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Criticality Safety and Fuel Performance Considerations for Enrichment Above 5 Weight Percent in the Uranium Dioxide Fuel Cycle

July 2020

Travis J Zipperer
David V Colameco
Ken J Geelhood

Prepared for the U.S. Nuclear Regulatory Commission
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Abstract

The U.S. Nuclear Regulatory Commission (NRC) is preparing for anticipated licensing applications requesting fuel enrichments of UO_2 ceramic fuels in excess of 5 weight percent (w/o) ^{235}U for commercial light-water reactors (LWRs). PNNL has been tasked with providing technical assistance to the NRC related to the review and approval of requests for increases above the current 5 w/o ^{235}U enrichment limit. This report will provide the agency with technical assistance to enhance the staff's knowledge base and to identify developmental needs required to support the licensing of greater than 5 w/o UO_2 ceramic fuels.

This report provides background on increasing enrichment of UO_2 LWR fuel beyond 5 w/o ^{235}U. The primary concern of increasing enrichment is the impacts to criticality safety. This report focuses on criticality safety as it relates to enrichment, conversion of uranium hexafluoride to UO_2, fuel assembly fabrication, transportation of fresh/spent fuel assemblies, reactor operations, and storage of fresh/spent fuel assemblies. The report provides a general summary of each of the steps in the UO_2 fuel cycle starting from transportation of enriched UF_6 through dry cask storage of spent fuel assemblies. This report also discusses other impacts of increased enrichment during reactor operation and the surrounding fuel cycle activities. Current needs regarding both data and analytical codes to support the increase above 5 w/o ^{235}U are identified and discussed.

Acronyms and Abbreviations

2D	two-dimensional
3D	three-dimensional
ANSI	American National Standards Institute
AOA	area of applicability
ATF	accident tolerant fuel
BWR	boiling-water reactor
CFR	Code of Federal Regulations
DOE	United States Department of Energy
DOT	United States Department of Transportation
Er_2O_3	erbium oxide
FPP	French Fission Production Programme
GDP	gaseous diffusion plant
Gd_2O_3	gadolinium oxide
GWd/MTU	giga-watt day per metric ton uranium
IAEA	International Atomic Energy Agency
ICSBEP	International Criticality Safety Benchmark Evaluation Project
IFBA	integral fuel burnable absorber
IRPhE	International Reactor Physics Experiment Evaluation
HEU	highly enriched uranium
H_2O	water
HF	hydrogen fluoride
HTC	*Haut Taux de Combustion* (French)
IEU	intermediate enriched uranium
ISG	Interim Staff Guidance
K	Kelvin
LEU	low enriched uranium
LHGR	linear heat generation rate
LOCA	Loss of Coolant Accident
LTA	Lead Test Assembly
LUA	Lead Use Assembly
LWR	light-water reactor
MCNP	Monte Carlo N-Particle® Transport Code System
NAC-LWT	NAC International Legal Weight Truck
NEI	Nuclear Energy Institute
NMSS	Nuclear Material Safety and Safeguards
NRC	U.S. Nuclear Regulatory Commission

NUREG	U.S. Nuclear Regulatory Commission technical report designation
NUREG/CR	Contractor-prepared NUREG
PWR	pressurized-water reactor
RAMPAC	Radioactive Material Packaging
RCA	radiochemical assay
SNM	special nuclear material
SRP	Standard Review Plan
TSUNAMI	Tool for Sensitivity and Uncertainty Analysis Methodology
UF_4	uranium tetrafluoride
UF_6	uranium hexafluoride
UO_2	uranium dioxide
UO_3	uranium trioxide
U_3O_8	triuranium octoxide
UO_2F_2	uranyl fluoride
$UO_2(NO_3)_2$	uranyl nitrate
USLSTATS	Upper Subcritical Limit Statistics
w/o	weight percent
ZrB_2	zirconium diboride

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1.0 Introduction

Since 2012, the United States Department of Energy (DOE) has funded projects aiming to develop accident tolerant fuels (ATFs). As these fuels have been studied more, it has become apparent that some of the concepts may require higher enrichment to maintain current plant operations [1]. In response, the U.S. Nuclear Regulatory Commission (NRC) is preparing for anticipated licensing applications requesting fuel enrichments of uranium dioxide (UO_2) ceramic fuels in excess of 5 weight percent (w/o) ^{235}U for commercial light-water reactors (LWRs). PNNL has been tasked with providing technical assistance to the NRC related to the review and approval of requests for increases above the current 5 w/o ^{235}U enrichment limit. This report provides a description of the current state of information available for the UO_2 fuel cycle with enrichments above 5 w/o, but below 20 w/o ^{235}U. Though there are ATF concepts that use new fuel pellet materials, this report will focus on the impacts to increases in enrichment on the fuel cycle for current and ATF LWR designs utilizing UO_2 ceramic fuels. The areas of the fuel cycle that will be covered are enrichment, conversion of uranium hexafluoride (UF_6) to UO_2, transportation, fabrication, reactor operation, and storage. Facilities that downblend highly enriched uranium (HEU) to low enriched uranium (LEU) for use in LWRs will not be discussed, as they are currently licensed to handle enrichments above 20 w/o ^{235}U. Effects of the higher enrichments in operating reactor conditions, reactor design basis accident conditions, spent fuel storage normal, off-normal and accident environments, and fresh and spent fuel transportation normal and hypothetical accident conditions are addressed. Criticality safety is the primary concern with increased enrichment. However, other impacts and needs have been identified as well. A review of the current available benchmarks is also provided.

Evaluation of the economics and regulatory aspects of increasing burnup and enrichment was reviewed previously by the Nuclear Energy Institute (NEI) [2]. The report identified three main issues and identified potential paths forward as shown in Table 1-1 below. The second issue is related to fuel fragmentation and increasing burnup and will not be discussed in this report.

The following subsections provide an overview of the United States UO_2 fuel cycle. Section 2.0 discusses benchmarking between 5 and 20 w/o ^{235}U. Subsequent sections go into details of the impacts that enrichments between 5 and 20 w/o ^{235}U will have on transportation (Section 3.0), enrichment and fuel fabrication (Section 4.0), reactor physics (Section 5.0), and storage (Section 6.0). Section 7.0 provides a summary of the findings.

Table 1-1. NEI identified issues and paths forward [2]

Issue	Path Forward
Regulation limits on enrichment to 5 w/o ^{235}U	Eliminate the limit of 5 w/o ^{235}U from 10 CFR 50.68. Provisions are available to request approval of alternative package designs that could be used for the shipment of uranium hexafluoride with uranium enrichments greater than 5 w/o under 10 CFR 71.55(b or c). Merits of a new or modified design that included special design features would be reviewed and approved under the provision of 10 CFR 71.55, including 10 CFR 71.55(c).[i]
Fuel rod burnup limits due to fuel fragmentation	Ongoing research is expected to result in a linear heat rate limit versus burnup to preclude fuel fragmentation. Tests are planned to demonstrate that high burnup rods do not rupture following a Loss of Coolant Accident (LOCA). Reactivity insertion accident will also need to be evaluated.
Fuel shipping packages are currently designed to accept fuel enriched up to 5 w/o ^{235}U	Container design modification and re-licensing would be required to ship fresh fuel in excess of 5 w/o ^{235}U.

[i] Currently, the NRC sees no need to pursue approvals under 71.55(c). Packages can either be approved meeting 71.55(b), or an exemption can be sought to requirement 71.55(c).

1.1 Uranium Dioxide Fuel Cycle

The current United States UO_2 fuel cycle consists of several operations starting with uranium mining and ending with disposal of spent nuclear fuel (see Figure 1-1). Mining of uranium is not regulated by the NRC; it is regulated by the Office of Surface Mining, U.S. Department of Interior, and individual state regulations. Outside of mining, the NRC regulates the facilities and activities of the uranium fuel cycle. These regulations are contained within Title 10, “Energy”, of the Code of Federal Regulations (CFR). Enriched uranium is special nuclear material (SNM) and falls under 10 CFR Part 70, “Domestic Licensing of Special Nuclear Material” [3]. This regulation applies to the handling and processing of SNM, such as enrichment and fuel fabrication facilities, at all levels of enrichment. Regulations for transportation of radioactive materials can be found in 10 CFR Part 71. Regulations for storage are found in 10 CFR Part 72. Regulations for gaseous diffusion plants (GDPs) can be found in 10 CFR Part 76. The United States currently does not have operational GDP facilities, and there are no current plans to operate such a facility.

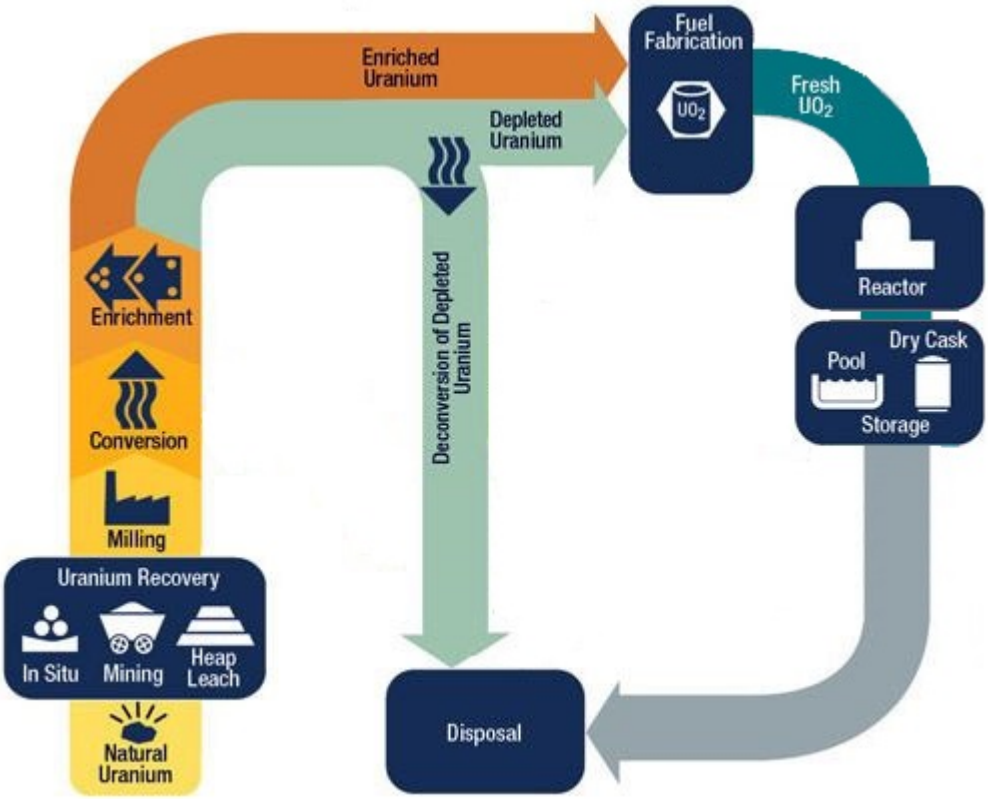

Figure 1-1. United States fuel cycle

1.2 Mining, Recovery, Milling, Conversion and Deconversion

The ^{235}U content of uranium ore is approximately 0.711% ^{235}U. This ore is mined with the uranium being recovered, milled, and converted to be input into the enrichment part of the cycle. Natural assay uranium does not pose a criticality concern as its ^{235}U content is too low to support a self-sustaining fission reaction absent the presence of heavy moderators more effective than water and a specific (concerted) geometrical configuration. Because the mining, milling, and conversion of uranium ore does not affect the ^{235}U content, it does not present a criticality concern. Honeywell's Metropolis facility is currently the only facility in the country that converts uranium ore into UF_6.

Deconversion of depleted uranium, the process of recovering the fluoride from depleted UF_6 by chemical conversion, is not expected to be impacted by the increase in enrichment. The ^{235}U content of depleted uranium is 0.3 w/o or less and like natural uranium does not present a criticality concern. Mining, recovery, milling, conversion, and deconversion will not be further discussed in this report.

1.3 Enrichment

Uranium has been enriched utilizing various methods that take advantage of the slight mass differences between isotopes of uranium. Gaseous diffusion and gas centrifuge enrichment processes have been used to provide the commercial enriched uranium supply. To achieve a higher enrichment, these processes are typically elongated with the material being enriched

passing through additional diffusors or centrifuges to achieve the desired enrichment. Currently there are no operating GDPs in the United States. The Paducah plant was the last operating GDP when it shut down in 2013. The only gas centrifuge commercial production plant in the U.S. currently is the URENCO USA facility in Eunice, NM. As of May 2020, the URENCO USA facility has received approval from the NRC for a license amendment, subject to an operational readiness review performed by the NRC, to increase operational limits from 5.0 w/o to 5.5 w/o ^{235}U [4]. One additional license has been granted by the NRC for construction of a commercial gas centrifuge facility, the American Centrifuge Plant in Piketon, Ohio. Construction of the plant is currently inactive [5].

An alternative method of enrichment to gas diffusion and gas centrifuge technology that has yet to be implemented on a commercial scale is laser enrichment. This method utilizes lasers to preferentially excite isotopes which are then separated by an electromagnetic field. A license has been granted by the NRC for construction and operation of the Global Laser Enrichment Facility in Wilmington, North Carolina. Construction of the facility is currently inactive [6].

1.4 Fuel Fabrication

During fuel fabrication, enriched UF_6 gas is converted into UO_2 powder either via a wet or dry chemical process. The UO_2 powder is pressed into pellets at room temperature and then sintered at high temperatures to achieve close to fully dense pellets. After sintering, the pellets are loaded into cladding. After the cladding is loaded with fuel and sealed, the new fuel rod is grouped with others into fuel assemblies. Burnable poisons are often added to some of the fuel rods to delay reactivity to later in the cycle. Burnable absorbers are added by either including gadolinium or erbium oxide (Gd_2O_3 or Er_2O_3) to the UO_2 powder prior to pressing and sintering, or by spraying a thin layer of zirconium diboride (ZrB_2) onto the surface of select pellets prior to loading in the cladding tubes. The general process flow of fuel assembly fabrication is shown in Figure 1-2.

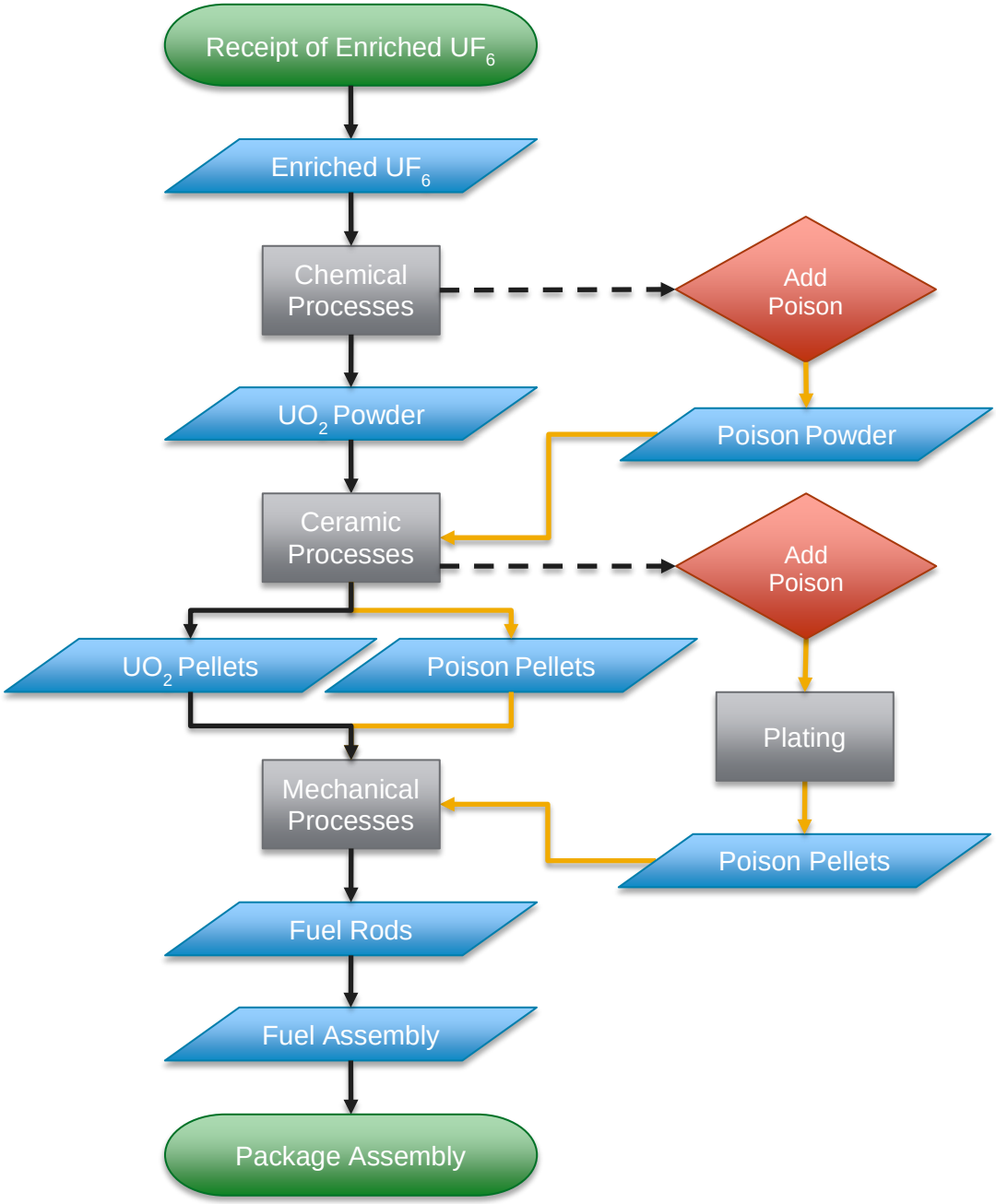

Figure 1-2. Fuel fabrication process

1.5 Fuel Transportation

Once fuel is ready for shipment it is loaded into an authorized package that meets the applicable standards for fissile material packages in 10 CFR Part 71 and shipped to its destination. There are two general types of packages for fissile materials; Type AF and Type BF. Type AF packages are designed for transport of fissile materials of limited radioactivity, as defined in Table A-1 of 10 CFR Part 71. LEU materials, such as UF_6 and fresh fuel, are typically

transported in Type AF packages. Type BF packages are designed to transport highly radioactive fissile material, such as spent fuel and reprocessed fuel. Both package types are reinforced to see during transport that:

1. Containment is maintained;
2. Radiation shielding is maintained; and
3. Criticality does not occur.

Both Type AF and Type BF packages must meet standard testing requirements to see that the above three requirements are met during normal conditions of transport and hypothetical accident conditions (as defined in 10 CFR 71.71 and 71.73, respectively).

1.6 Reactor Operation

During reactor refueling, preselected assemblies placed in the reactor during a previous cycle are either remotely extracted from the core and placed into a spent fuel pool to cool or are rearranged to a new location in the core. Fresh fuel is transferred from storage and loaded into the reactor. Once fully loaded the reactor goes through startup procedures that involve physics tests to determine if the operating characteristics of the core are consistent with the core design calculations.

1.7 Fuel Storage

Fuel storage for fresh fuel assemblies occurs in dry storage racks after final assembly in the fuel fabrication facility and prior to core reload either on racks or in the spent fuel pool at the reactor site. Spent fuel is placed in the spent fuel pool immediately after reactor discharge to allow for cooling. After sufficient cooling time has elapsed, spent fuel assemblies are placed in dry casks for longer term storage, typically onsite until they can be moved to a final waste repository.

2.0 Benchmarks

Benchmarks are used to validate that the methods used to model a problem, typically contained in software, provide the expected results with respect to an operating domain. Increased enrichment extends the operating domain which necessitates additional benchmarks for greater than 5 w/o ^{235}U. Benchmarks can be performed utilizing 1) experimental data such as laboratory critical experiments, 2) in-reactor measurements such as flux measurements, or post irradiation examination of fuel assemblies, such as gamma scans, and 3) field measurements, such as external radiation measurements of in-use dry cask storage. The differences between a method contained in software and measured data is typically evaluated by NRC in topical reports or other licensing actions. This section will discuss benchmarks as they relate to the UO_2 fuel cycle described in the introduction.

2.1 Criticality Safety

The safety of the public and personnel within the vicinity of nuclear materials is of the utmost importance. If not properly controlled, fissile materials can sustain a nuclear chain reaction, which is accompanied by a sudden release of radiation that is potentially deadly to local and co-located workers. Criticality within a reactor is highly controlled and desired, whereas criticality outside of a reactor is highly dangerous. Safety of those within the vicinity of these materials is made possible by controlling parameters such as mass, absorption, geometry, interaction, concentration, moderation, enrichment, reflection, and volume of the fissile material such that for a given operation inadvertent criticality will not occur under normal and abnormal conditions. Methods that analyze various spatial configurations of fissile materials and their surroundings must be benchmarked against critical experiments to see that the methods are predicting the effective neutron multiplication factor (k_{eff}, where a value less than 1 indicates subcriticality, equal to 1 is critical, and greater than 1 supercritical) as intended.

For criticality benchmarking, the geometry, materials, and neutron energy spectrum of system(s) of interest (i.e., area of applicability [AOA]) are validated utilizing a set of benchmark experiments based on guidelines provided in ANSI/ANS 8.24 [7]. One of the most extensive and respected sources of critical benchmarks is the International Criticality Safety Benchmark Evaluation Project (ICSBEP) Handbook [8]. It provides a database of thousands of vetted experiments (614 evaluations with 4,938 cases as of the 2019 release) that can be used as a basis for comparison in criticality benchmarking. While there are over 2,000 critical benchmark cases in the ICSBEP Handbook for systems with uranium enriched to less than 5.0 w/o, there are a few hundred in the range of 5 to 20 w/o uranium enrichment being considered for ATF. The benchmarks used to license higher enrichments would need to be reviewed to see that the increased enrichments and any other material changes (e.g., burnable absorber changes or cladding) or neutron spectrum changes can effectively establish an AOA for the methodology and/or code system being licensed for the higher enrichments.

There are sensitivity and uncertainty analysis methods that can be used to assist in the criticality benchmark validation process. Codes such as the Tool for Sensitivity and Uncertainty Analysis Methodology (TSUNAMI) in SCALE and Whisper in the Monte Carlo N-Particle® Transport Code System (MCNP) [9] [10] have been developed to perform these analyses. These methods have been introduced to augment expert judgement that is used in identification of applicable criticality benchmarks for criticality safety validation. These codes have not been heavily used for validation. However, the methodologies used, in tandem with current methods, can provide a

more formal and rigorous approach to determination of the applicability of critical experiments for criticality safety validation.

2.1.1 Sensitivity and Uncertainty Analyses

The sensitivity and uncertainty methods such as those described in NUREG/CR-6655 can be used as a supplement to benchmark validation [11]. These methods analyze the underlying physics of the models being evaluated and help determine if specific critical experiments are applicable to the system being evaluated. This is done by correlating various aspects of the experiments, such as the energy spectrum of the neutrons, with what is being modeled. Two geometric arrangements of material may look similar, but due to spacing, shapes, arrangements of moderators or absorbers nearby, the neutron spectra are actually very different. This can lead to use of a different range of the cross-section data and require additional benchmarks to validate the computational method for the system being evaluated. These methods can assist in determining the similarity of experiments needed to validate computational methods for a particular fissile material system and may be used to show that experiments that would not be considered applicable using traditional methods, are actually applicable based on sensitivity and uncertainty analysis. A critical experiment with 5 w/o ^{235}U enrichment can be compared to a system with 10 w/o enrichment to determine if there is enough similarity to use the lower enrichment experiment to validate computations with the higher enriched system. For example, the NRC guidance document, Nuclear Material Safety and Safeguards (NMSS) issues NMSS-0007 concluded that when utilizing the NUREG/CR-6655 method for developing criticality safety validations, benchmark experiments at 5 w/o are applicable to calculations up to 11 w/o for simple geometries with well-moderated neutron spectra [12].

2.1.2 Fuel Fabrication: Processing of UF_6 and UO_2

Fuel fabrication facilities must be able to safely receive and move UF_6 with enrichment greater than 5 w/o through their fabrication facility. Analysis of the configurations of the enriched material need to be performed to see that a criticality accident does not occur. Analysis using approved methods that are benchmarked would be required for licensing. The methods contained in facility analysis would be benchmarked against critical experiments.

The feedstock for production of UO_2 is typically UF_6. In most cases, operations involving UF_6 are conservatively considered moderated in criticality analysis. When UF_6 is in contact with moisture, uranyl fluoride (UO_2F_2) and hydrogen fluoride (HF) are formed. Although there are no available benchmarks for UF_6 other than at 93 w/o ^{235}U with only a few cases at 4.89 w/o and 4.94 w/o ^{235}U, other benchmarks containing similar uranium isotopics can be used to construct a validation that bounds the analysis for operations involving UF_6 and UO_2F_2 provided that the other parameters (material, geometry, neutron energy, moderation) are comparable to the system being analyzed and suitable statistical treatments are utilized. There may be instances where more conservative materials such as UO_2 or U metal are validated for similar operations that can be utilized in analysis of an operation. This however may be too restrictive and additional benchmarking may be required to allow for greater material throughput. Additionally, utilization of a physical argument such as the transparency of oxygen or fluorine to thermal neutrons may be a justifiable reason for similarities between compounds such as UO_2 or UF_4, provided the neutron energy ranges are similar between the experiment and operation.

During fuel processing and fabrication, there are numerous benchmark experiments that are upwards of 93 w/o ^{235}U to validate the onsite transportation, handling, storage, and operations in which UO_2 is present. However, taking credit for the reduced reactivity from some absorbers

and cladding may not be explicitly available due to the lack of absorbers present in the benchmark experiments for some enrichments in the range of 5 – 20 w/o as shown in Appendix A. Ignoring the absorbers or cladding typically adds conservatisms to the criticality analysis. If credit is taken for these absorbers or claddings not present in a given range, additional sensitivity analyses would be needed in the validation to justify their use.

Other compounds, such as U_3O_8, UO_3, UF_4 and $UO_2(NO_3)_2$ may also be present in fuel fabrication facilities. Benchmarks containing these compounds are limited in the range between 5 and 20 w/o ^{235}U. As with UF_6, other benchmarks containing similar uranium isotopics can be used to construct a validation that bounds the analysis for operations.

2.1.3 Transportation and Storage of Fresh Fuel

The Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages NUREG/CR-6361 utilizes 180 critical experiments as benchmarks ranging from 2.35 w/o to 5.74 w/o ^{235}U [13]. Review of the ICSBEP Handbook [8] shows that there are a limited number (less than ten cases) of experiments that match the geometry of a cylindrical fuel in a square lattice above 7 w/o ^{235}U (see Table 2-1 below).

Table 2-1. Evaluated criticality benchmark experiments for greater than 5 and less than 20 w/o ^{235}U

Evaluation Identification	UO_2 enrichment	Number of cases
IEU-COMP-THERM-009	18.31	2
LEU-COMP-THERM-018	7	1
LEU-COMP-THERM-024	9.83	2
LEU-COMP-THERM-047	7	2
LEU-COMP-THERM-076	7	3
LEU-COMP-THERM-078	6.90	15
LEU-COMP-THERM-080	6.90	11
LEU-COMP-THERM-081	6.6	1
LEU-COMP-THERM-096	6.90	19
LEU-COMP-THERM-097	6.9	24
LEU-COMP-THERM-098	5.74	6

NUREG/CR-6698 Table 2.3 provides guidance on the isotopic composition range allowed for a given benchmark experiment as well as other parameters important to validation [14]. For a system with 5-10 w/o ^{235}U, an experiment can have a range of ±2.5 w/o; for a system with 10-20 w/o, an experiment can have a range of ±5 w/o. If all materials of the system are present in the benchmark experiment, it would be possible to benchmark up to 9.5 w/o UO_2 with the 7 w/o ^{235}U benchmark experiments (with suitable statistical treatments for extending the range of applicability, per Section 5 of NUREG/CR-6698). Above 9.5 w/o and up to 20 w/o, there are less than ten benchmark cases total (more if triangular pitches are included; less if reflector, moderator, poisons, and pitch are not adequately represented). Although there is no guideline for the minimum number of critical experiments necessary for validation of a method for a given material or condition [14], the use of a few experimental benchmarks should be accompanied by a suitable technical basis to support acceptance of the validation results.

Appendix A provides a list of all available benchmarks containing UO_2 found in ICSBEP. The table utilizes the guidance on the isotopic compositions from NUREG/CR-6698 Table 2.3 to identify the moderators, reflectors, cladding, poisons, pitch type, and fuel geometry that are present in that given range. Figure 2-1 shows that there is a dramatic decrease in available benchmarks above 6 w/o, and they become increasingly limited as enrichment increases. This does not consider that not all the available benchmarks will be of use in validation to establish an AOA and not all materials may be covered by benchmarks within a range of interest. For example, if validating a system with UO_2 at an enrichment of 8 w/o ^{235}U, there are no benchmarks available that contain ZrB_2, gadolinium, or concrete.

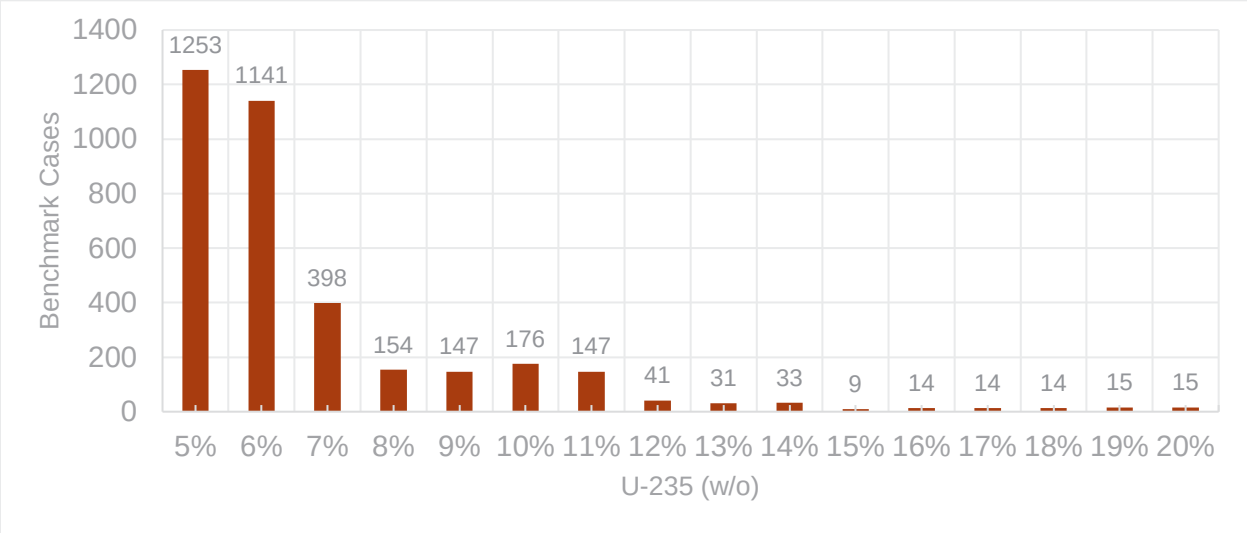

Figure 2-1. Number of UO_2 benchmark cases in ICSBEP with enrichments of 5 – 20 w/o ^{235}U

2.1.4 Refueling

Movement of fuel between the spent fuel pool and the reactor during refueling is a controlled operation. Fuel assemblies would be of similar size and dimensions, but the increased reactivity due to increased enrichment would necessitate a review of refueling operations to see that a criticality accident does not occur. Methods for validation of transporting and preloading of fuel will be like those described in Section 2.1.3. Methods used to analyze refueling operations would need to be approved for the extended operating domain resulting from increased enrichment. Benchmarks for simulations of reactor operation are discussed in Section 2.3.

2.1.5 Transportation and Storage of Spent Fuel

Criticality benchmarking of transportation and storage of spent fuel is typically done in one of two ways; 1) assume that the material is fresh fuel or 2) take credit for the depletion and treat as spent fuel. Treating the fuel as fresh and ignoring any integral poisons is usually conservative in comparison to treatment as spent fuel due to the decrease in reactivity as a result of the reduction in fissile nuclides and production of neutron-absorbing nuclides during fuel depletion. Although irradiation produces fissile actinides such as ^{239}Pu and ^{241}Pu, the reactivity effect of these fissile nuclides is outweighed by the depletion of ^{235}U and the production of neutron-absorbing actinides and fission products. Increases in the ^{235}U enrichment also affect the production of ^{239}Pu and ^{241}Pu which is discussed in Section 5.1.

Because critical experiments typically do not contain enriched uranium, plutonium, other actinides, and fission products in the same proportions contained in commercial spent fuel, a combination of critical experiments is necessary to validate the application of burnup credit. In addition to critical experiments, radiochemical assay (RCA) measurements of spent fuel samples are also typically used for validation of isotopic depletion codes, which are necessary to determine burned fuel composition. French *Haut Taux de Combustion* (HTC) [15] and French Fission Product Programme (FPP) experiments have been used as a basis for burnup credit validation in LWRs in conjunction with select experiments from the ICSBEP Handbook. These experiments, however, only address up to 5 w/o UO_2 fuel. Further evaluation for burnup credit is needed for greater than 5 w/o UO_2.

If taking credit for burnup, the requirements of ANSI/ANS 8.27-2015, "Burnup Credit for LWR Fuel," as well as the exceptions discussed in Regulatory Guide 3.71, Revision 3 can be used for validation. This standard provides criteria for accounting for reactivity effects of fuel irradiation and radioactive decay in criticality safety control of storage, transportation, and disposal of commercial LWR UO_2 fuel assemblies [16]. Burnup credit analysis requires validation of the composition of nuclides, ensuring their consistency within the fuel, and validation of the criticality calculation covering the nuclides used to determine k_{eff}. In addition to these requirements, recommendations are provided in NRC Interim Staff Guidance 8, Revision 3 (ISG-8), "Burnup Credit in the Criticality Safety Analyses of Pressurized-Water Reactor (PWR) Spent Fuel in Transportation and Storage Casks," for licensing basis limits and assumptions, code validation, and loading curve and burnup verification [17]. Currently ISG-8 only supports up to 5 w/o enrichment in ^{235}U and will need to be extended to higher enrichments and burnups.

Two NUREG reports, NUREG/CR-7108 and NUREG/CR-7109 [18] [19], provide an approach for validating actinide and fission product burnup credit for commercial spent fuel criticality safety analyses. The methods in these reports provide steps for determining isotopic compositions as well as criticality (k_{eff}) predictions of spent fuel. Extending use of these methods to higher enrichments is possible. However, it is important to note that when determining bias and bias uncertainties utilizing methods/tools such as Upper Subcritical Limit Statistics (USLSTATS) code, the accuracy of predicting the biases' estimates can be dependent on knowing the biases of the criticality experiments and may also require approximately 20 to 40 benchmark experiments to produce accurate bias estimates [20]. With the limited benchmarks above 5 w/o and the high degree of correlation suggested by guidance from NRC ISG-10 [21], additional efforts may be needed to determine the biases. It is important to note that there are general concerns about reliability of such methods in criticality safety bias predictions. Thus, it is recommended that if such methods are used, they be used in tandem with the trending analysis methods provided in NUREG/CR-6698.

2.2 Source Term and Shielding

Source terms define the type, energy, and strength of radiation from nuclear materials. They are used as a source to analyze how the materials and environment are affected by the radiation and to see that the containers and packages holding the nuclear material is providing the required protection to the public from radiation.

Validation of depletion and shielding codes via comparison to experimental data has generally not been required for every application to the extent that is needed for criticality. The expectation is that the use of reasonable procedures and well-established computer codes that have existing benchmarking data, in addition to requirements to confirm dose rates using measurement, have been enough to determine compliance with regulations related to dose and

dose rate limits for transportation and storage systems. Source term calculations can be performed in a conservative way by assuming a lower enrichment and therefore additional benchmarking may not be needed to support an application authorizing high enrichment fuel. NUREG/CR-6802 provides recommendations for shielding evaluations for transport and storage packages [22]. The NRC staff is currently engaged in research to update this document, which includes updating the validation for the current codes and determining if they remain applicable up to burnup as high as 70 GWd/MTU. Additional benchmarking may be needed for an analysis that assumes higher enrichment if it is in conjunction with higher burnup.

2.3 Reactor Operations and Simulation

Reactor simulation tools cover phenomena related to interactions of neutrons with materials for sustaining a nuclear reaction (neutronics), effects of radiation and heat dissipation within fuel elements (fuel performance), and the effects of radiation and heat dissipation outside of the fuel elements in the reactor coolant system (thermal hydraulics). Radiation safety calculations related to worker dose rates will be discussed after simulation. Neutronics, fuel performance, and thermal hydraulics have historically been modeled separately in different pieces of software due to computational resource limitations. A neutronics code would have simplified fuel performance and thermal hydraulic models within it, while a fuel performance code would have a simplified neutronics and thermal hydraulics model, and a thermal hydraulics code would have simplified neutronics and fuel performance models. During the reactor design process to develop a fuel loading pattern for a cycle of reactor operation, the groups work iteratively with each other to see that the reactor design meets all regulatory requirements. There have been recent submittals to apply coupled code predictions to reactor safety analysis.

Increasing enrichment affects neutronics, fuel performance and thermal hydraulics in different ways. The amount of heat produced in a length of fuel axially (linear heat generation rate (LHGR)) would change; the range of LHGRs in the higher enriched fuel should be examined for possible changes to the LHGR AOA. Most fuel performance codes are approved for up to 5 w/o enriched fuel but contain data from higher enriched fuel rods in their assessment databases from test reactors. Increased enrichment will likely result in an increase in the number and loading of integral fuel burnable absorbers (IFBA) or Gd_2O_3, because the amount of boron in the coolant is limited by its temperature reactivity coefficient. A review of the fuel performance and thermal hydraulic methodologies should be performed to see that the material properties and thermal conductivities are adequately accounted for in the methodology. Neutronics will be impacted by both neutron transport effects, and modifications needed to account for any changes in fuel performance and thermal hydraulics modeling within the methodology.

Historically, neutronic core simulation is a two-step process with a high fidelity two-dimensional (2D) fuel lattice neutronics calculation with reflective boundary conditions performed for each region of a fuel assembly of neutronic importance. These high fidelity, 2D results are then processed into data tables which describe the neutronic behavior of the fuel lattices which covers the range of reactor operating conditions such as temperatures of fuel, cladding, moderator and coolant, along with coolant boron concentrations, and control rod or blade insertions. This description of the neutronic behavior is then used as input to a lower fidelity three-dimensional (3D) core simulator. The 2D and 3D software applications are benchmarked together as a code system against plant data such as flux measurements and against post irradiation measurements such as gamma scans while the 2D software is benchmarked against additional experiments to see that the global reactivity and local flux and power distributions within the lattice were acceptable. Alternatively, software that encompasses an entire method and not part of a two-step process would still be benchmarked against the same benchmarks of

a two-step method. This is because all methods are simulating nuclear fuel and need to be benchmarked against measured data.

2.4 Characteristics of Additional Benchmarks

Fuel vendors continually innovate and bring new fuel designs to the market. Increases in the number of fuel rods in a fuel assembly, and changes to spacer grids, mixing vanes and debris catchers are a few examples. These new designs undergo a licensing process. Increasing enrichment may be analyzed with a similar licensing process to that used for new fuel designs, however the review of topical reports and other licensing actions would need to see that the physics of the increased enrichment is adequately being captured. In a two-step process described in Section 2.3, the higher fidelity lattice physics code provides an intranodal shape of power and flux to the core simulator for use in calculating peaking factors. The AOA of these methodologies encompasses various strengths and arrangements of absorbers that can be adequately modeled within an error range defined in the methodology's licensing. Enrichment above 5 w/o would require an assessment of the methodologies errors as compared to benchmarks in the licensing application.

For example, increases in current linear boron loadings for IFBA or Gd_2O_3 concentration for burnable absorber rods beyond the current AOA would need to be benchmarked to see that the higher absorption (blacker material) is adequately being modeled. Blacker materials may challenge current methodologies, and benchmarking will show if the methodology captures the dips and peaks in the flux shape. Likewise, new materials not in a methodology's AOA would need to be benchmarked. Due to the large number of possibilities and combinations of possibilities, each methodology's ability to model a new fuel design outside of its current AOA would need to establish a new extended AOA. The International Reactor Physics Experiment Evaluation (IRPhE) Handbook contains many benchmarks in addition to proprietary benchmarks a licensee would present for validation of a methodology's AOA [23].

3.0 Transportation

Information on transportation was derived mainly from the information provided on the Radioactive Material Packaging (RAMPAC) website for currently certified packages [24]. The intent of this section is to provide a general summary of the current state of certification for transportation of UF_6, UO_2 (loose pellets/powder), and fresh/spent fuel rods.

3.1 Uranium Hexaflouride – UF_6

Transportation of UF_6 must meet certain inspection, testing, and in-service requirements found in 49 CFR 173, Subpart I. Enrichments that exceed 1.0 w/o ^{235}U must have an overpack that meets the applicable standards for fissile material packages in 10 CFR Part 71. 49 CFR 173.420 requires that the UF_6 packages must be designed in accordance with American National Standards Institute (ANSI) N14.1 [25]. ANSI N14.1-2019 provides a list of standard UF_6 cylinder data (see Table 1 in ANSI N14.1-2019). Standard cylinders are available for UF_6 with enrichments above 5 w/o ^{235}U, however, the maximum mass fill limits are significantly less than limits at or below 5 w/o.

Currently, only model 30B, 48X, and 48Y containers are certified for use under United States Department of Transportation (DOT) regulations (additional versions of the 48 series are approved under International Atomic Energy Agency [IAEA]). The UX-30 overpack is also certified for use with the 30B and 30C cylinders. The 30B cylinder is permitted to have UF_6 enriched up to 5 w/o ^{235}U and the 48X and 48Y containers are permitted to have UF_6 with up to 4.5 w/o ^{235}U. The 30B cylinder maximum fill limit is 5020 lbs., and the 48X and 48Y fill limits are 21030 lbs.

The 8A cylinder could be used for enrichments up to 12.5 w/o ^{235}U. The cylinder mass fill limit for the 8A container is 255 lbs. The 5A and 5B cylinders can be used for enrichments up to 100 w/o ^{235}U. The cylinder mass fill limit for the 5A and 5B containers is 54.9 lbs. The significant reduction in mass fill limits for the 8A, 5A, and 5B containers will impose additional constraint and burden on throughput for greater than 5 w/o enriched UO_2, by limiting the feed material for fuel fabrication. Thus, new cylinders should be designed to higher mass limits for greater than 5 w/o enriched UF_6. Currently, several package designers are developing 30-inch cylinders with integral neutron absorbers, which would be able to transport higher enriched UF_6.

3.2 UO_2 Fuel

Excluding LWR fuel assemblies and rods, there are several packages that are currently certified to transport fresh UO_2 in excess of 5 w/o ^{235}U. The currently licensed packages can be found on the RAMPAC website [24]. With regards to both fresh and spent LWR fuel, there are no currently certified packages that can transport full length UO_2 LWR fuel rods/assemblies at greater than 5 w/o ^{235}U. Packages such as the NAC International Legal Weight Truck (NAC-LWT) cask can be certified to ship LWR fuel with greater enrichments, as it is already certified for shipment of higher enrichment research reactor fuel.

4.0 Enrichment and Fuel Fabrication

Enrichment facilities have processed enriched uranium up to and beyond 20 w/o in the past. If an enrichment facility is not licensed to enrich up to 20 w/o, then actions would need to be taken to license their facilities and the transportation methods used to deliver the enriched product to the fuel fabrication facility. All fuel enrichment facilities will need to use approved containers and configurations of containers capable of transporting greater than 5 w/o enriched product such as UF_6. All processes involving fissile material will need to be reevaluated for criticality safety.

The steps in fuel fabrication take UF_6 material and convert it into UO_2 powder. This powder is converted to a ceramic UO_2 pellet and then loaded into a rod, which is loaded into a fuel assembly. For fabrication operations, mass, volume, and cylindrical diameter of various uranium solutions need to be reviewed. This will provide a general yet applicable correlation between current approved operations and provide a picture of what changes to operations will result due to increases in enrichment. Although materials may not be in the forms that are used, they will provide a trend of what to expect as enrichment is increased.

Fissile material in process are oriented either as homogenous or heterogeneous in normal operation. Homogenous material is randomly dispersed and well mixed, such as powders, single phase solutions, and random oriented, tightly packed pieces. Heterogenous material contain a patterned orientation which can include storage arrays and, in the case of fuel fabrication, fuel assemblies.

4.1 Processes with Homogeneous UF_6 and UO_2

During the conversion of UF_6 to UO_2, uranium will be handled in several different chemical forms. These chemical forms may vary depending on the processes being used. The most common chemical forms in the UO_2 fuel cycle other than UF_6 and UO_2 are U_3O_8, UO_2-H_2O mixtures, UO_2F_2 solutions, uranyl nitrate ($UO_2(NO_3)_2$) solutions, and UF_6-HF. A set of calculations performed by Hiroshi Okuno et al. provide critical values for single units of various homogenous uranium materials [26]. Table 4-1 through Table 4-3 provide minimum values of the lower limit critical parameters for a range of enrichments from 5 to 20 w/o for homogenously mixed UO_2-H_2O mixtures, UO_2F_2 solutions and UF_6-HF. As enrichment is increased, the mass, volume, and diameter that is considered subcritical decreases. UO_2-H_2O is the most limiting for the UO_2 fuel cycle, but vessels containing each chemical form should be properly evaluated. Table 4-4 shows as an example the percent reduction in these parameters as enrichment increase for UO_2-H_2O mixtures. At an enrichment of 20 w/o, there is an ~85% reduction in mass, ~60% reduction in volume, and ~30% reduction in cylinder diameter for systems to remain subcritical.

These reductions will likely drive operations to minimize the enrichment to the desired needs and/or create multiple facilities that accommodate varying enrichments in order to better optimize throughput. Operations at a single facility should be cautious when adjusting enrichments on a batch basis as residual material may exist in the process, which may increase holdup accumulation rates of ^{235}U in system components, and it is not possible to visually distinguish between enrichments.

Some conditions may exist where initially the fissile material is homogenous but through some environmental, physical, or chemical means the material reorients and becomes heterogeneous. This can occur through material precipitation, phase separation, boiling/freezing

temperatures, and filtration. It is necessary to address these situations to see that concentrations do not exceed safety limits during normal and abnormal conditions.

Table 4-1. UO_2-H_2O mixture lower limit critical parameters [26]

^{235}U Enrichment (w/o)	Uranium Mass (kg U)	Sphere Volume (L)	Cylinder Diameter (cm)
5	31.1	23.7	24.2
6	23.5	20.2	22.8
7	18.5	17.6	21.7
10	11.1	13.6	19.7
20	4.62	9.50	17.0

Table 4-2. UO_2F_2 solution lower limit critical parameters [26]

^{235}U Enrichment (w/o)	Uranium Mass (kg U)	Sphere Volume (L)	Cylinder Diameter (cm)
5	32.4	29.9	26.3
6	23.8	24.2	24.4
7	18.7	21.1	23.1
10	11.1	15.5	20.8
20	4.60	10.2	17.6

Table 4-3. UF_6-HF lower limit critical parameters [26]

^{235}U Enrichment (w/o)	Uranium Mass (kg U)	Sphere Volume (L)	Cylinder Diameter (cm)
5	828	243	55.1
6	648	190	50.0
7	525	154	46.6
10	339	99.7	39.4
20	154	45.4	29.8

Table 4-4. Percent reduction in minimum values parameters with respect to UO_2-H_2O at 5 w/o ^{235}U

^{235}U Enrichment (w/o)	Reduction in Uranium Mass	Reduction in Sphere Volume	Reduction in Cylinder Diameter
6	24.4%	14.8%	5.79%
7	40.5%	25.7%	10.3%
10	64.3%	42.6%	18.6%
20	85.1%	59.9%	29.8%

4.2 Processes with Heterogenous Fissile Materials

Heterogenous systems such as storage, processing rods, or construction of the fuel assembly that contain fissile material act much in the same way as a homogenous system in that the critical limits for physical parameters of the material will be reduced with increased enrichment. However, the critical mass and volume of a system containing ^{235}U may be less for a heterogenous system than a homogenous system as seen in Figure 4-1. Above enrichments of 5 w/o, the differences between the heterogenous and homogenous system begin to converge to the same critical mass and the differences in critical volumes of the system stay relatively the same, as shown in Figure 4-2.

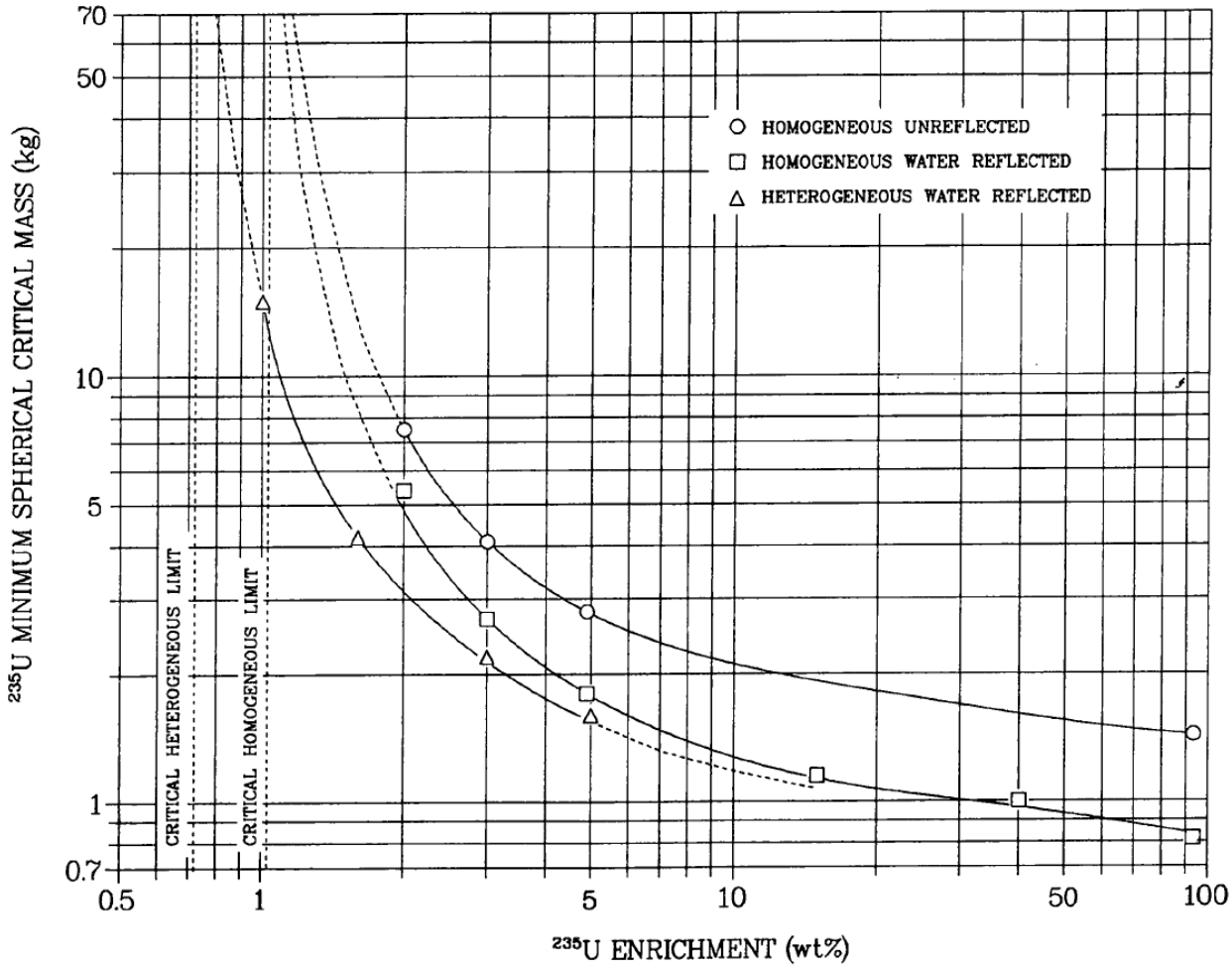

Figure 4-1. Minimum spherical critical masses as functions of ^{235}U enrichment in homogenous and heterogenous hydrogen-moderated systems [27]

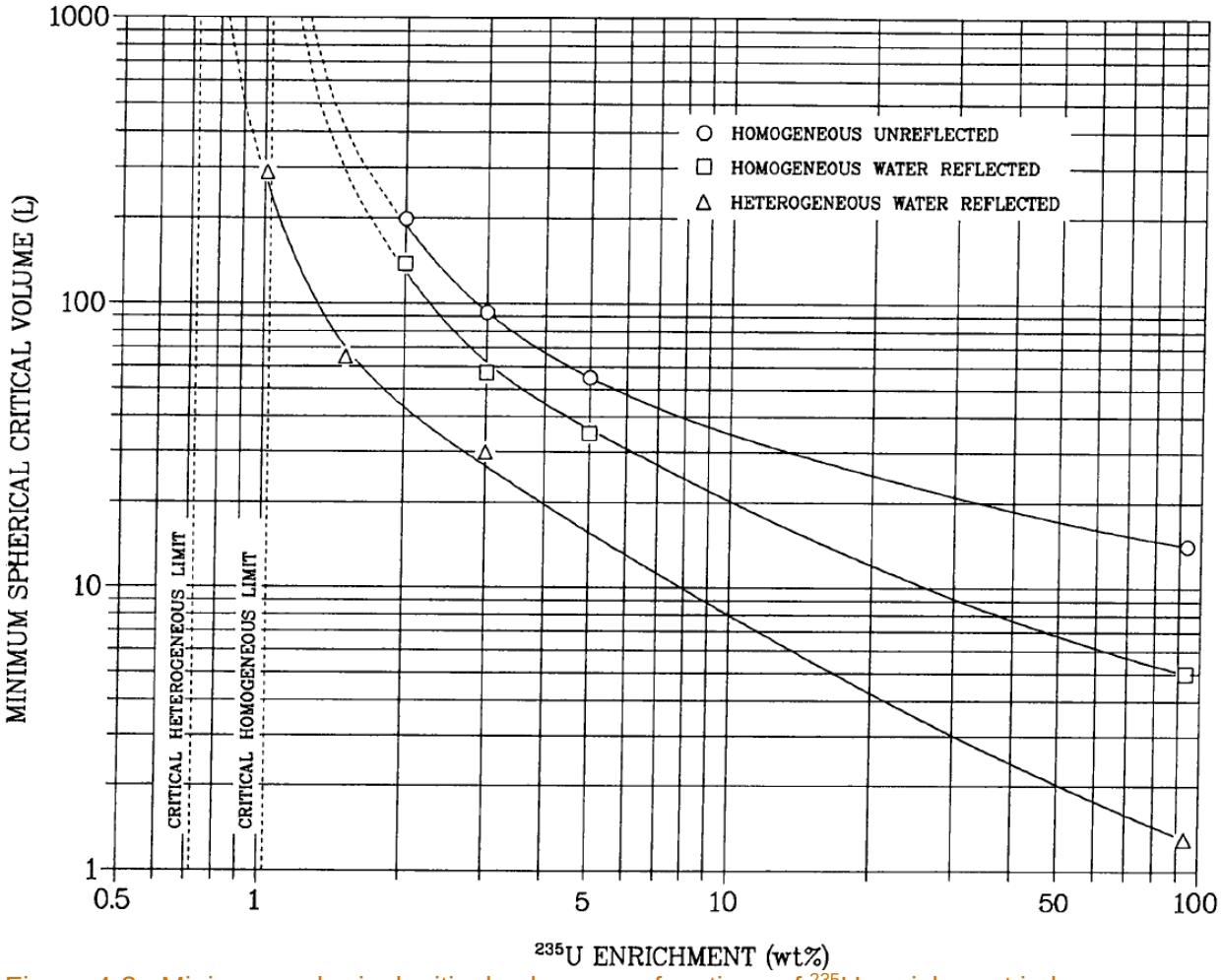

Figure 4-2. Minimum spherical critical volumes as functions of ^{235}U enrichment in homogenous and heterogenous hydrogen-moderated systems [27]

5.0 Reactor Operation

The interest in increase to greater than 5 w/o ^{235}U for fuel comes from several factors beneficial to reactor operation. Without the enrichment constraint, better control of local thermal peaking factors and reactivity coefficients can be achieved to optimize reactor core performance and minimize the amount of fresh fuel needed. With a decrease in needed fresh fuel, the cost of refueling goes down due to reduced fuel costs and decreased duration of refueling outages. Additionally, with increased enrichment, it is possible to more easily achieve extended fuel cycle length as well as power uprates of 10% or more [28]. Overall, there is better utilization of fuel with greater than 5 w/o ^{235}U.

5.1 Reactor Physics

The methodologies and software code systems used for reactor operation are evaluated based upon their ability to accurately predict both global characteristics such as eigenvalues for boiling-water reactors (BWRs) and critical boron concentration for PWRs, and local characteristics such as local flux, pin powers, peaking factors, etc. Safe and efficient reactor operation typically target a flat power shape radially across the core and axially. Variations during the cycle do occur such as with a “ring-of-fire” design where power is more efficiently produced by both flattening the power profile radially over much of the core while providing protection against fluence to the reactor vessel by having burned assemblies at the periphery of the core. Axial variations will occur due to thermal profiles resulting in differences in coolant densities or by design with control blades being withdrawn over the course of the cycle. In all cases the designer works to achieve the flattest possible power profile to better meet safety criteria.

Increasing enrichment enables core designers to achieve longer cycle lengths, power uprates, and reduced LHGRs by enabling new fuel designs that utilize a larger number of fuel rods with a reduced radius. Increasing enrichment would increase the reactivity in fuel assemblies. This increase in reactivity can be counteracted on a larger scale globally by affecting many assemblies or locally by affecting single assemblies and possibly targeting increased reactivity in specific fuel rods. Global reactivity control is achieved via control blade insertions in BWRs and soluble boron concentrations in PWRs, both of which are limited to the negative worths of the current control blades, and to the upper soluble boron limit due to the positive reactivity coefficient that is achieved at higher concentrations of boron. Local reactivity control is currently achieved using burnable absorbers incorporated into the fuel design. These burnable absorbers suppress the power and reactivity of the fresh fuel during the first cycle, shifting power production to twice burned fuel. Increased enrichment would do two things; first, require an increase in the strength of burnable absorbers; and second, increase the reactivity of fresh fuel going into the second cycle.

The use of stronger burnable absorbers leads to stronger gradients in the power and flux shapes within the fuel lattices. The increased reactivity of higher enriched fuel may initially result in stronger gradients in power and flux shapes within the core at the assembly to assembly level. This will necessitate an evaluation of methodologies to see that they can accurately predict core behavior given the increase in gradients in power and flux shapes. This can be accomplished by benchmarking against small reactor experiments to show that the 2D high fidelity code in a two-step process, or 3D high fidelity code, is able to accurately predict global reactivity and local power or flux shapes. Globally the two-step process of 2D high fidelity followed by a 3D low fidelity core calculation, or the 3D high fidelity code would have to show

that it can accurately predict core behavior. This can be accomplished by benchmarking against Lead Test Assemblies (LTA) or Lead Use Assemblies (LUA) to show that the codes can predict core behavior with a small number of higher enriched assemblies prior to fuel reloads of whole batches. The IRPHE also provides benchmarks in addition to vendor proprietary benchmarks [23]. In addition to steady state reactor operation, the methods used to predict core behavior must be evaluated based upon transient and accident scenarios. The use of higher enriched fuel may extend the current AOA, which would need to be evaluated for not only the neutronic core simulator but for fuel performance and thermal hydraulic methodologies.

Longer cycle lengths enabled by using higher enriched fuel will lead to an increase in burnup. Extended burnups will result in further hardening of the neutron spectrum (higher average neutron energy) due to the in-growth of plutonium over time. In LWRs as much as one third of the power produced comes from fissions of ^{239}Pu formed by ^{238}U transmutation over the course of the cycle. This hardening of the spectrum will require an evaluation of the current methodologies to account for this shift in spectrum.

Higher enriched fuel will also impact plutonium production rates and needs to be evaluated due to the complex interaction of many factors. Plutonium is produced via the parasitic absorption of neutrons by ^{238}U. Increasing ^{235}U reduces ^{238}U present in the fuel and thus reduces plutonium production rates; however, a hardened spectrum from additional absorbers will increase plutonium production. Additionally, the effects of greater fission product inventories in extended cycle fuels and flux spectral changes during operation impact plutonium production and are difficult to quantify without modeling. Given the number of parameters and their contrasting effects, it is important that increases in enrichment are accompanied by thorough benchmarking to show the codes can accurately predict core neutronic behavior.

5.2 Thermal Hydraulic and Thermal-Mechanical

With increased cycle lengths and power uprates, burnups are expected to exceed rod average burnup of 62 GWd/MTU, and fuel temperatures at core full power will likely increase. Increases in burnup and time in the core places more strain on the fuel cladding, requires higher corrosion resistance, increases internal pressure of the fuel rod as a result of an increase in gaseous fission products, and increases radiation-induced swelling that can result in fuel assembly/rod bow which can negatively impact control rod drop times due to friction [29]. A separate report, PNNL-29368, looks at fuel performance considerations and data requirements above 62 GWd/MTU [30].

In general, current thermal hydraulic methods should be applicable to fuel with greater than 5 w/o ^{235}U. Although excess reactivity in the fuel may challenge current thermal hydraulic limits, the current methods are expected to be appropriate for evaluating margin to these limits.

Likewise, with fuel thermal-mechanical codes and methods, the limits (e.g., cladding oxide thickness, cladding stress and strain) are expected to be applicable to fuel with greater than 5 w/o ^{235}U. The material properties and performance models are also expected to be applicable to higher enriched fuel. As mentioned in Section 2.3, thermal-mechanical codes have simplified neutronics models. Typically, the rod average linear heat generation and the axial distribution of power are input for each time step. These values would be taken from a neutronics model that has been approved for high enriched fuel.

The only modeling aspect of a thermal-mechanical code that is expected to be impacted by increased enrichment is the radial distribution of power within the pellets and the evolution of

that distribution with increasing burnup. There are two approaches that have been used to model the radial power distribution. The first is to use an approved neutronics code to generate look-up tables that the thermal-mechanical code can use. This approach is expected to be acceptable provided the neutronics code is validated and approved to the appropriate enrichment levels. The second approach is to develop simplified models that are incorporated within the thermal-mechanical code, such as the TUBRNP model [31] [32] in the NRC's FRAPCON code [33]. In this case, the thermal-mechanical code should be approved to the appropriate enrichment level.

In general, the radial power profile becomes less edge peaked with increasing enrichment. The only safety significance this has, is that the radial power profile has a small impact on the prediction of fuel temperature. As an example, Figure 5-1 shows a sample calculation of the radial power profile of a pellet at 40 GWd/MTU with various enrichment levels. It can be seen from Figure 5-2 that the differences in radial power profile have a moderate impact (36-66K) on the fuel centerline temperature prediction.

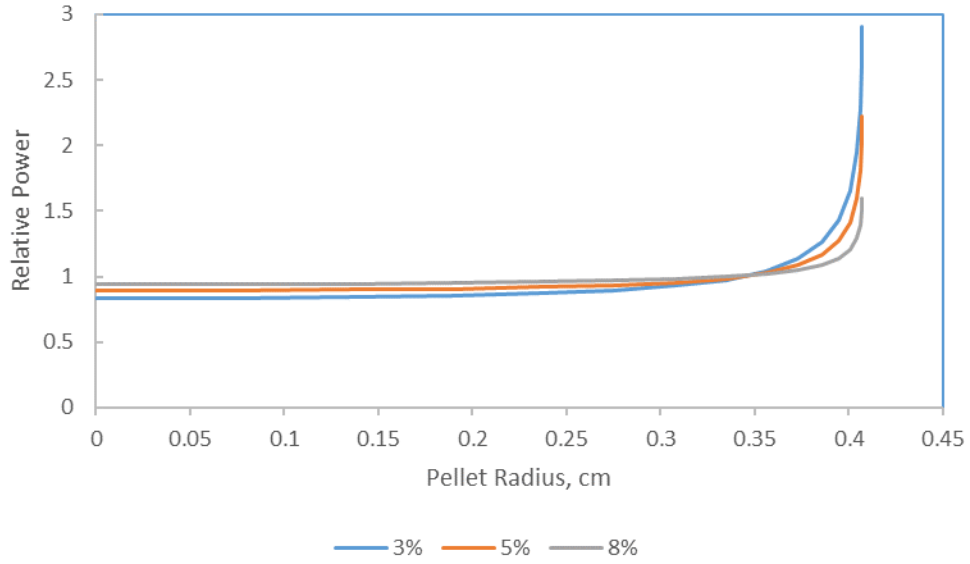

Figure 5-1. Radial power profile for UO_2 at 40 GWd/MTU with 3, 5, and 8 w/o ^{235}U

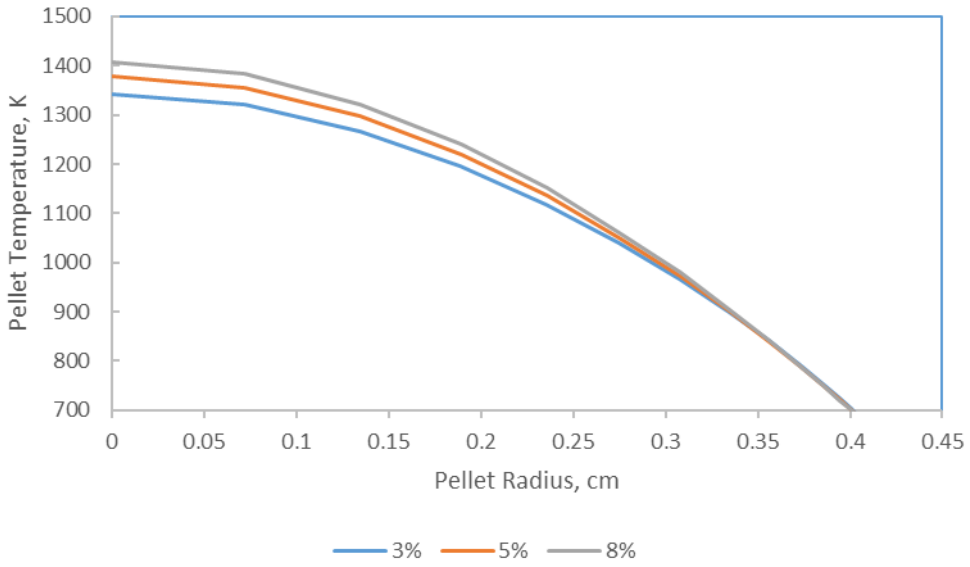

Figure 5-2. Centerline temperature prediction for UO_2 at 40 GWd/MTU with 3, 5, and 8 w/o ^{235}U

6.0 Storage

This section provides a brief summary of storage at reactor for both fresh and spent fuel as well as dry cask storage. Standard Review Plan (SRP) 9.1.1 establishes the criteria that the NRC staff utilize to determine if an applicant/licensee meets the NRC regulations.

6.1 Fresh Fuel Storage at Reactor

Prior to loading assemblies into the reactor, fresh fuel is stored temporarily in dry racks pending transfer into the spent fuel pool and then into the reactor core. For criticality safety, typically two independent events must be addressed with the storage rack filled with the most reactive assembly: 1) flooding using water, and 2) filling with low density hydrogenous fluid to optimal moderation. Given the higher reactivity of greater than 5 w/o ^{235}U fuel, it is expected that fewer assemblies will be permitted to be stored in these dry racks, the racks may be redesigned, or credit taken for integral poisons. However, it is also expected that fewer assemblies will be required to refuel the reactor at higher enrichments and criticality safety of the fresh fuel storage will likely not significantly impact operations.

Per 10 CFR 50.68 if controls are not in place to mitigate flooding, fresh fuel storage requires that the racks are loaded with fuel having the maximum fuel assembly reactivity and remain below a k_{eff} of 0.95 while flooded with unborated water, with a 95/95 probability/confidence level [34]. From the standpoint of criticality safety, unless there has been a detailed study of process conditions, there is no value of k_{eff} that provides a consistent value of safety that can be utilized in respect of prevention of criticality. As such, it is recommended that this control be revised to specify its AOA as well as what specific bias and/or bias uncertainty it intends to cover.

6.2 Spent Fuel Pool

Once the fuel has been used in the reactor, it is stored in the spent fuel pool. Depending on the density of the spent fuel allowed in the rack, the spent fuel pool may incorporate boron-10 as a neutron absorber to provide subcriticality. Additionally, neutron absorbers may be present in the storage racks themselves.

Spent fuel pools are subject to the requirements in 10 CFR 50.68 or 10 CFR 70.24 [34] [35]. Per 10 CFR 50.68, to assure subcriticality, credit may or may not be taken for the soluble boron in spent fuel storage, or per 10 CFR 70.24, controls may be implemented to detect and mitigate the consequences of an inadvertent criticality event. Per 10 CFR 50.68, not taking credit for soluble boron requires that the k_{eff} for the system remains below a reactivity of 0.95 with a 95/95 probability/confidence level when flooded with unborated water. Taking credit for soluble boron, the k_{eff} of 0.95 with 95/95 probability/confidence level must be met for a flooded system with borated water and remain below 1.0 with a 95/95 probability/confidence level when flooded with unborated water. As mentioned in Section 6.1, use of k_{eff} as a means of ensuring criticality safety is not recommended and the requirements should be revised to better reflect the AOA as well as the specific bias and/or bias uncertainty covered.

Fuel enrichment is one of several factors that affect reactivity in the spent fuel pool. Increases to enrichment will add reactivity to the spent fuel pool. Increasing enrichment is not novel to the commercial nuclear industry as fuel enrichments have progressively increased from 2-3 w/o to now 4-5 w/o ^{235}U. In conjunction with this increase, the center-to-center loading of fuel assemblies has decreased, introducing additional positive reactivity to the spent fuel pools and

resulting in the use of permanently installed neutron absorbers. To counter these reactivity effects, credit is taken for items such as ^{241}Pu decay, ^{241}Am buildup, axial blankets, integral burnable poisons on fresh fuel assemblies, and increased burnup.

Additional increases to enrichment will further complicate spent fuel pool analysis. Increases in enrichment will result in changes to the fuel and assembly, such as the fuel pellet diameter, fuel pellet density, amounts of removable and/or integral burnable absorbers, cladding, blanket length, enrichment distribution, average burnup, and burnup distribution. Changes to the core operating parameters due to power uprates may also occur, resulting in potentially more reactive fuel assemblies stored in the spent fuel pool. Increases in reactivity from fuel with enrichments above 5 w/o ^{235}U will drive the use of additional credits used in analyses such as burnup credit beyond the rod average of 60 GWd/MTU. NEA-6624 shows that enrichments above 5 w/o can reach very high burnup levels (80 GWd/MTU). It is mentioned that although burnup credit at these levels can be beneficial to spent fuel requirements, the higher the burnup (and cooling times) the more neutronic de-coupling affects the average burnup's reliability as an indicator of the true multiplication factor and should be considered [36]. Additionally, the lack of data on spent fuel at enrichments above 5 w/o ^{235}U will further exacerbate the issue by adding additional uncertainties to address in the analysis unless additional benchmarks are developed.

Engineering judgement has been utilized in lieu of the lack of experimental data for depleted fuel assemblies to determine the depletion uncertainties for spent fuel pool burnup credit calculations. Previously, it has been acceptably assumed that a 5% uncertainty in reactivity resulting from the uncertainty in depletion calculations is adequate [37] [38] [39]. In more recent years, additional quantification and/or justification has been requested by the NRC from licensee applicants for the 5% uncertainty assumption. This has driven the development of a series of experimental benchmarks for PWRs that can be used to quantify the biases and uncertainties associated with computations of depleted fuel reactivity for spent fuel criticality analysis up to 5 w/o ^{235}U [40]. Licensees can also apply the 5% uncertainty if lattice depletion codes are utilized in a manner that is consistent with nuclear design calculations previously performed for commercial power reactor licensing. The usefulness of these latter methods for increased fuel enrichments will be limited until a history of nuclear design calculations for these enrichments are available. These methods should be reassessed and extended to see that increases in enrichment do not impact the current guidance on the uncertainty in the depletion calculations.

6.3 Dry Cask

When spent fuel has cooled in the spent fuel pool for a few years (industry average is 10 years) and its radioactivity decreased enough to be removed, a dry storage cask is lowered into the spent fuel pool and filled with spent fuel. Once full, the cask is raised, drained, dried, and sealed. Then, it is placed outdoors onto a concrete pad. These containers have neutron absorbers in the spent fuel basket for criticality control while it is in the pool and flooded with water. It is expected that the reactivity of these systems will be higher for fuels that initially were enriched to 5 w/o ^{235}U or greater due to the amount of ^{235}U still present after depletion. If burnups also rise with enrichment, the additional heat output and neutron emissions from the spent fuel may potentially reduce the number of assemblies permitted in a cask.

The analysis for dry casks typically does not credit burnup. During loading of the cask, the analysis takes credit for soluble boron in the spent fuel pool for PWRs, and for BWRs assumes fresh fuel with no Gd_2O_3. Going to higher enrichments, PWR dry cask analysis could either increase the soluble boron loadings during cask loading (already done for some systems) or

take some level of burnup credit. A BWR dry cask analysis could utilize a conservative peak reactivity burnup credit analysis, similar to what is done in BWR spent fuel pools.

7.0 Conclusions

Exceeding the 5 w/o ^{235}U limit enrichment up to 20 w/o on fuel for commercial LWR's is feasible; however, there are gaps that need to be addressed. While above 5 w/o and below 20 w/o enriched fuels have fewer available benchmarks that are directly applicable than below 5 w/o, it is possible given the current state of critical benchmark experiments and use of suitable statistical treatments for extending the range of applicability of the validation to validate up to 9.5 w/o ^{235}U with little additional justification or work needed. Above 9.5 w/o ^{235}U, there are few directly applicable critical benchmarks available, and it would require more assumptions, uncertainties, and explanation to exceed that level unless additional benchmarks are produced or identified. For spent fuel, if burnup credit is sought, additional experiments are needed for burned fuel with higher enrichments and very high burnups.

For benchmarking of reactor operations analysis and codes, a review of the fuel performance and thermal hydraulic methodologies should be performed to see that the material properties and thermal conductivities are adequately accounted for in the methodology. Neutronics would be impacted by both the neutron transport effects from increased enrichment and the effects from possible changes in thermal hydraulic feedback from increased enrichment. Due to the large number of material combinations, evaluating each methodology's ability to model a new fuel design outside of its current AOA would be needed to establish a new extended AOA.

For transportation, packages with fuel enriched to greater than 5 w/o ^{235}U will need to be evaluated and approved for transportation for both UF_6 and UO_2 materials. Cylinders designed in accordance with ANSI N14.1 can be used for transporting up to 100 w/o ^{235}U enriched UF_6, however, loss in throughput due to mass limitations may occur. There are packages than can transport UO_2 materials with enrichments exceeding 5 w/o ^{235}U, however, none large enough to transport full assemblies are currently licensed. It is suggested that a currently licensed container such as the NAC-LWT be reevaluated for a new loading of greater than 5 w/o ^{235}U enriched fuel assemblies. Currently permissible upper limits for burnup will need to be increased given the longer in-cycle lengths.

Redesign of fuel fabrication facilities is expected given the additional geometry and volume constraints that accompany increased enrichment in ^{235}U. Caution should be taken if a process reduces uranium enrichment from previous enrichments given potential holdup in systems that may exceed the new limits.

For reactor operation, changes to the enrichment of the fuel will require adjustments to the core design that will increase the use of burnable absorbers. The increase in enrichment is driven by the desire to increase cycle lengths and power uprates. With increased cycle lengths and power uprates, more robust fuel elements will be required for reactor operation. It is recommended that the effects of changing fuel enrichment, burnable poisons, temperature, and fuel radius are appropriately accounted for in the analysis.

Storage of fresh and spent fuel assemblies will need to be evaluated at higher enrichments, and it is likely that more credit for poisons and burnup will be taken in the analyses. Given the longer cycle lengths in core, it is expected that higher burnups will be achieved, and additional analysis will need to be performed to permit the use of burnup credit at assembly average burnups in excess of 60 GWd/MTU. Requirements in 10 CFR 50.68 that utilize k_{eff} as a means of ensuring criticality safety are not recommended and should be revised to better reflect the AOA as well as the specific margin of safety the control intends to cover. Additionally, the 5% uncertainty

established in requirements for the uncertainty in depletion calculations should be reassessed for enrichments above 5 w/o.

Table 7-1 provides a summary of the issues identified in this report.

Table 7-1. Identified issues and suggested paths forward

Issues	Suggested Path Forward
Limited criticality benchmarks for greater than 5 w/o ^{235}U UO_2 for LWRs.	Identify or develop new benchmarks that can be used to expand the range of enrichments. Utilize sensitivity and uncertainty methods to assist in identifying gaps and necessary benchmarks.
Fuel shipping packages and storage casks are currently designed only to accept fuel enriched up to 5 w/o ^{235}U, fresh or spent. Higher burnups are expected for spent fuels with initial enrichments exceeding 5 w/o ^{235}U.	Shipping package design modification and licensing for greater than 5 w/o ^{235}U enriched fuel. New evaluations for storage areas with increased enrichments. Expansion of burnup credit to higher burnups.
Fuel Fabrication facilities designed for enrichments up to 5 w/o ^{235}U.	Reevaluate safety basis, redesign, or build new fuel fabrication facilities to permit higher enrichments.
Increase in the reactor reactivity due to higher enrichments will result in need for a higher strength of burnable absorbers.	Evaluation of methodologies to provide accurate predictions of core behavior given the increase in gradients in power and flux shapes.
Burnup expected to exceed current rod average limit of 62 GWd/MTU for reactor operation. This puts additional strain on the fuel cladding, requires higher corrosion resistance, increases internal pressure of the fuel rod as a result of an increase in gaseous fission products, and increases radiation induced swelling that can result in fuel assembly/rod bow which can negatively impact control rod drop times due to friction.	Additional data is needed for burnups above 62 GWd/MTU. See PNNL-29368 for specific needs [30].
10 CFR 50.68 requirements b.2, b.3, and b.4 utilize k_{eff} to provide safety margin for criticality safety. Changes to the fuel enrichment and materials will likely invalidate any study of process conditions to which the k_{eff} limits apply.	Revision of these requirements to better reflect the uncertainties bounded by the k_{eff} limits such as a margin of safety from a given bias and/or bias uncertainty anticipated in the evaluation of the system.
The analysis methods used to quantify the biases and uncertainties associated with computations of depleted fuel reactivity for spent fuel criticality analysis and using the 5% reactivity uncertainty established in requirements for the uncertainty in depletion calculations is only applicable up to 5 w/o ^{235}U.	Extend the analysis methods to enrichments above 5 w/o. As approved nuclear design calculations become available for greater than 5 w/o, utilize these to establish a reactivity uncertainty for the uncertainties in depletion calculations.

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Appendix A – ICSBEP UO_2 Benchmarks 5-20 w/o ^{235}U

The following table provides a list of the available benchmarks from the 2019 ICSBEP that are applicable for uranium between 5 and 20 w/o ^{235}U given the AOA specified in NUREG/CR-6698 Table 2.3 of ±2.5 w/o ^{235}U for uranium containing 5-10 w/o ^{235}U and ±5 w/o ^{235}U for uranium containing 10-20 w/o ^{235}U.

w/o	Lower	Upper	# of Cases	Moderators	Reflectors	Cladding	Solid Poison	Soluble Poison	Pitch	Fuel Geom
5.00%	2.50%	7.50%	1253	Lucite; Water	Borated Concrete, Water; Borated Water; Carbon Steel; Carbon Steel, Water; Concrete; Concrete, Water; Depleted Uranium, Water; Lead; Lead, Water; Lucite/Plexiglas (C, H, O); Lucite/Plexiglas (C, H, O), Water; Nickel; None; Polyethylene; Stainless Steel (Fe, Cr, Ni), Water; Water	Aluminum; None; Stainless Steel; Zircaloy; Zirconium-Niobium Alloy	Ag-In-Cd Alloy; Aluminum; Aluminum Oxide, Boron Carbide (B4C); Aluminum, Titanium; Boraflex (B, CH2, Si) Boral (B, Al, Na, Si); Borated Glass (Pyrex) (B, Si); Borated Stainless Steel (B, Fe, Cr, Ni); Boron Carbide (B4C); Cadmium; Cadmium, Copper; Cadmium, Stainless Steel (Fe, Cr, Ni); Copper; Copper, Stainless Steel (Fe, Cr, Ni); Dysprosium Alloy (Dy, Ti); Europium; Gadolinium; Gadolinium, Stainless Steel (Fe, Cr, Ni); Hafnium; Molybdenum; None; Polyethylene; Polystyrene (C, H); PVC (CH2, Cl); Rhodium; Stainless Steel (Fe, Cr, Ni); Titanium; Zircaloy (Zr, Fe, Sn, Cr); ZrB2	Boron; Cadmium; Gadolinium; None; Samarium	Square; Triangular	Cuboid; Cylindrical Rod(s); Mixed/Complex

6.00%	3.50%	8.50%	1141	Lucite; Water	Borated Concrete, Water; Borated Water; Carbon Steel; Carbon Steel, Water; Concrete; Concrete, Water; Depleted Uranium, Water; Lead; Lead, Water; Lucite/Plexiglas (C, H, O); Lucite/Plexiglas (C, H, O), Water; Nickel; None; Polyethylene; Stainless Steel (Fe, Cr, Ni), Water; Water	Aluminum; None; Stainless Steel; Zircaloy; Zirconium-Niobium Alloy	Ag-In-Cd Alloy; Aluminum; Aluminum Oxide, Boron Carbide (B4C); Aluminum, Titanium; Boraflex (B, CH2, Si) Boral (B, Al, Na, Si); Borated Glass (Pyrex) (B, Si); Borated Stainless Steel (B, Fe, Cr, Ni); Boron Carbide (B4C); Cadmium; Cadmium, Copper; Copper; Copper, Stainless Steel (Fe, Cr, Ni); Dysprosium Alloy (Dy, Ti); Europium; Gadolinium; Gadolinium, Stainless Steel (Fe, Cr, Ni); Hafnium; Molybdenum; None; Polyethylene; Polystyrene (C, H); PVC (CH2, Cl); Rhodium; Stainless Steel (Fe, Cr, Ni); Titanium; Zircaloy (Zr, Fe, Sn, Cr); ZrB2	Boron; Cadmium; Gadolinium; None; Samarium	Square; Triangular	Cuboid; Cylindrical Rod(s); Mixed/Complex

7.00%	4.50%	9.50%	398	Water	Borated Concrete, Water; Borated Water; Concrete, Water; Lead; Lead, Water; None; Polyethylene; Stainless Steel (Fe, Cr, Ni), Water; Water	Aluminum; None; Stainless Steel; Zircaloy; Zirconium-Niobium Alloy	Aluminum; Aluminum, Titanium; Boral (B, Al, Na, Si); Borated Glass (Pyrex) (B, Si); Borated Stainless Steel (B, Fe, Cr, Ni); Boron Carbide (B4C); Cadmium; Hafnium; None; Polyethylene; Polystyrene (C, H); Titanium	Boron; Gadolinium; None; Samarium	Square; Triangular	Cuboid; Cylindrical Rod(s)
8.00%	5.50%	10.50%	154	Water	Borated Water; Water	Aluminum; Stainless Steel; Zircaloy; Zirconium-Niobium Alloy	Aluminum; Aluminum, Titanium; Borated Glass (Pyrex) (B, Si); Boron Carbide (B4C); None; Titanium	None	Square; Triangular	Cylindrical Rod(s)
9.00%	6.50%	11.50%	147	Water	Borated Water; Water	Aluminum; Stainless Steel; Zircaloy; Zirconium-Niobium Alloy	Aluminum; Aluminum, Titanium; Borated Glass (Pyrex) (B, Si); Boron Carbide (B4C); None; Titanium	None	Square; Triangular	Cylindrical Rod(s)
10.00%	5.00%	15.00%	176	Water	Borated Water; Water	Aluminum; Stainless Steel; Zircaloy; Zirconium-Niobium Alloy	Aluminum; Aluminum, Titanium; Borated Glass (Pyrex) (B, Si); Boron Carbide (B4C); None; Titanium	Boron; None	Square; Triangular	Cylindrical Rod(s)

11.00%	6.00%	16.00%	147	Water	Borated Water; Water	Aluminum; Stainless Steel; Zircaloy; Zirconium-Niobium Alloy	Aluminum; Aluminum, Titanium; Borated Glass (Pyrex) (B, Si); Boron Carbide (B4C); None; Titanium	None	Square; Triangular	Cylindrical Rod(s)
12.00%	7.00%	17.00%	41	Graphite; Water	Graphite; Water	None; Stainless Steel	Borated Glass (Pyrex) (B, Si); Cadmium; Gadolinium; None	None	None; Square; Triangular	Annular Rod(s); Cylindrical Rod(s); Spherical Pebbles
13.00%	8.00%	18.00%	31	Graphite; Water	Graphite; Water	None; Stainless Steel	Cadmium; Gadolinium; None	None	None; Square; Triangular	Annular Rod(s); Cylindrical Rod(s); Spherical Pebbles
14.00%	9.00%	19.00%	33	Graphite; Water	Graphite; Water	None; Stainless Steel	Boron Carbide (B4C); Cadmium; Gadolinium; None	None	None; Square; Triangular	Annular Rod(s); Cylindrical Rod(s); Spherical Pebbles
15.00%	10.00%	20.00%	9	Graphite; Water	Graphite; Water	None; Stainless Steel	Boron Carbide (B4C); Cadmium; Gadolinium; None	None	None; Square; Triangular	Annular Rod(s); Cylindrical Rod(s); Spherical Pebbles
16.00%	11.00%	21.00%	14	Graphite; Water	Graphite; Water	None; Stainless Steel	Boron Carbide (B4C); Cadmium; Gadolinium; None	None	None; Square; Triangular	Annular Rod(s); Cylindrical Rod(s); Spherical Pebbles

17.00%	12.00%	22.00%	14	Graphite; Water	Graphite; Water	None; Stainless Steel	Boron Carbide (B4C); Cadmium; Gadolinium; None	None	None; Square; Triangular	Annular Rod(s); Cylindrical Rod(s); Spherical Pebbles
18.00%	13.00%	23.00%	14	Graphite; Water	Graphite; Water	None; Stainless Steel	Boron Carbide (B4C); Cadmium; Gadolinium; None	None	None; Square; Triangular	Annular Rod(s); Cylindrical Rod(s); Spherical Pebbles
19.00%	14.00%	24.00%	15	Graphite; Water; Wax (C, H)	Graphite; None; Water	Lacquer; None; Stainless Steel	Boron Carbide (B4C); Cadmium; Gadolinium; None	None	None; Square; Triangular	Annular Rod(s); Cylindrical Rod(s); Spherical Pebbles
20.00%	15.00%	25.00%	15	Graphite; Water; Wax (C, H)	Graphite; None; Water	Lacquer; None; Stainless Steel	Boron Carbide (B4C); Cadmium; Gadolinium; None	None	None; Square; Triangular	Annular Rod(s); Cylindrical Rod(s); Spherical Pebbles

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