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Status Report on Fuel Safety Implications of Extended Enrichment and High Reactivity/High Suppression Core Designs

A Report from the Working Group on Fuel Safety (WGFS)

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NUCLEAR ENERGY AGENCY
COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

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The Committee focuses primarily on the safety aspects of existing power reactors, other nuclear installations and new power reactors; it also considers the safety implications of scientific and technical developments of future reactor technologies and designs. Further, the scope for the Committee includes human and organisational research activities and technical developments that affect nuclear safety.

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List of abbreviations and acronyms

ATF	Accident tolerant fuel
BfS	Bundesamt für Strahlenschutz (Federal Office for Radiation Protection, Germany)
BWR	Boiling water reactor
CAPS	CSNI Activity Proposal Sheet
CSN	Spanish Nuclear Safety Council
CSNI	Committee on the Safety of Nuclear Installations (NEA)
FAST	Fuel Analysis under Steady-state and Transients
FIDES-II	Second Framework for Irradiation Experiments (NEA joint project)
FRG	Fission gas release
GRS	Gesellschaft für Anlagen- und Reaktorsicherheit (Germany)
IAEA	International Atomic Energy Agency
ICSBEP	International Criticality Safety Benchmark Evaluation Project
IFA	Instrumented fuel assembly
IFBA	Integral fuel burnable absorbers
IRSN	L'Institut de radioprotection et de sûreté nucléaire (Institute for Radiation Protection and Nuclear Safety, France)
IVNET	Innovative and Viable Nuclear Energy Technology
JAEA	Japan Atomic Energy Agency
JAERI	Japan Atomic Energy Research Institute
JMTR	Japan Materials Testing Reactor
KINS	Korea Institute of Nuclear Safety
KUCA	Kyoto University Critical Assembly
LAGER	Laser Ablation of Gadolinium Evolution Radially
LOCA	Loss-of-coolant accident
LWR	Light water reactor
MTC	Moderator temperature coefficient
NEA	Nuclear Energy Agency
NRA	Japan Nuclear Regulation Authority
NRC	Nuclear Regulatory Commission (United States)
OECD	Organisation for Economic Co-operation and Development
ONR	Office for Nuclear Regulation (United Kingdom)

ORNL	Oak Ridge National Laboratory (United States)
PIE	Post irradiation examinations
PWR	Pressurised water reactor
SFR	Sodium cooled fast reactors
SSM	Swedish Radiation Safety Authority
SÚJB	State Office for Nuclear Safety (Czechia)
UO_2	Uranium dioxide
VVER	Water-cooled water moderated energy reactor (Russian design)
WGFS	Working Group on Fuel Safety (NEA)

Executive summary

Rising international interest in increasing fuel burnup limits and fuel cycle length may require fuel enrichment above 5% and high reactivity/high suppression core designs. Based on this interest in extended fuel enrichment (5-8%), the Nuclear Energy Agency (NEA) Working Group on Fuel Safety (WGFS): 1) collected information on fuel enrichment limits for uranium dioxide fuel in light water reactors in NEA member countries through a questionnaire; 2) consolidated the information in a status report 3) utilised WGFS expertise to evaluate fuel safety implications of extended enrichment fuel and high reactivity/high suppression core designs; and 4) provided recommendations for collaborative analytical and experimental research on this topic. The task group focused on in-reactor behaviour of high reactivity/high suppression core designs. Issues related to the front-end (e.g. fuel fabrication) and back-end (e.g. spent fuel storage) of the fuel cycle are outside the scope of this report.

The questionnaire responses indicated that while there is limited operating experience in NEA member countries with fuel enrichment of 5-8%, extensive experimental data from research and test reactors could be used to validate neutronic and fuel performance codes and methods for extended enrichment fuel. From these responses, the group concluded that there are no major experimental gaps related to extended enrichment fuel behaviour.

The group also concluded that few design limits are directly related to fuel enrichment. However, fuel enrichment can impact core radial and axial power profiles, as well as the radial power profile across the fuel pellet. These changes may make it challenging to meet existing design limits. Furthermore, there is limited experience with extended enrichment fuel or gadolinia doped fuel behaviour in design basis accident conditions.

Therefore, the group recommends performing experiments on extended enrichment or gadolinia-doped fuel, particularly for reactivity-initiated accident or power ramp conditions. Such experiments could be proposed under the NEA Second Framework for Irradiation Experiments (FIDES-II) joint project. The group also recommends launching two code benchmark activities. The first benchmark activity would focus on fuel thermal mechanical behaviour and may involve tests on extended enrichment fuel material. This activity could be conducted under the auspices of the WGFS. The second benchmark activity would focus on the neutronic behaviour of a high reactivity/high suppression core with extended enrichment fuel. Similar efforts are currently underway in the United States; these core designs could potentially serve as the basis of the recommended benchmark activity. This activity should be conducted by an NEA group focused on neutronics or core design. Together, these recommended activities would verify that extended enrichment fuel is well understood and that existing design limits are still valid for fuel of 5-8% enrichment. The expert opinions collected in this report should be revisited during the activities to determine if they can be confirmed or not by the work.

1. Introduction

1.1. Background, objectives and scope

An international interest in operating fuel to higher burnup may require initial fuel enrichment above 5 weight percent (%).[1] It is expected that regulatory authorities will soon be faced with applications for fuel designs utilising enrichments of 5-8%. To prepare for these expected applications, a Committee on the Safety of Nuclear Installations (CSNI) activity was proposed for a "Status Report on Fuel Safety Implications of Extended Enrichment and High Reactivity/High Suppression Core Designs." The objective of this activity is to consolidate information regarding fuel enrichment limits of uranium dioxide (UO_2) fuel for light water reactors (LWRs) in NEA member countries and utilise the expertise of the NEA Working Group on Fuel Safety (WGFS)[2] to evaluate possible fuel safety implications of extended enrichment (5-8%) and high reactivity/high suppression core designs.[3]

The scope of this report is limited to UO_2 fuel in LWRs. However, there are two areas outside of this scope that must be acknowledged. First, research reactors in many countries today operate with fuel at enrichment levels above 5%. This experience demonstrates, to some extent, the technical feasibility of extended enrichment operation and supporting analysis requirements. At the same time, it must be recognised that the size, complexity and design of research reactors is significantly different than large LWRs (i.e. significantly lower pressures and temperatures, limited use of poisons), often allowing for a more simplified safety basis. In addition, in most cases the fuel used in research reactors is not UO_2. This report does not attempt to capture operating experience of highly enriched, non-UO_2 fuel in research reactors and the high enrichment UO_2 fuel used in sodium cooled fast reactors (SFR) is not discussed, for the fuel in SFRs has different loads (e.g. high linear heat generation rates, higher temperatures, fast neutron spectrum) and the fuel behaviour is characterised by specific phenomena (e.g. very high fission gas release, fuel restructuring, redistribution of fuel constituents). At the same time, there are examples of research and test reactors that utilise UO_2 fuel enriched slightly above 5% (e.g. the Halden Reactor in Norway) and examples of non-resident test articles of UO_2 fuel that have been irradiated in research reactors where conditions were established to simulate LWR conditions (e.g. loops within the Halden Reactor in Norway). In some cases, this experience may be useful to

[1] Unless otherwise noted, enrichments are presented in terms of weight percent. For simplicity, weight percent enrichment will be indicated using the % sign.

[2] The NEA WGFS was established to advance the understanding of fuel safety issues by assessing the technical basis for current safety criteria and their applicability to high burn-up and to new fuel designs and materials, including fuels with increased enrichment.

[3] The proposed activity, presented in a CSNI Activity Proposal Sheet (CAPS), includes high reactivity/high suppression/extended operating cycle core designs because of the relevance of these core designs to operation with extended enrichment. One of the biggest implications of operating with enrichments above 5% will be that the fuel will have an increased initial reactivity excess, which will need to be absorbed (e.g. by burnable poisons or other means). Some fuel designs, such as MOX, or countries operating on 24-month cycles, may already be dealing with core designs that require substantial power suppression early in life and this operating experience has relevance for operating with extended enrichment.

understand the performance of extended enrichment fuel and therefore will be discussed within the report. Second, there are ongoing research programmes in several countries related to the development of new fuel forms with extended enrichment for advanced reactor designs. This research may have relevance to the development of a safety basis for UO_2 fuel in LWRs. However, the design constraints and historical safety basis of large LWRs may prove far more restrictive than for new advanced reactors. This report does not attempt to capture ongoing research on extended enrichment fuel designs for advanced reactor designs, since the operational conditions in advanced reactors could be different compared to LWRs.

This report focuses on in-reactor fuel behaviour. Issues related to the front-end (e.g. fuel fabrication) and back-end (e.g. spent fuel storage) of the fuel cycle are not covered in this report. Such issues (e.g. criticality safety) are important and could be addressed by other NEA working groups.

While extended enrichment may be paired with increased fuel burnup limits in some member countries, this report does not attempt to evaluate the fuel safety implications of increasing fuel burnup limits. The fuel safety implications of increasing fuel burnup are considered as an important issue. They were discussed in the NEA report *Very High Burn-ups in Light Water Reactors* (NEA, 2006). Based on the latest knowledge and findings, the fuel safety implications of increasing fuel burnup should be addressed again, in another activity.

1.2. Procedure and organisation of the report

This report first presents a summary of international experience related to extended enrichment (Chapter 2). The summary follows the format and scope of a questionnaire used to collect information from members of the activity task group. The questionnaire can be found in Appendix A. The report then provides a summary of the evaluation of possible fuel safety implications of extended enrichment (5-8%) and high reactivity/high suppression core designs (Chapter 3). The evaluation was facilitated by a comprehensive table of fuel performance criteria utilised in the NEA *CSNI Technical Opinion Paper No. 19: Applicability of Nuclear Fuel Safety Criteria to Accident-Tolerant Fuel Designs* (NEA, 2022). The evaluation table can be found in Appendix B. The evaluation captures the qualitative determination of the extent of the public database and data gaps as well as members' consideration of where there are opportunities for collaborative research. Finally, the report presents conclusions and recommendations from the task group (Chapter 4).

2. Summary of international experience related to extended enrichment

To begin the activity, a questionnaire was developed to collect information regarding fuel enrichment limits and high reactivity/high suppression core designs in NEA member countries. The questionnaire was designed to capture experience generally with core designs that require high power suppression as a way to gain insight into one of the fuel performance concerns with enrichment above 5%. In other words, the activity will include one of the fuel performance aspects of high enrichment, defined broadly, to gain some insight despite a lack of specific operating or performance experience with extended enrichment. The questionnaire distributed to task group participants is provided in Appendix A and a summary of the replies is provided below.

2.1. Explicit regulatory limits on ^{235}U enrichment

Respondents were asked if there is an explicit regulatory limit on ^{235}U enrichment in their country. If there is a limit, they were asked to specify if the limit is in reference to fuel manufacturing operation, transportation, storage and/or use. All respondents reported that there are no explicit limits on ^{235}U enrichment in their country. However, in most countries existing safety analyses for either manufacturing operation, transportation, storage or use have assumed an enrichment limit of 5%. These analyses are part of the regulatory footprint and would have to be re-performed if ^{235}U enrichment exceeded 5%. Some countries have established performance-based criteria that could be used for such re-analysis. Most notably, in Japan the measures for the following criteria are required to be met for manufacturing fuel with enrichment greater than 5%:

1. Exposing the public to radiation should be avoided.
2. Effective measures (neutron absorber/shielding, mass/shape/volume control, etc.) to keep subcriticality for single- and multi-unit plants should be applied.
3. Detection and mitigation measures against criticality accidents should be secured.

Respondents from Japan also explained with regard to fuel manufacturing operation that when the enrichment exceeds 5%, criticality must be considered and accident scenarios and countermeasures are required. The transportation of fuel enriched beyond 5% may require a licensing issue for new shipping containers. Respondents from Japan also noted that there is no regulatory restriction on storage but domestic facilities, with some exceptions, are designed and licensed for enrichment of 5% or less. There are no regulatory restrictions on use but new fuel storage pools are designed to contain fuel of less than 5% enrichment. (Yamasaki & Unesaki, 2010)

Also notable is the existence of analysis simplifications when enrichment is below 5%. For example, the International Atomic Energy Agency (IAEA) SSR-6 “Regulations for the Safe Transport of Radioactive Material 2018 Edition” (IAEA, 2018) includes an allowance to disregard flooding in criticality assessments for packages containing uranium hexafluoride only, with a maximum uranium enrichment of 5% ^{235}U. Should transport of uranium hexafluoride with an enrichment > 5% be required, this allowance could not be utilised and package flooding would have to be considered in the criticality assessment or else the package would need to show it had multiple high standard water barriers.

2.2. Operating limits and practical restrictions related to enrichment

Respondents were asked if there are established operating limits (e.g. inherent reactivity feedback), or other practical restrictions that effectively limit ^{235}U enrichment in their country. If there are limits, they were asked if the limit was in reference to fuel manufacturing operation, transportation, storage and/or use.

Most countries reported that an enrichment limit of 5% has been integrated within the existing safety analysis or technical specifications in various ways. Increasing the enrichment would require revisiting these safety analyses and technical specifications to evaluate the implications of an increase. Multiple respondents speculated that other operating limits may interact with an enrichment limit, including:

- limits on inherent reactivity feedback operating limits;
- use of burnup credit in fuel storage;
- requirements for negative moderator temperature coefficient during power operation;
- limits on the maximum linear heat generation rate during operation;
- requirements for subcriticality during fuel handling;
- limits on local power peaking;
- burnup limits;
- storage capacity limits.

Respondents from Germany elaborated that the 5% limit is based on a non-proliferation rationale (Federal Office for the Safety of Nuclear Waste Management, 2022).

2.3. Ongoing activity to extend enrichment levels

Respondents were asked if there was active interest in their country to extend the current enrichment limit of LWRs. They were asked if there is any indication that an intended use of high enrichment fuel would be combined with any accident tolerant fuel (ATF) design.

At this time, there are active industry efforts to increase enrichment above 5% for LWR fuel in the United States and for water-water energetic reactor (VVER) designs in Russia. In both cases, the efforts are being pursued in combination with ATF designs. Respondents representing Czechia, France and the United Kingdom speculated there may be interested in the distant future. Respondents from Japan and France reported that national programmes have considered enrichment above 5% in the past, but both ultimately abandoned the idea. Respondents from the United Kingdom reported that enrichment above 5% is being considered for advanced reactor designs.

2.4. Qualification of neutronic calculation tools for enrichment of 5-8%

Respondents were asked if there are any efforts ongoing in their country to qualify existing calculations tools for fuel in the range of 5-8% enrichment.

Most respondents indicated that there were no active efforts to qualify existing tools for fuel in the range of 5-8% enrichment. However, respondents from Japan pointed to a past R&D project. Criticality experiments simulating ^{235}U enrichment of the level of 5-10% were conducted at KUCA (Kyoto University Critical Assembly) by using uranium of both 93% and natural enrichment (Shiroya et. al., 1988; Yamamoto et. al., 2007). Through these criticality experiments, the accuracy of code (Continuous-Energy Monte Carlo and deterministic) and nuclear data library was validated (Yamasaki et al., 2007; Nakajima et. al., 2012). The criticality experiment data were registered in the International Criticality Safety Benchmark Evaluation Project (ICSBEP) database (case name LEU-MET-THERM-005) developed under the framework of the NEA Nuclear Science Committee and distributed by the NEA Data Bank. Respondents from the United States pointed to ongoing efforts to qualify the SCALE code for increased enrichment and high burnup applications. Some lattice physics level studies were performed via SCALE and no unexpected or anomalous trends were found when increased enrichments (up to 8 wt.%) were analysed. Respondents from Russia reported that codes being used for neutronic design calculations for VVERs are certified for the enrichment range up to 6% ^{235}U and that validation was carried out for enrichment up to 6.5%, taking into account the increase in the burnable absorber content. Finally, respondents from Czechia (UJV) noted that research reactors today run with the fuel enriched up to 20% and therefore some calculational tools are qualified for higher enrichments already. They note that qualification up to 8% might need justification, but 5% is not perceived as a "hard limit" for current core design tools.

2.5. Ongoing or completed analysis of extended enrichment operation

Respondents were asked if there was any ongoing or completed analysis on the use of fuel with extended enrichment in their country.

Respondents from Hungary noted that reactor physics calculations were carried out for fast reactor fuel (ALLEGRO gas-cooled fast reactor design, UO_2 fuel with up to 20% enrichment). Respondents from Czechia (UJV) noted that analysis was completed for implementation of a 6-year fuel cycle in the VVER-440 reactor (Heraltova, 2015). Respondents from Japan noted that criticality experiments were conducted at KUCA using erbia-doped fuel and it was evaluated that the erbia-doped fuel with an enrichment of more than 5% could be equivalent in criticality safety to uranium fuel of 5wt% or less (Kuroishi & Yamasaki, 2008). The analysis conditions were a 4-loop pressurised water reactor (PWR) and enrichment of 6% (Innovative and Viable Nuclear Energy Technology [IVNET] Development Project, 2008). An analysis of the benchmark problem of reactor physics for the next generation fuels with a burnup of 70 GWd/t and an enrichment of about 6% was performed (Yamamoto et al., 2002). Criticality experiments were conducted at Toshiba NCA using low gadolinia concentration. In this experiment, no above-5% enrichment fuels were used. The aim of the experiment was to measure the reactivity of low-concentration gadolinia. This low-concentration gadolinia was studied to suppress the initial reactivity of above-5% enrichment fuel. Usually, gadolinia is used to control the change of reactivity by fuel burnup. On the other hand, this low-concentration gadolinia would be used to control criticality safety during the process of fuel fabrication so that above-5% enrichment fuels can be treated in a current fabrication facility without large-scale modification. Although this experiment was conducted for the study of above-5% enrichment fuel, only "under-5%" enrichment fuels (maximum 4.9%) were used (Kikuchi et al., 2009). Respondents from the United States noted that SCALE[4] is currently undergoing development and assessment work for increased enrichments. Four technical reports were issued from NRC-sponsored work at Oak Ridge National Laboratory (ORNL) that investigated the effects of high burnup and extended enrichment fuel. These reports looked at front end transportation canisters (Hall, et al., 2020), ATF (Hall, et al., 2021), PWR (Hall, et al., 2021) and boiling water reactor (BWR) (Cumberland, et al., 2021) fuel assembly designs. The PWR volume looked at a conventional Westinghouse 17x17 design

[4] www.ornl.gov/scale/references

and the BWR volume looked at a convectional GNF-2 10x10 BWR design. Respondents from Russia explained that an analysis of the use of fuel with an enrichment of more than 5% in ^{235}U for the implementation of 24-month fuel cycles of VVER-1000 and VVER-1200 was performed. The impact on the neutronic characteristics of the core, nuclear safety, accumulation and yield of fission products, activity of corrosion products, discharge and release of tritium, and decay heat was studied. Further, an analysis of nuclear safety during fuel fabrication was performed.

2.6. In-pile testing of extended enrichment fuels

Respondents were asked if there was any intention to carry out in-pile tests with extended enrichment fuel samples in their country and/or if there are any data available from past experiments characterising fuel behaviour in the range of 5-8% enrichment.

Respondents from Japan noted that there were some bilateral projects between the Halden Reactor Project and Japanese organisations, including the Japan Atomic Energy Research Institute (JAERI, predecessor to Japan Atomic Energy Agency or JAEA), as Instrumented Fuel Assembly (IFA)-208, 209, 224, 225, in which 5-8% enriched fuels were irradiated in the Halden Boiling Water Reactor (Yanagisawa, et al., 1978; Uchida, et al., 1976; Uchida & Ichikawa, 1975). But achieved burnup levels were low, less than 10 GWd/tU. Also, JAERI performed some in-pile tests at the Japan Materials Testing Reactor (JMTR) in the 1970s (Special Committee on Fuel Safety of Nuclear Safety Research Association, 1978; Special Committee on Fuel Safety of Nuclear Safety Research Association, 1980), but the cladding used for these tests was thinner than that for commercial reactors, and the achieved burnup levels were quite low (<1GWd/tU). Respondents from the United States also referenced data available from Halden, stating they expect historic tests to be sufficient for the validation of their fuel performance code (FAST) and neutronics code (SCALE) for enrichment in the range of 5-8%. Respondents from Russia reported that a series of tests was performed in 2020-2021 at the critical test facility of the Kurchatov Institute to determine the characteristics of compositions containing fuel with ^{235}U enrichment of 6.5% and erbium oxide absorber. The test compositions simulate the uranium-erbium fuel with respect to enrichment, average erbium content and neutron spectrum. Critical tests are planned with uranium-erbium fuel enriched to 6.5% of ^{235}U.

In addition to the JAERI irradiations referenced above, many highly instrumented irradiations with >5% enriched UO_2 fuel were conducted in the Halden Reactor over the years. Typical enrichments for these tests were in the 7-13% range, with fuel rod lengths normally in the 400-600 mm range. Operating conditions were typically 300-600 W/cm, with the obtained burnup range starting from fresh fuel up to beyond 90 MWd/kgU in some cases.[5] These tests were regularly highly instrumented to measure, for example, the fuel centre temperature, fuel rod pressure and fission gas release (FGR), and fuel stack or cladding length change. They will have contributed in-pile data for model evaluation and verification on thermal and mechanical analysis, fission gas release and pellet/clad mechanical interactions for fuel modelling codes used by industry.

For most of these tests, post irradiation examinations (PIE) were also conducted on a selection of test rods. The extent of PIE data available varied somewhat between the experiments, but examinations of note include axial gamma scanning (supplemented with

[5] Note that the higher enrichments used in these tests were needed to compensate for the low neutron flux of the Halden reactor. Thus, the experience with higher enrichment fuel in Halden may not be directly applicable to fuel with enrichment greater than 5% in a commercial light water reactor.

radial gamma scans in a few cases), rod puncturing and FGR analysis and basic ceramography (i.e. light optical microscopy).

It is worth noting that in addition to the bilateral research sponsored by JAERI at Halden mentioned above, the Halden Reactor operated with highly enriched fuel material as its “driver” fuel rods for decades. The Halden Reactor is powered by so called “driver” fuel assemblies that operate in heavy water boiling water reactor conditions. Driver fuel rods are UO_2 and resemble commercial fuel rods in their dimensions except that the active fuel length of a driver fuel rod is approximately one metre. The initial enrichment of driver fuel rods included values between 6 and 10%. Due to the nature of the experiments performed in the Halden Boiling Water Reactor, driver fuel assemblies generally operated at high linear heat rates (Holcombe et al., 2014). The operating experience gained by the Halden driver fuel may provide useful insight into fuel performance considerations for extended enrichment fuel.

2.7. New fuel testing requirements

Respondents were asked if there are any requirements for testing new types of high enrichment fuel including: a) material testing; b) testing during manufacturing; c) testing under conditions as close as possible to operating conditions in a reactor; and/or d) preliminary operation at research facilities.

The responses all generally described a similar situation, which is that there are no specified requirements outlined for testing to qualify high enrichment fuel. Rather, most countries have general requirements that apply to all new fuel types, whether ATF, evolutions of current fuel types or high enrichment fuel. In general, the requirement to demonstrate that fuel performance models reflect as-built performance dictates subsequent testing needs.

Several respondents offered additional discussion on the definition of test requirements for new fuel designs. Respondents from Czechia explained that tests for new fuel types are specified once the regulator receives the application describing what innovations/changes are proposed. They explained that in the past, reference in-pile operation of the same fuel type was required before a new design was approved. Respondents from the United Kingdom said that:

> The general expectation is that fuel must be supported by a robust program of pilot loadings and sufficient relevant operating experience and testing. It is expected that suitable operating limits have been defined which have sufficient safety margin to allow for both uncertainty in manufacturing parameters and transient events. It is expected that licensees have a systematic program of post-irradiation component examination and testing to ensure that the arguments made in the safety case remain valid.

Respondents from the Slovak Republic reported that there are no specific requirements for material testing or preliminary operation at research facilities. However, there are requirements related to testing during manufacturing (covered by the nuclear power plant operator) and testing under conditions as close as possible to operating conditions (included in the requirements of Slovak legislation). Respondents from Russia noted that when using new burnable absorbers, the supervisory body of the Russian Federation requires materials science research. They also note that there are requirements for trial operation of individual fuel assemblies with increased enrichment in operating reactors.

2.8. Relevant operating limits

Respondents were asked about specific operating limits that may be relevant to high reactivity/high suppression/extended operating cycle core designs. The specific operation limits included in the questionnaire were:

- assembly or fuel rod peaking factor;
- limits on burnable poisons;
- moderator temperature coefficient;
- cycle length with respect to instrument calibration (drift);
- shutdown margin.

In response to this question, respondents from France explained that enrichment was an input to neutronic calculations used to demonstrate that, when the reactor core is critical, the inherent reactivity feedbacks are able to guaranty the core stability. The neutronic conception and the safety assessment lead to the following limits:

1. A higher discharge burnup has consequences for fuel in the safety studies (fragmentation, relocation and dispersion during postulated design basis accidents, more frequent and ampler fuel assembly or rod bowing);
2. A higher boron concentration or supplementary rod cluster control assemblies might be required to ensure subcriticality in shutdown states.

Respondents from the United Kingdom also noted generally that all the operation limits identified by the question are considered as part of the reload design and safety case. For either existing or new plants, these parameters are usually limited as part of the safety case to demonstrate fault tolerance with the desired cycle length and load factor, but limits on individual parameters are not prescribed by the regulator.

Finally, respondents from Spain noted generally that there are no explicit requirements in any of the operating limits inquired, except those stated in the respective safety analysis as an input or assumption for the safety analysis. They noted that as a general requirement, it is legally stated that: "The analysis of accidents shall be conducted with safety margins such that the maintenance of the safety functions is guaranteed with due considerations to the uncertainties inherent to the processes involved."

2.9. Assembly or fuel rod peaking factor

Respondents from Czechia, Korea, Japan, the Slovak Republic and Russia reported that there are limits for assembly or fuel rod peaking factor. Respondents from Japan specified that for PWRs, a power peaking factor is evaluated for each reload core (Japan Electric Association Code, 2018). For BWRs, a maximum linear heat generation rate or a minimum critical power ratio is evaluated for each reload core (Japan Electric Association Code, 2018). Respondents from Germany explained that requirements are dependent on the safety assessment of every specific nuclear power plant and core loading. Respondents from Hungary, France, Sweden, the United Kingdom and the United States explained that there are no generic limits defined, though limits for each reload are established to demonstrate that safety criteria for operation and accidents are met.

2.10. Use of burnable poisons

Many countries authorise gadolinia-doped fuel and some respondents cited practical boundaries for gadolinia doping, though there are no established limits. With that said, in some countries fuel suppliers limit the enrichment of the gadolinia-doped rods to prevent them from becoming limiting in the safety analysis.

Respondents from Japan noted that fuel is generally designed to be within a gadolinia concentration of 10% or less. Respondents from Sweden explained that there are limits on different combinations of nodal enrichment and burnable absorber (uranium enrichment in combination with the amount of gadolinia within a rod). In Sweden there are no established maximal limits regarding the amount of burnable poison (number of rods or gadolinia concentration) since having annual cycles reduces the need for large amounts of burnable poison. Respondents from France reported that any limits on burnable poisons are practical and imposed by the code qualifications. Respondents from Hungary reported that VVER-440 fuel assemblies, which have a total of 126 fuel rods, are in use with zero, three or six gadolinia-containing rods and varying levels of enrichment. Examples can be found in (Végh et al., 2015).

Respondents from the United States reported that fuel designs utilising gadolinia and integral fuel burnable absorbers (IFBA) have been approved. As in other countries, fuel vendors may have established limits for the number of rods with gadolinia or burnable poisons, or the concentration of those materials within a fuel rod, in their approved methods. Nevertheless, there are no regulatory limits established that limit the use of burnable poisons in the United States.

2.11. Moderator temperature coefficient (MTC)

All respondents reported that there are limits in place in their country for MTC, specifically that it must be negative. Respondents from the United States elaborated that while there are no explicit limits for MTC, licensees must demonstrate that they meet safety criteria in their operational and accident analyses. NRC staff guidance states that “MTC should be non-positive over the entire fuel cycle when the reactor is at a significant power level” (US NRC, 2007).

2.12. Cycle length and considerations of instrument calibration

Currently, many countries operate with cycle lengths of approximately one year, but most are moving to longer cycle lengths. Respondents from Japan noted a common cycle length to be 13 months and respondents from Hungary reported that cycles lengths have recently increased to 15 months. Respondents from Czechia also reported that cycle lengths are starting to shift to 15-18 months. Respondents from Russia reported that a typical cycle length is already 18 months, specifying that all VVER-1000 plants are operating in the 18-month cycle and that VVER-1200 plants are in a process of transition to the 18-month fuel cycle (to be finished by 2025). Respondents from France note that 900 megawatt electric reactors (3-loops) remain on 12 months cycles; however, for 1 300 and 1 450 megawatt electric reactors (4-loops) the cycle lengths are around 18 months.

There are no explicit limits on cycle length based solely on instrument calibration requirements, but many respondents reported that there are requirements for instrument calibration on established frequencies. Respondents from France added that “most of the mandatory periodic tests can be performed during the cycle (with reactor at zero power if necessary). Hence, this aspect does not limit the cycle length in France." Respondents from Russia said:

> When an 18-month cycle was being introduced, there were some problems with justifying the duration between equipment tests. Regulations strictly required testing some equipment at least every 18 months. Current regulations state that the testing frequency shall be justified in the design. Thus, currently there are no strict requirements in Russian regulations regarding the instrument calibration frequency, system design shall contain all necessary justifications. During transition of the VVER plants to the 18-month cycle, we did not face any limitations originating from the instrumentation calibration and we do not expect such limitations for the 24-month cycle. Of course, implementation of the 24-month fuel cycle will require additional justification of the instrument calibration frequency.

Considering the link between cycle length and enrichment, respondents from Korea noted that "although the cycle length can be technically increased by increasing the fuel enrichment, the maximum cycle length can be determined by the excess reactivity, MTC (Moderator Temperature Coefficient) at the beginning of the cycle. The operating cycle length is currently set according to the periodic inspection, but the amount of the length is not restricted by excess reactivity or MTC."

2.13. Shutdown margin

Respondents reported that there are constraints on shutdown margin, and in many cases these constraints are established in the safety analysis. Respondents from Japan reported that under operational conditions, the shutdown margin must be designed to allow the core to go subcritical even when the single control rod with the highest worth is completely pulled out of the core and cannot be inserted (Japan Electric Association Code, 2018). Respondents from Sweden explained that the constraints are reactor-specific and established in safety analyses, where in controlled shutdown states, subcriticality should be at least 1%. Respondents from France said that shutdown margin must be shown to be sufficiently high in the safety demonstration of accident scenarios, such as a steam line break at zero power. Respondents from Hungary noted that shutdown margin cannot be less than 2% according to the design provided by the fuel supplier. Respondents from the United States explained that while there is no explicit limit on shutdown margin, General Design Criterion 27 requires that the reactor coolant systems have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods (General Design Criteria for Nuclear Power Plants, 2022). Respondents from Germany said that shutdown reactivity for boron injection systems is 5% for PWRs (Nuclear Safety Standards Commission, 2012). They explained that the point in time with the highest reactivity is to be selected. This does not always have to be at the beginning of the cycle (e.g. for BWRs with burnable neutron poisons the reactivity can increase at the beginning before it decreases again) (see Sections 3.2 (3), 3.2 (4) and 3.2 (7) of the Safety Requirements for Nuclear Power Plants (Federal Ministry for the Environment, Nature Conservation and Nuclear Society, 2015a), and Interpretation I-1 3(1) (Federal Ministry for the Environment, Nature Conservation and Nuclear Safety, 2015b).

2.14. Human error in core loading

Respondents were asked if there are requirements in place to specifically address the potential for "human error" in core loading. If so, respondents were asked if the limits relate in any way with the possibility that high reactivity/high suppression/extended operating cycle core designs may be more sensitive to core loading errors.

Respondents from Russia confirmed that the potential for human error in core loading must be addressed in the core loading process (see NP-006-16 of [Scientific and Engineering Centre for Nuclear and Radiation Safety, 2018]), explaining that typically the fuel design justification includes analysis of such an error aimed at demonstrating that the error is detected by monitoring systems and no negative consequences occur. Respondents from Germany explained that misloading is postulated in events on safety levels 2 (Anticipated Operational Occurrence) and 3 (Design Basis Accident) (see Annex 2 of SiAnf events D2-25, D3-17, S2-20 and S3-15 of [Federal Ministry for the Environment, Nature Conservation and Nuclear Society, 2015a]). In this way, human errors can be intercepted. Otherwise SiAnf 3.1 (13) (Federal Ministry for the Environment, Nature Conservation and Nuclear Society, 2015a) is still generally valid. All other respondents reported that there are no requirements specifically addressing the potential of human error in core loading. However, the potential of an error in core loading could be prevented by ensuring limits are met in reactor physics tests before operating (e.g. ANSI/ANS 19.6.1 "Reload Startup Physics Tests for Pressurized Water Reactor") as well as adherence to operating procedures intended to prevent human error in core loading.

2.15. Mixed cores and transition cores

Respondents were asked if there are particular restrictions or operating requirements for mixed cores and transition cores in their country.

Respondents reported that there are no specific restrictions or operating requirements for mixed cores and transition cores. Respondents explained that the requirements for mixed cores are the same as for homogeneous cores; the safety criteria must be fulfilled. The differences in flowrate, neutronic power (if any), etc. must be taken into account. Respondents from France noted that for transition cores, besides the "classical" safety demonstration, the main issue is often related to pellet-cladding interaction risk, as the burnup pattern within the core is often quite different from the equilibrium cycle (for which the demonstration related to the absence of pellet-cladding interaction risk is performed). Respondents from the Slovak Republic noted that if operating requirements are needed, they are established as necessary in co-operation with the fuel producer. Respondents from Spain noted that mixed cores are allowed but must be licensed specifically by the regulatory body.

3. Summary of safety implications evaluation

One objective of this activity was to utilise the expertise of the WGFS to evaluate possible fuel safety implications of extended enrichment (5-8%) and high reactivity/high suppression core designs. The evaluation was facilitated by a comprehensive table of fuel performance criteria utilised in the *NEA CSNI Technical Opinion Paper No. 19 - Applicability of Nuclear Fuel Safety Criteria to Accident-Tolerant Fuel Designs* (NEA, 2022). The evaluation table can be found in Appendix B. The key findings of the evaluation are captured below.

3.1. Impacts of extended enrichment on design limits

In the evaluation of extended enrichment, there were few instances where enrichment itself was expected to have a clear impact on an existing design limit. However, increased enrichment and additional burnable absorbers may change core-wide and individual fuel rod axial and radial power profiles and the excess reactivity in the core. These changes could impact the margin to applicable safety limits.

As one example, increasing the enrichment may impact fuel rod power peaking factors assumed in plant safety analyses, particularly for high burnup rods. Under existing enrichment limits, there is little residual fissile material in the fuel at high burnup; increasing the enrichment may allow these higher burnup rods to reach the failure limits in certain accident scenarios (e.g. during a reactivity-initiated accident).

As a second example, changing the enrichment would impact the fuel rod radial power profile. This would impact various fuel performance characteristics, including the fuel rod radial temperature profile, fission gas production and release, and fuel-cladding gap thermal conductivity. Furthermore, the addition of gadolinia to UO_2 fuel reduces its thermal conductivity,[6] while neutron absorption by zirconium diboride added to fuel produces helium gas that impacts the rod internal pressure. All these effects could impact the margins to fuel rod safety criteria (e.g. by reducing the margin to fuel centerline melt or by challenging rod internal pressure limits).

Overall, the task members noted that the impact of enrichment on the various fuel safety criteria may be minor, but the evaluation concludes that design-specific analysis and confirmation of the safety basis for various limits should be performed. In some cases, this may mean analysis to quantify the impact of enrichment and the analysis will show that the impact is insignificant. In these cases, the existing design limits would be applicable in the case of extended enrichment. In other cases, an analysis may reveal the impact of enrichment is not insignificant, though the impact could be bounded by the basis for the existing limits. In these cases, too, the existing analytical limit (e.g. deposited enthalpy limits during reactivity accidents, shutdown margin requirements) would be applicable in the case of extended enrichment. Even if there are many instances where the conclusion of such an analysis could be seen to confirm the adequacy of an existing analytical limit, the analysis must still be performed to reach that conclusion.

[6] At the same time, gadolinia rods produce less power than rods without gadolinia. In current core designs, the power reduction in gadolinia rods more than compensates for the reduced thermal conductivity, such that gadolinia rods are typically not limiting in terms of meeting fuel rod safety criteria. However, this may not be true for future high suppression core designs with increased enrichment fuel.

3.2. Opportunities for collaborative research

During the course of this review, the task group did not identify any major experimental data gaps for validation of the computer codes. The task group noted that numerous experiments have been conducted to study the neutronic and thermal mechanical behaviour of UO_2 fuel with enrichment above 5% and of gadolinia-doped fuel. Much of this data is available to NEA member countries. There is also extensive operating experience with fuel containing burnable absorbers such as gadolinia or zirconium diboride; however, much of this operating experience is not publicly available. It would be beneficial to provide a more thorough description of publicly available experiments involving enrichment above 5% or gadolinia-doped fuel. This task could be accomplished as part of a future NEA collaborative research activity.

Additionally, the task group has noted that there may be some value in performing additional experiments on extended enrichment fuel or gadolinia-doped fuel, particularly for high burnup fuel under design basis accident conditions. The group noted that two NEA joint projects are currently studying the behaviour of gadolinia-doped fuel during loss-of-coolant accident (LOCA) conditions: the Studsvik Cladding Integrity Project and the LOCA-MIR Joint Experimental Programme under the Framework for Irradiation Experiments. Furthermore, the proposed Laser Ablation of Gadolinium Evolution Radially (LAGER) project would look at the behaviour of gadolinia-doped fuel early in life to validate neutronics and fuel performance codes. Additional experiments could be proposed to study the behaviour of gadolinia-doped fuel or standard UO_2 fuel with enrichment greater than 5% in power ramp or reactivity-initiated accident conditions, especially for fuel at high burnup or with new cladding types. The SPARE project is currently harvesting material from the Halden Reactor and could potentially provide higher enrichment material for power ramp or design basis accident testing.

There are also opportunities to conduct neutronic code-to-code benchmark activities to assess the impact of enrichment on parameters like reactivity coefficients, shutdown margin, power peaking factors or decay heat. Such activities would require some design-specific information, which may be difficult to obtain from fuel vendors because of proprietary information concerns. However, there have been some recent activities in the United States related to high burnup and extended enrichment core designs that could serve as a starting point for collaborative research efforts (Hall, et al., 2020; Hall, et al., 2021; Hall, et al., 2021; Cumberland, et al., 2021; Capps, et al., 2021).

Further benchmarks could be conducted to assess the impact of enrichment on fuel thermal mechanical behaviour, including its effects on the radial power profile, fuel temperature profile and fission gas release. Adverse impacts on any of these parameters could reduce the margin to fuel safety limits (e.g. rod internal pressure limits, restrictions on fuel centerline melting).

4. Conclusions and recommendations

4.1. Conclusions

This report discussed past experience with extended enrichment and high reactivity/high suppression core designs in NEA member counties, as well as past experimental programmes dedicated to the behaviour of uranium dioxide fuel with enrichment above 5%. While there is little operating experience with extended enrichment fuel in commercial nuclear power plants, there have been several experimental campaigns that focused on the behaviour of extended enrichment uranium dioxide fuels. Furthermore, the Halden Boiling Water Reactor operated for years with driver fuel above 5% enrichment. Thus, there is some data available to qualify fuel performance and neutronics codes used to perform safety analysis. Based on this experience, the task group did not identify any major experimental gaps related to extended enrichment fuel or high reactivity/high suppression core designs; however, several potential areas of research have been identified, as discussed below.

The report also provided an evaluation of the fuel safety implications of extended enrichment (5-8%) fuel and high reactivity/high suppression core designs. The task group noted that there are few instances where enrichment has a direct impact and fuel safety limits; however, enrichment and high reactivity/high suppression core designs will impact core-wide axial and radial power profiles, which may pose a challenge to meeting existing limits. Furthermore, enrichment will impact the fuel radial power profile, which will in turn affect the pellet temperature profile, fission gas release and various other parameters that may impact the margin to design limits under normal and accident conditions.

4.2. Recommendations

The task group identified several potential areas for collaborative research. Among those areas, the following research activities should be prioritised. The expert opinions collected within this report should be revisited during the activities to determine whether they can be confirmed by the work.

1. Given the limited test data on gadolinia-doped fuel and uranium dioxide fuel between 5-8% enrichment, the group recommends performing additional experiments on gadolinia-doped and extended enrichment fuel under design basis accident conditions, especially for fuel at high burnup or with new cladding types. In particular, the group recommends performing power ramp or reactivity-initiated accident (RIA) tests to confirm that existing design limits are applicable to these fuel types. Such experiments could be proposed as part of the Second Framework for Irradiation Experiments (FIDES-II) NEA joint project. Fuel rods from lead test assemblies irradiated in commercial nuclear power plants in NEA member states or fuel from the Halden Boiling Water Reactor could potentially be used in these power ramp or RIA tests.
2. Further, the group recommends performing two benchmark studies regarding extended enrichment fuel.
    a. The first benchmark would focus on fuel performance calculations of a single rod with enrichment of 5-8%. The goal of the benchmark is to evaluate the impact of enrichment on significant figures such as fission gas release and fuel centerline temperature. The benchmark could be based on

existing experimental data, or it could be a purely computational exercise designed to evaluate the impact of enrichment on predictions of fuel performance codes used in NEA member countries. This benchmark could be conducted as a WGFS activity.

b. The second benchmark would focus on neutronic calculations for a high reactivity/high suppression core design with fuel enrichment above 5%. The goal of the benchmark would be to evaluate important core design parameters, including power peaking factors, shutdown margins and reactivity coefficients. As mentioned, there are related efforts underway in some NEA member countries (e.g. the United States); however, it would be beneficial to establish a benchmark problem to evaluate neutronics codes used by the NEA member countries and to identify potential challenges associated with meeting existing design limits for high reactivity/high suppression core designs with enrichment above 5%.

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Appendix A: Questionnaire in preparation for a status report on fuel safety implications of extended enrichment and high reactivity/high suppression core designs

Background:

The objective of this NEA Working Group on Fuel Safety (WGFS) activity is to consolidate information regarding fuel enrichment limits of UO_2 fuel for light water reactors (LWRs) in NEA member countries and utilise the expertise of the WGFS to evaluate possible fuel safety implications of extended enrichment (5-8%) and high reactivity/high suppression core designs. *The questionnaire below supports the first step: to collect information regarding fuel enrichment limits and high reactivity/high suppression core designs in NEA member countries.*

Discussion:

The CAPS includes high reactivity/high suppression/extended operating cycle core designs because of the relevancy of these core designs to operation with extended enrichment. One of the biggest implications of operating with enrichments above 5% will be that the fuel will have an increased initial reactivity excess, which will obviously need to be absorbed (e.g. by burnable poisons or other means). In fact, some fuel designs, such as MOX, or countries operating on 24-month cycles, may already be dealing with core designs that require substantial power suppression early in life. This CAPS will capture experience generally with core designs that require high power suppression as a way to gain insight into one of the fuel performance concerns with enrichment above 5%. In other words, the CAPS will include one of the fuel performance aspects of high enrichment, defined broadly, to gain some insight despite a lack of specific operating or performance experience with high enrichment.

Questionnaire:

Question	Response
1. Is there an explicit limit on ^{235}U enrichment established by a regulatory limit in your country? If so, is the limit in reference to fuel manufacturing operation, transportation, storage and/or use?	
2. Are there established operating limits (e.g. inherent reactivity feedback), or other practical restrictions, which effectively limit ^{235}U enrichment in your country? If so, is the limit in reference to fuel manufacturing operation, transportation, storage and/or use?	
3. Is there interest in your country to extend the current limit of LWRs? If so, is there any indication that an intended use of high enrichment fuel would be combined with any accident tolerant fuel (ATF) design?	
4. Are there any efforts ongoing in your country to qualify existing calculations tools for fuel in the range of 5-8% enrichment? If so, can you state the level of qualification or any determination of where enhancements would be needed?	
5. Is there any ongoing or completed analysis on the use of fuel with extended enrichment in your country? If yes, please specify the reactor type, fuel enrichment and analysed scenarios.	

6. Is there any intention to carry out in-pile tests with extended enrichment fuel samples in your country? Is there any data available from past experiments characterising fuel behaviour in the range of 5-8% enrichment?	
7. Are there any requirements for testing new types of high enrichment fuel including: a. material testing b. testing during manufacturing c. testing under conditions as close as possible to operating conditions in a reactor d. preliminary operation at research facilities	

As discussed above, the implications of fuel with 5-8% ^{235}U enrichment shall be considered broadly in this status report. The questions below focus on high reactivity/high suppression/extended operating cycle core designs.

Question	
8. Are there established limits, or any practical constraints, in your country for included, but not limited to:	
a. Assembly or fuel rod peaking factor	
b. Use of burnable poisons	
c. Moderator temperature coefficient	
d. Cycle length with respect to instrument calibration (drift)	
e. Shutdown margin	
9. Are there requirements in place to specifically address the potential for "human error" in core loading? If so, do these limits relate in any way with the possibility that high reactivity/high suppression/extended operating cycle core designs may be more sensitive to core loading errors?	
10. Are there particular restrictions or operating requirements for mixed cores and transition cores in your country?	

Appendix B: Safety implications evaluation table

The safety implications evaluation table for extended enrichment fuel and high reactivity/high suppression core designs can be found at: www.oecd-nea.org/r2023table. The table is based on the evaluation tables developed for the *NEA CSNI Technical Opinion Paper No. 19 - Applicability of Nuclear Fuel Safety Criteria to Accident-Tolerant Fuel Designs* (NEA, 2022). Note that some of the parameters evaluated in the 2022 publication tables were not relevant to fuel enrichment because they focus on cladding phenomena; these boxes have been greyed out in the extended enrichment table.