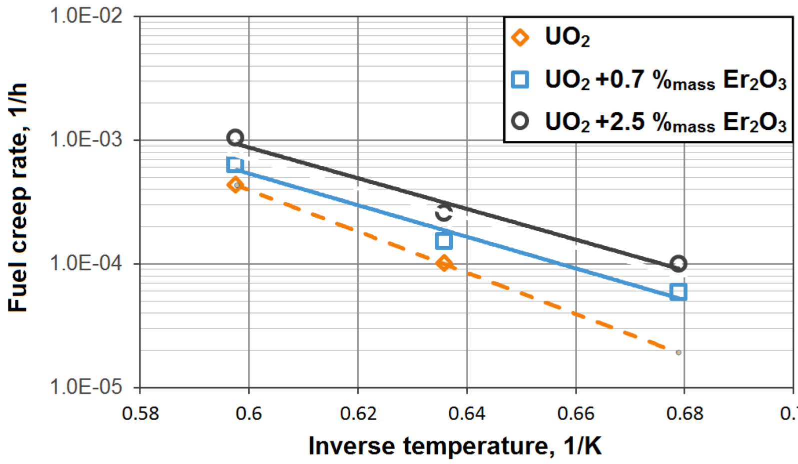

FIG. 11. Fuel creep rate versus the inverse temperature at a stress of 20 MPa for the samples with different content of the oxide of erbium, Reproduced courtesy of P.G. Demyanov et al., [47].

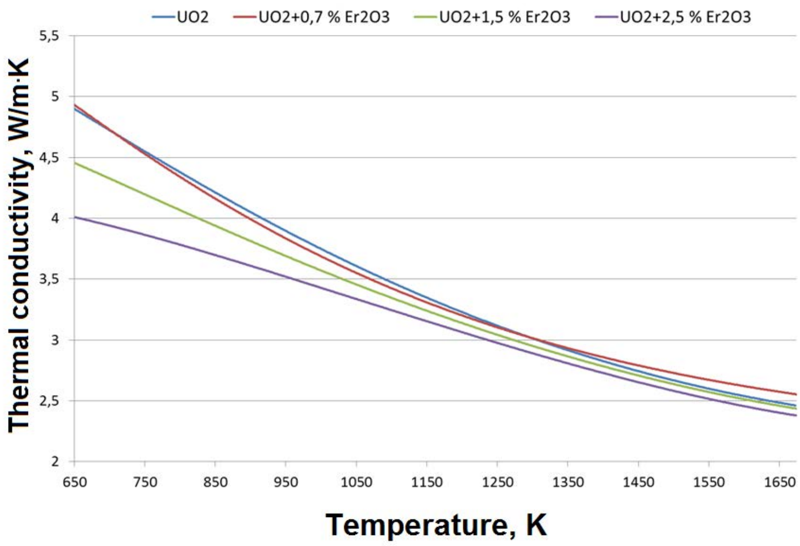

FIG.12. Thermal conductivities of UO_2 and Er-doped burnable absorbers as function of temperature, Reproduced courtesy of P.G. Demyanov et al., [47].

7. CONCLUSIONS

There are two main drivers for changing the current effective limit of 5% on enrichment in LWRs. The first is the desire to obtain the economic benefits deriving from enhanced burnup of the fuel, which can increase cycle length, decrease volume of high-level waste and in general reduce costs associated throughout the fuel cycle. The second driver derives from the March 2011 accident at Fukushima Daiichi nuclear power plant and the realization that there are alternative fuel designs that would be more tolerant of accident conditions. These designs are many and varied, but many of these proposals lead to an increase in parasitic neutron absorption in operation. Increasing the enrichment of the fuel has been shown to be an effective remedy to this effect, allowing designs with minimal changes to assembly geometry.

The 5% limit on enrichment derives from early studies carried out for the first commercial reactors. It was chosen to provide a large operational safety margin to the then current practice of burnups in the range of 20–30 MW·d/kgU, annual cycles with 60-day refuelling outages and enrichments around 3%. There is no firm technical reason for this enrichment limit.

Many LWRs intend to utilize fuel cycles up to 24 months with burnups over 50 MW·d/kgU. To achieve these limits, enrichments have now reached the notional limit and 4.95% enrichments are often used, there remains a small margin to the 5% limit to allow for some manufacturing tolerances.

Further increases in burnup are possible with core designs utilising higher enrichments and there are still benefits to be obtained with lower waste arising, improved utilisation of uranium and increased operation of a nuclear power plant through longer cycle lengths. VVER operators are exploring an increase to 6–7% enrichment, which seems to be technically possible.

The implementation of ATF will generally require increased enrichment to compensate for major design changes, for example using iron based cladding materials. Enrichments of around

10% have been explored, coupled with various burnable poison additives to control reactivity. However, some designs exist which would require up to 20% enrichment. A practical requirement for ATF fuel is that it at least matches the performance of current fuel.

The current limit of 5% enrichment has been used in many safety studies as a limit; criticality studies have generally used this limit to bound the calculations. This has meant that many design decisions in nuclear facilities have been made on this basis. Only in one area is there a statutory limit of 5% and that is in the use of transport flasks for UF_6, where the current flask designs are designed against that limit.

The use of higher enrichments is certainly feasible, but there are several areas where significant work is needed to demonstrate safety, or to make design changes that would allow the safe use, of such fuel. The one that seems to be the most pressing is the upgrading of either the safety case or design of transport flasks for the handling of enriched uranium hexafluoride or uranium metal. However, significant safety assessments are required throughout the fuel cycle from the enrichment plant to the manufacturer, use in a reactor, storage of the spent fuel pool, in long term storage and ultimate disposal. Overall, transport is considered as a key issue; it is also noted that progress toward facilitating this issue has been made as demonstrated by Russian investigators.

The papers presented at the technical meetings and the associated discussions suggest that the removal of the 5% limit is justified as long as all existing safety requirements are met through the safety case processes of calculation, verification, validation and uncertainty quantification. It is considered that use of up to 6% enrichment could be justified without any serious disruption to current practice, indeed many facilities are already licensed to this limit. Lead test assemblies are to be loaded in a commercial NPP at 7% enrichment and meeting the safety requirements is being demonstrated. There was concern for PWR designs that there was a potential for criticality under accident conditions for a single assembly at above 7% enrichment under current operating procedures. Issues such as this could be mitigated through design of a transport flask or by other technical means.

There were no technical requirements identified that would prevent the use of higher enrichments, though the extra requirements arising from the justification of a safety case will increase as the enrichment rises.

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The supplementary files for this publication can be found on the publication's individual web page at www.iaea.org/publications. These include:

PAPERS PRESENTED AT THE FIRST TECHNCIAL MEETING IN 2015

- Beyond 5% Enrichment Limit for PWR 17x17 and VVER-440 fuel – Criticality, Inventory and High Burnup Problems, V. Chrapciak, B. Hatala (Slovakia)

- Development of High Burnup Fuel for Next Generation Light Water Reactor (Total Performance of 5 wt.% –10 wt.% Enrichment High Burnup Fuel), K. Hiraiwa, T. Yoshizu, H. Maruyama (Japan)

- Feasibility Study of VVER-400 Fuel with Higher Enrichment, L. Heraltova (Czech Republic)

- Fuel Management and Economic Evaluation of 24-month Fuel Cycles, J. Wei, X. Fu, X. Xu, L. Wang, D. Cai (China)

- The Thinking about LWR Fuel Enrichment Beyond the 5% Limit, J. Zheng (China)

- Preliminary Study on Fuel Enrichment Increase for Accident Tolerant Fuel Concept, Q. Ren, D. Xu, T. Liu, Z. Lu (China)

- Use of Stainless-Steel as Alternative Clad Material in LWRs, K.K. Yadav, R. Kathikeyan, U. Pal (India)

- Preliminary Neutronic Assessment of Iron Based Alloy Fuel Cladding, A. Abe, T. Caluccio, P. Piovezan, C. Goiovedi, M.R. Martins (Brazil)

- Advanced LWR Fuel Designs with Significant Potential for Power Uprate, K. Shirvan (United States of America)

- The Impact of Increased Enrichment of VVER-400 Fuel on Licensing of Storage Capacities and Transport Means, J. Vaclav (Slovak Republic)

- Challenges and Drivers to High Enrichment, J. Wright (Sweden)

- Lightbridge Metallic Fuel, J. Malone (United States of America)

- Possible Characteristics of Fuel Assembly with Fuel Enrichment Beyond 5% Limit on the basis of TSV-A, Y. Kovbasenko, Y. Bilodid (Ukraine)

- Outcome of the World Nuclear Association Discussion on 'Beyond 5%', S. Tarlton (World Nuclear Association)

PAPERS PRESENTED AT THE SECOND TECHNCIAL MEETING IN 2018

- Prospects and Challenges of Using More Than 5 Percent Enriched Uranium Fuel at NPPs with VVER-1000 and VVER-1200 Reactors, Y. Shestakov, V. Orlov, A. Semenovykh (Russian Federation)

- Perspectives of HANHIKIVI-1 Nuclear Fuel Enrichment Beyond the 5% Limit, J. Kumpula (Finland)

- Perspectives for Implementation of VVER Nuclear Fuel Enriched Above 5%, A. Shaulskaya (Russian Federation)

- PJSC NCCP Technical Availability for Manufacture of VVER and PWR Nuclear Fuel with Enrichment in U-235 up to 7%, E. Vykhodtsev (Russian Federation)

- Fuel of High Enrichment – Directions of Administrative and Technical Preparation of the Entire Process Line at PJSC MSZ for Pilot and Series Manufacture, E. Bagdatyeva (Russian Federation)

- Nuclear Screening of Potential Candidate for Accident Tolerant Fuel, A. Abe, C. Giovedi, M. Martins (Brazil)

- Enrichment Problems in Accident Tolerant Fuel Concept, E.K. Kosourov, A.M. Pavlovichev, Yu.M. Semchenkov, A.I. Shcherenko, A.V. Ugryumov (Russian Federation)

- Experimental Studies of Uranium-Erbium Fuel, P.G. Demyanov, V.V. Novikov, V.I. Kuznetsov, E.N. Mikheev (Russian Federation)

- Economic Evaluation of a Mobile Nuclear Power Plant Using Uranium Over 5% Enrichment, S.R.C. Mello, A.R.M. Pinheiro, L.O. Freire, D.A. Andrade (Brazil)

- Design Optimization Studies and Evaluation of Safety Parameters for an LWR Core with Stainless Steel as Clad Material, K.K. Yadav, U. Pal, R. Karthikeyan (India)

- Light Water Reactor Fuel of above 5wt.% Enrichment to Reduce Production of Trans-Uranium, S. Wada, K. Hiraiwa, K. Yoshioka, R. Kimura, S. Sakurai, T. Sugita (Japan)

- Adaptability of RTOP Fuel Performance Code for Thermo-Mechanical 2D and 3D Simulations of Fuel Rods with Various Burnable Neutron Absorbers, V.V. Likhanskii, A.A. Sorokin, M.G. Chernetskiy, O.V. Vikhivskaya, K.E. Ulybyshev, D.V. Ivonin (Russian Federation)

- Safety Concerns Associated with Fuel Designs Having Enrichment Higher Than 5%: A Regulatory Perspective, S. Kumar (India)

- The Impact of Increased Enrichment of VVER-440 Fuel on Licensing of NPP Operation, Storage Capacities and Transport Means, J. Vaclav, M. Melicharek (Slovak Republic)

ABBREVIATIONS

ATF	Accident Tolerance Fuel.
BWR	Boiling water reactor.
CPR	Chinese pressurized reactor.
EFPD	Effective full power day.
EPR	European pressurized reactor.
HALEU	High Assay Low Enriched Uranium.
HCF	Helical cruciform-shaped fuel.
IXAF	Internally and externally cooled annular fuel.
LWR	Light water reactor.
MOX	Mixed oxide.
NPP	Nuclear power plant.
PWR	Pressurized water reactor.
RBMK	High Power Channel-Type Reactor.
RU	Reprocessed (or recycled) uranium.
SFP	Spent fuel pool.
SMR	Small modular reactor.
VVER	Water-water energy reactor (WWER).

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