

The Economic Benefits and Challenges with Utilizing Increased Enrichment and Fuel Burnup for Light-Water Reactors

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1 INTRODUCTION

In the United States, commercial light water reactors generate electricity using low-enriched uranium (LEU) fuel. On average, fuel costs comprise approximately 20% of nuclear power plants' total generating costs. Few other individual cost components have such a large impact on the economics of the nuclear fleet.¹ A site's fuel costs depend on two factors, the price of the fuel components (uranium feed, conversion, enrichment, and fabrication) and the efficiency of the core design. Fuel component costs are driven by supply and demand and are largely outside the control of a utility. The efficiency of a core design determines the quantity of nuclear material needed to meet a plant's energy objectives. While a utility can improve the efficiency of the core design, this efficiency is ultimately limited by the specific design constraints of the core design. Two of several constraints that have been shown to directly impact the core design efficiency are the uranium enrichment level and discharge burnup achieved by the core and/or fuel design. A review of the current fuel management practices, based on equilibrium cycle designs, has shown that 99% of the variation in fuel cycle efficiency is attributable to variations in enrichment and burnup. Many sites are currently constrained by the existing regulatory limits on one or both of these parameters.

With the increased interest in higher burnup cores, it is likely that within the next decade, both operating and advanced reactors will see a demand for fuel enriched greater than 5 weight percent (wt%) U-235. This white paper provides a study—including assumptions, economic projections, inflation and financial methodologies—that evaluates the technical, financial and regulatory issues associated with increasing the limits on uranium enrichment and on fuel burnup for current uranium dioxide (UO₂) fuel types. Revising these limits impacts a large portion of the nuclear fuel cycle as well as the licensing bases for both plant operators and fuel suppliers. While there are economic advantages to making these changes, they also require long-term capital investment and regulatory changes. Revising these limits will provide savings through additional cycle length flexibility, reduced high level waste storage and disposal requirements, and a positive benefit on the environmental impact of the fuel cycle. The final decision to pursue new limits must consider not only the expected benefits but the business risks associated with such an undertaking.

¹ NEI, Nuclear Costs in Context, October 2018.

2 ECONOMIC ANALYSIS

2.1 Technical Challenges

2.1.1 Burnup

Increasing the fuel burnup limit requires addressing a number of fuel mechanical design and reliability considerations. These include but are not limited to: rod internal pressure, cladding corrosion, rod and assembly growth, and cladding strain. While demonstrating acceptable fuel performance that satisfies all these design criteria represents a significant effort, it does not present an insurmountable technical challenge. The fuel suppliers are developing, or have developed, advanced materials or design features to mitigate these considerations. Additionally, some of the features of new accident tolerant or advanced technology fuel (ATF) designs may provide additional safety performance margins in these areas.

However, issues related to fuel fragmentation, relocation and dispersal (FFRD) during postulated design basis accidents (DBAs) remains a challenge. FFRD has been observed in some test reactor experiments under simulated light water reactor (LWR) conditions. However, the Electric Power Research Institute (EPRI) with support from the U.S. Department of Energy (DOE) and in collaboration with the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, is conducting separate effect tests which may potentially result in a full integrated test in 2022 using the restarted Transient Test Reactor (TREAT) at Idaho National Laboratory with the modifications for LOCA testing. This integral test will be performed under prototypical LWR conditions, which combine realistic fuel temperature profiles, appropriate linear power densities, and fission gas distributions appropriate to high burnup fuel and is expected to advance the understanding of FFRD to support potential extensions of existing burnup limits. Tests with next generation cladding and fuel pellet designs may also provide some additional insights on the benefits of the proposed ATF concepts. These pellets typically have a larger fuel pellet grain size which has been shown in separate effects tests to reduce normal and transient fission gas release behaviors during normal power operations and DBAs.

As an alternative approach, limiting cladding swelling and therefore the associated ballooning and rod burst opening size will potentially reduce fuel material relocation and dispersal while ensuring adequate safety margin. Some international regulators have already licensed plants to higher burnup levels using this criterion. Crediting this phenomenon will likely require reductions in the linear power density as a function of burnup. EPRI and DOE are evaluating the practicality of this approach as a backup to the fuel fragmentation research strategy.

2.1.2 Generic Safety Analysis

In addition to the execution of standard core reload safety analysis methods, some generic analysis methods may be impacted by fuel burnup limit increases (e.g., revised accident source terms and decay heat). A full review of the existing industry database is needed to determine if sufficient margin exists in the current analysis limits to safely support higher burnup limits. Potentially, additional tests with higher burnup fuel might be required. Additionally, some sites will likely require revisions to their current licensing basis with respect to fuel handling or other accident analyses that support higher accident source terms or the adoption of alternate source terms under 10 CFR 50.67 with appropriately revised regulatory guidance.

2.1.3 Dry Cask

Evaluations of dry cask fuel criticality, decay heat, and site boundary (both exclusion area and low population zone) dose limits will need to be performed to load higher burnup/enrichment fuel in dry casks. New certificates of compliance (COCs) are expected to be required to address the criticality requirements. Current decay heat limits are expected to be met with increased cooling times; however, methods will need to be developed to support extending the decay heat analysis for higher burnups and cooling times. While these changes to the design and licensing bases for dry cask systems will require effort, they do not pose a significant technical challenge. A significant reduction in spent fuel discharge inventory is expected with higher burnup/enrichments. This reduction will allow longer cooling times for the same spent fuel pool storage capacity. This increase in cooling time will partially offset the increase in heat load due to higher burnup. Sites with minimal storage capacity will need to develop an appropriate strategy prior to committing to higher burnup fuel. These strategies could include revising their fuel selection, fuel loading configurations or using a cask design with higher heat load limits. Implementation of these strategies will not occur until after 2035.

2.1.4 Enrichment

The largest technical challenge to increasing the enrichment limit is related to the controls and analysis to maintain criticality safety margins for fuel enrichment and, fabrication facilities, as well as storage and transportation systems. The transportation of enriched UF₆ from the fuel enricher to the fuel supplier is currently performed using Type 30B transportation packages. The current Type 30B design has sufficient margin to increase the allowable fuel enrichment limit to approximately 6 wt% U-235. However, this licensing basis is somewhat unique for transportation systems. It will be challenging to obtain the approvals to extend this analysis assumption to higher fuel enrichments. New transportation systems are being designed with fixed neutron absorbers to support an array of higher enrichments. These systems are expected to be licensed by 2022. Other fuel storage and transportation systems do not represent a significant technical barrier.

Fuel designed with higher enrichments will also include higher concentrations of fixed neutron absorbers to control in-reactor reactivity and power peaking. This is expected to largely offset the challenges for fuel storage and transportation. Additionally, higher enriched fuel will operate to higher burnups which also tends to offset criticality issues. New criticality analysis assumptions must be consistent with the expected fuel designs. Both the fuel designs and storage system margins are site specific. The strategy of using a higher concentration of fixed absorbers is generally expected to be effective however, some sites have limited design or storage flexibility and will elect to not adopt higher enrichment fuel designs. While relicensing any fuel system to meet modern criticality analysis standards poses regulatory challenges, as discussed in Section 3.4, these are considered to be manageable with the current technology and regulatory guidance.

2.2 Industry Demand and Economic Evaluation

2.2.1 Fuel Management

Fuel management plans were developed based on current fuel designs and analysis methods. Separate EPRI-sponsored analyses for boiling water reactors (BWR) and pressurized water reactors (PWR) were performed^{2,3} and are nearing publication in early 2019.

The BWR analysis² is based on a high power density BWR-6 reactor operating on a 24-month fuel cycle - the BWR-6s have a higher power density, more power per assembly, and therefore more restrictive design limits than BWR-4s. As such, the BWR-6 results are more conservative when extrapolated to BWR-4s. The fuel management plans produced equilibrium cycle designs using a 624 fuel bundle core, operating at 3299 MWt. Both the reference and high enrichment designs were developed based on designs which maximized the economics within the available design margins. The differences in the fuel management results are therefore attributable to the higher fuel enrichment and burnup assumptions. The results are summarized in Table 1. Note that the high enrichment cases included in Table 1 use pellet enrichments as high as 5.9 wt% U-235 and achieved peak pellet burnups near the target limit of 80 GWd/MTU. The current fuel design case results in significantly more margin to this target burnup limit than the new fuel case.

	Reload Batch Size	Batch Average Enrichment (%)	Peak Enrichment (%)	Batch Average Discharge Burnup (GWd/MTU)	Peak Pellet Burnup (GWd/MTU)	Fuel Utilization (gm-U-235/MW-Day)
Reference Case	256	4.18	4.9	48.19	62.9	0.8674
Higher Enrichment – Current Fuel Design	240	4.39	5.9	51.40	72.6	0.8540
Higher Enrichment – New Fuel Design ₁	216	4.78	5.9	57.11	79.7	0.8369

₁ A new fuel type is required to optimize the fuel performance at higher enrichments

Table 1: Summary of BWR Fuel Management Results

PWR fuel management analyses³ were performed for both high power and low power plants. Both sets are based on 18-month fuel cycles, the prevalent cycle length in the current PWR fleet. The high power design is based on a Westinghouse NSSS 4 loop, 193 fuel assembly core operating at 3469 MWt, based on the Westinghouse Robust Fuel Assembly (RFA) fuel design. The low power design is based on a Westinghouse NSSS 3 loop, 157 fuel assembly core operating at 2775 MWt, based on a Westinghouse Optimized Fuel Assembly (OFA) fuel design. The low power and high power cases were used to determine the fuel management efficiency's sensitivity to core power and core size. The efficiency was

² "Consulting Support for Increasing the Enrichment Limit – BWR Fuel Management Evaluation," September 2018.

³ "Westinghouse Fuel Management Scoping Study – Feasibility of Increasing U-235 Enrichment and Peak Pin Burnup Limits," November 2018.

determined to be proportional to core size. The high enrichment cases are based on pellet enrichments as high as 5.95 wt% U-235. Both the reference and high enrichment designs were developed using a consistent set of aggressive design limits so that the differences in the design results reflect increases in enrichment and burnup, not increases in power peaking or other limits. The results show consistent power peaking and boron levels. The economics (Table 7 and 8) are based on the difference in the fuel management results between the reference and high enrichment cases (Table 2 and 3) and do not credit a significant increase in design limits. However, when higher enrichment and burnups are applied to individual sites, more aggressive design limits may be needed to obtain the full benefit of higher burnup and enrichment. Therefore, existing limits in the areas of peaking factors, moderator temperature coefficient (MTC), shutdown margin (SDM), crud induced power shift (CIPS) risk, etc. may be challenged and will need to be addressed on a plant-specific basis. Any revised limits will need to be developed within existing safety criteria. The fuel management results are summarized in Tables 2 and 3.

	Reload Batch Size	Batch Average Enrichment (%)	Peak Enrichment (%)	Batch Average Discharge Burnup (GWd/MTU)	Peak Pin Burnup (GWd/MTU)	Fuel Utilization (gm-U-235/MWd)
Reference Case – average of odd and even cycles	76.5	4.6147	4.8	52.15	59.5	0.9002
High Enrichment Case	60	5.6400	5.8	65.36	74.0	0.8629

Table 2: Summary of PWR Fuel Management Results for High Power Plant

	Reload Batch Size	Batch Average Enrichment (%)	Peak Enrichment (%)	Batch Average Discharge Burnup (GWd/MTU)	Peak Pin Burnup (GWd/MTU)	Fuel Utilization (gm-U-235/MWd)
Reference Case – average of odd and even cycles	62.5	4.8360	4.95	54.97	61.1	0.8797
High Enrichment Case	52	5.6154	5.8	66.07	74.7	0.8499

Table 3: Summary of PWR Fuel Management Results for Low Power Plant

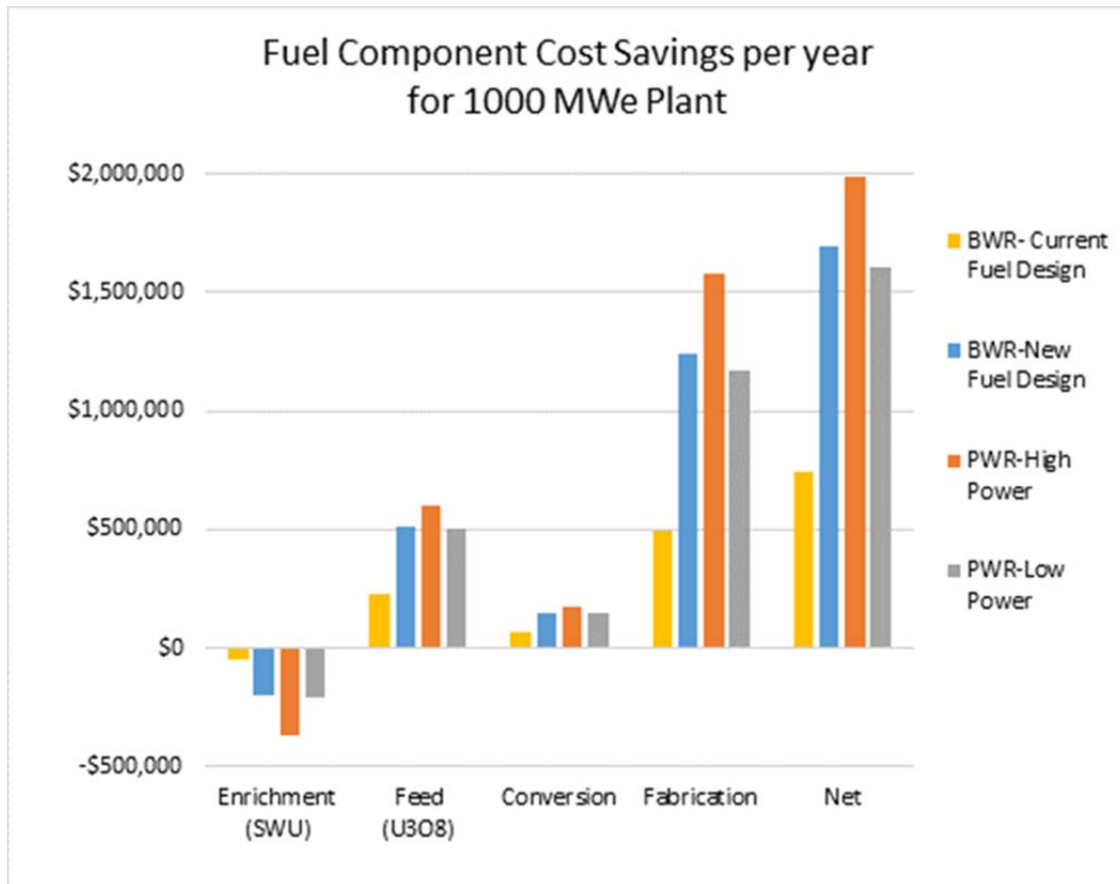
In order to apply these results to the entire fleet, fuel design data from the EPRI Fuel Reliability Database (FRED)⁴ and a recently held EPRI-sponsored fuel management workshop were used to construct a database of site-specific reference fuel designs. The burnup and fuel utilization changes

⁴ "Fuel Reliability Database (FRED) Version 5.0", 3002013234

associated with the use of higher enrichments were applied to the reference designs. This produced a set of site-specific high enrichment/burnup fuel designs. This approach supports the analysis of various strategies, such as a moderate burnup increase of 67 GWd/MTU. The actual moderate burnup increase that is achievable may vary somewhat from this assumption based on the fuel design specific performance in normal operation and postulated accident scenarios. Fuel component requirements (feed, enrichment, conversion and fabrication) were determined for each site. The results for the reload fuel cases described in Tables 1-3 are presented in Figure 1 based on projected fuel component costs in 2030. For all cases, enrichment expenses increased. Most of the savings come from fabrication (e.g., reduced number of fabricated assemblies for new reload batches) with a modest contribution from feed U3O8 stock (e.g., reduced amount of feed uranium for new reload batches). Therefore, these results are not very sensitive to future feed or enrichment market prices.

Consistent with current industry practice, fuel leases and their corresponding carrying cost are not included in the determination of fuel costs. The approach is consistent with fuel cost reported on FERC Form 1, "Electric Utility Annual Report" and informal surveys of key utilities. Currently only one utility, representing approximately 5% of the fleet, continues to employ fuel leases. However, the use of higher burnup fuel will impact the amortization of fuel costs. Smaller batch sizes will tend to extend the amortization while higher batch average power will tend to accelerate amortization. The net impact of these two opposing factors is core design specific. Also, if a site elects to adopt 24-month fuel cycles as part of adopting higher burnup designs, the amortization period is significantly reduced.

The sites are assumed to implement core designs using increased burnup/enrichment limits based on economic value to that site, over an 8-year period beginning in 2026. The fuel cost benefits continue until the expected retirement date of the plant. The current 60 year operating life and a potential 80 year operating life are both evaluated. The economic analyses of the 80 year operating life cases terminate in 2067, however these additional cycles have only a moderate impact on discounted cash flows and other project figures of merit.

Figure 1: Annual Fuel Component Cost Savings⁵

2.2.2 Capital Costs

Capital costs were estimated based on results from previous studies,^{6,7} input from fuel service supplier representatives, utility representatives, and engineering judgment. The cost for the modified Type 30B UF6 transportation systems were determined based on the assumption that an extension of the current moderator exclusion licensing basis to higher enrichments will not be approved by NRC or international regulators. A summary of the estimated capital costs is provided in Table 4. Note that the capital costs vary somewhat for different scenarios. Table 4 results are representative of the expected typical costs. For example, utility costs vary depending on the number of sites expected to implement higher enrichment/burnups. The timing of the capital expenditures is based on the expected regulatory approval dates.

⁵ Capital Cost are not included

⁶ "Strategy for Deployment of Advanced Fuels," RPT-3005895-000, September 28, 2011.

⁷ "Optimum Cycle Length and Discharge Burnup for Nuclear Fuel: Phase II: Results Achievable with Enrichments Greater than 5 w/o," 1003217, EPRI, September 2002.

Category	Capital Costs(\$M)
Enrichment	20.0
UF ₆ Transportation	21.3
Fabrication	136.0
Safety Analysis	10.0
Burnup Extension - DOE	7.5
Utility	79.5
Dry Storage	6.0
Total	280.3

Table 4: Summary of U.S. Fleet Required Capital Costs (estimated)

2.2.3 Fuel Component Unit Costs

The future fuel component costs (feed, enrichment, conversion and fabrication) are dependent on supply and demand for these components. A long term prediction of these costs was performed based on the projected global nuclear generation from several sources. These projections were provided to utility participants in the study and adjustments were made based on their feedback. Only the minimum unit costs were used in this evaluation. As noted previously, the fuel management savings are dominated by the fuel fabrication costs. Fabrication costs are somewhat different from other components since they include large technological and intellectual property barriers to entry (capital investment, design and manufacturing technology, regulatory approval). As such, fabrication prices are expected to increase as a result of investments necessary to manufacture higher enrichment fuel.

2.2.4 Discount Rate

The discount rate is used in capital allocation decisions to determine the current value of both costs and returns on investment which occurs in the future. The sum of these discounted cash flows provides the Net Present Value (NPV) of a proposed project. Projects which produce returns in excess of the required discount rate are expected to increase the value of the organization(s) sponsoring the project. The discount rate is also referred to as the hurdle rate. The appropriate discount rate is determined based on the weighted average cost capital (WACC). As shown in Table 5, the WACC is determined for three large nuclear utilities based on publicly available information. Utilities operating in regulated and deregulated markets were selected. The resulting WACC reflects the required return for a typical capital project of nominal risk. Since a nuclear related project generally involves higher risks, additional factors are needed to account for this additional risk. A nuclear risk premium was determined based on the accumulated interest payments for a large nuclear construction project. The regulatory and technical challenges necessary to modify the enrichment and burnup limits have a higher uncertainty than a construction project which is based on a mature design. As such, an additional 2.0% risk premium was determined to be appropriate. This results in a 9.93% (rounded to 10%) discount rate for this project.

	WACC
Utility A	4.92%
Utility B	6.21%
Utility C	4.86%
Average	5.33%
Nuclear Risk Premium	2.60%
Project Specific Premium	2.00%
Total Discount Rate	9.93%

Table 5: Estimated WACC for Representative Utilities

2.2.5 Escalation

All future cash flows were escalated to account for the expected impact of inflation. The escalation factors were determined based on published Department of Labor economic indices. These include indices for general and production employee compensation and the producer price index for industrial commodities. Applying different weighting functions results in escalation rates ranging from 1.92% to 2.62%, depending on the specific fuel cycle activity.

2.2.6 Economic Evaluation

Seven different scenarios were evaluated for both 60 and 80 year operating life assumptions. The scenarios are listed in Tables 7 and 8. These scenarios include individual BWR and PWR design results as well as various combinations of both designs. The PWR results consider two different burnup increase assumptions, a “Moderate Burnup” of 67 GWd/MTU and a “High Burnup” of 75 GWd/MTU. The “Moderate Burnup” case is estimated to correspond to a level that would be achieved if the LOCA no

rod burst criteria is applied, as discussed previously in Section 2.1.1. The BWR results only consider one burnup increase scenario (“High Burnup”) since the BWR designs restrict the linear power density for high exposure fuel assemblies so they are expected to demonstrate better FFRD performance. The BWR results include current designs and a future “next-generation” design which is optimized for higher enrichments.

All of these scenarios assume that the initial reloads begin operation in 2026 and the changes to the fuel cycle infrastructure occur in parallel to the testing and development of the increased burnup designs. An additional scenario was also evaluated which delayed the most significant capital investments until after the high burnup design’s testing and development (2025) were complete. This scenario reduces the risk associated with parallel activities but delays the initial reload date to 2029.

Fuel cost savings, capital costs, and dry cask savings were determined for each scenario. Escalation was applied to each area and the results were used to determine the NPV based on the project specific discount rate. In addition, the internal rate of return and the average annual savings per reactor were also determined. These results reflect the industry wide return that is available to compensate the stakeholders (e.g., fuel suppliers, DOE, utilities) of the project. The distribution among these groups is subject to commercial discussions that are beyond the scope of this study. However, to provide some perspective, estimates were made based on the following assumptions:

1. All capital costs were recovered with a rate of return equal to the project specific discount rate.
2. No dry cask cost saving flowed to utilities.
3. The fuel fabricator loss of revenue could result in significant fixed costs that are unrecovered. To account for this impact, a portion of the fabrication savings was accrued to the fabricators and the balance to the utilities.
4. DOE provided no significant financial support.
5. All feed savings and increases in enrichment cost are accrued to utilities.

The impact of these assumptions is illustrated in Tables 7 and 8 below. A discussion of these results is provided in the results section below.

In order to estimate the uncertainty of these results, perturbations in key input variables were performed. The input perturbations and the corresponding impact on the estimated annual savings per reactor, available to utilities, are summarized in Table 6. The individual perturbations were combined as independent variables to provide an estimate of the overall uncertainty. The Table 6 result corresponds to 13% of the Table 7 results for this parameter.

The impact of varying the number of participating plants was also evaluated. This evaluation determined the number of participating plants needed to recover the capital investment including the minimum required rate of return. This evaluation assumed only PWR units participate in higher burnup/enrichment projects. Only plants which operate in a challenging economic environment with a high expected return were assumed to participate. If fourteen (~25% of PWR fleet) of these high power units implement higher burnup/enrichment, then the minimum required rate of return is earned for the industry. The average annual saving for these fourteen units is \$0.9M/reactor.

	Perturbation (%)	Resulting Impact of Perturbation on Estimated Annual Savings per Reactor Available to Utilities (\$M) ₁
Uranium Feed and Enrichment Price	15	0.02
Fabrication Price	10	0.14
Vendor Capital Cost	25	0.07
Utility Capital Cost	33	0.01
Total		0.16

Table 6: Uncertainty in Economic Evaluation

₁ Excludes PWR Low Burnup case due to small impact

2.3 Results

2.3.1 Economic Results

A summary of the economic analysis results is included in Tables 7 and 8 below. Table 7 contains results assuming all sites operate to the end of their 60-year license life. This excludes sites that have already publicly announced early shutdown dates. Table 8 contains results for all sites assuming an additional 20-year license extension is obtained to reach 80 years of operation. Each of the evaluated scenarios results in a net present value that exceeds the discount rate (negative NPV). Additionally, the resulting internal rate of return is provided to demonstrate the expected return on the capital investment. This also exceeds the discount rate developed above (10%). However, note that there is significant variation in these quantities as well as the expected average annual savings for each reactor core. For PWR units the maximum return is typically 25% higher than the average while for BWR the maximum is 10% higher. The expected average annual savings are strongly dependent on the discharge burnup assumed in the analysis. Results vary between approximately \$3.83M/reactor and \$1.93M/reactor for peak pin burnup limits of 75 GWd/MTU and 67 GWd/MTU respectively [Table 7, Column 7]. Since this difference represents a significant uncertainty in the results, additional cases were evaluated which show the impact of changing the timing of licensing the revised burnup limits. These alternative scenarios largely mitigate the risk of licensing the higher burnup limit with only a minor impact on the value of implementing higher enrichment limits. The results from delaying the capital investment until high burnup is proven show slight increases in the average annual savings available to utilities. This is due to the elimination of initial, lower cost reloads, so the average value increases slightly. The net present value and internal rate of return are reduced, since the number of reloads is reduced in this case.

The development of ATF design concepts has the potential to extend the value of increased enrichments and fuel burnup levels by optimizing fuel design thermal limits, improving corrosion resistance of the fuel cladding and improving the fuel reliability over current designs. The specific performance of the

various ATF designs has not been fully determined at this time and therefore no quantifiable benefit regarding burnup and enrichment has been established.

2.3.2 High Level Waste

In addition to the direct economic value provided to utilities, an increase in fuel enrichment and burnup will reduce the volume of high level radiological waste generated and subject to long term storage and disposal. While the quantities vary between scenarios, reductions are projected for all the evaluated scenarios. For example, the 60 year operating life, moderate burnup PWR only scenario (“PWR Moderate Burnup” in Tables 7 and 8) is expected to reduce the dry cask storage requirements by approximately 200 dry casks while the high burnup PWR + New BWR fuel design is expected to reduce the number of dry casks by approximately 500.

2.3.3 Impact on Cycle Length

While virtually all U.S. BWR reactors operate on a 24-month fuel cycle, only about 20% of the PWR plants can operate economically on a 24-month fuel cycle with the current burnup and enrichment limits. With a modest increase in the burnup limit to 67 GWd/MTU, approximately 68% of the PWR fleet could operate on a 24-month cycle and all PWR plants could do so if the limit were raised to 75 GWd/MTU along with an increase in enrichment. The capability to achieve 24 month cycles is based on each site’s expected reload batch fraction determined from the revised burnup limits. Sites with a batch fraction above 50% are not considered economically viable candidates for 24-month cycle operation. The decision to extend cycle lengths depends on the net economic value of the increase in fuel cost, reduction in replacement power cost and reduction in the number of outages over the remaining life of the plant. Some capital cost must also be considered to extend surveillance intervals and instrument drift calculations. While some plants might not elect to change their cycle length, multi-unit sites and fleets with single unit sites that share key outage resources would realize additional operational benefits if they choose to transition to longer cycles.

As with previous industry increases in fuel enrichment and burnup, the plants would more efficiently use the available fuel. This results in a modest reduction in the demand for uranium ore and a corresponding reduction in uranium mining. While not quantified, some environmental benefits will be realized from this change in the fuel cycle.

2.3.4 Summary of Results

As described above, the risk-adjusted discount rate was determined to address the unique features of a complex nuclear project. Each of the identified scenarios provides a return that exceeds the required discount rate, indicating the project will exceed the required rate of return. However, this economic assessment is based solely on this specific project and does not evaluate its merits relative to other available options. Additionally, this project does require a commitment from a number of stakeholders and significant changes to fuel-related regulatory requirements. However, most PWRs would benefit from these changes and higher power BWR units would also realize tangible benefits.

	Fuel Cost Savings (\$M)	Dry Cask Savings (\$M)	Total Capital Expense (\$M)	Interest Expense (\$M)	Net Savings ₁ (\$M)	Average Annual Net Savings per Reactor ₁ (\$M)	Estimated Annual Savings per Reactor Available to Utilities ₂ (\$M)	Net Present Value (\$M)	Internal Rate of Return (%)	Reactor-Years Using Higher Enrichment
60 Year Operating Life Cases										
New BWR Fuel Design	643	244	143	44	700	2.89	1.13	31.78	13.0%	242
PWR High Burnup + New BWR Fuel Design	2480	1034	321	93	3099	3.66	1.46	251	19.2%	846
PWR High Burnup + Existing BWR Design	2069	871	291	84	2565	3.19	1.23	195	18.2%	804
PWR High Burnup	1837	790	240	70	2316	3.83	1.45	179	18.7%	604
PWR Moderate Burnup	847	409	212	67	977	1.93	0.45	11	10.7%	507
PWR Moderate Burnup with transition to High Burnup	1647	712	240	70	2048	3.39	1.24	119	15.6%	604
PWR + BWR Delay Capital Investment Until High Burnup proven	2120	867	311	106	2571	3.62	1.38	160	17.6%	711
¹ Fuel + Dry Cask - Capital - Interest ² Fuel - Capital - Interest - Estimated Fabrication Cost Impact										

Table 7: Summary of Results for the U.S. Fleet Based on an Assumed 60-Year Operating Life

	Fuel Cost Savings (\$M)	Dry Cask Savings (\$M)	Total Capital Expense (\$M)	Interest Expense (\$M)	Net Savings ₁ (\$M)	Average Annual Net Savings per Reactor ₁ (\$M)	Estimated Annual Savings per Reactor Available to Utilities ₂ (\$M)	Net Present Value (\$M)	Internal Rate of Return (%)	Reactor-Years Using Higher Enrichment
80 Year Operating Life Cases ₃										
New BWR Fuel Design	1734	966	157	47	2496	2.96	1.24	130	17.3%	844
PWR High Burnup + New BWR Fuel Design	9433	3540	342	98	12533	4.83	2.28	577	22.1%	2594
PWR High Burnup + Existing BWR Design	7882	2951	305	87	10441	4.05	1.89	461	21.2%	2578
PWR High Burnup	6687	2574	252	71	8938	5.11	2.31	384	21.2%	1750
PWR Moderate Burnup	2736	1365	230	70	3802	3.44	1.22	99	13.9%	1105
PWR Moderate Burnup with transition to High Burnup	7374	2824	254	71	9871	5.99	2.71	334	18.7%	1648
PWR + BWR Delay Capital Investment Until High Burnup proven	11890	3356	245	127	14875	6.29	3.60	650	24.3%	2366
¹ Fuel + Dry Cask - Capital - Interest ² Fuel - Capital - Interest - Estimated Fabrication Cost Impact										

Table 8: Summary of Results for U.S. Fleet Based on an Assumed 80-Year Operating Life

3 REGULATORY REVIEW

The degree to which existing regulations and guidance will need revision, or new regulatory requirements will have to be established and new guidance developed, depends on the level of departure from existing fuel designs. The figure below illustrates the current and potential future enrichment and burnup design limits.

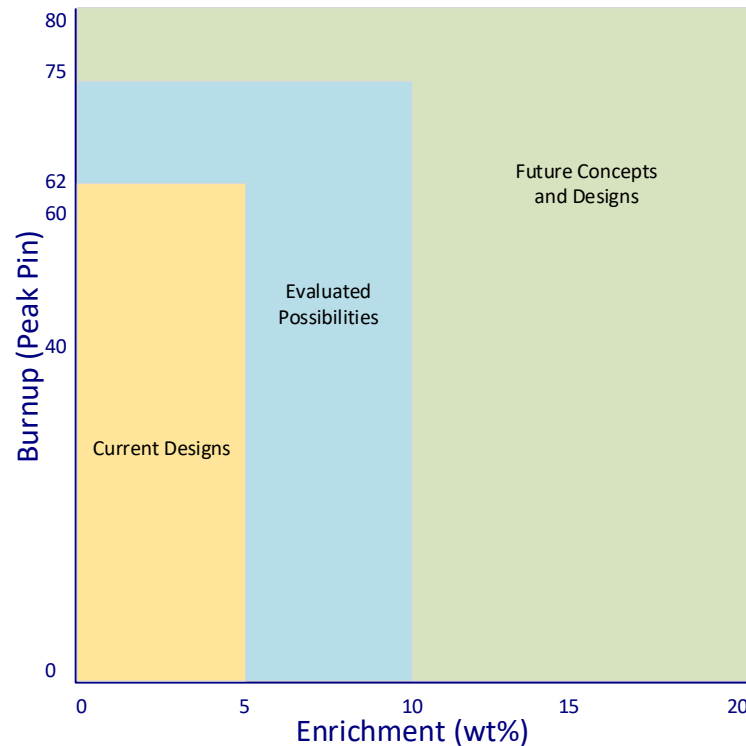


Figure 2: Comparison of Design Ranges

The regulations in Appendix A, “General Design Criteria for Nuclear Power Plants,” to U.S. Title 10 of the Code of Federal Regulations (CFR) Part 50, “Domestic Licensing of Production and Utilization Facilities,” provide principal design and performance requirements. Rulemaking and other regulatory guidance changes will be needed to support the expansion of the existing regulatory framework. The safety performance protected by the general design criteria (GDC) related to fuel design and overall fuel performance under normal and accident conditions as required per 10 CFR 50.46 will be maintained. In addition to the regulatory guidance related to the GDC, the utilization of fuel designs with higher burnup and higher enrichment combinations may also affect regulations associated with nuclear fuel transportation, material control and accounting, and physical protection.

One of the purposes of this white paper is to highlight the regulatory issues that need to be addressed to allow the manufacture and use of current UO_2 fuel types with higher burnup and higher enrichment combinations. This review addresses enrichments up to 10 wt% U-235. Individual licensees may elect to license their facilities at somewhat lower levels. The following sections provide an initial review of the associated regulations and design criteria pertaining to each stage of the fuel cycle.

3.1 Fuel Enrichment Facilities

U.S. fuel enrichment facilities are licensed under 10 CFR 70, “Domestic Licensing of Special Nuclear Material.” A review of the applicable regulations, particularly all of 10 CFR 70, shows that there is no specific restriction on the level of enrichment of special nuclear material (SNM) that a fuel enrichment facility is allowed to possess onsite. Specifically, a review of 10 CFR 70.24, “Criticality Accident Requirements,” does not include a limit on enrichment as is found in the corresponding regulation for nuclear power plants, i.e., 10 CFR 50.68, “Criticality Accident Requirements.” In fact, the NRC has already issued a license to Centrus for a fuel enrichment facility, the American Centrifuge Plant, which permits enrichment up to 10%; however, this facility has not been constructed. The NRC has also issued a license to General Electric-Hitachi Global Laser Enrichment for a fuel enrichment facility, which permits enrichment up to 8%.

It is expected that currently licensed fuel enrichment facilities will request license amendments to modify existing facilities to accommodate enrichments greater than 5.0 wt% U-235. The only currently operating U.S. fuel enrichment facility has an operating license that allows it to enrich up to 5.5 wt% U-235. In order for this facility to increase enrichment up to 10 wt%, the licensee would be required to apply to the NRC for an amendment to its license. As part of this amendment process, the licensee, at a minimum, would need to revise its Nuclear Criticality Safety (NCS) and Integrated Safety Analysis (ISA) calculations and evaluations to reflect the effects of higher enriched uranium in the facility and demonstrate compliance with NRC requirements. The licensee would also need to assess whether increasing the enrichment from the licensed limit of 5.5 wt% up to 10 wt% U-235 would necessitate a change to the conclusions in the NRC’s Environmental Impact Statement issued as part of the licensing of the facility. As a result of these analyses, some modification of the facility may be necessary. According to statements from senior company officials, all this can be accomplished. Getting approval to modify an existing license to enrich uranium up to 10 wt% U-235 is expected to take 12-18 months and require an environmental review.

The conclusion of this review is that there are no regulation or rulemaking changes, including criticality protection rules, needed to allow U.S. fuel enrichment facilities to enrich fuel up to 10 wt% U-235. Licensing U.S. fuel enrichment facilities for increased enrichments from 10 wt% U-235 would not require revisions or changes to the existing regulations. Although, NRC may still find a need to issue applicable guidance for its staff review of licensee or applicant submittals.

3.2 Fuel Fabrication Facilities

Fuel fabrication facilities are licensed under 10 CFR 70, “Domestic Licensing of Special Nuclear Material.” A review of the applicable regulations, particularly all of 10 CFR 70, shows that there is no specific restriction on the level of enrichment of special nuclear material that a fuel fabrication facility is allowed to possess on site. Specifically, a review of 10 CFR 70.24, “Criticality Accident Requirements,” does not include a limit on enrichment as is found in the corresponding regulation for nuclear power plants, i.e., 10 CFR 50.68, “Criticality Accident Requirements.”

The currently operating fuel fabrication facilities have operating licenses that allow them to produce fuel with an enrichment of up to 5 wt% U-235. In order to fabricate fuel enriched up to 10 wt% U-235, the licensee would have to apply to the NRC for an amendment to its license. As part of this amendment process, the licensee, at a minimum, would need to revise its Nuclear Criticality Safety (NCS) and Integrated Safety Analysis (ISA) calculations and evaluations to reflect the effects of higher enriched

uranium in the facility. The licensee would also need to assess whether increasing the enrichment of 5 wt% up to 10 wt% U-235 would necessitate a change to the conclusions in the NRC's Environmental Impact Statement issued as part of the licensing of the facility. As a result of these analyses, some modification of the facility may be necessary. According to representatives of these facilities, the actions required to obtain an amendment from the NRC are achievable. Note, these facilities have relatively small throughput and may not be able to support fleet-wide implementation of high enrichment without facility modifications.

Additionally, there are two U.S. facilities licensed to fabricate highly enriched fuel from existing HEU inventories, primarily for national defense use. The Nuclear Fuel Services facility (Erwin, TN) and BWXT Nuclear Operations Group plant (Lynchburg, VA) currently produced fuel for reactors requiring greater than 5.0 wt% U-235 (e.g., test, medical isotope and research reactors). These facilities produce fuel containing both high and low-enriched uranium, for use in the U.S. Naval Reactors program. They also blend down HEU to lower enrichments, which can be used for applications such as non-power reactors, as well as for LEU for use in existing LWRs. With their Category I fuel facility licenses; these facilities could produce fuel for HALEU reactors. Depending on the fuel manufacturing planned, these two sites might need only minor license amendments or none at all, to manufacture HALEU fuel.⁸

The conclusion of this review is that there are no regulation or rulemaking changes, including criticality rules, required to allow fuel fabrication facilities to make fuel with enrichments up to 10 wt% U-235. Licensing facilities for reactor fuel fabrication operations with enrichments up to 10 wt% U-235 would not require revisions or changes to the existing regulations. That being said, NRC might find a need to issue applicable guidance for its staff review of licensee or applicant submittals.

3.3 Fuel Transportation

The principal assurance of safety in the transport of nuclear materials is the packaging, which must mitigate against foreseeable accidents. Typical packaging of uranium hexafluoride (UF₆) consists of an inner steel cylinder that acts as a containment vessel, and an outer protective overpack. The overpack provides thermal protection to prevent overheating of the UF₆, which can cause hydraulic failure of the cylinder. The overpack also protects the cylinder from impacts. Unenriched UF₆ may be transported in bare cylinders, without the protective overpack, as authorized in DOT regulations. Protective overpacks are typically required only for the transport of enriched (fissile) UF₆. Design and performance standards for fissile UF₆ packages are stated in 10 CFR 71, and design and performance standards for non-fissile UF₆ packages appear in DOT regulations. ANSI N14.1 and USEC-651 contain information regarding overpacks. Choice of specific design features (e.g., overpacks) to meet regulatory standards is left to designers.

Uranium hexafluoride feed can continue to be transported from the conversion facility to the enrichment facility using currently approved cylinders; however, transporting uranium hexafluoride enriched above 5.0 wt% from the enrichment facility to the fuel fabrication facility presents a regulatory challenge. There is no U.S. DOT-approved, commercially viable cylinder (i.e., 30-inch and greater inner diameter) or overpack for material that is enriched to greater than 5.0 wt% U-235. Currently, shipments of uranium hexafluoride are made in 30B cylinders and UX-30 overpacks that are certified for material up to 5.0 wt% enrichment pursuant to Tables 2 and 3 of 49 CFR 173.417.

⁸ NEI, Addressing the Challenges with Establishing the Infrastructure for the front-end of the Fuel Cycle for Advanced Reactors, 2018.

Cylinders designed to hold uranium hexafluoride are qualified under DOT regulations; see 49 CFR 173.420 uranium hexafluoride (fissile, fissile excepted and non-fissile). This regulation applies to the packaging and shipment of any quantities greater than 0.1 kg of fissile, fissile excepted, or non-fissile uranium hexafluoride. It also includes requirements on the design, fabrication, inspection, and the testing and marking of the packages as well as the applicable Codes and Standards for manufacturing the cylinder under ANSI N14.1. ANSI N14.1 specifies the design and fabrication of the UF₆ cylinder. ANSI N14.1 - Table 1, "Standard UF₆ Cylinder Data," provides details for the various UF₆ cylinder models and indicates the maximum enrichment allowed for each model. Additionally, NUREG-1609, Standard Review Plan for Transportation Packages for Radioactive Materials, discusses that for the 30-in. cylinder, the UF₆ enrichment must not exceed 5.0 wt% U-235, along with NUREG 1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel," that states the initial enrichment shall not exceed 5.0 wt% U-235 for the licensing basis limit.

In addition to the DOT rules, containers must also meet the NRC requirements in 10 CFR Part 71. Specifically, 10 CFR 71.55(g)(4) states that packages containing uranium hexafluoride only are excepted from the requirements of paragraph (b) of the section provided that the uranium is enriched to not more than 5 wt% uranium-235. These provisions provide the necessary regulatory requirements to preclude an inadvertent criticality in the package. For enrichments above 5 wt% U-235, the exception provided in 10 CFR 71.55(g) for uranium hexafluoride packages will no longer apply. 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 wt% under §71.55(b) or §71.55(c). Merits of a new or modified design that included special design features would be reviewed and approved under the provisions of §71.55, including §71.55(c). Therefore, if the industry moves to enrichments greater than 5 wt% U-235, fuel shippers would need to get approval for new packages that would meet the normal fissile material package standards in §71.55(b), or could include special design features that would enhance nuclear criticality safety for transport for approval under the provisions of §71.55(c).

Daher-TLI has developed a replacement for the current 30B system, the DN-30. The DN-30 has been under development for the last 10 years. The EU regulatory authority (France) completed their review and issued approval for the DN-30 in December 2018. The initial license allows fuel enriched to 5.0 wt% U-235. Daher-TLI has also had two pre-submittal meetings with the U.S. NRC and submitted the design for approval in August 2018 with an expected approval in mid-2019.

The company has also developed a modified version of the DN-30 capable of carrying fuel enrichments up to 20 wt% U-235. This design uses the same DN30 overpack with a modified UF₆ cylinder. The DN-30-20 cylinder contains stainless steel tubes that will incorporate neutron absorbers to support enrichments of 10 wt% U-235 or 20 wt% U-235. This configuration has a total capacity of 1600 kg, somewhat less than the current 30B (2267 kg). The configuration has been analyzed without credit for moderator exclusion, consistent with ANSI.N14.1.2001 and ISO 7195. Extending the license to these higher enrichments is not expected to require additional accident testing. The licensing would require a revised criticality safety analysis. A completed preliminary design demonstrated acceptable performance over a range of moderator configurations, including the presence of allowable impurity levels ("heels"). Establishing the final configuration(s), and submitting and obtaining regulatory approval is expected to take 12-24 months. This design provides a viable option to addressing the transportation of UF₆, enriched over 5.0 wt% U-235, in a timeframe consistent with an initial reload in 2026.

Packaging designed to hold new fuel assemblies is approved under DOT and NRC regulations; see 49 CFR 173.417, “Authorized Fissile Materials Packages” and 10 CFR 71 “Packaging and Transportation of Radioactive Material.” This packaging is designed to transport fissile material that does not meet the fissile material exemption standards in 10 CFR 71.55 and has a total radioactivity less than an A1 or A2 quantity pursuant to 49 CFR 173.435 (e.g., fresh, unirradiated nuclear reactor fuel). For new fuel assemblies, typical packaging consists of a metal outer shell, closed with bolts and a weather-tight gasket. An internal steel strongback, shock-mounted to the outer shell, supports one or two fuel assemblies fixed in position on the strongback by clamps, separator blocks, and end support plates. Depending upon the type of fuel, neutron poisons are sometimes used to reduce reactivity. If the package is used to transport individual fuel rods, a separate inner container is often employed. The contents of the package are unirradiated uranium in fuel assemblies or individual fuel rods. Because the majority of these packages are for commercial reactor fuel, the uranium is typically in the form of Zircaloy-clad uranium dioxide pellets. The principal function of the package is to provide criticality control. The metal outer shell of the packaging retains the assemblies within a fixed geometry relative to other such packages in an array and provides impact and thermal protection. Shielding requirements are not significant because of the low radioactivity of unirradiated fuel.

The conclusion of this review is that the licensing basis of current UF₆ package designs restricts enrichment to less than 5.0 wt% U-235. Also, new fuel shipping packages are currently designed to accept fuel enriched up to 5.0 wt% U-235. Container design modification and re-licensing would be required to ship uranium hexafluoride and fresh fuel in excess of 5.0 wt% U-235.

3.4 Criticality Issues

A significant factor in the licensing of any enrichment or fabrication facility is criticality. As licensees look for ways to optimize facilities and storage/transport packages, computer software methods and data used in establishing the criticality safety of systems with fissile material become more important.⁹ For enrichments up to 11 wt% U-235, criticality benchmark data is described in NMSS-0007, an NRC guidance document. NMSS-0007 identified the need to develop and confirm the adequacy of methods, analytical tools, and guidance for criticality safety software to be used in licensing nuclear facilities.

“Computer codes used for criticality calculations must be benchmarked against critical experiments that represent the specific fissile materials, configurations, moderation, and neutron-poisoning conditions that represent the facility being licensed. However, it is well recognized that existing critical benchmark experiments will never precisely match these conditions. In addition, there are fewer benchmark experiments that are available at higher enrichment ranges [e.g., between 5 to 20 percent and lower-moderation (i.e., H/X, where H is hydrogen and X is fissile media)] ranges, that could be of future interest to potential applicants. Methods are needed to extend the range of applicability of current benchmark experiments via sensitivity/uncertainty (S/U) analysis techniques.”

NMSS-0007 goes on to state that:

“NMSS has performed extensive work with Oak Ridge National Laboratory (ORNL) to further develop criticality safety computer codes [e.g., Standardized Computer Analyses for Licensing Evaluation (SCALE)] to address these challenges. The final reports for the S/U methods were

⁹ Resolution of Generic Safety Issues: NMSS-0007. Criticality Benchmarks Greater than 5% Enrichment (Rev. 2) (Section 6 of NUREG-0933, Main Report with Supplements 1–34) December 2011.

published in November 1999 as Volumes 1 and 2 of NUREG/CR-6655. The reports cover the following subjects: (1) methodology for defining range of applicability including extensions of enrichments from 5 to 11 percent; (2) test applications and results of the method; (3) test application for higher enrichments using foreign experiments; (4) feasibility study for extending the method to multidimensional analyses, such as transport casks and reactor fuel.”

NMSS-0007 concludes that the results of the test applications of the ORNL methods show:

“.... for simple geometries with neutron spectra that are well moderated (high H/X), benchmark experiments at 5 percent enrichment are applicable to calculations up to 11 percent enrichment.”

For enrichments above 11 wt% U-235, the Oak Ridge National Laboratory (ORNL) “method provides sensitivity and uncertainty information, to help designers allow adequately large margins to cover the lack of benchmark validation.” Guidance to the NRC staff is provided in Fuel Cycle Safety and Safeguards-Interim Staff Guidance-10, Revision 0¹⁰, which clarifies the minimum margin of subcriticality for safety required for a license application or an amendment request under 10 CFR Part 70, Subpart H.

Criticality accident requirements for the spent fuel pool of a nuclear power plant are found in 10 CFR 50.68 or 10 CFR 70.24. These regulations specify requirements for licensees to maintain either a criticality monitoring system (10 CFR 70.24) or design margin to criticality accidents (10 CFR 50.68). 10 CFR 50.68 includes acceptance criteria which ensure that adequate safety margins are maintained and requires the maximum enrichment of fresh fuel to be limited to 5 wt% U-235. This additional limitation does not protect the criticality safety margins, since many other independent parameters impact safety margins. These include, among others, the presence of integral neutron poisons in the fuel assembly, poison inserts in the storage rack and fuel storage configurations. The specific combinations of parameters which do protect criticality safety margins are described in each facility’s licensing bases. With the potential need to increase enrichment levels for both existing commercial LWR reactors and advanced reactors, the appropriate action is to remove specific enrichment limits from 10 CFR 50.68. Retention of the necessary enrichment limits in the facility licensing bases, including Technical Specifications, provides adequate protection for criticality safety margins. Also, the use of exemption requests as a parallel path contingency to support the timeline for 10 CFR 50.68 rulemaking is an option.

In addition to the regulations and guidance documents described above, several guidance documents were identified wherein the use of enrichments up to 10% or the burnup assumptions would need to be addressed.

Section 7.5.2 of NUREG-1536¹¹ states, in part, “Although the burnup of the fuel affects its reactivity, many criticality analyses have assumed the cask to be loaded with fresh fuel (the fresh fuel assumption). Alternatively, the NRC staff has provided guidance for limited burnup credit for intact fuel. This guidance is currently limited to burnup credit available from actinide compositions associated with UO₂ fuel of 5.0 wt% or less enrichment that has been irradiated in a PWR to an assembly-average burnup value not exceeding 50 GWd/MTU [increased to 60 GWd/MTU in draft NUREG-2215] and cooled out-of-reactor for a time period between 1 and 40 years.”

With respect to criticality at the reactor sites, there is an opportunity to improve the analyses by relaxing a number of excessive conservatisms. Based upon a review of the NRC sources and discussions

¹⁰ USNRC, Fuel Cycle Safety and Safeguards-Interim Staff Guidance-10, Revision 0 (FCSS-ISG-10, Revision 0), ML061650370

¹¹ USNRC, NUREG 1536 Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility, Revision 1, July 2010

with several subject matter experts, it is clear that numerous excessive conservatisms have crept into these criticality analyses which do not exist in other areas of design, analysis, and operation. This could potentially allow higher enrichments above 5.0 wt% U-235 to be utilized without significant redesign of structures, systems, or components. These conservatisms could include, for example, additional credit for fission product poisons or credit for BWR burnup beyond peak reactivity. Additionally, the design of cores using higher enriched fuel would also require increased concentrations of neutron poisons to meet core limits, and fuel will achieve higher burnup, which also reduces reactivity. The additional poison concentrations and higher burnup are expected offset the reactivity effects of higher enrichment. As is the current case, some assemblies will not meet the enrichment burnup requirements, so storage in highly poisoned racks or in checkerboard or other configurations will be required.

3.5 Safety Analysis (High Burnup & High Enrichment)

This topic considers both an increase in pellet enrichment and an increase in fuel rod average burnup beyond currently approved limits. Many utilities need the combination of increased fuel burnup and increased enrichment to achieve their economic goals. The discussion below focuses on the issues identified during the regulatory review that would need to be addressed to support an enrichment increase up to 10 wt% U-235 and an incremental increase in burnup to 75 GWd/MTU range.

Proposed rulemaking for 10 CFR 50.46c would limit fuel rod burnups due to NRC concerns with fuel fragmentation after cladding rupture. It is assumed that the fuel fragmentation question will be addressed by a licensee seeking higher fuel rod burnups even if rulemaking does not proceed. There are efforts currently underway to address this concern. EPRI is conducting research that is expected to result in a linear heat rate limit versus burnup to preclude fuel fragmentation. Additionally, fuel vendors are looking at how calculations can demonstrate that high burnup rods do not rupture following a LOCA. Reactivity insertion accidents will also need to be evaluated. It is expected that most sites will not exceed PCMI or temperature limits, so fuel fragmentation is not expected to be an issue. EPRI is also evaluating the industry needs in this area.

Utilities (with the exception of those non-GDC plants who cite requirements specific to their licensing basis) and vendors typically use NUREG-0800¹² to demonstrate how they meet General Design Criteria 10 (see 10 CFR 50, Appendix A). The software and methods used to demonstrate compliance are approved by the NRC. For PWRs, the NRC limits the applicability of codes and methods to no more than 62 GWd/MTU. Resolution of the fuel fragmentation issue is needed to permit higher burnups. There may also be an implied limitation of 5.0 wt% U-235 enrichment for codes and methods, if not specific limitations and conditions, which would require them to be relicensed for a higher enrichment. Any necessary changes to the current fuel performance codes and methods should be able to support a modest burnup increase in the 67 - 68 GWd/MTU range with low regulatory risk. Burnups beyond 68 GWd/MTU may require additional fuel performance data to validate the software models. At the EPRI sponsored workshop on Burnup and Enrichment in December 2018, the NRC was generally supportive of moderate burnup increases and open to a phased approach to higher burnup increases afterwards.

There are several Regulatory Guides (Reg Guide) that contain fission product data for dose analysis calculations. For example, Reg Guide 1.183 and Reg Guide 1.195 both state that the release fractions listed in the guides have been determined to be acceptable for use with currently approved LWR fuel with a peak rod burnup up to 62 GWd/MTU provided that the maximum linear heat generation rate

¹² NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition"

does not exceed 6.3 kW/ft peak rod average power for burnups exceeding 54 GWd/MTU. Existing regulations limit the use of these release fractions to enrichments below 5 wt% U-235. These regulatory guides will need to be revised to extend their applicability to support higher burnups and enrichments for implementation of higher lead rod burnup limits.

Addendum 1 to Volume 1 of NUREG-1437, “Generic Environmental Impact Statement for License Renewal of Nuclear Power Plants,” (NRC 1999), states that the environmental impacts would be small for transporting spent fuel enriched with up to 5.0 wt% U-235 with an average burnup for the peak rod of up to 62 GWd/MTU. It also states that, if peak fuel burnup is projected to exceed 62 GWd/MTU and/or fuel is enriched to higher levels, license renewal applicants must submit an assessment of the implications for the environmental impact values reported in 10 CFR 51.52. Utilities would need to consider whether any licensing actions are needed for current operating licenses or renewed licenses with respect to environmental impact statements for higher burnups and enrichments.

As discussed in 10 CFR 50.61 (Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events), calculations for vessel fluence include assumptions for core loading patterns. Vessel fluence only considers neutrons over 1 MeV, but it is not clear how the energy spectrum will shift due to the combination of higher burnup, increased enrichment, and more/different burnable absorbers. From a programmatic standpoint, even if the end of life fluence is increased only slightly, the plant’s current pressure-temperature curves (which reside in either the plant Technical Specifications or the Pressure Temperature Limits Report for ITS plants) will need to be re-analyzed to determine if the valid Effective Full Power Year limit has decreased. This is especially a concern for plants considering second license renewal (SLR).

In the Standard Technical Specifications (4.2.1, Fuel Assemblies), allowable fuel pellets are those with an initial composition of natural or slightly enriched uranium dioxide (see Ref. 25, for example). Therefore, no changes to this Standard Technical Specification would be needed.

In conclusion, there are several areas, such as fission product inventories for dose analyses and the licensing of fuel performance codes, which will need to be addressed as change management items. While there are efforts currently underway to address fuel fragmentation, resolution of the NRC’s fuel fragmentation concern is critical to moving forward with increased fuel rod burnups.

3.6 Material Control and Accounting (MC&A)

An MC&A program is the way a facility operator conducts a sustainable, effective graded safeguards program for the control and accounting of nuclear materials, to detect and deter theft and diversion of SNM. The MC&A program implements a defense-in-depth approach to ensure that all accountable nuclear materials are in their authorized location and being used for their intended purposes, such that single component failures will not result in significant vulnerabilities. The goal of MC&A is to (1) maintain current knowledge of the location of SNM and resolve any discrepancies and (2) prevent undetected access resulting in unauthorized changes to quantities of SNM at a site that might ultimately result in diversion of SNM. MC&A also complements international treaty obligations by accounting for SNM at facilities and reporting the quantity of SNM at those facilities, as appropriate, to the International Atomic Energy Agency (IAEA). As provided by 10 CFR Part 70.22(b), the rule applies to each applicant for a license to possess special nuclear material, to possess equipment capable of enriching uranium, to operate an uranium enrichment facility, to possess and use at any one time and location special nuclear material in a quantity exceeding one effective kilogram (with exceptions for certain end users licensed

under other regulations). The entity seeking a license must provide an application which contains a full description of the applicant's program for control and accounting of such special nuclear material or enrichment equipment that will be in the applicant's possession under license to show how compliance with the requirements of the applicable 10 CFR 74 requirements are accomplished.

NRC provides clear guidance on compliance with 10 CFR 74 (Material Control and Accounting of Special Nuclear Material) in the form of two documents prepared by the NRC, NUREG-1065 and NUREG-1280 for facilities involved with Category III and Category I SNM quantities defined in 10 CFR 70.4, respectively. These documents provide guidance on establishing a Fundamental Nuclear Control Plan (FNMCP) that describes how the requirements of 10 CFR 74.31 section (c) System Capabilities (1) through (8) are met. MC&A nuclear measurement non-destructive assay (NDA) is described in chapter 3 of NUREG-1065 and chapter 4 of NUREG-1280. NDA based on interrogation of U-238 (or total U) are not directly affected by enrichment changes of the type being contemplated. However, NDA systems based on interrogation of U-235 will be directly affected by an enrichment change and therefore those systems may need to be requalified.

In accordance with 10 CFR 74.31(b), each applicant for a license, and each licensee that, upon application for modification of its license, would become newly subject to the performance objectives of 10 CFR 74.31(a) of that section, shall submit an FNMC plan describing how the requirements of paragraph (c) of that section will be met. The FNMC plan shall be implemented when a license is issued or modified to authorize the activities being addressed in paragraph (a) of this section, or by the date specified in a license condition.

In conclusion, the regulations and associated guidance do not have assay-specific limits, and no rule changes in MC&A would be required to allow for an assay of up to 10 wt% U-235. However, a modification to the facility license may be needed, updating the FNMC plan, because NDA systems based on interrogation of U-235 will be affected when enrichment of U-235 is increased.

3.7 Physical Protection of HALEU Plants and Materials

10 CFR 70.22(k) requires license applicants seeking to possess SNM of 10kg or more of low strategic significance (except those who are licensed to operate a nuclear power reactor pursuant to 10 CFR 50) to include a physical security plan that demonstrates how the applicant will meet the requirements of 10 CFR 73.67(f). The plan must address how and where the material is to be stored, how access is controlled, and provisions for a watchman or offsite response force to respond to all unauthorized penetrations or activities. Additionally, written response procedures dealing with threats of theft of these materials must be established and maintained.

The physical protection requirements are generally graded based on the risk of the material being used for malevolent purposes. The principal RGs used in licensing Category I, II and III facilities are Reg Guide 5.52, "Standard Format and Content of a Licensee Physical Protection Plan for Strategic Special Nuclear Material at Fixed Sites (Other than Nuclear Power Plants)" (NRC, Rev. 3, 1994); Reg Guide 5.55, "Standard Format and Content for Safeguards Contingency Plans for Fuel Cycle Facilities" (NRC, 1978b); and Reg Guide 5.59, "Standard Format and Content for a Licensee Physical Security Plan for the Protection of Special Nuclear Material of Moderate or Low Strategic Significance" (NRC, Rev. 1, 1983).

This review did not identify any regulatory obstacles for the current licenses for fuel enrichment or fuel fabrication facilities when enriching fuel up to 6 -7 wt% U-235. Current licenses of fabrication and enrichment facilities allow for possession of SNM of low strategic significance which, by definition,

includes greater than or equal to 10,000 grams of SNM with enrichments up to 10 wt% U-235. Therefore, no modifications to the facilities' security plans and no revisions or changes to existing physical protection regulations would be required to support enrichment increases up to 10 wt% U-235.

3.7.1 HALEU in Transit

Performance objectives of the physical protection systems in transit are described in §73.67(g) for Category III materials. Category III SNM material, also referred to as SNM of low strategic significance, is defined as 10,000 grams or more of U-235 contained in uranium enriched above natural but less than 10 percent U-235. In ways similar to the fixed facility physical protection requirements, physical protection requirements for material in transit are also graded based on risk. The conclusion of this review is that there are no regulatory obstacles for HALEU in transit when increasing enrichments up to 10 wt% U-235.

4 CONCLUSIONS

There are a number of technical challenges that will need to be addressed to demonstrate compliance with existing safety criteria prior to implementing higher burnup and higher enrichment combinations with current fuel designs. One of the challenges associated with the initial deployment of higher burnup/enriched fuels for the current LWR fleet is that neither the designers nor fuel producers can proceed past a certain point without the other. Early in the process, developers of either cannot be certain that the other is actually going to reach commercial deployment. The length of time it takes to gain commercial support for funding, address technical and regulatory issues and then construct the necessary fuel cycle infrastructure creates special challenges for realizing these changes. In order to address these challenges it becomes imperative that the federal government and the industry work together to realize the benefits associated with the use of higher burnup/enriched fuels.

Each of the identified scenarios in the economic analysis discussed in Section 2.0, provides positive economic benefits for the U.S. nuclear fleet. The use of higher burnup fuel designs results in a reduction in high level waste and the corresponding investment in dry cask storage systems. Additionally, most PWRs could increase their operating schedule flexibility potentially resulting in fewer refueling outages and increased energy production; however, these benefits are plant dependent and not credited in the economic analysis. This project also requires a commitment from a number of stakeholders including fuel suppliers, plant operators, and government agencies. Fuel management benefits are dominated by fuel fabrication savings for utilities. Estimated fuel fabrication price increases are included in the analysis; however, the actual increases are subject to commercial negotiations. Commitment by at least sixteen large PWR commercial reactors is estimated to provide the minimum required return for the project.

To support this initiative, changes to the regulatory framework will be needed to change or eliminate the maximum enrichment limit of 5.0 wt% U-235. With the potential need to increase enrichment levels for both existing commercial LWR reactors and advanced reactors, removal of specific enrichment limits from 10 CFR 50.68 is the most appropriate action. The specific combinations of parameters that protect criticality safety margins are described in the facilities' licensing basis documents. These parameters include, among others, the presence of integral neutron poisons in the fuel assembly, poison inserts in the storage rack, borated water requirements, and fuel storage configurations. Retention of the necessary enrichment limits in the facility licensing basis, including Technical Specifications, provides adequate protection for criticality safety margins to support removal of the specific enrichment limit from 10 CFR 50.68.

Federal government support of development of Lead Use Assemblies (LUAs) to be used for testing will be needed. At present the only U.S. source for uranium enriched to greater than 5.0 wt% U-235 would be uranium produced from down-blending government owned high enriched uranium (HEU). Some test reactor fuel may be available to provide LUA quantities. While DOE could theoretically provide a limited source of greater than 5.0% enriched U-235 through down-blending, this option is constrained by the availability of HEU. Thus, this approach could be a stop-gap strategy but cannot be relied upon as a long-term fuel source.¹³

¹³ "Addressing the Challenges with Establishing the Infrastructure for the frontend of the Fuel Cycle for Advanced Reactors," Nuclear Energy Institute, January 2018.

U.S. government assistance will also be needed to amend facility licenses and make necessary modifications to support an increase in enrichment for both fuel enrichment and fuel fabrication facilities in addition to license changes associated with transportation issues. Amending an existing license to enrich up to 10 wt% U-235 is expected to take 1-1.5 years and the estimated time to modify a facility to handle enrichments from 5-10 wt% U-235 is 1 year.

NRC staff has recognized through stakeholder interactions that requests for increased fuel burnups, beyond the current licensed limits, are very likely to occur. As such, it has begun assessing the current knowledge and experimental data associated with high burnup fuels, beginning with NUREG/CR-6744, "Phenomenon Identification and Ranking Tables (PIRTs) for Loss-of-Coolant Accidents in Pressurized and Boiling Water Reactors Containing High Burnup Fuel."¹⁴ NRC staff has also indicated that it expects industry decisions on targeted maximum burnups will direct future plans in regards to an associated increase in enrichment to efficiently achieve the desired burnup. So, along with work associated with increased fuel burnups, the staff is beginning an assessment of what enrichment increase the current knowledge and database could also support with respect to licensing accident tolerant fuels.¹⁵ Future work in this area will identify the next steps and implementation strategy.

Issue	Potential Path Forward
Regulations limiting enrichment to 5 wt% U-235.	Initiate new rulemaking to eliminate the limit of 5.0 wt% U-235 from 10 CFR 50.68 based on retention of the necessary enrichment limits in the facility licensing bases, including Technical Specifications, providing adequate protection for criticality safety margins. 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 wt% under §71.55(b) or §71.55(c). Merits of a new or modified design that included special design features would be reviewed and approved under the provisions of §71.55, including §71.55(c).
Resolution of fuel fragmentation issue	Ongoing research is expected to result in a linear heat rate limit versus burnup to preclude fuel fragmentation. Additionally, tests are planned to demonstrate that high burnup rods do not rupture following a LOCA. Reactivity insertion accidents will also need to be evaluated.
New fuel shipping packages are currently designed to accept fuel enriched up to 5.0 wt% U-235.	Container design modification and re-licensing would be required to ship fresh fuel in excess of 5.0 wt% U-235.

Table 9: Key Issues and Path Forward

¹⁴ ADAMS Accession No. ML013540584

¹⁵ ADAMS Accession No. ML18261A414

5 LIST OF OTHER REFERENCES

- 1) ANSI N14.1-2001
- 2) 10 CFR Part 50, Domestic Licensing of Production and Utilization Facilities
- 3) 10 CFR Part 51, Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions
- 4) 10 CFR Part 70, Domestic Licensing of Special Nuclear Material
- 5) 10 CFR Part 71, Packaging and Transportation of Radioactive Material
- 6) 10 CFR Part 73, Physical Protection of Plants and Materials
- 7) 10 CFR Part 74, Material Control and Accounting of Special Nuclear Material
- 8) 10 CFR Part 75, Safeguards on Nuclear Material- Implementation of US/IAEA Agreement
- 9) 49 CFR Part 173.420, Uranium hexafluoride (fissile, fissile excepted and non-fissile)
- 10) Argonne National Laboratory web site. [Web.ead.anl.gov/uranium/guide/uf6/index.cfm](http://web.ead.anl.gov/uranium/guide/uf6/index.cfm)
- 11) NRC Website, <https://www.nrc.gov>
- 12) Resolution of Generic Safety Issues: NMSS-0007. Criticality Benchmarks Greater than 5% Enrichment (Rev. 2) (Section 6 of NUREG-0933, Main Report with Supplements 1–34) December 2011
- 13) DOE-STD-1194-2011, Nuclear Materials Control and Accountability, June 2011
- 14) NUREG 1520, Rev. 2, “Standard Review Plan for Fuel Cycle Facilities License Applications,” June 2015
- 15) U.S. Nuclear Regulatory staff guidance, Fuel Cycle Safety and Safeguards-Interim Staff Guidance-10, Revision 0 (FCSS-ISG-10, Revision 0), ADAMS Accession No. ML 061650370
- 16) NUREG-1280, Rev. 1, “Standard Format and Content Acceptance Criteria for the Material Control and Accounting (MC&A) Reform Amendment,” 10 CFR Part 74, Subpart E,” Issued April 1995 (for high enriched uranium facilities)
- 17) NUREG-1065 Rev. 2, “Acceptable Standard Format and Content for the Fundamental Nuclear Material Control (FNMC) - Plan Required for Low-Enriched Uranium Facilities,” Issued December 1995
- 18) NUREG-1748, “Environmental Review Guidance for Licensing Actions associated with NMSS Programs,” issued August 2003 (provides general procedures for the environmental review of licensing actions regulated by the Office of Nuclear Material Safety and Safeguards)
- 19) Draft NUREG 2215, “Standard Review Plan for Spent Fuel Dry Storage Systems and Facilities,” draft report for comment November 2017

- 20) NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel," March 2000
- 21) NUREG- 1437, "Generic Environmental Impact Statement for License Renewal of Nuclear Power Plants," (NRC 1999)
- 22) Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000
- 23) Regulatory Guide 1.195, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors," May 2003
- 24) NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition"
- 25) NUREG-1431, "Standard Technical Specifications — Westinghouse Plants"
- 26) USNRC, Fuel Cycle Safety and Safeguards-Interim Staff Guidance-10, Revision 0 (FCSS-ISG-10, Revision 0), ADAMS Accession No. ML061650370
- 27) ADAMS Accession No. ML013540584 - NUREG/CR-6744, "Phenomenon Identification and Ranking Tables (PIRTs) for Loss-of-Coolant Accidents in Pressurized and Boiling Water Reactors Containing High Burnup Fuel," December 2001.
- 28) ADAMS Accession No. ML18261A414 - Project Plan to Prepare the USNRC for Efficient and Effective Licensing of Accident Tolerant Fuels, September 2018.
- 29) "Strategy for Deployment of Advanced Fuels," RPT-3005895-000, September 28, 2011.
- 30) NUREG-1609, "Standard Review Plan for Transportation Packages for Radioactive Material," March 31, 1999.
- 31) 49 CFR Part 173.417, Authorized fissile materials packages



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