APPENDIX C

EVALUATION CRITERIA AND METRICS

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C. EVALUATION CRITERIA AND METRICS

The Charter for the Evaluation and Screening of Nuclear Fuel Cycle Options [Attachment 1, Appendix A] authorizes the Fuel Cycle R&D (FCR&D) Program to develop appropriate evaluation metrics for the evaluation and screening of nuclear fuel cycle options. In general, a very broad definition of these high-level criteria may include topics that are inherently not relevant at the level of a nuclear fuel cycle. Therefore, the Evaluation and Screening Team (EST) developed focused descriptions of the criteria that are appropriate for this evaluation of nuclear fuel cycle options considering the scope of the study as described in detail in Appendix A.

Content and Structure of Appendix C

This Appendix describes the development of the Evaluation Metrics and the methods used for developing the Metric Data to evaluate and screen each of the 40 fuel cycle Evaluation Groups listed in Appendix B. Appendix C is organized as follows:

One individual section (i.e., C-1 through C-9) is devoted to each of the nine Evaluation Criteria. Each section includes a definition of the Criterion, background discussion and considerations which went into the development of the metrics for the Criterion. Most details on the methodologies for calculation or estimation of each metric are included here while further details are provided in Appendix D as the Metric Data is developed. For each metric, the bin structure utilized for the Metric Data is provided. References for each section are listed at the end of the section.

The structure provides the entire discussion for each Criterion and the associated Evaluation Metrics, one criterion at a time. The EST developed the Evaluation Metrics, with input and review from groups external to the study, including government, industry, and universities, as discussed in Appendix A. The Independent Review Team reviewed and DOE-NE/5 approved the final set of Evaluation Metrics. The development of all Evaluation Metrics is presented here in Appendix C.

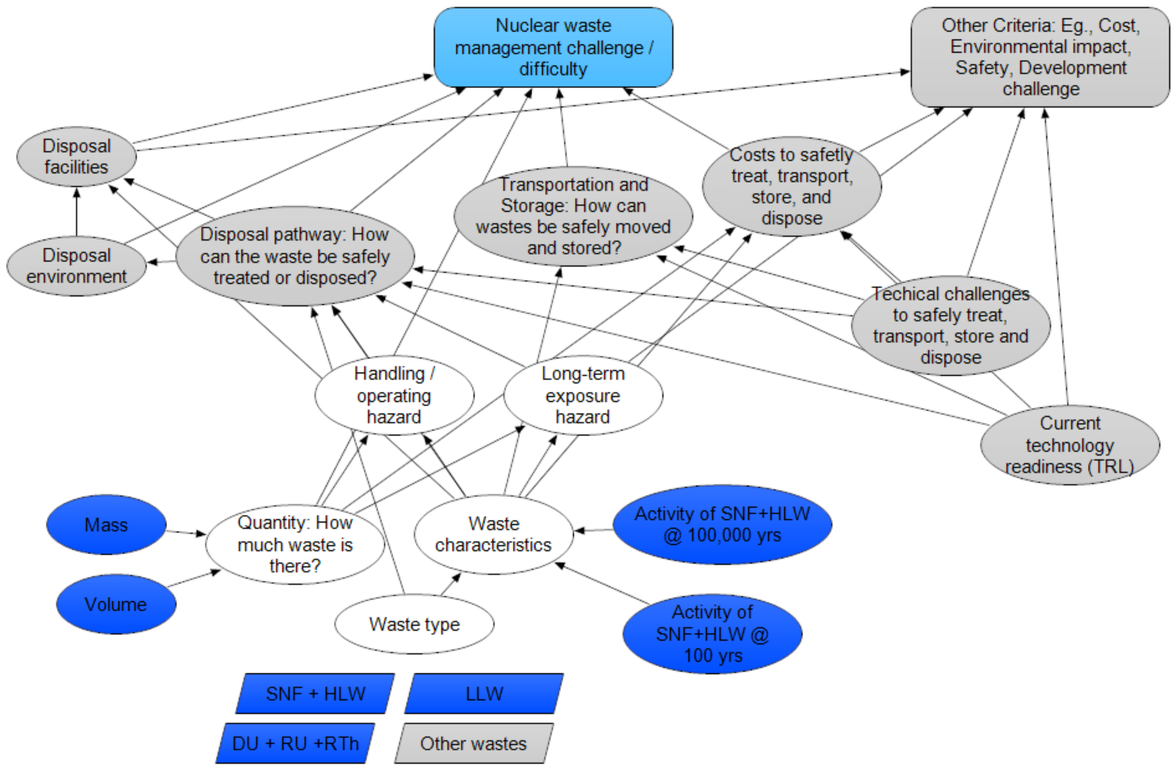
C-1. Nuclear Waste Management Criterion

The definition developed by the EST for the Nuclear Waste Management Criterion is:

Nuclear Waste Management - *A broad definition of nuclear waste management includes the safe and effective storage, transportation, and disposal of all radioactive material considered waste. For the purpose of this fuel cycle evaluation and screening, the assessment of the Evaluation Groups for the Nuclear Waste Management Criterion focused on the generation of the radioactive wastes requiring disposal, including any spent nuclear fuel, high-level waste, excess fuel material and low-level waste. Since adequate disposal capability is required by any fuel cycle, as described in Appendix A, in the context of this study the premise was that appropriate disposal would be available for any nuclear fuel cycle. By concentrating on waste generation, the study focused on the effects that fuel cycle options may have on the available disposal paths.*

C-1.1 Background on Nuclear Waste Management

The management of radioactive waste from nuclear energy production is an important consideration for civilian use of nuclear technology, and is often identified as a key issue for the future of nuclear energy. While all technologies create wastes that must be managed, the potentially concentrated radioactive nature of nuclear waste is unique – and results in a high level of societal concern on how it is managed. Figure C-1.1 is an influence diagram showing some of the key factors that affect the challenges of nuclear waste management, and the relationships among those factors. These factors are discussed below.



Note: Each oval represents a factor, element, or question related to nuclear waste management. Rounded rectangles represent different high level Evaluation Criteria for the Evaluation and Screening. Dark blue indicates factors for which Evaluation Metrics were defined, and white indicates factors related to nuclear waste generation that are strongly driven by the characteristics of fuel cycle. Grey indicates some of the other factors relevant to nuclear waste management that are not included explicitly in the Evaluation Metrics for this Criterion.

Figure C-1.1. Influence Diagram Showing Some of the Factors Affecting the Challenge of Nuclear Waste Management.

All nuclear energy production systems generate a variety of radioactive wastes: only the relative amounts and characteristics of the wastes differentiate between fuel cycles. Typical nuclear wastes (shown in the dark blue boxes at the bottom left of Figure C-1.1 under the oval labeled “Waste type”) include:

- **Spent Nuclear Fuel** – All nuclear energy systems use some sort of fissionable fuel, and create irradiated fuel. When this irradiated fuel is destined for disposal without reuse, it is designated ‘spent nuclear fuel’ (SNF). Intact SNF retains the physical configuration used in the reactor and contains all the fission, activation, and transmutation products (including actinide and transuranics) produced during irradiation as well as the unused portions of the initial fuel. SNF is typically highly radioactive and requires shielding to reduce potential radiation exposure of workers and the public during handling, storage, transport and disposal operations. SNF also generates significant decay heat from short-lived radionuclides.
- **High-Level Waste** –The Nuclear Waste Policy Act of 1982 defines high level radioactive waste (HLW) as “the highly radioactive material resulting from the reprocessing of spent nuclear fuel, including liquid waste produced directly in reprocessing and any solid material derived from such liquid waste that contains fission products in sufficient concentrations”. Transuranic elements (TRU) not recovered for use in the fuel cycle is accounted for under HLW. Most commonly, fissile materials that can be chemically separated are recovered during reprocessing to make new reactor fuel (e.g., ^{239}Pu in uranium-based fuel cycles and ^{233}U in thorium-based fuel cycles) – and these materials are only present in the HLW as small ‘processing losses’. Wastes may include minor actinides if they are not recycled. HLW is typically highly radioactive and also requires

shielding during handling, storage, transport and disposal operations. HLW may also generate significant decay heat from short lived radionuclides.

- **Low-Level Waste** – A variety of wastes with lower levels of radioactivity are produced in the nuclear fuel cycle. In the U.S. these wastes are referred to as 'low-level waste' (LLW), and are radioactive material other than SNF or HLW that the U.S. Nuclear Regulatory Commission (NRC) determines is LLW. LLW typically includes operational wastes with low levels of contamination such as used equipment, cleaning supplies, personal protection equipment, air filters, reagents, etc., as well as 'wastes deemed incidental to reprocessing'. LLW is categorized by radioactive material content into classes: A, B, C and 'greater than class C' (GTCC).
- **Depleted Uranium, Recovered Uranium, and Recovered Thorium** – Reactor fuel is often made with uranium enriched in the fissile isotope ^{235}U, i.e., the ^{235}U content is greater than that in nature. As a consequence, the enrichment process produces large quantities of uranium that is depleted in ^{235}U. This 'depleted uranium' (DU) is less radioactive than natural uranium. In reprocessing irradiated fuel for recycle (with the irradiated fuel destined for reprocessing designated "Used Nuclear Fuel", or UNF), the uranium in the fuel is often recovered. This 'recovered uranium' (RU) may be reused, re-enriched or disposed as waste. It is more radioactive than natural uranium due to short-lived uranium isotopes produced during irradiation and is contaminated with residual fission products not removed in processing. These features are also similar for recovered thorium (RTh), with the exception that the recovered thorium cannot be enriched since thorium has no fissile isotopes. A characteristic of DU/RU/RTh is the slow accumulation of radioactive decay products of uranium and thorium – which can result in the relative hazard level increasing over long times rather than decreasing as is common for most other radioactive materials.
- **Other Wastes** – There can be other wastes created in some nuclear energy systems. Examples include mixed LLW, where the LLW contains regulated toxic elements and is referred to as 'mixed waste', activated structural components, and "TRU waste" defined as waste containing more than 100 nanocuries of alpha-emitting transuranic isotopes per gram of waste with half-lives greater than 20 years, except for (A) high-level radioactive waste, (B) waste that the Secretary of Energy has determined, with concurrence of the Administrator of the Environmental Protection Agency, does not need the degree of isolation required by the disposal regulations, or (C) waste that the Nuclear Regulatory Commission has approved for disposal on a case-by-case basis in accordance with part 61 of title 10 Code of Federal Regulations (CFR). The Waste Isolation Pilot Plant (WIPP) project was authorized under Section 213 of the DOE National Security and Military Applications of Nuclear Energy Authorization Act of 1980 (Pub. L. 96-164; 93 Stat. 1259, 1265), to demonstrate the safe disposal of radioactive waste materials generated by atomic energy defense activities. Thus, by law, WIPP can accept only radioactive waste generated by atomic energy defense activities of the United States. Non-defense TRU waste is accounted for in the respective categories of HLW or LLW including GTCC waste depending on the specific constituents. As a result we are not considering TRU waste as a separate category.

SNF and HLW represent the most radioactive wastes, and for that reason are typically of most concern to the public and decision makers. The quantity and the characteristics of all wastes determine what type of disposal pathway is necessary, and the pathway and the facilities required to implement that pathway contribute to the overall challenge of nuclear waste management. This is illustrated in the influence diagram with the white ovals for "waste quantity" and "waste characteristics" and the arrows indicating that they influence the disposal pathway (and the pathway influences the required facilities and both affect the nuclear waste management challenge).

The characteristics of SNF and HLW (and to some extent GTCC) result in a disposal requirement for significantly more robust isolation as compared to LLW or DU disposal. While LLW is often the largest volume of waste, commercial facilities for shallow land burial of LLW exist. DU/RU (and RTh) can also represent a large mass of waste if these materials are not recycled, but has levels of radioactivity lower than either SNF or HLW, but potentially higher than LLW. Accordingly, three categories of waste were considered separately in the Evaluation and Screening. “Other wastes” as described above (shown in grey in the figure) are not considered.

Important Technical Considerations for Development of Nuclear Waste Management Metrics

The development of waste management metrics for use in the Evaluation and Screening was a joint effort between the Fuel Cycle Options (FCO) and Used Fuel Disposition (UFD) campaigns, with additional input from the Separations and Waste Forms (SWF) campaign in the DOE Office of Fuel Cycle Technologies. The following describes the perspectives of the three campaigns used in guiding the development of the metrics:

- **FCO -** From the perspective of the FCO campaign which is conducting this study, the metrics should be able to estimate waste management benefits of alternative nuclear energy systems relative to the Basis of Comparison (once-through use of uranium-based fuel in light-water reactors (LWRs) followed by disposal of the spent fuel and low-level wastes). Since the intent of the Evaluation and Screening is to identify the potential for substantial improvement between nuclear energy systems rather than minor enhancements, it is essential that any evaluation metrics reflect such potential. Because this evaluation is on complete nuclear energy systems at a functional level as described in Appendix A, and not on any of the potential implementing technologies, the metrics should represent inherent characteristics of the fuel cycle at this functional level.
- **UFD -** From the perspective of the UFD campaign, the metrics should be able to represent the important features of radioactive waste management but not attempt to characterize any specific disposal environment or any relationship between the fuel cycle and the characteristics of any particular disposal environment. The position of the UFD campaign is that an acceptable repository could be developed for any suitable geologic environment, and such relationships would be considered as repository designs are developed. This position influenced the scope of this Study, specifically leading to a decision not to explore the effects of different disposal environments on fuel cycle performance, as discussed in Appendix A (Section A-1.3.3)
- **SWF -** From the perspective of the SWF campaign, the metrics should represent the important features of waste generation, treatment and immobilization.

Many of the important issues for UFD and SWF are related to the technologies selected for implementation and are independent of fuel cycle characteristics. Evaluation metric development focused on waste management issues that reflected the inherent characteristics of the fuel cycles based on the functions performed and not on characteristics associated with any specific implementing waste management technologies. The important waste characteristics affecting nuclear waste management are evaluated at the functional level (i.e., characteristics related to generation of waste) than the details of separation technologies, waste form production, waste handling, storage and disposal. It is intended that the results of the fuel cycle Evaluation and Screening provide a context for future decisions on which technologies are pursued to support potentially promising fuel cycle options, and to inform R&D priorities. So, it is important that the fuel-cycle level waste metrics be able to connect to the important issues for technology R&D.

As shown in the Figure C-1.1 and discussed above, the quantity and characteristics of the waste produced are key contributors to the transportation, storage, and disposal challenges. Quantity can be measured in mass, volume or piece count, but these characteristics may be suitable or not for this study depending on whether they are inherent characteristics of the fuel cycle or dependent on the implementing technologies. For example, mass is often used as a measure for intact SNF, such as tons of fuel per GWe-yr. Volume is typically used for LLW, although volume can be specific to the technologies used in implementing a fuel cycle. Similarly, although HLW volume may be an important characteristic, HLW volume is also specific to the technologies used. Piece count is usually used for the number of fuel assemblies or packages that must be handled. The challenge was to develop metrics that would be informative on the issue of nuclear waste management yet not be related to any specific technology used in the fuel cycle.

There have been many evaluations of the important issues for SNF/HLW management [C-1.1 to C-1.9]. In consultation with the UFD campaign, and in consideration of the needs of the Evaluation and Screening study, it was determined that some of these issues could be used to determine applicable metrics for the criterion of nuclear waste management, while others were used to provide supporting information for other high-level criteria such as Financial Risk and Economics, or Development and Deployment Risk, illustrated by the grey shaded nodes in the influence diagram (Figure C-1.1).

C-1.2 Metric Development for the Nuclear Waste Management Criterion

Numerous factors are relevant to the overall challenge of managing nuclear waste, as is evident from the discussion above and from Figure C-1.1. Given the purpose of this Evaluation and Screening, the EST focused on identifying Evaluation Metrics that are both relevant to the nuclear waste management challenge and are directly tied to the fundamental characteristics of fuel cycles. The five Evaluation Metrics focus on waste type, waste quantity, and waste characteristics. They are shown by the four blue ovals and three highlighted waste categories in Figure C-1.1, and are described in additional detail below. In order to provide a consistent basis for comparing fuel cycles, all metrics were normalized to a per-unit-energy-generated basis.

As described above, nuclear waste was divided into several groups that have unique characteristics that impact waste management and reflect differences in the general approach for disposal. HLW and SNF require long-term isolation from the environment such as that provided by deep geologic disposal. There is no current U. S. specified approach for GTCC, and the potential GTCC generation for a future fuel cycle is highly dependent on the technology selected and facility designs. For the purpose of this evaluation, GTCC was grouped with low-level wastes. The level of isolation required by RU, DU and recovered thorium (RTh) is not known this time, and a disposal pathway for these materials has not been determined. As mentioned above, LLW is generally managed with near-surface disposal. As a result, these groups of wastes adequately discriminate waste stream characteristics for comparison of fuel cycles.

Waste quantity: Evaluation metrics were identified for the quantity of each waste stream. As discussed above, the mass of SNF+HLW is a characteristic of the fuel cycle at the functional level while volume of SNF+HLW is dependent on the technologies chosen to implement the fuel cycle. For this reason, one metric is:

- Mass of SNF+HLW per energy generated

Similarly, the other two metrics on waste quantity are:

- Mass of DU+RU+RTh disposed per energy generated
- Volume of LLW produced per energy generated

Waste characteristics*:* The high level of radioactivity of the SNF+HLW drives the requirements for shielding during handling / storage and isolation for disposal. One Evaluation Metric was selected for its ability to inform on the operational and handling challenges associated with SNF and HLW. While such a metric could be expressed in terms of heat generation, radiotoxicity or radiation field, all of these are caused by the total radioactivity, or "activity". As the activity (or any of these other measures) changes with time, it was necessary to select a representative time for operations and handling for disposal. Many waste management approaches include a significant delay prior to disposal to allow the initial very high level of activity to decay. A time of 100 years after discharge from a reactor was selected as being representative for the metric:

- Activity of SNF+HLW at 100 years per energy generated

Similarly, a total activity at a representative time for geologic disposal is relevant to a range of issues including the radiotoxicity of the disposed wastes, and is dependent on the fundamental characteristics of the fuel cycle. A time of 100,000 years was chosen to represent the long-term isolation challenge, leading to the metric:

- Activity of SNF+HLW at 100,000 years per energy generated

Summary

For the purpose of the fuel cycle Evaluation and Screening, the EST considered these metrics to adequately address the intent of the criterion 'Nuclear Waste Management' given that they are sufficient to inform on the quantity and the relative hazard level of nuclear waste generated by a fuel cycle. The metrics are appropriate for an evaluation of fuel cycles at the functional level and are not based on specific technologies. The following sections describe each metric for the Nuclear Waste Management Criterion in detail.

C-1.3 Mass of SNF+HLW Disposed per Energy Generated

Definition of Metric – This metric is the mass of SNF+HLW disposed per energy generated. The mass includes all heavy metals and fission products derived from the initial fuel materials but does not include any structural materials for the SNF or additional waste form materials for the HLW since these are determined by the technologies chosen for implementation.

The mass of SNF focuses on the fuel material and includes the remaining unused fuel material, fission products from nuclear fission of the fuel, and transmutation products from neutron absorption, but does not include any structural materials associated with the intact spent fuel. The mass of HLW includes any highly-radioactive materials separated from UNF that need to be disposed as waste. HLW mass of the separated materials destined for disposal is used instead of other characteristics that are dependent on choice of technology such as volume, which can vary from essentially no impact for consolidation of the recovered materials to a large increase in volume when a waste form matrix is used. The HLW can include the fission products and any unrecycled transmutation products (including actinides and TRU). Any unused fuel material separated from the UNF, the RU and/or RTh, is not included with the HLW, but is grouped with the DU using the same disposal path. The metric is the sum of SNF and HLW.

Appendix D-2.1 provides detailed information on how the Metric Data for this metric were calculated from the reactor and fuel cycle mass and isotopic data, and other information contained in Appendix B. The approach for developing the bin ranges for the Metric Data and the resulting bins are also described in Appendix D. For convenience, Table C-1.1 provides the bins developed for the Mass of SNF+HLW Disposed per Energy Generated metric.

Table C-1.1. Metric Bins for Mass of SNF+HLW Disposed per Energy Generated.

Bin ID	Data Range (t/GWe-yr)	Bin Description
A	< 1.65	Mass of SNF+HLW disposed per energy generated < 1.65 t/GWe-yr; 1.65 t/GWe-yr is approximately the HLW mass that would result from processing of LWR SNF to separate and recover all uranium
B	1.65 to < 3	Mass of SNF+HLW disposed per energy generated from 1.65 t/GWe-yr to < 3 t/GWe-yr
C	3 to < 6	Mass of SNF+HLW disposed per energy generated from 3 t/GWe-yr to < 6 t/GWe-yr
D	6 to < 12	Mass of SNF+HLW disposed per energy generated from 6 t/GWe-yr to < 12 t/GWe-yr
E	12 to < 36	Mass of SNF+HLW disposed per energy generated from 12 t/GWe-yr to < 36 t/GWe-yr; contains the basis of comparison (EG01)
F	≥ 36	Mass of SNF+HLW disposed per energy generated equals or greater than 36 t/GWe-yr

C-1.4 Activity of SNF+HLW (@100 years) per Energy Generated

Definition of Metric – This metric is the activity of SNF+HLW at 100 years per energy generated, where SNF+HLW is as defined for the metric of mass of SNF+HLW.

As discussed above, the activity of the SNF+HLW at 100 years is being used as the indicator for radiation and decay heat generation during handling and storage, including disposal operations. Once the activity has been obtained from the isotopic compositions of the SNF and HLW, informing on the radiation from these wastes, decay heat at that time can be derived and used to inform on the relative impact on geologic disposal. Geologic repository loading is often limited by decay heat, including the decay heat at time of placement or the integrated decay heat from closure up until the time of peak temperature, which is determined by the heat removal paths from the repository. While not an exact surrogate, the decay heat at 100 years gives a relative indication of the challenge of disposing of such wastes. In developing this metric, the EST in consultation with external input (from Campaigns and external meetings) concluded that the activity is a more fundamental characteristic of the generated wastes.

Appendix D-2.2 provides detailed information on how the Metric Data for this metric were calculated from the reactor and fuel cycle mass and isotopic data, and other information contained in Appendix B. The approach for developing the bin ranges for the Metric Data and the resulting bins are also described in Appendix D. For convenience, Table C-1.2 provides the bins developed for the Activity of SNF+HLW at 100 years per Energy Generated metric.

Table C-1.2. Metric Bins for Activity of SNF+HLW at 100 years per Energy Generated.

Bin ID	Data Range (MCi/GWe-yr)	Bin Description
A	< 0.67	Activity of SNF+HLW at 100 years < 0.67 MCi/GWe-yr.
B	0.67 to < 1.05	Activity of SNF+HLW at 100 years ≥ 0.67 MCi/GWe-yr and < 1.05 MCi/GWe-yr; the lower bound for this bin is approximately 50% less than the activity for the Basis of Comparison.
C	1.05 to < 1.60	Activity of SNF+HLW at 100 years ≥ 1.05 MCi/GWe-yr and < 1.60 MCi/GWe-yr; Bin C contains the Basis of Comparison and the bin range is approximately ±20% of the Basis of Comparison.
D	1.60 to < 2.00	Activity of SNF+HLW at 100 years ≥ 1.60 MCi/GWe-yr and < 2.0 MCi/GWe-yr; the upper bound for this bin is approximately 50% greater than the activity of the Basis of Comparison.
E	≥ 2.00	Activity of SNF+HLW at 100 years ≥ 2.00 MCi/GWe-yr.

C-1.5 Activity of SNF+HLW (@100,000 years) per Energy Generated

Definition of Metric – This metric is the activity of SNF+HLW at 100,000 years per energy generated, where SNF+HLW is as defined for the metric of mass of SNF+HLW.

The purpose of the metric is to inform on the long-term hazard from the disposal of SNF and HLW. The effectiveness and acceptability of SNF+HLW disposal is determined by the hazard that such disposal poses to the public and the environment, i.e., the biosphere. The radiotoxicity of any hazardous materials that reach the biosphere determines this hazard, usually expressed in terms of a dose to affected individuals. The materials released from a geologic repository and transported to the biosphere are dependent on the geologic environment in which the repository is placed, the surrounding geologic environment, the design of the repository and the challenges to confining the hazardous materials within the repository, not a function of the fuel cycle. However, the amount of material placed in the repository and its characteristics are a function of the fuel cycle, and represent the available source for any potential releases from the repository. Activity is the basic characteristic, yet experience has shown that what matters when evaluating the effectiveness of isolation from the environment is the radiotoxicity of material released from the repository that gets to the biosphere under degradation of waste packaging or intrusion scenarios. The EST had to be able to inform on the relative performance of fuel cycles in this regard, without focusing on repository design and pathway analyses that were beyond the scope of this Evaluation and Screening study. An EST analysis suggested that activity could be used as a surrogate for radiotoxicity in the long-term period for the purpose of comparing fuel cycle options as far as the impact on disposal is concerned. The EST was also cognizant that activity or radiotoxicity of the emplaced wastes is relevant for assessing the risk associated with the disposed materials.

Appendix D-2.3 provides detailed information on how this metric is calculated from the reactor and fuel cycle mass and isotopic data, and other information contained in Appendix B. The approach for metric binning and the developed bins are also contained/described in that Appendix D. Table C-1.3 provides the bins developed for the Activity of SNF+HLW at 100,000 years per Energy Generated metric.

Table C-1.3. Metric Bins for Activity of SNF+HLW at 100,000 years per Energy Generated.

Bin ID	Data Range (MCi/GWe-yr)	Bin Description
A	< 5.0 x 10^{-4}	Activity of SNF+HLW at 100,000 years < 5.0 x 10^{-4} MCi/GWe-yr
B	5.0 x 10^{-4} to < 1.0 x 10^{-3}	Activity of SNF+HLW at 100,000 years ≥ 5.0 x 10^{-4} MCi/GWe-yr and < 1.0 x 10^{-3} MCi/GWe-yr
C	1.0 x 10^{-3} to < 2.3 x 10^{-3}	Activity of SNF+HLW at 100,000 years ≥ 1.0 x 10^{-3} MCi/GWe-yr and < 2.3 x 10^{-3} MCi/GWe-yr; Bin C contains the Basis of Comparison and the bin range is approximately ±40% of the Basis of Comparison.
D	2.3 x 10^{-3} to < 5.0 x 10^{-3}	Activity of SNF+HLW at 100,000 years ≥ 2.3 x 10^{-3} MCi/GWe-yr and < 5.0 x 10^{-3} MCi/GWe-yr
E	5.0 x 10^{-3} to < 1.0 x 10^{-2}	Activity of SNF+HLW at 100,000 years ≥ 5.0 x 10^{-3} MCi/GWe-yr and < 1.0 x 10^{-2} MCi/GWe-yr
F	≥ 1.0 x 10^{-2}	Activity of SNF+HLW at 100,000 years ≥ 1.0 x 10^{-2} MCi/GWe-yr

C-1.6 Mass of DU+RU+RTh Disposed per Energy Generated

Definition of Metric – This metric is the sum of the depleted uranium (DU), recovered uranium (RU), and recovered thorium (RTh), i.e., DU+RU+RTh, produced in a fuel cycle that is unused (not recycled) and needs to be disposed. These materials are mainly the product of enrichment or reprocessing activities in the fuel cycle. This mass is calculated per energy generated.

The DU+RU+RTh mass has been separated out from the SNF+HLW mass because the EST recognizes that nuclear waste can be divided into several groups that have unique characteristics that impact waste management. These groups also suggest the general approach for disposal. The HLW, SNF and TRU in general will require isolation from the environment such as that provided by deep geologic disposal. As noted above, the disposal pathways for RU, DU, and RTh have not been determined at this time. If a lesser amount of isolation proves acceptable for these materials, then differences in SNF+HLW and RU, DU, and RTh would indicate the relative size of the challenge for each waste disposal path. This grouping was created to recognize the waste stream characteristics for comparison of fuel cycles. The quantity of each characteristic waste stream is needed. The RU, DU and RTh disposed are typically measured in mass of heavy metal, and can be normalized to energy generation. The metric focuses on the masses that are inherent to a fuel cycle, not other characteristics such as volume that are determined by the choice of technology.

Appendix D-2.4 provides detailed information on how this metric is calculated from the reactor and fuel cycle mass and isotopic data, and other information contained in Appendix B. The approach for metric binning and the developed bins are also described in that Appendix D. Table C-1.4 provides the bins developed for the Mass of DU+RU+RTh Disposed per Energy Generated metric.

Table C-1.4. Metric Bins for Mass DU+RU+RTh Disposed per Energy Generated.

Bin ID	Data Range (t/GWe-yr)	Bin Description
A	< 1	Mass DU+RU+RTh disposed < 1 t/GWe-yr
B	1 to < 40	Mass DU+RU+RTh disposed from 1 t/GWe-yr to < 40 t/GWe-yr
C	40 to < 80	Mass DU+RU+RTh disposed from 40 t/GWe-yr to < 80 t/GWe-yr
D	80 to < 120	Mass DU+RU+RTh disposed from 80 t/GWe-yr to < 120 t/GWe-yr
E	120 to < 200	Mass DU+RU+RTh disposed from 120 t/GWe-yr to < 200 t/GWe-yr; contains the Basis of Comparison (EG01)
F	≥ 200	Mass DU+RU+RTh disposed equals or greater than 200 t/GWe-yr

C-1.7 Volume of LLW per Energy Generated

Definition of Metric - This metric evaluates the quantity of Low Level Waste (LLW) arising from each of the fuel cycles under evaluation. The quantity of low-level waste was determined as the number of cubic meters per gigawatt year electric (m^3 / GWe-y.) This volume included both the Class A through Class C LLW and the greater than Class C (GTCC) wastes. The waste estimates included generation from operations as well as the Decontamination and Decommissioning (D&D) of the facilities at the end of their useful life. The volume of LLW for this analysis did not include the volume of naturally occurring radioactive materials (NORM).

The estimated volumes were developed from published historic data where it exists or extrapolated from existing historic data.

Considerable progress has been made over the last 30 years at reducing the volume of LLW. No attempt was made to project how much additional volume reduction could be achieved in the future.

Metric Development

A systematic approach was used to develop the LLW estimates used for this metric. The Analysis Example for each Evaluation Group was used to inform on this metric. The first step involved identification of the fuel cycle processes that were used in one or more of the Analysis Examples. The five primary categories of operations examined were enrichment, fuel fabrication, reactor operations, reprocessing of used fuel, and recycle fuel fabrication. Tables C-1.5 to C-1.9 summarize the operations that were identified for each of the five primary categories of operations. For example, Table C-1.5

which examines uranium enrichment identified four classes of operations; enrichment to less than 5%, enrichment from 5% to less than 20%, re-enrichment of recycle uranium, and no enrichment required. Initially it was not known if the quantity of low-level waste generated for the two levels of enrichment would be different or could be described as noticeably different. Also shown in Tables C-1.5 to C-1.9 are the associated Analysis Examples for the Evaluation Groups that utilize each of these operations. This provides a crosswalk between the type of operation and the individual Analysis Example. For some operations, assumptions needed to be made about the type of technology that could be used to allow development of an LLW estimate, such as aqueous reprocessing for separations. The assumptions were applied consistently across the Analysis Examples to avoid any effects that could arise from using different assumptions for the same separations in different Analysis Examples. The color coding of the individual operations was developed during the subsequent analysis in which LLW generation rates were assigned to each of the operations. Green indicates that a value was directly identified in the literature for the generation rate while yellow was assigned to those operations where rates were extrapolated by subject matter experts from data found in the literature with a reasonably high level of confidence. Red was assigned to values where no applicable data was found and considerable uncertainty exists in the extrapolation.

Table C-1.5. Enrichment.

Operations	Used by the Analysis Example for the Evaluation Group
No Enrichment (other than startup)	EG03, EG04, EG06, EG07, EG08, EG09, EG10, EG12, EG14, EG19, EG20, EG23, EG24, EG26, EG28, EG29, EG30, EG33, EG34, EG38, EG40
Enrichment to < 5wt%	EG01, EG13, EG15, EG16, EG17, EG21, EG22, EG25, EG31, EG32, EG35, EG36, EG37
Enrichment from 5wt% to < 20wt%	EG02, EG05, EG11, EG18, EG27, EG39
Enrichment of RU to < 20wt%	EG27

Table C-1.6. Fresh Fuel Fabrication.

Operations	Used by the Analysis Example for the Evaluation Group
HWR UO2 fuel	EG03, EG12
PWR UO2 fuel	EG01, EG13, EG15, EG16, EG17, EG21, EG22, EG31, EG32, EG35, EG36, EG37
ADS NU metal blanket fuel	EG04, EG07, EG09
Coated U kernels in graphite compacts	EG02
PWR U-Th oxide fuel	EG18
ADS Th metal blanket	EG11, EG40
MSR Th fuel salt	EG06, EG10, EG26
Coated U and Th kernels in graphite compacts	EG05
Thorium, Lithium, Deuterium	EG06, EG08
SFR LEU metal driver	EG11
ADS Pb metal target	EG07, EG40

Table C-1.7. Reactors.

Operations	Used by the Analysis Example for the Evaluation Group
PWR	EG01, EG12, EG13, EG14, EG15, EG16, EG17, EG18, EG21, EG22, EG25, EG29, EG30, EG31, EG32, EG33, EG34, EG35, EG36, EG37, EG38, EG39, EG40
HWR	EG03, EG12, EG19, EG20
HTGR	EG02, EG05
SFR	EG04, EG09, EG11, EG14, EG15, EG23, EG24, EG27, EG28, EG29, EG30, EG31, EG32, EG37, EG38
ADS	EG07, EG16, EG33, EG34, EG35, EG36, EG39, EG40
MSR	EG10,EG 26
FFH	EG06, EG08

Table C-1.8. Reprocessing Operations.

Operations	Used by the Analysis Example for the Evaluation Group
Aqueous: PWR UO2 Product 1: RU, Pu Waste 1: RU, MA, FP	EG12, EG13, EG15, EG16, EG17, EG19, EG20, EG21, EG22, EG25, EG29, EG31, EG32, EG33, EG35, EG37, EG39
Aqueous: PWR UO2 Product 1: RU, Pu Product 2: MA Waste 1: RU, FP	EG30, EG34, EG36, EG37
Aqueous: PWR Th, RU oxide Product 1: Th, U3, MA Waste 1: FP	EG25, EG38
Aqueous: PWR Th oxide Product 1: Th Product 2: U3 Product 3: TRU Waste 1: FP	EG39
MA Dispersion matrix: MA dispersion fuel Product 1: MA Waste 1: FP	EG36, EG39
MSR Separations Product 1: Th, U3, TRU, FP Waste 1: FP	EG10, EG26
Pyro: ADS metal blanket Product 1: TRU Product 2: RU Waste 1: FP	EG14, EG23, EG24, EG27, EG29, EG30, EG32, EG33, EG34, EG35
Pyro: SFR metal blanket Product 1: RU, Pu Product 2: RU, TRU Product 3: RU Waste1 : MA / FP	EG14

Pyro: SFR Th metal driver and blanket Product 1: U3, Th, TRU Product 2: U3, Th Waste1 : FP	EG27, EG28, EG38, EG40
SFR metal fuel reconditioning Product 1: RU/TRU/FP Waste 1: FP	EG09, EG11
Aqueous: PWR Th oxide Product 1: Th Product 2: U3 Waste 1: FP, MA	EG18, EG37, EG40

Table C-1.9. Recycle Fuel Fabrication.

Operations	Used by the Analysis Example for the Evaluation Group
Shielding Glove Box - Recycle U - oxide	EG39
Shielding Glove Box - Recycle Pu - oxide	EG12, EG13, EG14, EG19, EG21, EG29, EG31, EG33, EG36, EG37
Shielding Glove Box - Recycle U - Metal	EG14, EG23, EG27, EG28, EG29, EG30, EG33, EG34, EG38
Shielding Glove Box - Recycle Pu – Metal	EG14, EG15, EG16, EG17, EG23, EG29, EG33, EG35
Remote - Th / U3 – oxide	EG18, EG25, EG37, EG39, EG40
Remote - Pu/TRU+ - oxide	EG20, EG22, EG25, EG30, EG34, EG37, EG38
Remote - Th / U3 – Metal	EG27
Remote - Pu/TRU+ - Metal	EG9, EG11, EG24, EG28, EG30, EG32, EG34, EG36, EG38, EG39
Shielded Glove Box – Tritium	EG6, EG8
MSR	EG10, EG26

Development of reference data

A number of references and sources were used for the development of the specific multipliers used to estimate the volume of LLW. These include open literature, DOE funded studies, expert opinion, input from DOE Industry teams under an “Indefinite Delivery/Indefinite Quantity contract” (IDIQ teams), etc. The specific multipliers and their origin are shown in Tables C-1.10 through C-1.14. These multipliers are comprised of four components: LLW volume per unit energy for processing, LLW volume per unit energy for D&D, GTCC volume per unit energy for process, and GTCC volume per unit energy for D&D. In the case of enrichment, the unit is per SWU (Separative Work Unit -the amount of separation done by an enrichment process) required. For the production of new fuel, recycle fuel, or reprocessing the volume is per ton of heavy metal processed. For reactor operations the volume is per gigawatt year electric. The “Source” column of these tables indicates the origin of the multiplier data, and the “Notes” column provides insight regarding the adjustments or approach that was used to develop the estimate when literature values were not available.

In Table C-1.10, the LLW volumes for enrichment of fresh uranium were based on Ref. C-1.10. This report indicated that the volume of LLW during production was 0.04 m³ per kSWU and the D&D low-level waste was 0.4 m^3 per kSWU. The LLW volumes for the enrichment of recycle uranium were developed based on the enrichment of fresh uranium because no data could be found at an industrial scale for this operation. It was estimated that the volume of low-level waste for production would be twice that of natural uranium and that 1/10 of that volume would be produced as GTCC waste arising from the carryover of fission products from reprocessing. The same ratios were used for the resulting D&D waste at the end of life for the facility.

Table C-1.11 provides the volume multipliers for the fabrication of fresh fuel. No low-level waste was identified from the fabrication of heavy water reactor fuel because no enrichment was required and that the waste would only contain naturally occurring radioactive materials (NORM). Similar logic was used for the fabrication of ADS natural uranium metal blanket fuel and SFR natural uranium metal driver fuel. For the thorium fuels (ADS thorium metal blanket, SFR thorium metal blanket, MSR thorium salt fuels) the volume of low-level waste was again assumed to be zero because this was naturally occurring radioactive material. The volume of LLW arising from the production of graphite uranium kernels was assumed to be 1.5 times that of the low-level waste arising from the production of uranium oxide fuels. The same assumption was used for coated uranium/thorium kernels. The extra volume arises from contaminated graphite. The volume of low-level waste for the ADS metal targets was assumed to be zero because these were fabricated from non-radioactive materials. The volume for tritium was estimated from production data at SRS.

The OECD report provided data for the low-level and GTCC waste from a number of reactor types (LWR, PWR, HWR, HTGR, SFR, ADS, and FFH) for both production and D&D [C-1.10]. These values are shown in Table C-1.12. No data could be identified for the molten salt reactor; thus it was assumed to be the same as an LWR. There are a number of ways in which one could envision the waste would be higher than for an LWR but it was decided not to penalize this reactor type due to the lack of available data.

Table C-1.13 provides the multiplier data for the reprocessing of the used fuel. The waste generation estimates for reprocessing are based on several reports developed by the Used Fuel Disposition Campaign [C-1.14, C-1.15]. The basis for all of the aqueous processing estimates are built upon the estimate for the processing of uranium oxide and recovery of a single product stream containing a mixture of recycled uranium (RU) and plutonium (Pu). The remaining uranium, minor actinide, and fission products are disposed of in a combined waste stream. This estimate was then increased with increasing complexity of the reprocessing operations as shown in the notes column in the table. A similar approach was used for the electrochemical processing; here again the basis for the electrochemical estimate was from the same Used Fuel Disposition Campaign reports [C-1.14, C-1.15] and was based on the recovery of two product streams, recycled uranium and a combined transuranic stream. The fission products are sent to waste. The D&D portion of this estimate is made up of two parts; the low-level waste volume was estimated to be 75% of the aqueous COEX waste and the GTCC waste was assumed to be twice that of the COEX waste.

Table C-1.14 provides the multiplier data for the fabrication of recycle fuel. The recycle fuel was broken into two primary categories; the first category required the use of shielded glovebox facilities for the fabrication and the second class required the use of hot cells for the fabrication of the recycle fuel. Very limited data was available for these operations - the best source of data was for the processing of MOX fuel. These operations are conducted in shielded glove boxes. MOX fuel fabrication data compiled by OECD indicates that 0.6 m^3 per ton of heavy metal per year of short lived low level waste is produced and an equal amount of long-lived low-level waste is produced [C-1.10]. This long-lived low-level waste was placed in the GTCC category. The D&D waste was assumed to be six times the waste generated during the D&D of a uranium oxide fuel fabrication plant. Additional adjustments for other fuel types made in shielded glove boxes include:

- For MOX the operational waste is estimated to be twice that of UOX production waste and approximately 6 times more for D&D waste.
- The total volume of waste for recycle uranium was assumed to be equal to that of MOX fuel, but there would be no GTCC waste produced.
- The fabrication of metal fuel was assumed to result in 25% more waste than for comparable oxide fuels due to casting operations.

The added complexity of remote operations associated with thorium and U233 fuels, minor actinide, or transuranic fuels was projected to be 150% of the comparable shielded waste. The addition of transuranics also increased the quantity of GTCC waste. The re-fabrication of molten salt fuels was included as part of the integral processing of the molten salt fuel at the reactor and is contained in the reprocessing estimate. The fabrication of the recovered tritium into fuel for the fusion portion of a fission-fusion hybrid (FFH) was estimated based on data obtained by personal communication.

Discussion of how value of metric was calculated using this data.

Table C-1.15 contains an example of the low-level waste calculations. This example is for the Analysis Example for Evaluation Group #1 (EG01), a once through LWR fuel cycle, which serves as the Basis of Comparison for this report. In this case the fuel cycle requires that 188.628 tons of natural uranium must be mined for each 1 GWe-yr produced. Uranium mining results in no low-level waste as the waste is classified as a naturally occurring radioactive material (NORM). The reactor requires 21.915 tons of uranium enriched to 4.21%. The tails are 0.25%. The enrichment operation requires 137.3 kSWU. The appropriate multipliers are shown in the table and the resulting waste is calculated by multiplying the kSWU by the multiplier to arrive at the volume of waste - 60.4 m^3 of LLW per GWe-yr. In a like manner, the low-level waste arising from the fabrication of the fresh fuel generates a total of 13.8 m^3 of waste per GWe-yr produced. The reactor generates the largest fraction of the total waste in this example, that volume of waste is 324.6 m^3 per GWe-yr. The totals for operations and D&D sum to 391.2 m^3 of LLW and 7.6 m^3 of GTCC for a total volume of LLW of 398.8 m^3/GWe-yr. Similar calculations were completed for each Analysis Example corresponding to each Evaluation Group.

Binning of the Metric Data

The calculated data derived from the Analysis Examples of the 40 Evaluation Groups were then used as the basis for binning the Metric Data for each Evaluation Group into a metric bin. By using bins, it is expected that the Metric Data to be used for evaluating an Evaluation Group relative to the Basis of Comparison is sufficiently representative of the capabilities of fuel cycles within the group. Details on the *volume of LLW per energy generated* metric calculation approach, the binning process, and the metric bins for the 40 evaluation groups, are described in Appendix D-2.5.

Table C-1.10. LLW Waste Generation Volume Multipliers for Enrichment.

Operations	Process				D&D				Source	Notes
	LLW	Units	GTCC	Units	LLW	Units	GTCC	Units		
Enrichment to < 5wt%	0.04	m^3/kSWU	0	m^3/kSWU	0.4	m^3/kSWU	0	m^3/kSWU	OECD report [C-1.10]	Need tails, enrichment, qty
Enrichment from 5wt% to < 20wt%	0.04	m^3/kSWU	0	m^3/kSWU	0.4	m^3/kSWU	0	m^3/kSWU	OECD report [C-1.10]	Need tails, enrichment, qty
Enrichment of RU to < 20 wt%	0.08	m^3/kSWU	0.008	m^3/kSWU	0.8	m^3/kSWU	0.08	m^3/kSWU		SME - 2 X LLW waste due to FP carryover. GTCC is 10% of LLW.
No enrichment required	0	m^3/kSWU	0	m^3/kSWU	0	m^3/kSWU	0	m^3/kSWU		

Table C-1.11. LLW Waste Generation Volume Multipliers for Fresh Fuel Fabrication.

Operations	Process				D&D				Source	Notes
	LLW	Units	GTCC	Units	LLW	Units	GTCC	Units		
HWR UO2 fuel	0	m^3/tHM	0	m^3/tHM	0	m^3/tHM	0	m^3/tHM	SME	NU --> no LLW, no GTCC
LWR UO2 fuel	0.6	m^3/tHM	0	m^3/tHM	0.03	m^3/tHM	0	m^3/tHM	OECD report [C-1.10]	
PWR UO2 fuel	0.6	m^3/tHM	0	m^3/tHM	0.03	m^3/tHM	0	m^3/tHM	OECD report [C-1.10]	
ADS NU metal blanket fuel	0	m^3/tHM	0	m^3/tHM	0	m^3/tHM	0	m^3/tHM	SME	NU --> no LLW, no GTCC
Coated U kernels in graphite compacts	0.9	m^3/tHM	0	m^3/tHM	0.045	m^3/tHM	0	m^3/tHM	OECD report [C-1.10]; SME	Assume similar to UOX but graphite will add additional volume to waste. Add 50% to volumes to account for graphite
PWR U-Th oxide fuel	0.6	m^3/tHM	0	m^3/tHM	0.03	m^3/tHM	0	m^3/tHM	OECD report [C-1.10]	Assume same as UOX
ADS Th metal blanket	0	m^3/tHM	0	m^3/tHM	0	m^3/tHM	0	m^3/tHM	SME	NTh --> no LLW, no GTCC
MSR Th fuel salt	0	m^3/tHM	0	m^3/tHM	0	m^3/tHM	0	m^3/tHM	SME	NTh --> no LLW, no GTCC
Coated U and Th kernels in graphite	0.9	m^3/tHM	0	m^3/tHM	0.045	m^3tHM	0	m^3/tHM	SME	Assume same as U -kernel

compacts										
Thorium, Lithium, Deuterium	0	m^3/tH3	0	m^3/tH3	0	m^3/tH3	0	m^3/tH3	SME	Assumed NTh --> no LLW, Lithium and Deuterium target wastes excluded from this analysis
SFR LEU metal driver	0.75	m^3/tHM	0	m^3/tHM	0.0375	m^3/tHM	0	m^3/tHM	OECD report [C-1.10]; SME	25% increase due to additional process wastes like molds & crucibles
ADS Pb metal target	0	m^3/tHM	0	m^3/tHM	0	m^3/tHM	0	m^3/tHM	SME	Hands on - Non rad liquid lead spallation neutron source

Table C-1.12. LLW Waste Generation Volume Multipliers for Reactors.

Operations	Process				D&D				Source	Notes
	LLW	Units	GTCC	Units	LLW	Units	GTCC	Units		
PWR	112	m^3/GWe-yr	2.6	m^3/GWe-yr	205	m^3/GWe-yr	5	m^3/GWe-yr	OECD Report [C-1.10]	
HWR	140	m^3/GWe-yr	23	m^3/GWe-yr	205	m^3/GWe-yr	5	m^3/GWe-yr	OECD Report [C-1.10]	
HTGR	56	m^3/GWe-yr	2.6	m^3/GWe-yr	205	m^3/GWe-yr	15	m^3/GWe-yr	OECD Report [C-1.10]; IAEA TECDOC-1521 [C-1.11]	Operational waste and D&D low level waste based on OECD report. D&D GTCC waste based on IAEA data for German HTGR -THTR-300 assuming that graphite will be GTCC.
SFR	56	m^3/GWe-yr	2.6	m^3/GWe-yr	205	m^3/GWe-yr	15	m^3/GWe-yr	OECD Report [C-1.10]; SME	Limited data available for D&D of SFR. Estimated D&D waste was assumed to be no greater than that from HTGRs.
ADS	112	m^3/GWe-yr	5.2	m^3/GWe-yr	205	m^3/GWe-yr	20	m^3/GWe-yr	OECD Report [C-1.10]; SME	Operational waste was estimated in OECD report as comparable to SFR. However, additional system will likely

										increase operational waste generation leading to a doubling of SFR estimate. D&D low level waste assumed comparable but GTCC waste increased by 33% to address activated spallation targets/ window/ accelerator components
MSR	112	m^3/GWe-yr	2.6	m^3/GWe-yr	205	m^3/GWe-yr	5	m^3/GWe-yr	SME	Assumed same as LWR Waste from salt processing is included in reprocessing table.
FFH	130	m^3/GWe-yr	0	m^3/GWe-yr	205	m^3/GWe-yr	5	m^3/GWe-yr	OECD Report [C-1.10] Stacey et al [C-1.12] Hoffman [C-1.13]	Fusion reactor operational waste based on an annualized generation of total waste reported by Stacey et al. Hoffman indicted that it was possible to avoid GTCC waste. GTCC waste from the fission portion of the operation of the FFH might be comparable to other fission reactors. However, based on lack of data for FFH this was estimated as zero thus not adversely impacting options using this style reactor. D&D waste assumed comparable to other reactor types based on lack of data.

Table C-1.13. LLW Waste Generation Volume Multipliers for Reprocessing.

Operations	Process				D&D				Source	Notes
	LLW	Units	GTCC	Units	LLW	Units	GTCC	Units		
Aqueous: PWR UO2 Product 1: RU, Pu Waste 1: RU, MA, FP	7.04	m^3/tHM	1.04	m^3/tHM	1.9	m^3/tHM	1.4	m^3/tHM	2012 UFD Update report[C-1.14] IAEA TRS 462 for D&D [C-1.15]	This represents the COEX process and serves as basis for the following aqueous processing estimates. UOX - two streams typically shown from reprocessing. All waste was typically shown in single stream but excess RU from COEX will be an additional stream. D&D wastes based on data reported by IAEA [C-1.14] for WAK facility assuming 40 yr life. Assumes 1 actinide stream (RU, Pu), 1 non-actinide stream (RU) and one primary waste stream (MA, FP).
Aqueous: PWR UO2 Product 1: RU, Pu Product 2: MA Waste 1: RU, FP	7.392	m^3/tHM	2.08	m^3/tHM	1.995	m^3/tHM	2.8	m^3/tHM	2012 UFD Update report[C-1.14] IAEA TRS 462 for D&D [C-1.15] SME	UOX - three streams shown from reprocessing - COEX with additional TRU stream. This contains 2 actinide streams (RU, Pu and MA), 1 non-actinide stream (RU), and 1 waste stream (FP). LLW is 1.05 times that of the co-extraction type

										aqueous operations and GTCC is doubled.
Aqueous: PWR Th, RU oxide Product 1: Th, U3, MA Waste 1: FP	8.8	m^3/tHM	1.3	m^3/tHM	2.375	m^3/tHM	1.75	m^3/tHM	2012 UFD Update report[C-1.14] IAEA TRS 462 for D&D [C-1.15] SME	Th - two streams shown from reprocessing. Assume more aggressive processes required resulting in more failed equipment. 1.25 x COEX.
Aqueous: PWR Th oxide Product 1: Th Product 2: U3 Product 3: TRU Waste 1: FP	10.56	m^3/tHM	1.3	m^3/tHM	2.85	m^3/tHM	1.75	m^3/tHM	2012 UFD Update report[C-1.14] IAEA TRS 462 for D&D [C-1.15] SME	Th - four streams from reprocessing. More complex than previous case. Added complexity results in more LLW. LLW is 1.5 times COEX case. GTCC is same as previous case.
MA Dispersion matrix: MA dispersion fuel Product 1: MA Waste 1: FP	8.7	m^3/tHM	3.1	m^3/tHM	1.425	m^3/tHM	2.8	m^3/tHM	2012 UFD Update report[C-1.14] IAEA TRS 462 for D&D [C-1.15] SME	Assume Echem Zr dispersion MA fuel/blanket. Assume similar waste generation as that of E-chem processing.
MSR Separations Product 1: Th, U3, TRU, FP Waste 1: FP	0.176	m^3/tHM	0.026	m^3/tHM	0.000475	m^3/tHM	0.00035	m^3/tHM	2012 UFD Update report[C-1.14] IAEA TRS 462 for D&D [C-1.15] SME	This assumes on-line processing of fuel at reactor. Actual material mass flow through salt processing loop is ~400 times more mass per year than PWR for same energy produced. LLW was assumed to be 10

										times the waste generation volume during processing compared with the volume of wastes arising from the processing of PWR fuel that generated a comparable amount of energy due to larger mass flow but simpler process without need to handle cladding and other hardware. Since processing was simpler the D&D waste was assumed to be 10 times less than for COEX type processing. This also credits collocation with reactor.
Pyro: ADS metal blanket Product 1: TRU Product 2: RU Waste 1: FP	8.7	m^3/tHM	3.1	m^3/tHM	1.425	m^3/tHM	2.8	m^3/tHM	2013 UFD Update report [C-1.16] IAEA TRS 462 for D&D [C-1.15] SME	This is the basic Echem case and used as basis of estimates for other Echem variants. Echem - 3 streams from reprocessing. 1 actinide, 1 non-actinide stream and 1 primary waste stream. D&D waste based on COEX LLW. Assumed a reduction of 25% for the LLW due to potentially more compact facility size but twice the volume of

										GTCC based on relative increase in operation GTCC waste compared to COEX type operation which is only 33% of the Echem operational GTCC waste.
Pyro: SFR metal blanket Product 1: RU, Pu Product 2: RU, TRU Product 3: RUWaste1 : MA / FP	9.135	m^3/tHM	3.875	m^3/tHM	1.425	m^3/tHM	3.5	m^3/tHM	2013 UFD Update report [C-1.16] IAEA TRS 462 for D&D [C-1.15] SME	E-chem - 3 or 4 streams from reprocessing. 2 actinide, optional non-actinide RU stream, and 1 primary waste stream. Assumed that LLW was 1.05 times previous Echem case resulting from 2 actinide products and 1.25 times GTCC waste again due to additional actinide stream.
Pyro: SFR Th metal driver and blanket Product 1: U3, Th, TRU Product 2: U3, Th Waste1 : FP	8.7	m^3/tHM	3.1	m^3/tHM	1.425	m^3/tHM	2.8	m^3/tHM	2013 UFD Update report [C-1.16] IAEA TRS 462 for D&D [C-1.15] SME	E-chem Th - 3 streams from reprocessing. Similar to Echem - no change
SFR metal fuel reconditioning Product 1: RU/TRU/FP Waste 1: FP	6.525	m^3/tHM	1.55	m^3/tHM	1.068	m^3/tHM	1.4	m^3/tHM	2013 UFD Update report [C-1.16] IAEA TRS 462 for D&D [C-1.15] SME	Lower waste volumes for SFR metal reconditioning vs. full EChem processing. LLW is 75% of Echem and GTCC is 50%

Aqueous: PWR Th oxide Product 1: Th Product 2: U3 Waste 1: FP, MA	8.8	m^3/tHM	0.78	m^3/tHM	2.375	m^3/tHM	1.05	m^3/tHM	2012 UFD Update report[C-1.14] IAEA TRS 462 for D&D [C-1.15] SME	Th - 3 streams from reprocessing. More complex than COEX case but all MA sent to waste. Added complexity results in more LLW. LLW is 1.25 times COEX case. GTCC is 75% of previous case.

Table C-1.14. LLW Waste Generation Volume Multipliers for Recycle Fuel Fabrication.

Operations	Process				D&D				Source	Notes
	LLW	Units	GTCC	Units	LLW	Units	GTCC	Units		
Shielded Glove Box - Recycle U - oxide	1.2	m^3/tHM	0	m^3/tHM	0.2	m^3/tHM	0	m^3/tHM	SME	Same total volume of waste as for MOX fuel fab. None of waste would be GTCC. Note that as a comparison this assumption results in an operational LLW stream that is twice that of fresh UOX fuel production. D&D waste is ~6 times that of fresh UOX fabrication facility D&D due to higher levels of contamination
Shielded Glove Box - Recycle Pu - oxide	0.6	m^3/tHM	0.6	m^3/tHM	0.1	m^3/tHM	0.1	m^3/tHM	OECD Report [C-1.10]; SME	Assume same total operation waste but 50% is GTCC. D&D data limit – Based on limited OREOX data presented in [C-1.10] for small scale facility LLW was ~0.2 m^3/t and GTCC was ~0.1 m^3/t. Assume larger facility would reduce LLW by 50%. GTCC would remain same. This is ~ 6 times that of fresh UOX

										fabrication facility D&D waste.
Shielded Glove Box - Recycle U - Metal	1.5	m^3/tHM	0	m^3/tHM	0.25	m^3/tHM	0	m^3/tHM	SME	Increase LLW by 25% as a metal fuel fab adjustment factor over recycle UOX
Shielded Glove Box - Recycle Pu - Metal	0.75	m^3/tHM	0.75	m^3/tHM	0.125	m^3/tHM	0.125	m^3/tHM	SME	Increase LLW by 25% as a metal fuel fab adjustment factor over recycle MOX
Remote - Th / U3 - oxide	1.8	m^3/tHM	0	m^3/tHM	0.3	m^3/tHM	0	m^3/tHM	SME	Increase LLW by 50% as a remote fuel fab adjustment factor over recycle UOX
Remote - Pu/TRU+ - oxide	0.6	m^3/tHM	1.2	m^3/tHM	0.1	m^3/tHM	0.2	m^3/tHM	SME	Increase GTCC by 100% as a remote fuel fab adjustment factor over recycle MOX in shielded glove box due to added handling complexity.
Remote - Th / U3 - Metal	2.25	m^3/tHM	0	m^3/tHM	0.375	m^3/tHM	0	m^3/tHM	SME	Increase LLW by 25% as a metal fuel fab adjustment factor over recycle UOX
Remote - Pu/TRU+ - Metal	0.75	m^3/tHM	1.5	m^3/tHM	0.125	m^3/tHM	0.25	m^3/tHM	SME	Increase LLW by 25% as a metal fuel fab adjustment factor over recycle MOX
Shielded Glove Box – Tritium	17900	m^3/tH3	0	m^3/tH3	895	m^3/tH3	0	m^3/tH3	SME	No data could be found on comparable facilities of this size. SME estimate based on projected waste from large scale operational facility. SME estimated D&D 5% of annual operational waste.
MSR	0.0	m^3/tHM	0.0	m^3/tHM	0.0	m^3/tHM	0.0	m^3/tHM	SME	Included as part of reprocessing

Table C-1.15. Example Case – Evaluation Group 01 – Basis of Comparison.

FUEL CYCLE PROCESS		**Analysis Example Values**	**Units**	**Enrichment (%)**	**Tails (%)**	**Basis for Calculation**	**Units**	**Multipliers**					**LLW Estimates**			
								Process		**D&D**		**Units**	**Process**		**D&D**	
								LLW	**GTCC**	**LLW**	**GTCC**		**LLW (m^3)**	**GTCC (m^3)**	**LLW (m^3)**	**GTCC (m^3)**
Fuel Material Supply - Uranium	Mined Uranium	188.628														
Uranium Enrichment	Enrichment to < 5wt%	21.915	t Enriched U / GWey	4.21	0.25	137.3	kSWU	0.04	0	0.4	0	m^3/kSWU	5.5	0.0	54.9	0.0
Fuel Fabrication from New Resources (hands on)	LWR UO2 fuel	21.915	t Enriched U / GWey			21.915	tHM	0.6	0	0.03	0	m^3/tHM-yr	13.1	0.0	0.7	0.0
Reactors - Stage 1 (critical and sub-critical)	LWR	1	GWe fleet			1	GWe-yr	112	2.6	205	5	m^3/GWe-yr	112.0	2.6	205.0	5.0
Total LLW (m^3 /GWe-yr)													**130.6**	**2.6**	**260.5**	**5.0**
LLW	**391.2**															
GTCC	**7.6**															
Grand Total	**398.8**															

References for C-1.

C-1.1. Harry H. Hess, et al, “The Disposal of Radioactive Waste on Land”. National Academy of Sciences – National Research Council Publication 519, 1957; Library of Congress catalog card number 57-60047.

C-1.2. “Alternatives for Managing Wastes from Reactors and Post-Fission Operations in the LWR Fuel Cycle”, 5 volumes, ERDA-76-43, Report Coordinated by Battelle, Pacific Northwest Laboratories, United States Energy Research and Development Administration, May 1976.

C-1.3. “Interagency Review Group on Nuclear Waste Management: Subgroup Report on Alternative Technology Strategies for the Isolation of Nuclear Waste”, October 1978 (DRAFT), TID-28818, DOE 46 FR26677, 1981.

C-1.4. “Final Environmental Impact Statement – Management of Commercially Generated Radioactive Waste”, U. S. Department of Energy, Office of Nuclear Waste Management, DOE/EIS-0046F, 3 volumes, October 1980.

C-1.5. “Nuclear Waste Policy Act of 1982 (Public Law 97-425) as amended.

C-1.6. “Rethinking High-Level Radioactive Waste Disposal”, A Position Statement of the Board on Radioactive Waste Management, National Research Council, National Academy Press, 1990

C-1.7. “Blue Ribbon Commission on America’s Nuclear Future – Report to the Secretary of Energy”, January 2012.

C-1.8. R. Wigeland, et al, “Options Study – Phase II”, FCRD-TIO-2010-000167, September 2010.

C-1.9. “Red Impact – Impact of Partitioning, Transmutation, and Waste Reduction Technologies on the Final Nuclear Waste Disposal – Synthesis Report,” Forschungszentrum Jűlich, GmbH, FP6 Contract No: F16W-CT-2004-002408, September 2007

C-1.10. OECD, “Advanced Nuclear Fuel Cycles and Radioactive Waste Management,” Nuclear Energy Agency, Organization for Economic Co-operation and Development, 2006.

C-1.11. IAEA, “Characterization, Treatment and Conditioning of Radioactive Graphite from Decommissioning of Nuclear Reactors,” IAEA-TECDOC-1521, International Atomic Energy Agency, Vienna, 2006.

C-1.12. Weston M. Stacey, Nolan E. Hertel, Edward A. Hoffman, “Radioactive waste produced by demonstration and commercial fusion reactors extrapolated from ITER and advanced databases,” Fusion Engineering and Design, 29, 198-206, 1995.

C-1.13. E. A. Hoffman, “Low Activation Tokamak Reactors,” Masters Thesis, Georgia Institute of Technology, Georgia, 1995.

C-1.14 Robert H. Jones Jr. and Joe T. Carter, “Comparison of Waste Estimates from Recycling and Fuel Fabrication,” FCRD-UFD-2013-000178, June 2013.

C-1.15. IAEA, “Managing Low-Level Radioactive Material from the Decommissioning of Nuclear Facilities,” Technical Report Series No. 462, International Atomic Energy Agency, Vienna, 2008.

C-1.16 Robert H. Jones, “Low Level Waste Disposition – Quantity and Inventory”, FCRD-USED-2010-000033, Rev. 3, June 2013.

C-2. Proliferation Risk Criterion

For the purposes of the fuel cycle Evaluation and Screening Study, the definition of Proliferation Risk is as follows:

Proliferation Risk *– A broad definition of proliferation risk includes the risk of a nation ("host-state") obtaining nuclear weapons where that nation currently does not possess nuclear weapons. Proliferation risk involves a number of factors, both technical and non-technical, including considerations such as facility location and host state intent.*

It is essential to recognize that since the United States already possesses nuclear weapons, there is no question of proliferation risk by the U.S. from the domestic use of nuclear power. However, domestic R&D might well involve collaboration with foreign partners, including non-nuclear weapons states, which makes relevant the question of whether that collaboration poses nonproliferation considerations. Also, if the U.S. embarks on a path to develop an alternative fuel cycle, other countries may wish to pursue the same or similar approaches. Proliferation risk is a foremost consideration in exports of domestically developed nuclear technology, facilities, and materials.

In general, assessing proliferation risk is a complex and challenging endeavor, primarily because it involves both technical and socio-political considerations, with the dominant factor being facility location. Since most of these factors are beyond the scope of the E&S Study, there was no attempt at an assessment of proliferation risk in the E&S Study, and efforts focused only on the evaluation of technical differences between fuel cycle options at the physics-based functional level (this study did not consider any specific implementing technologies as described in the Main Report and in Appendices A and B).

C-2.1 Fuel Cycle Assessments

A considerable amount of effort over several decades has been devoted to the subject of the proliferation risk from the civilian nuclear fuel cycle. [C-2.1 to C-2.18] These previous studies show that host state proliferation concerns typically center around uranium enrichment and reprocessing facilities rather than power reactors because such facilities entail the capability to alter nuclear materials and make them attractive for proliferant activities.

To be successful, nonproliferation agendas require a combination of policy and technological innovation. Policy alone may make unrealistic assumptions of what is technically feasible, whereas technology alone may overlook opportunities for institutional arrangements that reduce proliferation risks. [C-2.18] Two technical considerations include material attractiveness and IAEA safeguards.

> *Material Attractiveness* – the usefulness of a material for proliferant activities
>
> *IAEA Safeguards* - a series of technical measures designed to provide credible assurances to the international community that nuclear material remains in peaceful use.

The technical measures supporting efficient and effective safeguards implementation depend on the choice of technology, facility design, and operations. This E&S Study only evaluated the fuel cycle options at a physics-based functional level where facility characteristics are not specified and it was not possible to provide any insight on the ability to apply safeguards. As a result, for the Proliferation Risk criterion, the E&S Study focused only on informing on the difference in material attractiveness between fuel cycles. This is only one attribute, however, and it is important to keep the larger IAEA safeguards context in mind if the facilities are in a non-nuclear weapons state [C-2.18]:

> There are several paths that a Host State attempting to proliferate might take to avoid a “timely warning” from the IAEA. For example, a Host State might try to divert safeguarded material from the reprocessing plant without detection by the IAEA. A Host State might also try to misuse or alter the facility to produce undeclared nuclear material outside of safeguards, again without detection by the IAEA. The expertise gained in operating a safeguarded reprocessing facility

would also provide the Host State with the necessary experience to build and operate a separate, clandestine facility.

Finally, the Host State might simply withdraw from the nuclear non-proliferation treaty, expel IAEA inspectors, and overtly seize the product material, possibly with further purification if needed. In this case, the timely warning from IAEA safeguards would no longer be relevant and it would only be pressure from the international community that could stop the Host State from proliferating.

C-2.2 Evaluation of Material Attractiveness

This report describes the results for the Evaluation Groups based on the metric of material attractiveness under normal operating conditions:

- *Material attractiveness - normal operating conditions*; for normal operations, the highest material attractiveness over all sensitive points was used to compare the attractiveness of materials that could be potentially diverted from the fuel cycle. The assumption was made that the fuel cycle and facility designers would take steps to lower material attractiveness for normal operations.

It is useful to reiterate the purpose of the Evaluation and Screening, which was to evaluate differences between fuel cycles at the functional fuel cycle level. For the Proliferation Risk criterion, it was not possible to evaluate relative proliferation risk for the E&S Study due to the many aspects of proliferation risk that are outside the scope of the E&S Study. At the physics-based fuel cycle functional level, it was only possible to evaluate material attractiveness, which is only one of many considerations for proliferation risk. There was no consideration of the implementing technologies, facility designs, or any other specific choices made in implementing a fuel cycle. Consideration of these additional factors and the effects they may have on the ability to apply efficient and effective safeguards could be very important as part of the R&D on alternative fuel cycles.

For evaluating the material attractiveness under normal operating conditions for each fuel cycle option Evaluation Group, the EST started with the material composition information presented for the Analysis Example described in Appendix B. Since lower material attractiveness was considered as desirable for hindering proliferant activities, the specific design choices made in the Analysis Example as well as other fuel cycles contained within the Evaluation Group were reviewed to determine if different design choices would lower material attractiveness.

The evaluation approach for this criterion benefitted from input from other activities within DOE, including the DOE National Nuclear Security Administration (NNSA) and the Material Protection, Control, Accounting, and Control Technologies (MPACT) campaign within FCR&D.

References for C-2.

C-2.1. D.E. Lilienthal, et al, “A Report on the International Control of Atomic Energy,” U.S. Department of State, Publication 2498, March 16, 1946.

C-2.2. B.M. Baruch, presentation to the United Nations Atomic Energy Commission, June 14, 1946.

C-2.3. “Treaty on the Non-proliferation of Nuclear Weapons,” INFCIRC/140, International Atomic Energy Agency, April 22, 1970.

C-2.4. “IAEA Safeguards, An Introduction,” IAEA/SG/INF/3, International Atomic Energy Agency, Vienna, Austria, 1981.

C-2.5. R. Skjoldebrand, “The International Fuel Cycle Evaluation – INFCE,” IAEA Bulletin, Vol. 22, No. 2, 1980.

C-2.6. "INFCE: Summary Volume," International Fuel Cycle Evaluation Vol. 9, International Atomic Energy Agency, Vienna, 1980.

C-2.7. "Nuclear Proliferation and Civilian Nuclear Power – Report of the Nonproliferation Alternative Systems Assessment Program," DOE/NE-0001/1, U.S. Department of Energy, June 1980.

C-2.8. "Nuclear Wastes: Technologies for Separations and Transmutation," Committee on Separations Technology and Transmutation Systems, National Research Council, National Academies Press, Washington, D.C., 1996.

C-2.9. "Technological Opportunities to Increase the Proliferation Resistance of Global Civilian Nuclear Power Systems (TOPS)," Report by the TOPS Task Force of the Nuclear Energy Research Advisory Committee (NERAC) of the U.S. Department of Energy, January 2001.

C-2.10. "A Technology Roadmap for Generation IV Nuclear Energy Systems," U.S. Department of Energy Nuclear Energy Research Advisory Committee and the Generation IV International Forum, December 2002.

C-2.11. "Evaluation Methodology for Proliferation Resistance and Physical Protection of Generation IV Nuclear Energy Systems, Revision 6," The Proliferation Resistance and Physical Protection Evaluation Methodology Working Group of the Generation IV International Forum, GIF/PRPPWG/2011/003, September 15, 2011.

C-2.12. R. Bari, et al, "Proliferation Risk Reduction Study of Alternative Spent Fuel Processing Technologies," Proceedings of the INMM 50th Annual Meeting, Tucson, AZ, July 12-16, 2009.

C-2.13. M. Zentner, et al, "An Expert Elicitation Based Study of the Proliferation Resistance of a Suite of Nuclear Power Plants," Proceedings of the INMM 51st Annual Meeting, Baltimore, MD, July 11-15, 2010.

C-2.14. R. Bari, "Proliferation Resistance and Physical Protection (PR&PP) Evaluation Methodology: Objectives, Accomplishments, and Future Directions," submitted to ANS Journal of Nuclear Technology, 2012.

C-2.15. "Draft Global Nuclear Energy Partnership Programmatic Environmental Impact Statement," Office of Nuclear Energy, U.S. Department of Energy, DOE-EIS-0396, October 2008.

C-2.16. "Draft Nonproliferation Impact Assessment for the Global Nuclear Energy Partnership Programmatic Alternatives," Office of Nonproliferation and International Security, National Nuclear Security Administration, December 2008.

C-2.17. "Improving the Assessment of the Proliferation Risk of Nuclear Fuel Cycles," The National Academies Press, Washington, D.C., 2013.

C-2.18. "Reprocessing and Mixed Oxide Fuel Use in Nuclear Reactors," Report to Congress, U.S. Department of Energy, November 2012.

C-3. Nuclear Material Security Risk Criterion

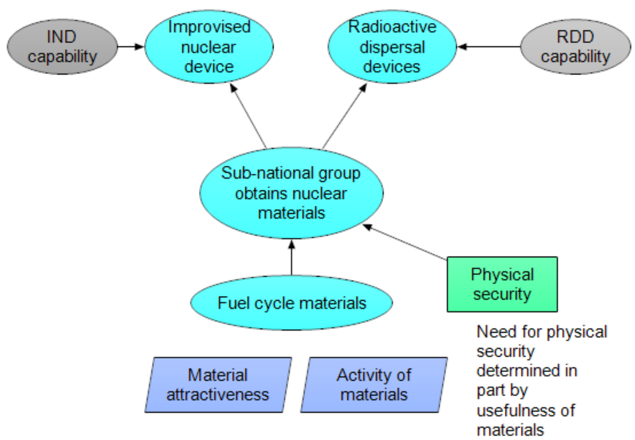
For the purposes of the fuel cycle Evaluation and Screening, the Nuclear Material Security Risk Criterion is as follows:

Nuclear Material Security Risk *– A broad definition of nuclear material security risk includes the risk posed by the threat of nuclear materials from civilian nuclear facilities being used by "sub-national" or terrorist groups in nuclear explosive devices (NED, including improvised nuclear devices, or IND) and radiological dispersal / exposure devices (RDD/RED).*

This section explores the issue of nuclear material security risk, the relationship of the civilian use of nuclear power to this risk, and the principles that can be used in evaluating the comparative risk between nuclear energy systems, i.e., the focus is not on evaluating the nuclear material security risk itself but on the differences in nuclear material security risk from one nuclear energy system to another. The goal in the E&S Study was to identify appropriate evaluation metrics for comparisons between nuclear energy systems at the physics-based functional level as explained in Appendices A and B. The ongoing GenIV Proliferation Resistance and Physical Protection (PR&PP) Evaluation Methodology project is the latest effort in the long history of the subject and reflects the continuing uniformity of approach for nuclear material security risk considerations. [C-3.1, C-3.2]

C-3.1 Background on Nuclear Material Security Risk

Historically, nuclear material security risk is discussed as one of the factors underpinning the need for "physical protection," reflecting the nature of the problem and the solutions implemented to manage this risk. According to Ref. C-3.1, physical protection (robustness) is that characteristic of a nuclear energy system that impedes access to materials suitable for nuclear explosive or radioactive devices and the sabotage of facilities and transportation by sub-national entities and other non-host-state adversaries. Figure C-3.1 shows the factors considered in developing the metrics to inform on this criterion.



Note: Each oval represents a factor, element, or question related to nuclear material security risk. Aqua indicates factors related to the risk that are strongly driven by the characteristics of fuel cycle. Green indicates technology-specific factors affecting the risk. Grey indicates other factors relevant to the risk that are not included in the Evaluation Metrics for this Criterion since there is no fuel cycle contribution.

Figure C-3.1. Influence Diagram of the Considerations for the Nuclear Material Security Risk Criterion.

The threats addressed by physical protection include radiological sabotage, material theft, and information theft. The nuclear material security risk criterion given in the Study Charter refers to the threat of material theft. The anticipated target of theft is typically expected to be the special nuclear materials with the potential for use in a nuclear device. However, much or all of the radioactive materials handled in a nuclear energy system may be a target depending on the goals of the adversary [C-3.1, C-3.2], such as:

- Theft of nuclear material from facilities or in transport
- Theft of hazardous radioactive material from facilities or in transport

The comparison of nuclear material security risk between nuclear energy system options includes an evaluation of the potential target materials as they exist for normal operations. Further, the other aspects of physical protection relevant to nuclear material security risk are a function of specific facility designs and operations, including physical barriers and assumptions made about the protective force and adversary force capabilities. These were not considerations in this E&S Study of fuel cycles, and as a consequence, it was not possible to evaluate nuclear material security risk; the E&S Study could only inform on the materials available from the fuel cycle.

The two possible goals of acquiring INDs or RDDs/REDs are discussed separately in the following since the targeted materials are not necessarily the same. The attractiveness and availability of IND-usable materials containing highly-enriched uranium (HEU) with ^{235}U content greater than 20%, ^{233}U, or plutonium depends on the characteristics of the fuel cycle and, if reprocessing is used, on the specific reprocessing approach.

However, any radioactive material can be used for RDDs/REDs since the purpose is disruption or exposure rather than physical destruction. Spent nuclear fuel or high level waste (SNF or HLW) may be the most desirable targets since these are the most highly radioactive materials in the nuclear energy system. Other highly radioactive materials include fission products, activation products in structural materials, and some of the actinide elements.

C-3.2 Evaluation of Targets for Malevolent Use

As described above, there are two categories of materials that could be the targets for malevolent uses, attractive materials and highly radioactive materials, resulting in the two metrics used in the E&S Study to inform on the Nuclear Material Security Risk criterion.

The first metric informs on the attractiveness of the fuel cycle materials and is the same as the metric used for informing on the Proliferation Risk criterion, the material attractiveness under normal operating conditions:

- *Material attractiveness - normal operating conditions*; for normal operations, the highest material attractiveness over all sensitive points was used to compare the attractiveness of materials that could potentially be taken from the fuel cycle. The assumption was made that the fuel cycle and facility designers would take steps to lower material attractiveness for normal operations.

Further discussion for this metric is given in Appendix C-2.2.

The second metric considers the activity of fuel cycle materials which would be relevant for RDDs or REDs. Since such devices benefit from the highest radioactivity, for the purposes of this E&S Study, the materials evaluated were spent fuel or HLW, the most radioactive materials in the fuel cycle, at 10 years after discharge.

- *Activity of SNF + HLW (@ 10years) per energy generated*; the mass includes all heavy metals and fission products derived from the initial fuel materials but does not include any structural materials for the SNF or additional waste form materials for the HLW since these are determined by the technologies chosen for implementation (the same definition of SNF+HLW used for the Nuclear Waste Management metrics).

Development of evaluation metrics for this criterion benefitted from input from other activities within DOE, including the MPACT campaign and NNSA since they are both involved in evaluating the risk posed by the threat of the nuclear materials from civilian nuclear facilities being used by "sub-national" or terrorist groups for INDs or RDD/RED, and in the development of supporting technologies or approaches that would deter or prevent such actions.

References for C-3.

C-3.1. "Evaluation Methodology for Proliferation Resistance and Physical Protection of Generation IV Nuclear Energy Systems, Revision 6," The Proliferation Resistance and Physical Protection Evaluation Methodology Working Group of the Generation IV International Forum, GIF/PRPPWG/2011/003, September 15, 2011.

C-3.2. R. Bari, "Proliferation Resistance and Physical Protection (PR&PP) Evaluation Methodology: Objectives, Accomplishments, and Future Directions," submitted to ANS Journal of Nuclear Technology, 2012.

C-4. Safety Criterion

For the purposes of the fuel cycle Evaluation and Screening, the Safety Criterion is defined as follows:

***Safety** - A broad definition of safety includes the ability to build and operate an entire fuel cycle in a manner that adequately protects workers and the public. For this fuel cycle evaluation and screening, the safety criterion is focused on the challenge of meeting safety requirements for nuclear facilities, based on the premise that all commercial nuclear facilities are regulated and must meet such safety requirements.*

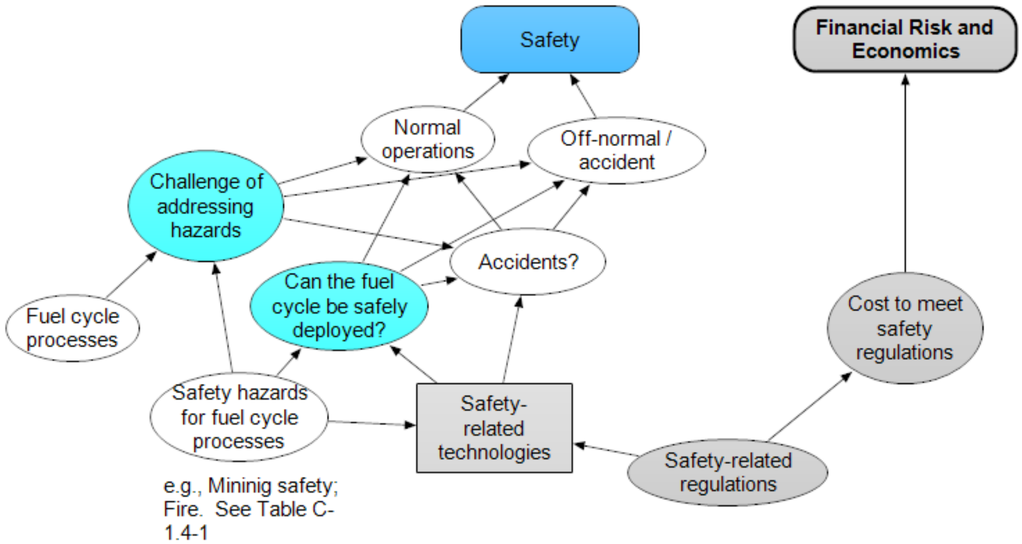
C-4.1 Background on Safety

In its simplest form "safety," in the context of nuclear energy, means ensuring a sufficient level of protection to the public, work force and environment from any adverse impacts that may result from the use of nuclear and radioactive materials in a nuclear energy system. The focus is typically on the potential for radioactive releases during the operation of nuclear facilities in normal and off-normal (accident) conditions, but may also include the potential for exposure during all stages of the nuclear fuel cycle, from fuel acquisition through nuclear waste disposal. The key consideration for characterizing fuel cycle safety is that the level of safety required is defined in NRC and other government regulations, and all commercial nuclear facilities are required to meet specific safety standards. These safety requirements are established in the U.S. Code of Federal Regulations, such as 10 CFR Part 50 "Domestic Licensing of Production and Utilization Facilities," 10 CFR Part 52 "Licenses, Certifications, and Approvals for Nuclear Power Plants," and 10 CFR Part 70 "Domestic Licensing of Special Nuclear Materials." These regulations are used by government organizations to develop rules and requirements, most of which have been developed for the deployment of the current fuel cycle facilities associated with LWRs.

Public concern regarding reactor and fuel cycle safety has risen because of the events at the Fukushima-Daiichi plant in Japan resulting from an earthquake and the associated tsunami. Additional areas of public concern can be related to the risk of deliberate releases as may be the result of terrorist activity. Both of these concerns are addressed through the establishment of regulations as discussed above.

All nuclear facilities and all elements of the nuclear fuel cycle must demonstrate that they have sufficient and appropriate design features such that they will meet regulatory requirements before they can be licensed for construction and operation. All reactor concepts and fuel cycle facilities would have taken safety into account in the design and will have all safety features (e.g. criticality controls, emergency core cooling, shutdown systems) required to meet NRC regulations. While the costs to achieve the level of safety required by regulations may vary across fuel cycles, safety-related costs are technology and facility design dependent, and would be included as part of the overall cost of each facility since all safety issues must be resolved before a facility is licensed for construction and operation. In this Study, general safety-related costs would be included in the facility cost estimates used for the Levelized Cost of Electricity as discussed in Section C-9, similar to other costs such as design, licensing activities, and site-specific costs. It is not possible to identify the fraction of the cost devoted to these activities in such generic facility cost estimates.

The approach for considering safety in this study was to understand the challenges presented by the hazards inherent in each part of the fuel cycle based on an understanding of the fuel cycle safety hazards and considering the experience in addressing these safety challenges, including identification of any safety challenges that cannot be addressed through research development and demonstration. The main factors and considerations are illustrated in Figure C-4.1, and further elaborated below. In the figure, factors shown in grey were not used directly as metrics, but either informed the development of metrics, or were more closely related to other Evaluation Criteria. As a result, the relative safety of a fuel cycle option compared to the current U.S. fuel cycle is determined by the characteristics of the challenges of the safety hazards, not the number of challenges or the complexity of the fuel cycle.



Note: Each oval represents a factor, element, or question related to safety of a fuel cycle. Rounded rectangles represent the Evaluation Criteria related to these questions. Aqua indicates factors for which Evaluation Metrics were defined, white indicates factors that are considered in development of Metric Data. Grey indicates factors not included explicitly in the Evaluation Metrics for this Criterion.

Figure C-4.1. Influence Diagram Illustrating the Primary Factors Related to the Safety Criterion.

C-4.1.1 Nuclear Safety Considerations

The principle means for ensuring the safety of nuclear facilities and operations is to control and minimize the exposure of the public, the workforce and the environment to radioactive material. Nuclear safety is achieved by a set of technology-specific design and operating decisions (represented by the grey bold box in the influence diagram):

- Facility design
- Use of appropriate operating conditions to minimize exposure to personnel and the public
- Prevention of accidents
- Mitigation of accident consequences

There are several levels safety that will be required by the NRC of any nuclear facility; first is the ability to operate the facility safely under normal operating conditions, second is to prevent off-normal, or accident conditions, and lastly, if accidents do occur, to limit the consequences in terms of risk to personnel, the public, and the environment.

- **Nominal Operations:** During all operations, the radiation exposure is kept below regulatory limits, and as low as reasonably achievable (ALARA). Such regulations would apply to all facilities and operations, such as a reprocessing plant, a waste repository, or any other fuel cycle facility. Uranium and thorium mining also needs to be considered, where the radiation exposure to miners is also regulated.
- **Off-Normal/Accident:** The NRC requires any facility to be designed to be able to prevent and mitigate accidents and their consequences, from more frequent events of low consequence that are anticipated to occur within the lifetime of the plant, down to very unlikely events that are not expected to occur within the lifetime of the plant but can have very serious consequences. The specific accidents that must be addressed are technology dependent and includes both internal and

external events. Accidents that must be considered for LWRs, for example, are reflected in the NRC Standard Review Plan Chapter 15.[C-4.3]

In order to achieve all safety objectives, the ability to predict accurately the operational and accident phenomena is required. However, the only way to evaluate the safety of a given facility to ensure that safety requirements are satisfied, as in the licensing review process, is to complete a detailed and thorough safety analysis for:

- (i) all planned normal operations of the facility
- (ii) performance under anticipated operational occurrences
- (iii) design basis events
- (iv) severe accidents of low probability.

Such analyses depend on the specific facilities and technologies used to implement a fuel cycle, and are typically not related to the fuel cycle itself. However, an exception occurs if there are aspects of a fuel cycle that are fundamentally more or less safe based on reactor physics and other basic principles.

C-4.2 Metric Development for the Safety Criterion

For the purposes of evaluation and screening, metrics for informing on Safety and the approach for estimating those metrics was based on the consideration of the processes involved in fuel cycles, the safety hazards associated with those processes, and the ability to address the challenges for those safety hazards. Additional parameters were considered in the development of the Safety metrics including the potential impact on costs associated with the fuel cycle. In this E&S, costs and financial aspects are included in the Development and Deployment Risk and Financial Risk and Economics Criteria, rather than in the Safety Criteria, specifically in the metrics of Development Cost, Deployment Cost and the capital cost components of the LCAE calculations. A detailed cost evaluation requires specification of technologies in which there are different approaches to addressing safety challenges (e.g. active vs passive systems). Given that this E&S study is being performed at the fuel cycle level without specifying specific technologies, identification of costs related to addressing safety challenges is not possible.

C-4.2.1 Fuel Cycle Processes

In pursuing the development of potential safety metrics for a fuel cycle, the safety challenges associated with each of the processes in the fuel cycle was considered. There were differences in the safety challenges between fuel cycle facilities such as enrichment, fuel fabrication, or reprocessing plants and the reactors, such as the examples given in Ref. C-4.1, with some specific examples as follows:

- Compared with a reactor, the materials in other fuel cycle facilities may be processed and then stored in a more readily-dispersible form, e.g., uranium ore powder in drums or high level waste as a liquid in storage tanks. The processing of materials necessitates transfer of materials within a facility and between facilities and changes in chemical form. In contrast, the fuel in a reactor is typically, but not always in a concentrated, solid form inside the sealed cladding.
- Fuel cycle facilities may use large amounts of hazardous chemicals that are often toxic, corrosive and/or combustible.
- Chemical processing of fissile material can lead to accidental release of hazardous chemicals or radioactive materials.

There are many facilities for any fuel cycle, even in a once-through fuel cycle. In order to consider the safety of a given fuel cycle, the fuel cycle was divided into its constituent parts and the specific safety challenges associated with each part were identified. The assessment did not attempt to quantify the level of risk for each part, but illustrated the different nature of the risks for each facility or operation. Regardless of the risks, all facilities must be designed with sufficient protection of personnel, the public, and the environment so that regulations are met; different safety features may be needed for the different types of facilities. For this study, a common set of fuel cycle processes, shown in Table C-4.1, was developed. The mapping of the fuel cycle processes to the Evaluation Groups is presented in Table C-4.2, where a √ mark indicates that the particular process is part of the Evaluation Group.

Table C-4.1. Set of Fuel Cycle Processes.

Process ID	Description
FS-1	Fuel supply - Mined uranium
FS-2	Fuel supply - Mined thorium
UE-1	Uranium enrichment < 5 wt. %
UE-2	Uranium enrichment >5 wt. %
FF-1	Fuel fabrication with unirradiated uranium (Contact handled)
FF-2	Fuel fabrication with unirradiated thorium or uranium/thorium (Contact handled)
FF-3	Recycle fuel fabrication with RU/Pu (Glove Box handled)
FF-4	Recycle fuel fabrication with RU/TRU (Remote handled)
FF-5	Recycle fuel fabrication with U3/Th/TRU (Remote handled)
RX-0	Reactor: Thermal-critical (no development required)
RX-1	Reactor: Thermal-critical (fuel development required)
RX-2	Reactor: Thermal-critical (all other thermal reactors)
RX-3	Reactor: Fast-critical
RX-4	Reactor: Sub-critical
RP-1	Reprocessing with RU/Pu product
RP-2	Reprocessing with RU/TRU product
RP-3	Reprocessing with U3/Th/TRU products
ST-1	Storage of fuel cycle materials
TR-1	Transport of fuel cycle materials
DS-1	Management and packaging of DU, RU, RTh
DS-2	Management and packaging of discharged fuel
DS-3	Preparation and packaging of high level waste

Table C-4.2. Mapping of Fuel Cycle Processes to Evaluation Groups.

Evaluation Group	Fuel Material Supply		Uranium Enrichment		Fuel Fab from New Resources		Reactors (critical and sub-critical)					Reprocessing			Recycled Fuel Fab			Storage	Transport	Disposal		
	FS-1	FS-2	UE-1	UE-2	FF-1	FF-2	RX-0	RX-1	RX-2	RX-3	RX-4	RP-1	RP-2	RP-3	FF-3	FF-4	FF-5	ST-1	TR-1	DS-1	DS-2	DS-3
EG01	✔		✔		✔		✔											✔	✔	✔	✔	
EG02	✔			✔	✔			✔										✔	✔	✔	✔	
EG03	✔				✔		✔											✔	✔		✔	
EG04	✔				✔					✔								✔	✔		✔	
EG05	✔	✔		✔		✔		✔										✔	✔	✔	✔	
EG06		✔				✔					✔							✔	✔		✔	
EG07	✔				✔						✔							✔	✔		✔	
EG08	✔					✔					✔							✔	✔		✔	
EG09	✔				✔					✔			✔			✔		✔	✔	✔	✔	✔
EG10		✔				✔			✔					✔				✔	✔		✔	✔
EG11	✔	✔		✔		✔				✔				✔			✔	✔	✔	✔	✔	✔
EG12	✔				✔			✔				✔			✔			✔	✔	✔	✔	✔
EG13	✔		✔		✔			✔				✔			✔			✔	✔	✔	✔	✔
EG14	✔				✔			✔		✔		✔			✔			✔	✔		✔	✔
EG15	✔		✔		✔		✔			✔		✔			✔			✔	✔	✔	✔	✔
EG16	✔		✔		✔		✔				✔	✔			✔			✔	✔	✔	✔	✔
EG17	✔	✔	✔		✔			✔				✔			✔		✔	✔	✔	✔	✔	✔
EG18	✔	✔		✔		✔		✔						✔			✔	✔	✔	✔	✔	✔
EG19	✔							✔				✔			✔			✔	✔	✔		✔
EG20	✔							✔					✔			✔		✔	✔	✔		✔
EG21	✔		✔		✔			✔				✔			✔			✔	✔	✔		✔
EG22	✔		✔		✔			✔					✔			✔		✔	✔	✔		✔
EG23	✔									✔		✔			✔			✔	✔			✔
EG24	✔									✔			✔			✔		✔	✔			✔
EG25	✔	✔		✔	✔			✔						✔			✔	✔	✔	✔		✔
EG26		✔				✔			✔					✔				✔	✔			✔
EG27	✔	✔		✔		✔				✔				✔			✔	✔	✔	✔		✔
EG28		✔				✔				✔				✔			✔	✔	✔			✔
EG29	✔				✔			✔		✔		✔			✔			✔	✔			✔
EG30	✔				✔			✔		✔			✔			✔		✔	✔			✔
EG31	✔		✔		✔		✔			✔		✔			✔			✔	✔	✔		✔
EG32	✔		✔		✔		✔			✔			✔			✔		✔	✔	✔		✔
EG33	✔				✔			✔			✔	✔			✔			✔	✔			✔
EG34	✔				✔			✔			✔		✔			✔		✔	✔			✔
EG35	✔		✔		✔			✔			✔	✔			✔			✔	✔	✔		✔
EG36	✔		✔		✔			✔			✔		✔			✔		✔	✔	✔		✔
EG37	✔	✔	✔			✔	✔			✔				✔			✔	✔	✔	✔		✔
EG38		✔				✔		✔		✔				✔			✔	✔	✔			✔
EG39	✔	✔		✔		✔		✔			✔			✔			✔	✔	✔	✔		✔
EG40	✔	✔				✔		✔			✔			✔			✔	✔	✔			✔

C-4.2.2 Safety Hazards

When considering the operation of a fuel cycle system, there is a large range of hazards that exist and which must be addressed to ensure safe operations. Consideration of safety in this study focused specifically on those hazards associated with the potential release of radioactive material and exposure to workers, public, and the environment. Based on these considerations, a review of the hazards that occur throughout the fuel cycle was performed and a list of these hazards developed. Categories of these hazards are shown in Table C-4.3, including a description of the hazard as well as experience with each of these hazard categories. These descriptions support an assessment and understanding of the challenge of addressing the hazards.

Hazards unrelated to radiation safety were not considered in this assessment. While important, those hazards are not a unique feature of nuclear energy, are not a function of the characteristics of nuclear fuel cycles, and are commonly addressed across a broad range of industrial applications. For example, the issue of transportation safety is frequently raised regarding the transportation of fresh and used nuclear fuel. Transportation-related risks depend on facility siting decisions and transport methods, and the largest transportation hazards are not necessarily radiological, but rather can be associated with other issues such as the potential for traffic accidents. Similarly, standard industrial safety (e.g., slips, trips, falls, etc.) was not considered when comparing fuel cycle options.

Table C-4.3. Fuel Cycle Hazard Categories.

Hazard Category	Description of Hazard	Experience with the Hazard Category
Mining safety	Mining is carried out in a number of ways, whether that is in-situ leaching, deep shaft, or open cast mining. Hazards (nuclear and non-nuclear related) are associated with the ore extraction, the refining/processing and the uranium extraction.	Uranium mining is carried out worldwide today with a significant number of years of operating experience of different mine conditions, ore deposit concentrations etc.
Radiological Exposure	Exposure to radiation and radioactive materials can result in a radiological dose. Radiological exposure is a function of time, type of radiation (alpha, beta, gamma or neutron) and intensity of the radiation.	Based on decades of experience in monitoring, processing and handling nuclear material, dose limits and robust working practices have been established. This is the case for fresh and irradiated fuels, and separated materials e.g. Pu, reprocessed uranium, minor actinide, waste streams.
Chemical	Chemical burns, reactions (e.g. strong oxidation/reduction, exothermic reaction) and toxicity. Since much of the nuclear fuel cycle is chemical processing, a number of strong acids and alkalis are used in order to extract/separate one material from another.	From the first uranium ore mined, fuel manufactured etc., chemical processing has been central to nuclear fuel cycle operations and as such, a vast experience base has been gained including on small and large scales, in extremely harsh environments (including high radiation fields, temperatures etc). Chemicals used in the nuclear industry to date include high molar nitric acids (up to 12 M), solvents (e.g. tri-butyl phosphate) and strong alkalis.
Fire	Fire is a hazard of many industries but it is primarily the concern over the source of the fire that is important here. The differentiation in origin due to the nuclear processes e.g. pyrophoric materials, chemicals used in processing etc.	Fire safety is a key part of the safety case for nuclear plants, including consideration of materials used to put out fires (e.g. to avoid uncontrolled criticality), sources of fires (e.g. uranium metal) and heat sources. There are sophisticated and highly regulated fire protection systems, equipment and procedures to ensure safety at nuclear facilities and protection of safety systems. Nuclear facilities use various systems and features, including fire protection barriers, physical separation, and fire detection and suppression equipment to meet these requirements.
Explosion	Explosion hazard is related to material processes using combustible gases (e.g. hydrogen used in reduction processes) and generation of combustible gases.	Use of potentially explosive gases in a nuclear environment is a common practice in all fuel manufacturing plants and operating reactors. Detection and control systems as well as safe working practices are well established.
Corrosive gases	In addition to the chemical hazards (see above), specific corrosive gases such as HF and UF_6 are used throughout many nuclear fuel cycles. These gases are a hazard to operators and to equipment due to their strongly corrosive nature.	Design of facilities and operations in order to safely manage and control corrosive gases are well demonstrated in the modern nuclear fuel cycle e.g. conversion and enrichment plants.
Criticality	Inadvertent criticality in the nuclear fuel cycle resulting in radiological exposure and explosions. Processing and handling of fissile material is a criticality hazard, with the potential for significant release of energy and radiation.	Criticality safety is a well-proven field in nuclear engineering, developed over many decades of experience in using and controlling fissile material. Requirements including safe by geometry or shape, moderation, reflection, mass etc. are the cornerstone of the experience base. Experience of using enrichments of up to 5wt% U235 or 10 wt% Pu are common within the nuclear industry today, with experience of up to 20 wt% and beyond in other

		industrial processes including naval fuel and test reactors.
Respiratory	Although common to many other industrial processes, the additional respiratory hazard from nuclear is the inhalation of nuclear material, especially where any powders are concerned that contain nuclear material e.g. uranium in fuel powder, contamination from cleanup of plants etc. This raises the risk concern over alpha particle radiation and resulting exposure.	Depending on the material type and the source of radioactive material, a range of respiratory protection is regularly used in the nuclear industry, in particular personal protective equipment such as dust masks, full face masks etc. Working practices to avoid airborne contamination are common, including use of glove boxes and containment, negative pressures etc.
Decay Heat	Nuclear materials continue to generate heat after the intended nuclear reactions for power generation are terminated. Removal of decay heat is a key part of reactor safety during operations, and after reactor shutdown and in the spent fuel pools (both at reactor and at fuel processing facilities). Maintaining sufficient cooling to avoid fuel failure or melt and thereby the release of radioactive material is a fundamental issue for nuclear operations, particularly in reactor operations.	All reactors have active and/or passive decay heat cooling systems, and spent fuel pools are used to remove the decay heat during periods of storage. Means by which to remove decay heat, use of backup systems and a move to passive systems demonstrates the level of experience in dealing with this significant hazard successfully. Decay heats of typically 200-300MWth are typical in modern sized reactors.
High temperatures and pressures	Both the generation of electricity and the processing involved in the nuclear fuel cycle involve the production or the use of high temperatures and pressures. During normal conditions, most reactors are operating at high temperatures and/or pressures. Monitoring of these for reactor safety is key to ensure barriers to radioactive release (e.g. containment) is maintained.	Operating temperatures and pressures are monitored within reactors on a routine basis, and the inspection of the pressure vessel to ensure integrity using a range of examination techniques is a routine practice. Similarly in the processing of nuclear material under high temperatures and pressures (e.g. fuel reprocessing etc.), experience has been gained at an industrial scale in a nuclear environment. Coolant temperatures of up to 1800 degF have been experienced in some experimental reactors, with temperatures more typically 1200 degF in commercial reactors (e.g. UK AGRs).

C-4.2.3 Safety Metrics

As illustrated in Figure C-4.1, several factors were identified that inform on the safety of fuel cycle options. Nuclear safety is managed during design and operation, and the safety requirements for licensing ensure that potential exposures to personnel, the public, and the environment are within applicable regulatory limits. Accordingly, in developing metrics to inform on the Safety Criterion, the EST focused on the relative challenge in meeting and managing the safety requirements, and on identifying whether any of these challenges are insurmountable. The following two metrics for the Safety criterion were established:

- *Challenges of Addressing Safety Hazards*
- *Safety of the Deployed System*

The Challenges of Addressing Safety Hazards metric considered the relative difficulty of addressing the safety hazards inherent in the fuel cycle and was based on the experience in addressing each hazard, as identified above in Section C-4.2.2. As an example, the U.S. fuel cycle does not include several of the processes presented in Section C-4.2.1, such as reprocessing, yet there is experience with operations and facilities within DOE and in other countries in addressing the hazards associated with reprocessing. The approach as implemented in Appendix D for this metric was to consider the identified safety hazards for each fuel cycle process to determine if the challenge associated with that hazard had ever been addressed and if there are other hazards present for which experience is not available.

The Safety of the Deployed System metric represents a determination if there were any safety issues that could not be addressed for the fuel cycle option being considered. This determination built upon the Challenges of Addressing Safety Hazards metric by considering any identified challenges and making a determination on whether they could be successfully addressed.

C-4.3 Challenges of Addressing Safety Hazards

Definition of Metric – This metric is a comparative assessment of the overall challenges in addressing the safety hazards for an Evaluation Group relative to the challenges of addressing safety hazards for the Basis of Comparison, considering previous experience in addressing safety hazards in current and past industrial processes.

The challenges of addressing safety hazards were considered relative to the Basis of Comparison based on a common set of fuel cycle hazard categories. Any new or unique hazard categories for the fuel cycle processes that were not present in the common set of hazard categories were identified and it was determined if there was any relevant experience in addressing these additional hazards. The hazard categories associated with each fuel cycle process were identified and the resulting mapping of the fuel cycle hazard categories discussed above to the fuel cycle process list is shown in Table C-4.4. The check mark in the table indicates the fuel cycle process includes the checked hazard category.

Table C-4.4. Mapping of Fuel Cycle Hazard Categories to Fuel Cycle Process.

Process ID	Fuel Cycle Process	Mining Safety	Radio-logical Exposure	Chemical	Fire	Explosion	Corrosive Gases	Criticality	Respira-tory	Decay Heat	High Temperatures and Pressures	New Hazards Identified
FS-1	Fuel supply - Mined uranium	✔	✔	✔					✔			
FS-2	Fuel supply - Mined thorium	✔	✔	✔					✔			
UE-1	Uranium enrichment < 5 wt. %		✔	✔			✔	✔	✔			
UE-2	Uranium enrichment >5 wt. %		✔	✔			✔	✔	✔			
FF-1	Fuel fabrication with unirradiated uranium (Contact handled)		✔	✔	✔	✔	✔	✔	✔			
FF-2	Fuel fabrication with unirradiated thorium or uranium/thorium (Contact handled)		✔	✔	✔	✔	✔	✔	✔			
FF-3	Recycle fuel fabrication with RU/Pu (Glove Box handled)		✔		✔	✔	✔	✔		✔		
FF-4	Recycle fuel fabrication with RU/TRU (Remote handled)		✔		✔	✔	✔	✔		✔		
FF-5	Recycle fuel fabrication with U3/Th/TRU (Remote handled)		✔		✔	✔		✔		✔		
RX-1	Reactor: Thermal-critical (limited level of development req'd, eg. PWR, HWR)		✔	✔	✔	✔		✔		✔	✔	
RX-2	Reactor: Thermal-critical (all other thermal reactors)		✔	✔	✔	✔		✔		✔	✔	
RX-3	Reactor: Fast-critical		✔	✔	✔	✔		✔		✔	✔	
RX-4	Reactor: Sub-critical		✔	✔	✔	✔		✔		✔	✔	✔
RP-1	Reprocessing with RU/Pu product		✔	✔	✔	✔		✔		✔		
RP-2	Reprocessing with RU/TRU product		✔	✔	✔	✔		✔		✔		
RP-3	Reprocessing with U3/Th/TRU products		✔	✔	✔	✔		✔		✔		
ST-1	Storage of fuel cycle materials		✔					✔		✔		
TR-1	Transport of fuel cycle materials		✔					✔		✔		
DS-1	Disposal of DU, RU, RTh		✔	✔								
DS-2	Disposal of Discharged Fuel		✔					✔		✔		
DS-3	Disposal of High Level Waste		✔	✔				✔		✔		

Based on the review of each process and the corresponding hazard categories, only the Sub-critical Reactor (RX-4) was identified as having safety hazards that have challenges that have not been previously addressed. Two additional hazards have been identified for these systems which are composed of either an ADS or a FFH. The additional hazards identified are as follows:

1) Sub-Critical Operations – applies to both ADS and FFH

Coupling of neutron source with fission blanket: Sub-critical systems operate with an external neutron source that must create neutrons within the blanket system. In an ADS system, this involves injecting a beam of protons into a target located within the fission blanket. For FFHs, the typical approach is to allow the fusion neutrons to migrate from the fusion chamber, through appropriate barriers and walls and into the fission blanket. This coupling has not been previously demonstrated and may have potential safety issues with control of proton beams, the fusion/fission barrier systems and maintenance, and the interaction of beam power with blanket coolant flow rate.

Neutron source: Neutron source excursions (ADS beam fluctuations and inadvertent operation at above design power and FFH plasma disruptions, magnetic field interactions with fission blankets) represent additional hazards that have not been previously addressed through demonstration. Systems for feedback of operating power to control of the neutron source power have not been demonstrated.

Sub-critical operation: The operational dynamics of a subcritical system are different from those of a critical system and largely driven by the neutron source, feedback coefficients, and degree of subcriticality. Extended operations of a source-drive subcritical system at significant power levels has not been previously demonstrated and potentially introduces new accident categories that must be considered.

2) Large-scale tritium handling – only applies to FFH

For FFH systems, significant amounts of tritium will be produced, processed and used as fusion fuel. Although there is experience with tritium production, the experience is for far smaller quantities than required in a fusion system. D-T fusion requires burning an estimated 56 kg of tritium per GWt-year, which is a much larger quantity when compared with the total current non-military tritium production rate of ~1.8 kg/yr. [C-4.2] The challenge in handling tritium is primarily ensuring that it can be contained to prevent release from the nuclear facilities and handling represents a challenge because of tritium permeability and the need for addressing all of the tritium transport pathways. In addition tritiated vapors (HTO, HT) are corrosive and can lead to leakage over time if not addressed. Tritium management issues occur in the tritium recovery from lithium targets, tritium target manufacturing (injection pellets for magnetic confinement and targets for inertial confinement).

Note that identifying additional hazard challenges did not imply that the corresponding processes that have these hazards could not be deployed safely. Rather, it acknowledged that there were additional hazards that must be considered and addressed before deployment. The determination if fuel cycle processes could be safely deployed is considered by the Safety of the Deployed System Metric, discussed in Section C-4.4.

The challenges of addressing safety hazards for an evaluation group were determined by considering the full set of hazards for all fuel cycle processes that exist in the Evaluation Group, and making a judgment about whether those hazards are more challenging, less challenging, or comparably challenging to the hazards that are addressed in the Basis of Comparison. Table C-4.5 provides the bin structure for this metric:

Table C-4.5. Bin Structure for the Challenge of Addressing Safety Hazards Metric.

Bin	Bin Description
A: Potentially much less challenging that the current US nuclear fuel cycle	Identified hazards are potentially much less challenging to address than those hazards that have been encountered and addressed through past R&D and/or current and past industrial processes.
B: Potentially less challenging than the current US nuclear fuel cycle	Identified hazards are potentially less challenging to address than those hazards that have been encountered and addressed through past R&D and/or current and past industrial processes.
C: Potentially similar in challenge to the current US nuclear fuel cycle	Identified hazards are potentially similar in challenge to address than those hazards that have been encountered and addressed through past R&D and/or current and past industrial processes.
D: Potentially more challenging than the current US nuclear fuel cycle	Identified hazards are potentially more challenging to address than those hazards that have been encountered and addressed through past R&D and/or current and past industrial processes.
E: Potentially significantly more challenging than the current US nuclear fuel cycle	Identified hazards are potentially significantly more challenging to address than those hazards that have been encountered and addressed through past R&D and/or current and past industrial processes.

C-4.4 Safety of the Deployed System

Definition of Metric – The metric is defined as a determination of whether a fuel cycle can be deployed safely. It builds from the same information used to identify the challenges in meeting safety requirements: if any one of the safety hazards identified was determined to be insurmountable (it has never been successfully addressed, and there is no evidence that it can be safely addressed), then the fuel cycle was determined as “unable to be safely deployed.” If all of the identified safety hazards could be addressed, then the fuel cycle was considered to be able to be deployed safely. This metric therefore represents a yes/no result to the question “Can the fuel cycle be safely deployed?”

The information needed to make the determination if the fuel cycle can be deployed safely is a result of the Challenge of Addressing Safety Hazards metric in which the hazards for a given fuel cycle were identified and evaluated. Fuel cycles that were determined to be “more challenging” than the Basis of Comparison have additional hazards that have not been previously addressed through demonstrated experience. For these additional hazards, a further evaluation was made to determine if they represented hazards that could not be addressed and therefore represented an insurmountable safety issue. This necessarily relied on a judgment that a viable approach would likely exist for addressing these additional challenges based on extensions of existing technologies, research, or other approaches. If no approaches were identified, then for the purposes of the Evaluation and Screening study, these fuels cycles were determined to not be able to be deployed safely.

For the two additional hazards that were discussed in Section C-4.3, it was anticipated that a reasonable research and development program would result in the development of suitable approaches for the operation of sub-critical reactors, coupling of neutron sources to the sub-critical reactor, and the handling of large quantities of tritium. There is on-going research in these areas and the necessary R&D and development costs were considered in the Development Cost, Development Time, and Deployment Cost fuel cycle process data. Note also that continuous improvement in the safety of nuclear reactor systems is an important goal reflected in most reactor R&D programs, such as Generation IV. Therefore, while the standard for this metric is to be able to address identified safety hazards and meet identified regulatory requirements, additional R&D may be needed to support further enhancements to reactor safety.

References for C-4.

C-4.1. IAEA2008. "Guidance for the Application of an Assessment Methodology for Innovative Nuclear Energy Systems, INPRO Manual – Safety of Nuclear Fuel Cycle Facilities", IAEA-TECDOC-1575 Rev.1 (November 2008)

C-4.2 The MITRE Corporation, "Tritium," JSR-11-345 (November 2011).

C-4.3 U.S. Nuclear Regulatory Commission, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition — Transient and Accident Analysis, NUREG-800 (June 1987)

C-5. Environmental Impact Criterion

For the purposes of the fuel cycle Evaluation and Screening, the Environmental Impact Criterion is defined as follows:

Environmental Impact - *A broad definition of environmental impact may include the range of environmental disturbance, resource withdrawals from and emissions into the environment. For the purpose of this fuel cycle evaluation and screening, the assessment of environmental impact is focused on environmental disturbance and emissions due to fuel acquisition, energy production and waste disposal.*

C-5.1 Background on Environmental Impact

Environmental Impacts can encompass a wide range of factors, such as the exhaustive list addressed in a comprehensive 'environmental impact statement' or 'environmental assessment'. The broad range of issues can include everything taken from or released to the environment, and may also include related issues such as operational safety and cost. Many factors that might be considered under Environmental Impacts either have their own specific criterion, or are better addressed under another criterion.

The influence diagram for the environmental impact criterion is shown in Figure C-5.1. The production of nuclear energy inevitably will result in disturbances to the environment. The development of the specific metrics for use in the evaluation of this criterion focused on two primary questions:

- How large is the demand of the fuel cycle on the natural resources used?
- How large is the quantity of the emissions released to the environment by the fuel cycle?

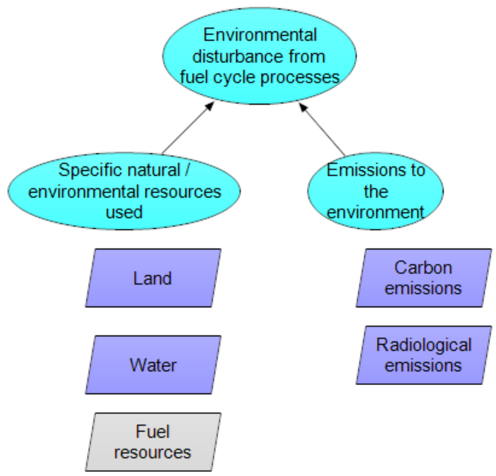
The first of these questions focuses on the demand for natural resources, with the exception of the fuel resources themselves, required to support the nuclear fuel cycle. The second relates to the releases back into the environment from the facilities associated with the nuclear fuel cycles.

C-5.2 Metric Development for the Environmental Impact Criterion

The question of "how much" or "how large" was the demand for natural resources was addressed by the area of land required to site the facilities and the volume of water used. In a similar way the quantity of emissions released to the environment was addressed by the mass of CO_2 released and the radiologic exposure to the workers which was used as an indicator of potential public radiologic exposure. These considerations resulted in the four quantifiable metrics to represent environmental impact:

- km^2 of land use per energy generated
- Volume of water use per energy generated
- Radiological exposure - total estimated worker dose per energy generated (as leading indicator for public dose potential)
- Mass of carbon emission - CO_2 released per energy generated

Some additional factors that might normally be considered under 'Environmental Impacts' were addressed under other high-level criteria. For example, energy use is often considered as a measure of environmental impact. But, in this evaluation, the financial costs of energy use was considered directly within the Financial Risk and Economics Criterion while the CO_2 emissions were considered here as an environmental impact. Other Criteria that overlapped with the Environmental Impacts Criterion included Resource Utilization and Nuclear Waste Management. The Resource Utilization criterion included metrics for quantity of fuel material (uranium and/or thorium) used by a fuel cycle per unit energy produced while the impacts associated with mining, milling, and refining were included with the Environmental Impacts criterion. The Nuclear Waste Management criterion considered the generation of radioactive wastes requiring disposal while the Environmental Impacts included environmental aspects associated with the storage, transport, and emplacement of wastes as shown by the influence diagram, Figure C-5.1. Radioactive releases to the environment resulting from acts of terrorism are addressed in the Nuclear Material Security Criterion.



Note: Each oval represents a factor, element, or question related to environmental impact. Blue parallelograms represent factors for which Evaluation Metrics were defined and are driven by the characteristics of fuel cycle. Grey parallelograms indicates other factors relevant to environmental impact that are not included explicitly in the Evaluation Metrics for this Criterion because they are covered under other criterion.

Figure C-5.1. Influence Diagram for Environmental Impact.

The fact that some metrics in Environmental Impacts tended to correlate with some of the other Criteria was a consequence of overlap between the Criteria themselves. This was acknowledged and addressed, to the extent practical, within the definitions of the individual metrics.

Many environmental impacts are significantly influenced by local considerations and design choices that are not intrinsic to the particular fuel cycle process employed. For example, the non-radioactive composition of waste streams for a fuel cycle can be strongly influenced by design choices; investment in treatment and/or recycle of waste streams is driven by resource availability and other economic and regulatory considerations; and resulting impacts to local populations are strongly influenced by site-specific parameters (i.e. local climatology, hydrology, etc.).

Consequently, environmental metrics were selected and the impact estimates developed with the objective of minimizing the bias introduced by factors that are not intrinsic to the process being evaluated. A further objective was to avoid the need to make estimates based on inadequate quantity or quality of data.

Relationship to other Criteria

This Evaluation and Screening has a defined set of Criteria of which Environmental Impacts is just one. Many of the factors that might be considered under Environmental Impacts either had their own specific Criterion, or were better addressed under another Criterion as explained below:

Nuclear Waste Management

Waste management is an important component of environmental impacts. However, this Study has a separate criterion for nuclear waste. The Nuclear Waste Management Criterion is specifically focused on issues related to the disposition of all the radioactive waste streams from the fuel cycle. Therefore, the potential emissions (e.g., releases) from nuclear waste disposal were not included in the Environmental Impacts criterion, while other emissions were considered here. This resulted in emissions from mining fuel resources (including incidental release of radioactive material from mining and milling) being considered in Environmental Impacts, but the radioactive waste streams, including depleted uranium from enrichment, being considered under Nuclear Waste Management.

Safety

Operational safety issues are addressed under the Safety Criterion, and are not included in the Environmental Impacts criterion.

Resource Utilization

Resource utilization is an important component of environmental impacts. However, this evaluation has a separate criterion for Resource Utilization. For the purpose of this evaluation, the Resource Utilization Criterion was focused on the quantity of fuel resource material (uranium and/or thorium) used by a fuel cycle. Therefore, the quantity of uranium and/or thorium used was not included in the Environmental Impacts Criterion, while other resources were included here. This results in the resources used for mining being considered in this criterion, but not the fuel material itself.

Approach to Metric Evaluation

A common approach was used to evaluate each of the four metrics that were identified. Each of the environmental impacts was estimated by developing an impact factor that represents the impact per unit of production (e.g. MTHM, SWU, or GWe-yr) from each fuel cycle process. As shown in Figure C-5.2, each fuel cycle process requires water, material, and energy. Process operations also produce by-product streams. Impacts for each fuel cycle process were developed by quantifying typical input and output streams. These impacts were then converted to impact factors by normalizing to unit process output (with the exception of the 'Storage, Transport, and Disposal' step, which was, by necessity, normalized per unit of input). These impact factors were then scaled to estimate impacts for a broad range of fuel cycles by applying the mass flows to the applicable fuel cycle process steps. For each fuel cycle, the resulting impacts were further converted to a net impact per unit electricity produced.

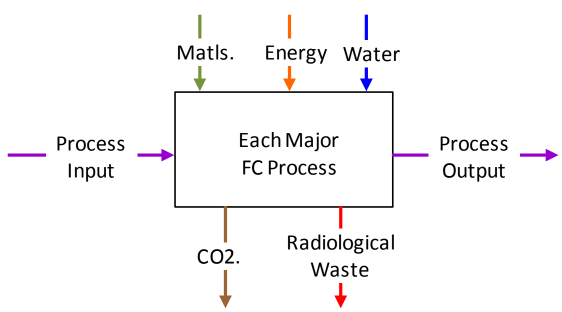


Figure C-5.2. Material and Energy Balance for Each NFC Phase.

Figure C-5.3 is a block flow diagram showing the various nuclear fuel cycle (NFC) process steps along with the candidate options that may be employed in each process. The diamonds in Figure C-5.3 do not indicate decision points but simply splits of the material flow process.

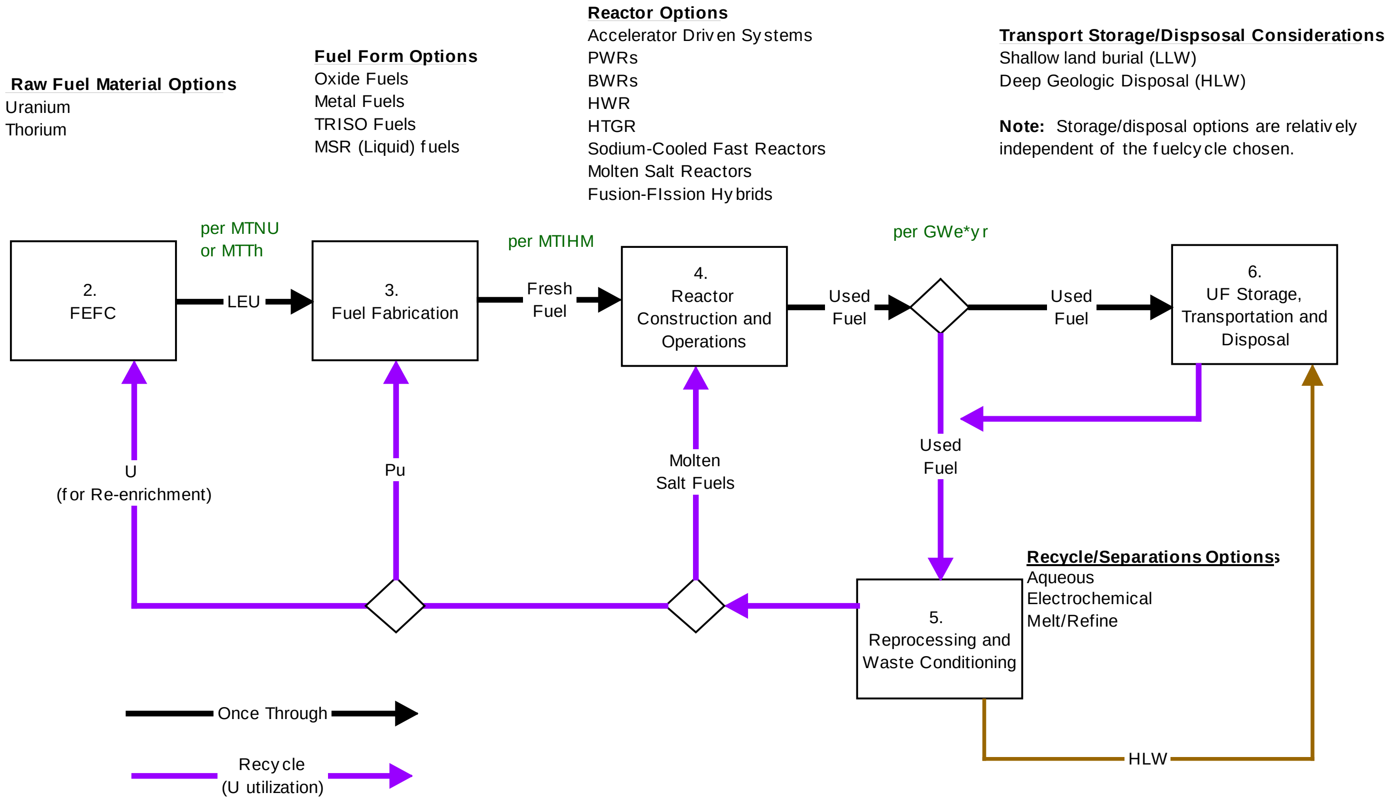


Figure C-5.3. Illustration of Range of Candidate Technologies and Options for Phases of the Nuclear Fuel Cycle.

The front-end of the nuclear fuel cycle (FEFC) includes uranium mining, milling and refining, conversion, enrichment, and fuel fabrication (conversion and enrichment are not necessary for Th fuels). Impact estimates, based on existing operational data and supplemented by engineering judgment when needed, were provided for each of the processes that compose a fuel cycle.

Impact estimates for the processes in the FEFC were summarized from [C-5.1]. Estimates for the remaining phases of the fuel cycle are developed in this section. Estimated impacts were those associated with steady state process operations and, with the exception of reactor operations, did not include the construction or decommissioning impacts. The CO_2 impacts associated with reactor construction were included because of the substantial embodied energy associated with construction materials and also because CO_2 emissions associated with reactor operations were considered negligible.

Subject matter experts from within the FCRD Fuel Cycle Technology campaigns knowledgeable on each of the fuel cycle phases were consulted when identifying the candidate processes that could be used within each fuel cycle phase. These options are shown in Figure C-5.3. Impact estimates were not developed for each of these candidate technologies. Rather, options for each fuel cycle process were grouped into those with similar impacts. Impacts were first estimated based on a reference process based on the current light water reactor (LWR) fuel cycle. Separate impacts were developed for alternate options only if data were available to support development of a separate impact estimate and when the estimate resulted in a significantly different impact with respect to one of the four metrics described above.

The impact estimates, or factors, are intended to provide a best estimate of the actual impact rather than a conservative or bounding estimate. When developing impact estimates, the following guidelines were used.

- Estimates were based on the presumption of a mature nuclear fuel cycle operating at steady state, which meant that fuel was not consumed from stockpiled reserves but was supplied from natural resources, new plants came on line as aging plants retire, etc.
- Estimates were 'forward-looking' in that they were not unduly based on past technologies or practices. Estimates were based, to the extent possible, on contemporary data from operating plants and/or mature designs supported by peer reviewed documentation. However, estimates did not attempt to anticipate technology developments that would cause the impacts to evolve going forward (and in the case of fuel choice, did not attempt to forecast the effects of resource depletion and new discoveries).
- Efforts were made to avoid estimates based on site-specific or other local factors that did not represent the industry as a whole. Examples of this site-specific bias would be things such as unusually high investment in water conservation due to local scarcity, local hydrology or atmospheric conditions that affect doses, etc. When needed, generalized assumptions were made to eliminate the effects of site-specific factors such that the resulting impact estimates were based on the operation itself rather than on the facility location or the administrative controls applied to the operation. However, it was not always possible to identify and completely compensate for local factors due to limited data.
- Significant effort was not expended pursuing impact estimates for a part of a fuel cycle if it could be reasoned to be negligible with respect to the impact summed across the complete fuel cycle. For example, water use for reactor cooling dominates water use for all other fuel cycle processes with the exception of the FEFC for once-through fuel cycles.

Because the fuel cycle processes for advanced fuel cycles may include technologies for which there is little or no data and/or operational experience, it was expected that there would be areas where there was insufficient data and/or understanding to support a credible estimate. When sufficient data was not available to make a defensible estimate of the impact, surrogate processes were used to represent the impacts.

C-5.3 Land Use per Energy Generated

Definition of Metric - Land Use was defined to include land not available for other purposes as a result of the fuel cycle processes contained in the nuclear fuel cycle such as mining, reactors and fuel fabrication plants. This included both temporary and permanent land occupied by facilities and within the exclusion area (i.e. inside the fence). Land use was measured in square kilometers and amortized over the operating facility's lifetime to obtain a metric for land used per unit of production (e.g. MTHM, SWU, GWe-yr, etc.). Land uses associated with obtaining material inputs used in fuel cycle activities was not included in the scope of the present document. Land use impacts that represent ~1% or less of that associated with FEFC processes were considered negligible.

Land use was separated into two categories: permanent and non-permanent in order to differentiate land use that could be rehabilitated for unrestricted use from that which would remain restricted for the foreseeable future. For this steady state assessment, this distinction was not relevant because it was assumed that new facilities were brought on line as aging facilities were decommissioned and the land was rehabilitated.

Determination of Impact Factors

<u>Front-end of the Nuclear Fuel Cycle</u>

The FEFC included subprocesses for extracting the raw ore and preparing it in a form suitable for the fuel fabrication process. For uranium-based fuels, this included mining, milling, conversion, enrichment, and deconversion of DUF_6 tailings from the enrichment process. Land use impact factors for FEFC processes included land needed to manage the associated waste/by product streams. Impact factors for mining and milling were based on a mix of underground, open pit, and in-situ mining representative of the current industry and normalized per metric tonne natural uranium (MTNU). Because milling operations are typically co-located with, and part of the mining operation, the land use impact factor for mining also included the land associated with the milling process.

For underground mining, the majority of site operations takes place underground, and most waste rock and over burden is backfilled into the mine once mining operations have ceased. At the conclusion of mining; tailings, equipment that was not able to be salvaged, waste rock, and overburden that may contain hazardous elements, are typically buried at the mine site and covered permanently with enough clay and soil to reduce both gamma radiation and radon emissions to levels near those naturally occurring in the region, and enough rock to resist erosion. A vegetation cover is then established to control run-off from erosion processes. Nonetheless, tailing confinements represent permanent land use because radioactive and toxic elements will remain for long into the future. Land use data for underground mining is based on the Cigar Lake uranium mine. Available data lists only the total land used in site operations and does not distinguish between the natures of the land use. It was assumed that 80% of the land used for mining operations would be used for tailings and/or other long-term needs and could not thus be reclaimed for other uses.

Open-pit mines are large open excavations that can disturb large areas of earth, referred to as overburden, in order to reach the underlying ore body. The management of overburden results in significant land use while the mine is active, although much of the land used for open pit mining may eventually be reclaimed. For the purpose of this study, it was assumed that 50% of the land used for mining operations would be used for tailings and/or other long-term needs and could not thus be reclaimed for other uses. Data for open pit mines was based on the Mary Kathleen, Nabarlek, Ranger, and Rössing mines.

For in-situ leaching (ISL), mining operations have minimal land disturbance. There is only a uranium-containing solution, and no ore, or overburden to be removed as is the case in conventional mining. Upon decommissioning, wells are sealed or capped, pipes and process facilities removed, any evaporation pond restored with vegetation, and much of the land can be returned to its previous uses. While the operating surface facilities may be removed when the operations cease, the portion of land occupied by tailings from the ISL processing would be considered permanent use. Tailings from ISL result from the precipitation of unwanted elements during the milling process. Since there is no ore, these tailings represent a very small volume of waste relative to other U mining operations. For the purpose of this study, it was assumed that 5% of the land was for permanent use and 95% for non-permanent use. Data for in-situ mining was based on the Rosita, Kingsville, Holiday/El Mesquite, and Beverly mines.

U_3O_8 yellowcake from the milling process is converted to uranium hexafluoride (UF_6) for use in enrichment operations. The major suppliers of conversion capability are BNFL (United Kingdom), Cameco (Canada), AREVA subsidiary Comurhex (France), ConverDyn (U.S.), and Minatom (Russia). Most facilities employ a 'wet' process that begins with a solvent extraction (SX) step followed by a fluorination step (F). Given its prevalence, the wet process was taken as the reference technology. Incomplete data were available for the French Malvesi and Tricastin plants, along with Cameco's Canadian facilities and two smaller, now-retired US plants, the General Atomics Sequoyah plant in Oklahoma and the DOE Fernald facility in Ohio. While both the Sequoyah and Fernald facilities would not satisfy today's environmental standards, historical data from these plants concerning, for example, energy, land and water use was utilized in the case where more modern data was not available. Data for land use was taken from Ref. C-5.2 and normalized per MTNU.

Two enrichment technologies are deployed at industrial scale: gaseous diffusion and gas centrifuge. As diffusion plants in the US and France are being retired in favor of centrifuge technology, they are largely of historical interest and the centrifuge enrichment process was selected as the basis for estimating environmental impacts. Reference data for centrifuge enrichment was based on facilities operated by Urenco, operator of plants in the United Kingdom, Germany, the Netherlands and the United States. In particular, an EIS was published in 2005 for the Louisiana Energy Services Urenco USA enrichment facility (formerly known as the National Enrichment Facility) [C-5.3]. Impacts from enrichment were normalized per separative work unit (SWU) and could thus be re-scaled for fuel cycles employing fuels of different enrichments (i.e. requiring more SWU per unit fuel). A third technology, the Silex process being developed by GE-Hitachi, has recently been licensed by the NRC. Although this technology was not selected as the reference technology because it has not yet been deployed, it is expected to be more representative of the impacts of next-generation enrichment technologies. Data from the EIS [C-5.4] indicate that the SILEX process will use approximately 65% of the water and 17% of the land relative to centrifuge enrichment.

Deconversion of depleted uranium hexafluoride (DUF_6) tailings generated during enrichment produces yellowcake (DU_3O_8) that in the Evaluation and Screening is either a resource or a waste, depending on the fuel cycle. The recovered DU_3O_8 powder is in a more chemically stable form than DUF_6 and is generally suitable for disposal as low-level radioactive waste (LLRW). Construction of deconversion facilities at Portsmouth and Paducah was completed in 2010, and as of 2012 the plants have commenced limited operations. The Paducah facility is expected to have an annual throughput of 18,000 MT DUF_6/year and operate for 25 years, while the Portsmouth facility will have an annual throughput of 13,500 MT DUF_6/year and 18-year operational lifetime. These two facilities were taken as reference plants. Data for all impact categories considered in this study were averaged between the two plants. The basis unit for deconversion was tonnes uranium in the DU_3O_8 product.

Unlike uranium, thorium does not require isotopic enrichment. However, because thorium is a fertile rather than a fissile material, sustainable Th-based fuel cycles require fissile material for startup, reprocessing to recover ^{233}U, and appropriate used fuel processing and refabrication processes or the use of an externally-driven system. The demand for thorium has historically been very small because it is only used in small quantities for specialty applications such as catalysts, gas lantern mantles, and welding rods. Nonetheless, substantial amounts of thorium are extracted from the earth because thorium occurs in ores containing currently valuable materials such as the rare earth elements (REE), titanium, and iron. As a consequence, large quantities of thorium are currently being separated in impure form and managed as a waste. Since thorium can be readily obtained as a byproduct of existing mining operations for the foreseeable future, most of the environmental and safety impact resulting from thorium production can be attributed to mining and production of other valuable materials and would occur whether thorium were recovered for use in a fuel cycle or not. Thus, the appropriate metric for recovering thorium was the incremental impact resulting from the added thorium recovery operations. Consequently, the land use associated with Th recovery and extraction was considered to be negligible.

FEFC land use impacts are summarized in Table C-5.1. Additional detail associated with development of the impact factors can be found in Ref. C-5.1.

Table C-5.1. Summary of Land Use Impact Factors for the FEFC.

	Uranium Fuels					Thorium Extraction and Refining
	Mining	Milling	Conversion	Enrichment	Deconversion	
Normalization Unit	MTNU		MTNU	SWU	MTDU	MTTh
Land Use	2.8E-04		3.3E-06	9.0E-09	9.3E-05	negligible
Permanent (km^2)	3.1E-05		2.6E-08	0	0	negligible
Non-Permanent (km^2)	2.4E-04		3.3E-06	9.0E-09	9.3E-05	negligible

Fuel Fabrication

Primary data for the land footprint of the fuel fabrication process are obtained from [C-5.5] for UOX fabrication and from [C-5.6] for MOX fabrication. The Westinghouse Columbia UOX fuel fabrication plant occupies 1,160 acres of land. Given a throughput of 1,150 MTIHM/yr and a facility lifetime of 40 years, land use at the Westinghouse facility is 102 m^2/MTIHM. The Areva MELOX MOX fuel fabrication facility occupies 35 acres of land [C-5.6]. Given a throughput of 195 MTIHM/yr and a facility lifetime of 40 years, land use at the MELOX facility is 18 m^2/MTIHM. The discrepancy between the land footprint of the two facilities arises from their locations; the Areva MELOX facility is located on their Marcoule Nuclear Site, currently home to the Phénix prototype fast breeder reactor, while the Westinghouse facility is standalone. The MELOX land use may thus be artificially small as it benefits from infrastructure and land exclusion areas shared with the Phénix facility. Therefore, the larger Westinghouse land use number was used for both types of facilities.

Reactor Construction and Operations

Estimating land use for reactor construction and operations presented a unique challenge for two reasons. First, little or no land used for reactor construction and operations is permanent. Although considerable land is committed to the production of nuclear energy, the vast majority of the land is unmodified and merely serves as a buffer zone around the plant. This, along with actual land occupied by the reactor plant and supporting facilities, would be available for other use following reactor decommissioning.

Second, the temporary land occupied is not proportional to the energy produced but to the energy production capacity. A 1GW$_e$ reactor with a 60-year plant lifetime and a 90% capacity factor will produce 54 GWe-yr of energy. So, the land impact per GWe-yr is 1/54th of the land that is actually committed to support the associated reactor. Although defined this way, this metric does not provide a ‘snapshot’ of the land tied up in nuclear plants at any given time, it does provide a land use value for comparison with other environmental impacts based on impacts per unit energy produced. Note however that, if the same amount of land was used for a reactor over ‘n’ reactor lifetimes, the land use for reactor operations would be reduced by a factor of 1/n.

The land use impact factor derived below is based on the land committed to a typical reactor plant amortized over the energy produced during its lifetime (km^2/GWe-yr). This is essentially the equivalent of assuming that the land within the exclusion area of a reactor was not recoverable following decommissioning of the reactor. So, in one sense, one might argue that this land use impact factor significantly underestimates (i.e. by a factor of 54) the land that is actually occupied by nuclear power plants at any given time. However, one could also argue that essentially no land is consumed by the process and it therefore significantly overestimates land use. The approach taken was considered by the EST to be a reasonable compromise.

Land use for reactor construction and operations was chosen as the exclusion area associated with the reactor facility. For example, Figure C-5.4 depicts the exclusion area (red circle) and site boundary (black polygon) of Crystal River Unit 3 [C-5.7]. The site boundary may be determined by factors not directly related to reactor needs and may include areas open for public use. For example, Highway 19 runs through the eastern portion of the Crystal River site. The exclusion area is the land withdrawn from public use and was chosen as the measure of land used for reactor construction and operations.

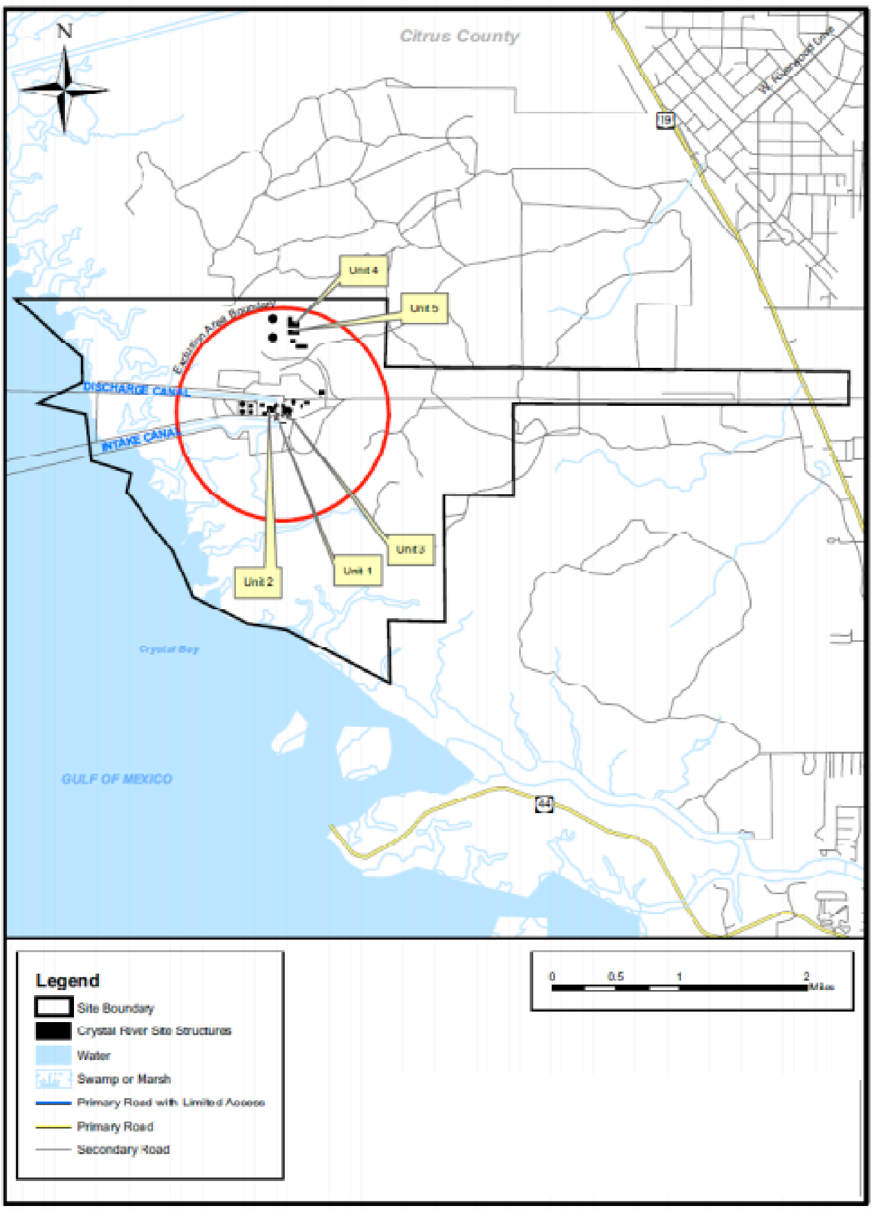


Figure C-5.4. Crystal River Unit 3 Site Boundary and Exclusion Area. [C-5.7]

More generally, Ref. C-5.8 reviews land use impacts for relicensing currently existing reactors. Within Ref. C-5.8, 19 sites explicitly state their exclusion area. The land use for these 19 sites was averaged to estimate the land used for a generic 1 GW_e generating facility as 3.93 km^2. The chosen reference facility, Wolf Creek, which has a generating capacity of 1,165 MW(e), has a circular exclusion area with a radius of 0.75 miles [C-5.9]. Assuming a 60 year lifetime and a capacity factor of 0.9, the resulting land use is 0.0727 km^2/GWe-yr.

Reprocessing and waste conditioning

The La Hague reprocessing facility sits on a 300 ha site. Given an annual capacity of 1,700 MTIHM processed per year and a 40-year facility lifetime [C-5.6], the land use at the La Hague plant is 44 m^2/MTIHM. This land use estimate may be somewhat low due to the fact that production over the facility lifetime will be less than 40 times the 1700 MTHM per year capacity. This is unlikely to be significant since the land use for reprocessing is only a very small fraction of the land impacts associated with the nuclear fuel cycle.

Storage, Transport, and Disposal

Interim Surface Storage - The land footprint of the Independent Spent Fuel Storage Installation (ISFSI) facility would encompass the storage pads themselves and supporting structures within a 40 hectare restricted access area. An isolation perimeter surrounding the restricted area as well as facilities supporting fail access to the site contribute to the total land use reported in the EIS, 120 hectares (1.2 km^2) [C-5.10]. Land use per MTIHM was thus estimated at 3.0E-05 km^2/MTIHM.

Deep Geologic Repository

The proposed Yucca Mountain repository was used as an example of a deep geologic repository for this study. This proposed repository consisted of a land withdrawal area of 150,000 acres (600 km^2), Figure C-5.5. [C-5.11].

Ref. C-5.12 was used to determine a generic land footprint for geologic disposal, which examined potential generic waste emplacement approaches for a 140,000 MTIHM repository. Chapter 4 of Ref. C-5.12 provided information on disposal approaches including panel dimensions plus related access, disposal, and service drifts and the number of panels needed to dispose 140,000 MTIHM for 5 representative disposal concepts. The footprint for these concepts ranged from 7 to 45 km^2, with an average of 19.5 km^2. Based on this information, a footprint of 20 km^2 was used to represent a geology-independent generic repository. In addition to the land footprint, 40 CFR 191 requires a maximum setback distance of 5 km. Based on this regulatory guidance a 5 km setback from the repository footprint was assumed for determination of the total controlled area.

For simplicity, it was assumed that repository footprint and controlled area were circular. Thus, the radius of the controlled area was calculated as:

$$5km + \left\{\sqrt[2]{20km^2/\pi}\right\} = 7.52km$$

There is no established upper limit for repository size. Ref. C-5.12 indicated the feasibility of developing a repository of up to 140,000 MTIHM in multiple geologies. Ref. C-5.13 estimated a similar capacity of 150,000 MTIHM for spent fuel disposal in Yucca Mountain with minimal perturbations to the then-existing design (loading remains at ~75 MTIHM/acre, drift footprint increased somewhat to 8 km^2). Therefore, a capacity of 140,000 MTIHM was assumed. The land use impact factor for geologic disposal was calculated as

$$\frac{\pi * (7.52km)^2}{140{,}000\ MTIHM} = .00127\frac{km^2}{MTIHM}$$

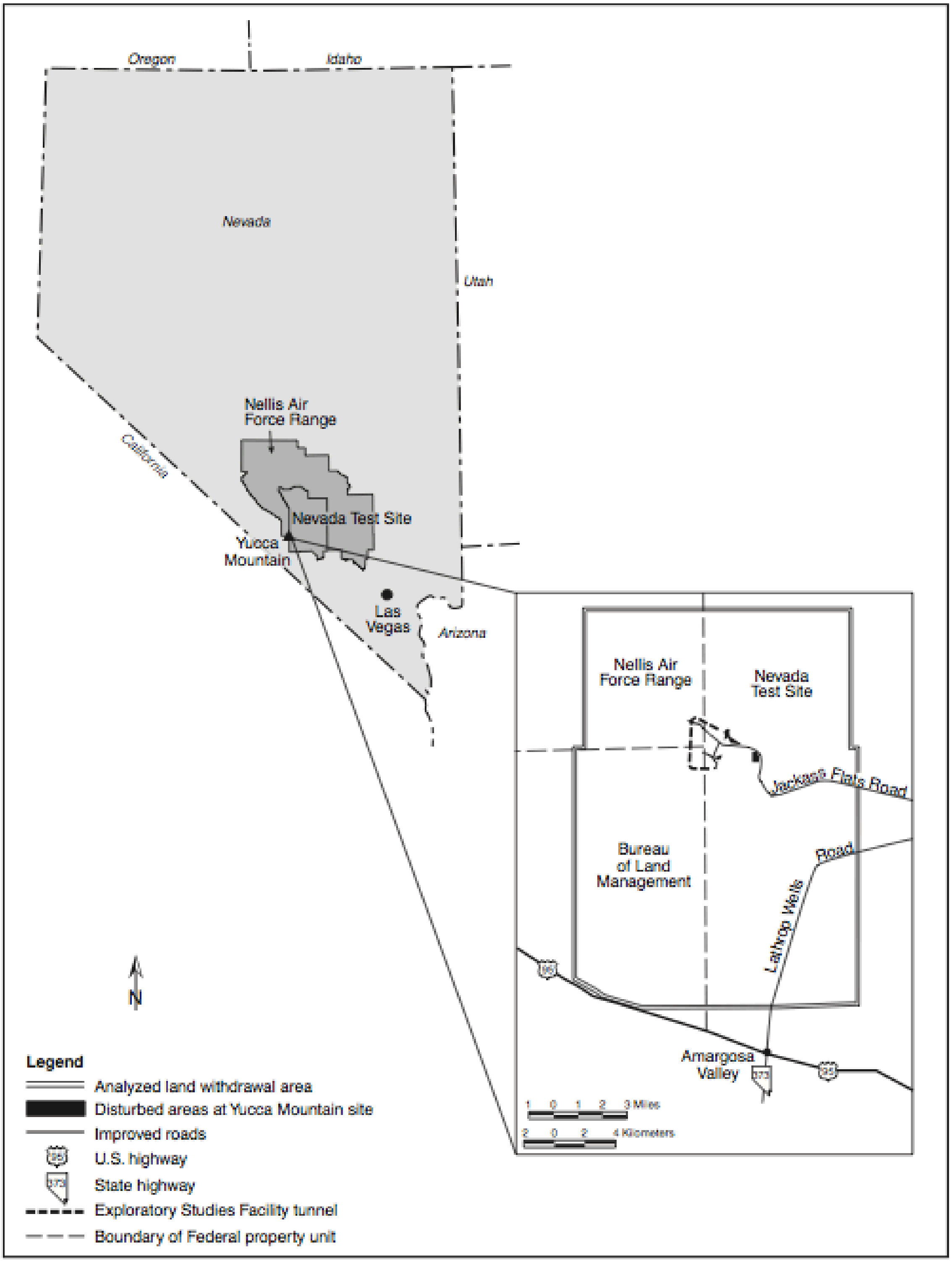


Figure C-5.5. Land Withdrawal Area. [C-5.11]

However, because the waste package density in a repository is limited primarily by the heat loading, this impact factor is adjusted to account for the decay heat in the disposed material. Ref. C-5.12 is based on disposal of LWR spent fuel with burnup of 40-60 GWd/MT so the waste packing density of disposed materials for the different fuel cycles must account for relative differences in the decay heat produced. The radiological activity of SNF+HLW at 100 years after discharge was used as an adjustment factor on the area required for disposal. The adjustment factor equals the activity of the disposed material divided by the activity of direct disposed PWR UOX fuel, the Basis of Comparison. This factor was applied to the facility footprint, prior to addition of the setback area. The resulting equation for the land use impact factor was then calculated as follows:

$$\frac{\pi}{140{,}000\ MTHM} * \left\{5km + \sqrt[2]{(20km^2/\pi) * \left(\frac{Act}{Act_{EG01}}\right)}\right\}^2$$

Where Act and Act_{EG01} represent the 100-year activity for the SNF and HLW for each Evaluation Group and the Basis of Comparison respectively.

Summary

Table C-5.2 provides a summary of the multipliers that were used to convert the material and processing requirements from each process in the fuel cycle into the corresponding land use needed to support the fuel cycle, as shown in the example in Table C-5.3.

Table C-5.2. Summary of Land Use Impacts.

Portion of Nuclear Fuel Cycle	**Multiplier**	**Units**
Front End of Fuel Cycle		
Mining / Milling – Uranium	2.8E-04	km^2/MTNU
Conversion – Uranium	3.3E-06	km^2/MTNU
Enrichment – Uranium	9.0E-09	km^2/SWU
Deconversion – Uranium	9.3E-05	km^2/MTDU
Extraction and Refining - Thorium	negligible	km^2/MTNTh
Fuel Fabrication		
UOX	1.02E-04	km^2/MTIHM
MOX	1.02E-04	km^2/MTIHM
Reactor		
Reactor Construction	See note 1	km^2/GWe-yr
Reactor Operation	7.27E-02	km^2/GWe-yr
Reprocessing and waste conditioning	4.41E-05	km^2/MTIHM
Disposal and Transportation (See note 2)		
Shallow Land Burial	9.74E-6	km^2/MT waste
Geologic Repository	1.50E-03	km^2/MTIHM
Interim Storage	3.0E-05	km^2/MTIHM

1. Land use for construction is absorbed into that used during reactor operations.
2. Based on impacts for disposal of DU given in [C-5.14]

Example of Land Use Calculation for Basis of Comparison

Table C-5.3 contains an example of the land use calculations for the Analysis Example for Evaluation Group #1 (EG01), a once-through LWR fuel cycle, which serves as the Basis of Comparison for this study. In this case, the fuel cycle requires that 188.628 tons of natural uranium must be mined for each 1 GWe-y produced. Uranium mining was calculated to require 0.053 km^2/GWe-yr. Conversion was calculated to require 0.00062 km^2/GWe-yr. The reactor required 21.915 tons of uranium enriched to 4.21%. The tails are 0.25%. The enrichment operation required 137.6 kSWU. The appropriate multipliers are shown in the table and the resulting land use was calculated by multiplying the kSWU by the multiplier to arrive at the appropriate land use of 0.00120 km^2/GWe-yr. In a like manner, the land use for each of the functions of the fuel cycle was calculated as shown in the table. The total land use was

0.17 km^2/GWe-yr. The reactor land usage accounts for the largest fraction of the total land usage in this example. Similar calculations were completed for the Analysis Example of each Evaluation Group.

Table C-5.3. Example Calculation of Land Use for EG01.

	Land Use Impact Factor	units	EG01 per GWe-yr	units	EG01 Land Use (km2/GWe-yr)	Notes
Front-end Activities						
Mining	2.8E-04	km2/MTNU	188.6	MTNU/GWe-yr	5.3E-02	Based on mine and mill occupying the same site
Milling						
Conversion	3.3E-06	km2/MTNU	188.6	MTNU/GWe-yr	6.2E-04	
Enrichment	9.0E-09	km2/SWU	137613.5	SWU/GWe-yr	1.2E-03	
Deconversion	9.3E-05	km2/MTDU	166.7	MTDU/GWe-yr	1.6E-02	
Fuel Fabrication						
Fuel Fabrication	1.0E-04	km2/MTIHM	21.9	MTIHM/GWe-yr	2.2E-03	
Reactor Construction and Operation						
Reactor Construction and Operations	7.3E-02	km2/GWe-yr	1.0	GWe-yr/GWe-yr	7.3E-02	
Recycling / Reprocessing						
Recycling/ Reprocessing	4.4E-05	km2/MTHM	0.0	MTHM/GWe-yr	0.0E+00	Based on aqueous reprocessing
Storage, Transport, and Disposal						
Wet Storage	N/A					Wet storage impacts included as part of reactor operations
Dry Storage	N/A					Assumed no dry storage needed for steady state system
Transport	N/A					Land use for transport is considered negligible
LLW Disposal	9.7E-06	km2/MTHM LLW disposed	166.7	MTHM/GWe-yr	1.6E-03	
SNF Disposal	1.3E-03	km2/MTHM SNF disposed	21.9	MTHM/GWe-yr	2.8E-02	see calculation of fuel-cycle-specific impact factor
Total Land Use (km2/GWe-yr)					**1.7E-01**	

Binning of the Metric Data

The calculated data derived for the Analysis Examples for the 40 Evaluation Groups were then used as the basis for binning the Metric Data for each Evaluation Group into a metric bin. An Evaluation Group was placed in a different bin than was indicated by the calculated data for the Analysis Example if assumptions for the associated Analysis Example were considered as not being representative of the fuel cycles within the Evaluation Group. By using bins, it is expected that the Metric Data to be used for evaluating an Evaluation Group relative to the Basis of Comparison is sufficiently representative of the capabilities of fuel cycles within the group. Details on the *land use per energy generated* metric calculation approach, the binning process, and the metric bins for the 40 evaluation groups, are described in Appendix D-2.10.

C-5.4 Water Use per Energy Generated

Definition of Metric **-** Water use was defined to include water used a result of the nuclear fuel cycle processes (NFC) contained in the nuclear fuel cycle such as mining, reactors and fuel fabrication plants and is measured in megaliters (MLs) per unit production (e.g. MTHM, SWU, GWe-yr).

Net water withdrawals represent the total water withdrawn by the NFC process minus any water returned to its source at equal or better purity and within allowable temperature limits. Many operations recycle large percentages of their water to reduce the overall consumption. Net water withdrawals account for any recycling or water treatment processes withdrawals by including only the volume of water withdrawn from outside sources and not returned to outside sources in equivalent or better purity. When net withdrawal data is unavailable, gross water withdrawals (the total volume of water taken from outside sources) were used and noted as such to recognize the potential for recycling in the operation.

The degree to which an operation invests in recycling water is determined largely by economic and/or regulatory consideration. It should however be recognized that water usage for virtually any facility can be reduced by additional investment in water cleanup and recycling processes. However, the extent to which additional investment is justified is commensurate with the availability of water and/or the applicable regulations. Hence, the water usage varies widely even for a given technology and is thus more properly represented by what is technically and economically achievable than what is currently achieved for a particular facility. Consequently, if the available water usage data for a particular fuel cycle process was found to vary broadly, the water usage from a plant with a strong recycling program was used. This is representative of what can be achieved if water usage is considered important.

Determination of Impact Factors

<u>Front-end of Nuclear Fuel Cycle</u>

The FEFC includes subprocesses for extracting the raw ore and preparing it in a form suitable for the fuel fabrication process. For uranium-based fuels, this includes mining, milling, conversion, enrichment, and deconversion of DUF_6 tailings from the enrichment process. Impact factors for mining and milling are based on a mix of underground, open pit, and in-situ mining representative of the current industry. Because milling operations are typically co-located and part of the mining operation, the water use impact factor for mining also includes usage associated with the milling process. FEFC water use impacts are summarized below in Table C-5.4. Additional detail associated with development of the impact factors can be found in [C-5.1].

Table C-5.4. Summary of Water Use Impact Factors for the FEFC.

	Uranium Fuels					Thorium Extraction and Refining
	Mining	Milling	Conversion	Enrichment	Deconversion	
Normalization Unit	MTNU		MTNU	SWU	MTDU	MTTh
Water Use (ML) net	8.5E-01		6.5E-02	2.9E-05	5.3E-04	1.1E-01

Fuel Fabrication

Operational water withdrawal for both the Areva FBFC (Franco-Belgian Fuel Fabrication) Romans facility and MELOX MOX facility is given in Ref. C-5.15 and Ref. C-5.16, respectively. Water use at the Romans facility averaged 141,000 L/MTIHM over the 2005-08 period, while water use at the MELOX facility averaged 521,000 L/MTIHM over the 2008-10 period.

Reactor Construction and Operations

Ref. C-5.17 estimates water use for reactors utilizing three cooling technologies. This data is summarized in Table C-5.5. Water consumption associated with a cooling tower was selected since it can be applied at any generic site without requiring a large water source (i.e. once-through) or an on-site cooling pond.

Table C-5.5. Summary of Water Withdrawals and Consumption for Reactor Operations. [C-5.17]

	Water Withdrawal [ML/GWe-yr]	Water Consumption [ML/GWe-yr]
Once-through	8.32E+05 – 2.01E+06	1.31E+04
Pond cooling	1.66E+04 – 3.68E+04	1.31E+04 to 2.37E+04
Cooling towers	2.63E+04 – 3.68E+04	2.37E+04

Reprocessing and waste conditioning

Operational water withdrawal for the La Hague facility is given in [C-5.18]. Water use at La Hague averaged 483,000 L/MTIHM over the 2007-09 period.

Storage, Transport, and Disposal

Interim Surface Storage - Water use, as reported in [C-5.19], is minimal for the passively-cooled ISFSI. It is considered below threshold relative to the water use associated with reactors and other fuel cycle technologies.

Deep Geologic Repository

Estimated water usage for the construction, operation, and closure based on calculations of the proposed Yucca Mountain repository is given in [C-5.11, Table 4-11] and replicated in Table C-5.6. Converting the total water use to units of ML (1.23 ML per acre-foot) gives a total water use of 10,000 ML. Dividing this by the 70,000 MTIHM YMP capacity yields an estimated water use of 1.43E-01 ML/MTIHM. Water use was assumed to scale with the mass of material disposed.

Table C-5.6. Water Use for Repository Construction, Operation, and Closure. [C-5.11]

Phase	**Duration (yr)**	**Water Demand (acre-feet/yr)**	**Total Water Demanded during Phase (acre-feet)**
Construction	5.00E+00	1.60E+02	8.00E+02
Operation & Monitoring			
Operations Period			
Emplacement and Development	2.20E+01	2.30E+02	5.06E+03
Subsequent emplacement only	2.00E+00	1.80E+02	3.60E+02
Monitoring Period			
Initial decontamination	3.00E+00	2.20E+02	6.60E+02
Subsequent monitoring & caretaking	7.30E+01	6.00E+00	4.38E+02
Closure	1.00E+01	8.10E+01	8.10E+02
		Total Water Use:	**8.13E+03**

Summary

Table C-5.7 provides a summary of the multipliers that are used to describe the water use impact for each element of the nuclear fuel cycle for this metric, applied in the same manner as the multipliers for land use to correlate fuel cycle activities with the corresponding water use.

Table C-5.7. Summary of Water Use Impacts.

Portion of Nuclear Fuel Cycle	**Multiplier**	**Units**
Front End of Fuel Cycle		
Mining / Milling – Uranium	8.5E-01	ML/MTNU
Conversion – Uranium	6.5E-02	ML /MTNU
Enrichment – Uranium	2.9E-05	ML /SWU
Deconversion – Uranium	5.3E-04	ML/MTDU
Extraction and Refining - Thorium	1.1E-01	ML/MTNTh
Fuel Fabrication		
UOX	1.41E-01	ML/MTIHM
MOX	5.21E-01	ML/MTIHM
Reactor		
Reactor Construction	See note 1	ML/GWe-yr
Reactor Operation	2.37E+04	ML/GWe-yr
Reprocessing and waste conditioning	4.83E-01 (see note 2)	ML/MTIHM
Disposal and Transportation		
Shallow Land Burial	2.3E-04	ML/MT waste
Geologic Repository	1.43E-01 (See note 3)	ML/MTIHM
Interim Storage	negligible	ML/MTIHM
1. Assumed to be negligible relative to water use during reactor operations. 2. Water use estimate reflects potable plus raw water consumption at the reference facility 3. In [C-5.11], water use is estimated as total raw water withdrawals. Data on any planned water recycling program is needed to generate an estimate of net water usage		

Example of Water Use Calculation for Basis of Comparison

Table C-5.8 contains an example of the water use calculations for the Analysis Example for Evaluation Group #1 (EG01), a once-through LWR fuel cycle, which serves as the Basis of Comparison for this report. In this case, the fuel cycle requires that 188.628 tons of natural uranium must be mined for each 1 GWe-y produced. Uranium mining was calculated to require 160 ML/GWe-yr. Conversion was calculated to require 0.00062 ML/GWe-yr. The reactor required 21.915 tons of uranium enriched to 4.21%. The tails were 0.25%. The enrichment operation required 137.6 kSWU. The appropriate multipliers are shown in the table and the resulting water use was calculated by multiplying the kSWU by the multiplier to arrive at a water use of 4 ML/GWe-yr. In a like manner, the water use for each of the functions of the NFC was calculated as shown in Table C-5.8. The total water use was 24,000 ML/GWe-yr. The reactor water usage accounts for virtually all of the total water usage in this example. Similar calculations were completed for the Analysis Example of each Evaluation Group.

Binning of the Metric Data

The calculated data derived for the Analysis Examples of the 40 Evaluation Groups are then used as the basis for binning each Evaluation Group into a metric bin. An Evaluation Group was placed in a different bin than was indicated by the calculated data for the Analysis Example if assumptions for the associated Analysis Example were considered as not being representative of the fuel cycles within the Evaluation Group. By using bins, it is expected that the Metric Data to be used for evaluating an Evaluation Group relative to the Basis of Comparison is sufficiently representative of the best capabilities of fuel cycles within the group. Details on the water use per energy generated metric calculation approach, the binning process, and the metric bins for the 40 Evaluation Groups, are described in Appendix D-2.11.

Table C-5.8. Example Calculation of Water Use for EG01.

	Water Use Impact Factor	units	EG01 per GWe-yr	units	EG01 Water Use (ML/GWe-yr)	Notes
Front-end Activities						
Mining	8.5E-01	ML/MTNU	188.6	MTNU/GWe-yr	1.6E+02	Based on mine and mill occupying the same site
Milling						
Conversion	6.5E-02	ML/MTNU	188.6	MTNU/GWe-yr	1.2E+01	
Enrichment	2.9E-05	ML/SWU	137613.5	SWU/GWe-yr	4.0E+00	
Deconversion	5.3E-04	ML/MTDU	166.7	MTDU/GWe-yr	8.8E-02	
Fuel Fabrication						
Fuel Fabrication	5.2E-01	ML/MTIHM	21.9	MTIHM/GWe-yr	1.1E+01	
Reactor Construction and Operations						
Reactor Construction and Operations	2.4E+04	ML/GWe-yr	1.0	GWe-yr/GWe-yr	2.4E+04	
Recycling / Reprocessing						
Recycling/ Reprocessing	4.8E-01	ML/MTHM	0.0	MTHM/GWe-yr	0.0E+00	Based on aqueous reprocessing
Storage, Transport, and Disposal						
Wet Storage	N/A					Wet storage impacts included as part of reactor operations
Dry Storage	N/A					Assumed no dry storage needed for steady state system
Transport	N/A					Water use for transport is considered negligible
LLW Disposal	2.3E-04	ML/MTHM LLW disposed	166.7	MTHM/GWe-yr	3.8E-02	
HLW Disposal	1.4E-01	ML/MTHM HLW disposed	21.9	MTHM/GWe-yr	3.1E+00	
Total Water Use (ML/GWe-yr)					**2.4E+04**	

C-5.5 Carbon Emission - CO_2 Released per Energy Generated

Definition of Metric **-** CO_2 emissions were defined to include emissions as a result of the fuel cycle processes contained in the nuclear fuel cycle such as mining, reactors and fuel fabrication plants and is measured in kg CO_2 per unit production (e.g. MTHM, SWU, GWe-yr).

CO_2 is the most significant green-house gas (GHG) resulting from the nuclear fuel cycle. Estimates of CO_2 emissions were developed based on the direct and embodied energy consumed within each of the nuclear fuel cycle processes. Direct energy is from electricity and other energy carriers used as a source of heat (e.g. distillate fuels, natural gas, coal, etc.). Embodied energy includes the energy used to produce materials and chemicals consumed within the process. For both direct and embodied energy, CO_2 emissions were calculated by disaggregating the energy consumption into the appropriate energy carriers and multiplying each by the applicable carbon intensity factor.

Determination of Impact Factors

Front-end of Nuclear Fuel Cycle

The FEFC includes subprocesses for extracting the raw ore and preparing it in a form suitable for the fuel fabrication process. For uranium-based fuels, this includes mining, milling, conversion, enrichment, and deconversion of DUF_6 tailings from the enrichment process. Impact factors for mining and milling are based on a mix of underground, open pit, and in-situ mining representative of the current industry. Because milling operations are typically co-located and part of the mining operation, the CO_2 emissions impact factor for mining also includes emissions associated with the milling process. FEFC CO_2 emissions impacts are summarized below in Table C-5.9. Additional detail associated with development of the impact factors can be found in [C-5.1].

Table C-5.9. Summary of CO_2 Emission Impact Factors for the FEFC.

	Uranium Fuels					Thorium Extraction and Refining
	Mining	Milling	Conversion	Enrichment	Deconversion	
Normalization Unit	MTNU		MTNU	SWU	MTDU	MTTh
CO2 Emissions (kg)	8.3E+04		2.2E+04	2.8E+01	-3.2E+03*	2.0E+04
Occupational Radiological (person*mSv)	5.2E-01	9.0E-01	8.8E-02	3.1E-05	2.9E-02	3.8E+00
* CO_2 emissions are negative due to the large amount of recovered embodied energy in the HF product stream.						

Fuel Fabrication

Reference data for direct energy used for UOX fabrication is taken from the Areva FBFC Romans facility. Over the 2005-2008 period, the Romans facility fabricated an average of 524 MTIHM/yr; average direct energy consumption was 212 GJ(e)/MTIHM and 73 GJ(t)/MTIHM [C-5.15]. Ref. C-5.20 provides an estimate of 723 GJ(e)/MTIHM and 2,440 GJ(t)/MTIHM for the embodied energy in process materials for the UOX fuel fabrication process.

Production at the reference Areva MELOX MOX fabrication facility averaged 129 MTIHM/yr over the 2008-10 period. The average direct energy consumption was 1,060 GJ(e)/MTIHM and 0.13 GJ(t)/MTIHM. [C-5.16] Ref. C-5.20 estimates 761 GJ(e)/MTIHM and 2,720 GJ(t)/MTIHM for the embodied energy in process materials for MOX fuel fabrication.

The primary contributor to embodied energy during UOX and MOX fuel fabrication is the Zircaloy material input. Zircaloy is employed in fuel cladding due to its transparency to neutrons and corrosion-resistant properties. The age of the data source [C-5.20] is recognized. Some environmental impact information related to modern Zircaloy production at Areva's Cezus Zircaloy plant as well as three others in the Zircaloy production chain is available in [C-5.21]. While energy use is available for the Cezus facility, Zircaloy environmental impacts were not able to be isolated as each plant in the chain also produces secondary products not related to nuclear fuel cladding.

Thermal energy is provided through natural gas for both UOX and MOX fuel fabrication. Carbon emissions were calculated from the energy use information described previously using carbon intensities.

Reactor Construction and Operations

With the negligible exception of diesel generator operation, energy use during operations is absorbed into net electrical efficiency. Therefore, the energy use and resulting CO_2 emissions associated with reactor operations are those which result from reactor construction. Energy use in reactor construction is comprised of direct energy consumed in construction operations and energy embodied in building materials and equipment.

Since no source providing direct construction energy use was available, construction energy intensity was assumed to be adequately represented by the construction sector averages reported by the US Energy Information Administration (EIA). The EIA tabulates direct energy coefficients that provide sector energy use per dollar of capital cost expended. To implement this approach, an overnight capital cost of \$4,000 \$(2012)/kW_e was obtained from [C-5.22]. This cost estimate was combined with the direct energy coefficient data (2.03E-3 GJ per \$(2005)) for construction in the nonmanufacturing sector [C-5.23].

Embodied energy was estimated from an inventory of the commodity inputs to reactor construction. Ref. C-5.17 reviews a study in which Oak Ridge National Lab (ORNL) estimated the commodity inputs for construction of a generic 1 GW_e plant. Energy and carbon intensities for these commodities were obtained from Ref. C-5.24. A calculation of the energy use and carbon emissions associated with reactor construction was made. This value of 6.29×10^8 kg CO_2 was amortized over the electrical energy produced over a 60-year lifetime of a plant operating with a 90% capacity factor ($1GW_e$ * 60 yrs *.9 = 54 GWe-yr) to obtain the net energy use and CO_2 impact per unit electrical energy produced ($6.29 \times 10^8/54 = 1.16 \times 10^7$).

Although no data was available for construction of fission-fusion hybrid or accelerator driven reactor systems, it was reasonable to expect that additional CO_2 impacts will be associated with construction of the additional infrastructure and supporting facilities. The additional CO_2 emissions were estimated based on existing LWRs. In the absence of any firm data, a scaling factor of 150% was judged to be a reasonable estimate.

Reprocessing and waste conditioning

Reference data for energy use in used fuel reprocessing is obtained from the Areva La Hague facility. La Hague processed an average of 938 MTIHM/yr over the 2007-09 period. Direct energy use at La Hague averaged 1,740 GJ(e)/MTIHM and 816 GJ(t)/MTIHM [C-5.16], all in the form of natural gas. Average annual chemical consumption at La Hague was also obtained from Ref. C-5.16 and was needed to estimate the energy embodied in material inputs to reprocessing and other operations. Inputs outside of chemicals gave rise to negligible impacts in this respect.

Storage, Transport, and Disposal

Interim Surface Storage - Four contributors to energy use were identified. These were:

1) fabrication of the high density concrete storage pads and overpack,
2) construction of the supporting buildings and container transfer equipment,
3) manufacture of the spent fuel storage, transport, and disposal containers, and
4) direct energy use during the operation phase.

Of these contributors, design and operational data reported in Ref. C-5.25 showed that construction of buildings and equipment (#2) as well as direct operational energy use (#4) were negligible (on the order of 1% or less) contributors to total energy use. Therefore, they were not considered further. Concrete requirements for the storage pads and overpacks are also provided in Ref. C-5.25. The reference design called for 4,000 dual purpose canisters (DPCs) to be emplaced at the facility. Each DPC would require an overpack of 1000 yd^3 in extent plus a storage pad of high density concrete measuring a further 67 yd^3. The waste packages themselves are designed for continued use when the waste is subsequently transferred to a deep geologic repository (DGR). Therefore, the energy and CO_2 intensities for waste package fabrication developed in the DGR section was used here as well. Note that if a fuel cycle incorporates both long term surface storage and DGR disposal, the waste package impacts should be deducted from the DGR disposal category in order to avoid double counting.

Table C-5.10 summarizes the reference facility design data taken from Ref. C-5.25. Table C-5.11 steps through the energy and CO_2 intensity calculations and provides overall results.

Table C-5.10. Interim Spent Fuel Storage Facility Data from Ref. C-5.25.

Item	Unit	Value
Capacity	MTIHM	40,000
Operating Lifetime	yr	40
Concrete required for pads	yd^3	2.68E5
Concrete required for overpacks	yd^3	4.00E6
Total concrete required*	m^3	3.26E6
Total concrete mass**	kg	7.83E9
Concrete mass per unit of capacity	kg/MTIHM	1.96E5
* at 0.76455 m^3 per yd^3 ** at 2400 kg/m^3 for high density concrete		

Table C-5.11. Calculation of CO_2 Impact Factor for Interim Surface Storage.

Item	Unit	Value	Source
Energy intensity, concrete manufacture	GJ/kg	1.11E-3	[C-5.24]
Energy use, concrete manufacture	GJ/MTIHM	2.17E2	Calculated
Emission factor, concrete manufacture	kg CO2/GJ	143.2	[C-5.24]
CO2 emissions, concrete manufacture	kg CO2/MTIHM	3.11E4	Calculated
* Energy and associated CO_2 emissions impacts from materials and fabrication of storage canisters is included in those calculated for the waste package below.			

Deep Geologic Repository

As described above, analysis information that had been developed for the proposed repository at Yucca Mountain was used for the example of a deep geologic repository in the Study. Major energy use processes associated with constructing, operating, and closing the repository were divided into four categories: (1) initial excavation, (2) operation of ventilation fans for active cooling during repository operations, (3) material inputs and fabrication for the engineered barrier design, and (4) transportation associated with emplacement of waste packages into the final repository. Of these major energy use processes, transportation associated with emplacement of (and transport to the repository site of) waste packages were considered negligible.

The CO_2 impact metric thus has three significant components. Excavation and closure impacts will scale with the decay heat production if the excavation requirements (e.g., number, length or spacing of tunnels) are tied to a constraint on the heat per unit of tunnel length in the repository. If no such constraint exists, then this impact would likely scale with mass or number of packages to be disposed. The operational active cooling impact, if present for a disposal concept, clearly scales with heat generation. Waste packaging impacts increase with the number of packages, but the waste packing density inside a single package is in turn coupled to the decay heat of the waste. Therefore, it would be appropriate to modify the geologic disposal impact factors estimated below (kg CO_2/MTIHM) to account for any packaging or emplacement constraints associated with decay heat.

Excavation

Excavation energy use was estimated as shown in Table C-5.12.

Table C-5.12. Energy Estimate for Repository Excavation and Backfill.

	value	Units	Source
Material extracted for YMP excavation	4.40E+06	m^3	[C-5.11]
Density of material extracted[1]	3.00E+00	MT/m^3	[C-5.26]
MT of material extracted	1.32E+07	MT	calculated
Energy required per MT extracted[2]	4.45E-01	GJ/MT	[C-5.27]
Excavation energy for YMP	5.87E+06	GJ	calculated
Backfill energy for YMP[3]	5.87E+06	GJ	calculated
YMP capacity	70,000	MTIHM	
Normalized excavation energy and backfill	1.68E+02	GJ/MTIHM	calculated

1. Excavated rock density values from mining operations vary widely; a typical value for a western US coal mine is 2.4 MT/m^3.[C-5.28] A density of 3 MT/m^3 is chosen as a conservative estimate.
2. Energy intensity of a hypothetical underground room-and-pillar coal mine.
3. Backfilling of drifts with previously excavated material is assumed to be equal to the initial excavation energy.

According to Ref. C-5.27, 78% of the energy used in the coal mining operation is obtained through electricity, with the remaining energy share obtained through distillate fuels. The CO_2 emissions associated with this energy use are estimates as shown in Table C-5.13.

Table C-5.13. CO_2 Emissions Estimate for Repository Excavation and Backfill.

	GJ/MTIHM	kg CO_2/GJ	kg CO_2/MTIHM
Total energy	1.68E+02		
Energy from electricity (78%)	1.31E+02	168	2.20E+04
Energy from distillate fuels (22%)	3.69E+01	79	2.92E+03
		Total CO_2	2.49E+04

Repository Operation

Using the analysis information that had been developed for the proposed repository at Yucca Mountain, the greatest energy expenditure during operations of that repository would have been the ventilation fans used for active cooling of the repository. Reference C-5.29 determined 3 fans, operating at an annual cost of $7.3 million total, were required to cool the 70,000 MTIHM in Yucca Mountain. Reference C-5.28 assumed an electricity cost of $0.1 per kWh. Under this assumption, energy consumption for active cooling is 376 GJ per MTIHM. Ref. C-5.11 states that electricity during repository operations will be provided through the Nevada Test Site electric power distribution system; applying the U.S. average carbon intensity of 168 kg CO_2 per GJ for electricity, carbon emissions due to repository operation were estimated at 63,200 kg CO_2 per MTIHM.

Waste Package Fabrication and Material Inputs

Two major material inputs to the repository exist: (1) titanium drip shields and (2) waste packages. These two engineered barriers were intended to confine the waste and to protect the waste package from contact with water. The total estimated cost of fabricating drip shields and waste packages was obtained from Ref. C-5.30 at $21 billion ($30,000 per MTIHM for the YMP 70,000 MTIHM capacity). Using these figures, the carbon emissions for fabrication of the drip shields and waste packages is calculated as shown in Table C-5.14. Because Ref. C-5.30 combines the fabrication costs of drip shields and waste packages, the energy use for fabricating these could not be disaggregated.

Ref. C-5.31 estimates a total of 3,087 MT of titanium is used to fabricate drip shields for the repository. Material inputs for waste packages are determined from Ref. C-5.31 and Ref. C-5.32. Energy (GJ per kg material) and carbon (kg CO_2 per kg material) intensity coefficients are obtained from Ref. C-5.25 and applied to obtain the final energy and carbon intensities given in the transportation and disposal portion of Table C-5.15.

Table C-5.14. CO_2 Emissions for Fabrication of Waste Packages and Drip Shields.

	GJ/$	$/MTIHM	GJ/MTIHM	kgCO_2/GJ	kg CO_2/MTIHM
Electricity	3.82E-04	2.92E+05	1.12E+02	168	1.87E+04
Natural Gas	5.27E-04	2.92E+05	1.54E+02	51	7.85E+03
Distillate Fuels	1.99E-06	2.92E+05	5.81E-01	79	4.59E+01
LPG	3.98E-06	2.92E+05	1.16E+00	51	7.90E+01
Coal	1.69E-05	2.92E+05	4.94E+00	89	4.39E+02
				Total	**2.71E+04**

Summary

Table C-5.15 provides a summary of the multipliers that are used to describe the CO_2 emissions for each element of the nuclear fuel cycle for this metric, applied in the same manner as the multipliers for land use described above to correlate fuel cycle activities with the associated emissions.

Table C-5.15. Summary of CO_2 Impacts.

Portion of Nuclear Fuel Cycle	**Multiplier**	**Units**
Front End of Fuel Cycle		
Mining / Milling – Uranium	8.3E+04	kg/MTNU
Conversion – Uranium	2.2E+04	kg/MTNU
Enrichment – Uranium	2.8E+01	kg/SWU
Deconversion – Uranium	-3.2E+03 (See note 1)	kg/MTDU
Extraction and Refining - Thorium	2.0E+04	kg/MTNTh
Fuel Fabrication		
UOX	2.85E+05	kg/MTIHM
MOX	4.45E+05	kg/MTIHM
Reactor		
Reactor Construction	1.16E+07	kg/GWe-yr
Reactor Operation	See note 2	kg/GWe-yr
Reprocessing and waste conditioning	5.15E+05	kg/MTIHM
Disposal and Transportation		
Shallow Land Burial	1.82E+00	kg/MT waste
Geologic Repository		
Excavation & Closure	2.49E+04	kg/MTIHM
Operations	6.32E+04	kg/MTIHM
Waste Packages and Drip Shields		
Fabrication	2.71E+04 (see note 3)	kg/MTIHM
Waste Package materials	2.91E+04 (see note 4)	kg/MTIHM
Drip Shield materials	1.30E+03	kg/MTIHM
Interim Storage		
Concrete Manufacture	3.11E+04	kg/MTIHM
Storage Package Fabrication	(see note 5)	kg/MTIHM
Storage Package Materials	(see note 5)	kg/MTIHM

1. CO_2 emissions are negative due to the large amount of recovered embodied energy in the HF product stream.
2. Assumed to be negligible relative to reactor construction
3. Includes fabrication of storage, transport, and disposal canisters, waste packages, and drip shields packages because energy use data could not be disaggregated.
4. Includes materials for storage, transport, and disposal canisters.
5. Energy and associated CO_2 emissions associated with materials and fabrication of the container used for storage and transport is was included in the disposal packaging estimates that formed the basis for the CO_2 estimate for the waste package shown in Table C-5.14 above.

Example of CO_2 Emissions Calculation for Basis of Comparison

Table C-5.16 contains an example of the CO_2 emissions calculations for the Analysis Example for Evaluation Group #1 (EG01), a once-through LWR fuel cycle, which serves as the Basis of Comparison for this report. In this case the fuel cycle required that 188.628 tons of natural uranium must be mined for each 1 GWe-y produced. Uranium mining was calculated to result in CO_2 emissions of 1.6 x 10^7 kg CO_2/GWe-yr. Conversion was calculated to result in CO_2 emissions of 4.1 x 10^6 kg CO_2/GWe-yr. The reactor required 21.915 tons of uranium enriched to 4.21%. The tails were 0.25%. The enrichment operation required 137.6 kSWU. The appropriate multipliers are shown in the table and the resulting CO_2 emissions were calculated by multiplying the kSWU by the multiplier to arrive at the CO_2 emissions of 3.9 x 10^6 kg CO_2/GWe-yr. In a like manner, the CO_2 emissions for each of the remaining functions of the NFC were calculated as shown in the table. The total CO_2 emissions were 4.5 x 10^7 kg CO_2/GWe-yr. For reference, the CO_2 emissions are also provided in another commonly-used unit, g CO_2/kWh. The mining and reactor construction and operation account for ~64% of the total CO_2 emissions in this example. Similar calculations were completed for the Analysis Example of each Evaluation Group.

Table C-5.16. Example Calculation of CO_2 Emissions for EG01.

	CO_2 Emissions Impact Factor	units	EG01 per GWe-yr	units	EG01 CO_2 Emissions		Notes
					kg CO_2/GWe-yr	g CO_2/kWh	
Front-end Activities							
Mining	8.3E+04	kg CO_2/MTNU	188.6	MTNU/GWe-yr	1.6E+07	1.82	Mine and mill at the same site
Milling							
Conversion	2.2E+04	kg CO_2/MTNU	188.6	MTNU/GWe-yr	4.1E+06	0.47	
Enrichment	2.8E+01	kg CO_2/SWU	137613.5	SWU/GWe-yr	3.9E+06	0.45	
Deconversion	-3.2E+03	kg CO_2/MTDU	166.7	MTDU/GWe-yr	-5.3E+05	-0.06	Negative due to energy embodied in HF by-product
Fuel Fabrication							
Fuel Fabrication	2.9E+05	kg CO_2/MTIHM	21.9	MTIHM/GWe-yr	6.2E+06	0.71	Impact factor for oxide fuels
Reactor Construction and Operations							
Reactor Construction and Operations	1.2E+07	kg CO_2/GWe-yr	1.0	GWe-yr/GWe-yr	1.2E+07	1.37	
Recycling / Reprocessing							
Recycling/ Reprocessing	5.2E+05	kg CO_2/MTHM	0.0	MTHM/GWe-yr	0.0E+00	0.0	No reprocessing
Storage, Transport, and Disposal							
Wet Storage	N/A						Included as part of reactor operations
Dry Storage	N/A						Assumed not needed for steady state system
Transport	N/A						Negligible
LLW Disposal	1.8E+00	kg CO_2/MTHM LLW disposed	166.7	MTHM/GWe-yr	3.0E+02	3.4E-05	
HLW Disposal	1.4E+05	kg CO_2/MTHM HLW disposed	21.9	MTHM/GWe-yr	3.2E+06	0.36	
Total CO_2 Emissions					**4.5E+07**	**5.12**	

Binning of the Metric Data

The calculated data derived for the Analysis Examples of the 40 Evaluation Groups were used as the basis for binning each Evaluation Group into a metric bin. An Evaluation Group was placed in a different bin than was indicated by the calculated data for the Analysis Example if assumptions for the associated Analysis Example were considered as not being representative of the fuel cycles within the Evaluation Group By using bins, it is expected that the Metric Data to be used for evaluating an Evaluation Group relative to the Basis of Comparison is sufficiently representative of the best capabilities of fuel cycles within the group. Details on the carbon emission – CO_2 released per energy generated metric calculation approach, the binning process, and the metric bins for the 40 evaluation groups, are in Appendix D, Section D-2.12.

C-5.6 Radiological Exposure - Total Estimated Worker Dose per Energy Generated (as Leading Indicator for Public Dose Potential)

Definition of Metric **-** Radiological dose to workers is defined as the collective annual dose, measured in person-mSv, to all plant workers for the fuel cycle processes contained in the nuclear fuel cycle such as mining, reactors and fuel fabrication and reprocessing plants.

As explained in section C-5.2, the dose to workers is also considered to be a satisfactory surrogate to represent the potential exposure to the local ecology and the off-site public. The use of worker exposure is not an implication that worker exposure is more important than public or environmental exposure – rather it is the factor most readily estimated for generic processes and facilities without design and location specificity, and worker exposure is a predecessor to potential public or environmental exposure.

Determination of Impact Factors

<u>Front-end of Nuclear Fuel Cycle</u>

The FEFC includes subprocesses for extracting the raw ore and preparing it in a form suitable for the fuel fabrication process. For uranium-based fuels, this includes mining, milling, conversion, enrichment, and deconversion of DUF_6 tailings from the enrichment process. Impact factors for mining and milling were based on a mix of underground, open pit, and in-situ mining representative of the current industry. FEFC radiological impacts include radioactive emissions from naturally occurring radioactive gases such as radon that are entrained in the ore, and mildly radioactive and hazardous material residue in mine and mill tailings. Such emissions by their nature are not entirely controllable and are thus somewhat different from the tightly regulated emissions from operating facilities such as reactors or used-fuel handling facilities which are controlled to extremely low levels.

FEFC radiological worker dose impacts are summarized below in Table C-5.17. Additional detail associated with development of the impact factors can be found in Ref. C-5.1

Table C-5.17. Summary of Radiological Worker Dose Emission Impact Factors for the FEFC.

	Uranium Fuels					Thorium Extraction and Refining
	Mining	Milling	Conversion	Enrichment	Deconversion	
Normalization Unit	MTNU		MTNU	SWU	MTDU	MTTh
Occupational Radiological (person-mSv)	5.2E-01	9.0E-01	8.8E-02	3.1E-05	2.9E-02	3.8E+00

<u>Fuel Fabrication</u>

Occupational radiological impact metrics were quantified for fuel types based on the fabrication plant design and operating approach:

- Hands-on: For fuels having small amounts of penetrating radiation and low inhalation radiotoxicity (e.g., low enriched uranium [LEU] fuels in any form)
- Glove box: For fuels having small-to-moderate amounts of penetrating radiation or substantial inhalation radiotoxicity (e.g., fuels containing Pu and/or Th)
- Hot cell: For fuels having high amounts of penetrating radiation (e.g., fuels or targets containing minor actinides or ^{233}U).

Normalized worker exposure estimates for each of these fabrication approaches are given in Table C-5-18 and discussed in the text that follows.

Table C-5.18. Radiological Impacts to Workers for Fuel Fabrication.

Fuel Fabrication	Normalized Impacts (person-mSv/MTIHM)		
	Hands-On	Glove Box	Hot Cell
Technology Basis	*LEU Fuels*	*Pu, Th fuels*	*U-233, MA fuels*
Occupational Radiological Dose	*1.43*	*11.66*	*0.38*

Hands-On Fuel Fabrication

There are two main technological methods to produce LEU oxide fuel – wet and dry. Both methods are analogous to the conversion methods presently in use, as the beginning stages must reverse the original conversion process by changing low-enriched UF_6 to LEU oxides. Operations at LEU-oxide fuel fabrication facilities (wet and dry) that lead to occupational exposure are external exposure from UF_6 cylinders and LEU fuel, and inhalation of UO_2 powder [C-5.33, C-5.34].

Currently, there are three U.S. facilities that produce LEU oxide fuel for use in commercial power plants [C-5.35, C-5.36]. The three facilities are:

- Areva NP Inc. – Richland, Washington (previously Framatome ANP)
- Westinghouse Electric Company, LLC – Columbia, South Carolina
- Global Nuclear Fuel Americas LLC – Wilmington, North Carolina

All of the currently operating U.S. facilities use the dry-process. The production capacities of each facility are listed in Table C-5.19 [C-5.36]. The collective occupational doses to workers in these facilities are reported to the Nuclear Regulatory Commission (NRC) annually. The NRC then summarizes the doses and statistics on nuclear fuel cycle facilities in the NUREG-0713 annual series. Collective doses to fuel fabrication plant workers for the last ten years where data is/are available are given in Table C-5.20 [C-5.19, C-5.39, C-5.40, C-5.41, C-5.42, C-5.43, C-5.44, C-5.45, C-5.46, and C-5.47]. Total collective radiological impacts to workers are normalized by the mass of fuel produced and are given in Table C-5.21. The average of the three facility-specific doses was used for the total collective radiological impact.

Table C-5.19. U.S. LEU-Oxide Fuel Fabrication Facilities and Annual Production Capacities.

Country	**Facility**	**Location**	**Capacity (MTIHM/year)**	**Capacity ($MTUF_6$/year)**	**Notes**
USA	Westinghouse Electric Company LLC - Columbia Fuel Fab Facility	Columbia, South Carolina	1150	1700	Uses Dry Process
USA	Areva NP Inc.	Richland, Washington	700	1035	Uses Dry Process
USA	Global Nuclear Fuel - Americas, LLC	Wilmington, North Carolina	1200	1775	Uses Dry Process
Sources: [C-5.34, C-5.36, C-5.37, C-5.38]					

Table C-5.20. Occupational Radiological Impact Data from LEU-Oxide Fuel Fabrication Facilities from Years 2000-2010.

Year	Facility Annual Worker Collective Dose [person-mSv/year]		
	Westinghouse Electric Company	**Areva NP, Inc. - Richland**	**Global Nuclear Fuel - Americas, LLC.**
2000	6154.67	1221.37	1126.91
2001	7251.77	1052.24	860.00
2003	2454.92	951.23	572.63
2004	2361.40	855.38	700.70
2005	1912.00	341.80	599.84
2006	2624.57	803.47	589.94
2007	1827.42	728.51	495.66
2008	1587.14	668.84	734.59
2009	1512.54	897.01	480.03
2010	1419.00	999.76	491.68
10-year Average:	2910.54	851.96	665.20
Notes: The 2002 annual report was not available online through the NRC's website			
Sources: [C-5.19, C-5.39, C-5.40, C-5.41, C-5.42, C-5.43 , C-5.44, C-5.45, C-5.46, and C-5.47]			

Table C-5.21. Occupational Radiological Health Impacts for LEU-Oxide Fuel Fabrication Facilities.

Parameter	Facility-Average Worker Collective Dose [person-mSv/MTIHM]		
	Westinghouse Electric Company	**Areva NP, Inc. - Richland**	**Global Nuclear Fuel - Americas, LLC.**
10-yr Average Annual Collective Dose [person-mSv/year] from Table 2.10-3	2.91E+03	8.52E+02	6.65E+02
Annual Production Capacity [MTIHM/year]	1.15E+03	7.00E+02	1.20E+03
Facility-Specific Normalized Metric [person-mSv/MTIHM]	2.53E+00	1.22E+00	5.54E-01
Average Normalized Collective Dose [person-mSv/LEU-Oxide MTIHM]	**1.43E+00**		

This estimate is likely somewhat conservative relative to what is achievable in future processes. This conclusion is based on the fact that the three sets of plant data differ by a factor of ~10, indicating that much better performance is achievable than is indicated by the average dose. It should be noted however that there is potentially some non-conservatism in that the normalized doses were based on the plant rated capacity and actual production was likely somewhat lower. Nonetheless, based on the considerable data available for hands-on LEU fuel fabrication, confidence in this worker dose estimate was considered high.

Glove Box and Hot Cell

The value for collective dose to workers from glove-box fuel fabrication was taken from Ref. C-5.48 and is based on measured doses at AREVA's MELOX plutonium MOX fuel fabrication plant in France.

The value for hot-cell fuel fabrication is assumed to be the same as that for a reprocessing facility and is based on data for the AREVA La Hague reprocessing facility.[C-5.47] This assumes that the hot cell fuel fabrication facility would be designed using the same standards, operating philosophy, and maintenance philosophy as the reprocessing plant – essentially 100% containment of radionuclides, remote operation and maintenance, and sufficient shielding so as to yield very low external dose rates.

The study of Ref. C-5.48 was used for these estimates and compared the radiological impacts of LWR/LEU oxide once-through and reprocessing fuel cycles. The study was conducted by a multi-national group of technical experts with oversight from the NEA's Committee on Radiation Protection and Public Health. Table C-5.22 provides the worker doses reported in Ref. C-5.48 and the conversion factors for renormalizing them to units of MTIHM.

Table C-5.22. Collective Doses to Workers from Glove Box and Hot Cell Fuel Fabrication.

Fuel Fabrication	Glove Box	Hot Cell
Technology Basis	*Pu, Th fuels*	*U-233, MA fuels*
Reported Occupational Radiological Dose (person-mSv/MWe-yr)	*0.43*	*0.014*
Conversion Factor (MW_eyr/MTIHM)*	27.1	27.1
Normalized Occupational Radiological Dose (person-mSv/MTIHM)	***11.66***	***0.38***
* Fuel burnup in [C-5.48] was 30 GW_td/MT. A thermal efficiency of 0.33 was assumed – leading to a conversion factor of 27.1 MWe-yr/MTIHM		

Confidence levels in the estimate are high for glove box fabrication of MOX fuels, and medium for other fuels using glove box fabrication. Confidence in the estimate was lower for hot cell fabrication because the experience base is very limited.

It should be noted that the hot cell value calculated above is based on the electricity generated from the 30 GW_td/MT SNF fed to the reprocessing plant that was assumed in Ref. C-5.48. However, for fabrication the conversion factor should be based on the electricity that will be produced by the fuel that is fabricated, which will vary depending on which representative fuel and reactor are being considered. If this dose represents a significant fraction of the total NFC dose, it would be appropriate to recalculate this impact using the fuel burnup applicable to the specific NFC under consideration.

Other Considerations

It is assumed that occupational impacts from fabrication of fuels having similar key radiological characteristics (i.e., penetrating radiation and radiotoxicity) are similar irrespective of differences in enrichment, fuel form (oxide, carbide, metal), or structure (zirconium-based, SS, graphite). For example, fabrication of HTGR LEU fuel would have the same occupational impacts as LWR LEU-oxide fuel. There is no identified, defensible base of experience with fuels other than LWR UOX and MOX on which to base occupational doses for the other fuels.

In some reactor systems (e.g., MSRs and some dry processing) fuel reprocessing and fuel fabrication are integrated into a single hot cell facility. A reasonable assumption is that the value for hot cell worker impacts should be applied only once, i.e., the worker impact should be accounted for either in the reprocessing or fabrication step, but not both.

Reactor Construction and Operations

Metrics for radiological impacts to workers were quantified for eight candidate reactor technologies shown in Table C-5.23: boiling water reactor (BWR), pressurized-water reactor (PWR), high-temperature gas-cooled reactor (HTGR), a heavy-water moderated reactor (HWR), sodium-cooled fast reactor (SFR), a molten-salt reactor (MSR), an accelerator-driven subcritical reactor (ADS), and a fission-fusion hybrid reactor (FFH). The normalized radiological impacts to workers for each reactor system are given in Table C-5.23 and discussed in the following subsections.

Collective occupational dose estimates from reactor decommissioning have been excluded based on NEA conclusions concerning the relatively small contribution of such doses compared to collective doses received during normal operations [C-5.48]. Ref. C-5.48 states that, "Annual collective occupational exposures during decommissioning of all stages of the fuel cycle, including reprocessing, have been very small, particularly in comparison with worker doses from other stages of the fuel cycle. This is due to the long time period over which decommissioning is conducted, and due to the radiation protection means applied during work activities. Additionally, these doses would be further reduced if normalized with respect to electricity production." As a cross-check for this NEA conclusion, the Trojan Nuclear Plant was decommissioned while incurring 5910 person-mSv. Trojan was a 1.095 GWe PWR. Projecting this class of reactor into the future with a 90% capacity factor and a 60-year reactor life yields 1.095*60*0.9 = 59.130 GWe-yr, which, when divided into 5910 person-mSv yields 100 person-mSv/GWe-yr which is not negligible but not large compared to reactor doses ranging from 500 to 2800. The impact of decommissioning other facilities will be even smaller, as they may each support several reactors. [NOTE: Putting a reactor in safe storage before decommissioning is estimated to lower worker dose by ~6x, per].

Table C-5.23. Radiological Impacts to Workers for Candidate Reactor Systems.

Reactor Operations	Normalized Impacts (person-mSv/GWe-yr)							
	PWR	BWR	HWR	SFR	HTGR	MSR	ADS	FFH
Occupational Radiological Dose	730	1570	1830	1200	730	490	2770	2060
Confidence Level	High	High	High	Med	Low	Low	Very low	Very low
Analogous System Used?	No	No	No	No	Yes (PWR)	Yes (Reprocessing)	Yes (SFR + BWR)	Yes (MSR + BWR)

It should be noted that the normalized collective worker doses in Table C-5.23 are (directly or by analogy) based on handling LEU fuel during fuel receipt and refueling. At least the SFR, and possibly all of these reactors, are likely to handle fresh MOX fuel containing reactor-grade Pu that is much more radioactive than LEU and which is expected to take longer to unload from the required secure transports. The associated dose increase has not been calculated because data to support disaggregation of the portion of LEU-fueled reactor dose resulting from fuel receipt and refueling has not been found. As an example of the potential implications, the penetrating radiation dose rates from reactor-grade Pu and Th are about 100 times that from LEU. For the purposes of illustration, if 1% of the presently measured LEU reactor collective dose to workers comes from fresh fuel handling (likely an over-estimate, but used here for arithmetic convenience), then use of Th or Pu fuels would roughly double the total reactor collective dose unless significant additional radiation protection measures are taken.

BWR and PWR

The NRC has been collecting data on occupational doses and power production at U.S. nuclear power reactors for more than 30 years. The data is analyzed and published in a series of reports with the latest edition at the time of this analysis being Ref. C-5.47 containing data through 2010. The information provided includes collective occupational dose and electrical energy production by reactor type (BWR or PWR) for the U.S. fleet. The value adopted in this report is the most recent 3-year average of the collective worker dose for BWRs and PWRs divided by their electricity production taken from Tables 4.1 and 4.2 of Ref. C-5.47, respectively. The historical trend of collective dose for workers has been declining as improved technologies and worker radiation protection measures have been brought to bear. [C-5.50] As a consequence, the 3-year average from 2008 to 2010 is used because this more recent data better represents future impacts of commercialized systems. The recent data from the NRC reports are given in Tables C-5.24 and C-5.25 for BWRs and PWRs, respectively, as well as historical data to provide some perspective on the decline in collective dose trends.

Table C-5.24. Occupational Dose and Energy Production Data from U.S. BWRs from Years 1994-2010.

Year	No. of Individuals with Measurable Dose	Annual Collective Dose (person-mSv)	Average Measurable Dose per Individual (mSv)	Electricity Generated (GWe-yr)	Average Collective Dose per GWe-yr (person-mSv/ GWe-yr)
1994	39171	120980	3.09	22.139	5.46E+03
1995	35686	94710	2.65	24.737	3.83E+03
1996	37792	94660	2.50	24.3222	3.89E+03
1997	34021	76030	2.23	22.8661	3.33E+03
1998	32899	68292.96	2.07	23.7812	2.87E+03
1999	31482	64344.3	2.04	26.9626	2.39E+03
2000	31186	60896.76	1.95	28.4769	2.14E+03
2001	28797	48353.97	1.68	28.7304	1.68E+03
2002	30978	61077.67	1.97	29.46	2.07E+03
2003	30759	56594.34	1.84	29.0944	1.95E+03
2004	33948	54509.82	1.61	29.4248	1.85E+03
2005	33544	59959.75	1.79	29.3868	2.04E+03
2006	34159	49897.61	1.46	30.2384	1.65E+03
2007	37515	53884.16	1.44	30.1893	1.78E+03
2008	34642	45224.13	1.31	31.2483	1.45E+03
2009	36207	52828.69	1.46	30.7627	1.72E+03
2010	37214	48076.56	1.29	31.2746	1.54E+03
				10-year average (2001-2010)	**1.77E+03**
				3-year average (2008-2010)	**1.57E+03**
Sources: [C-5.50] (Table 4.1: Summary of Information Reported by Commercial Boiling Water Reactors 1994-2010)					

Table C-5.25. Occupational Dose and Energy Production Data from U.S. PWRs from Years 1994-2010.

Year	No. of Individuals with Measurable Dose	Annual Collective Dose (person-mSv)	Average Measurable Dose per Individual (mSv)	Electricity Generated (GWe-yr)	Average Collective Dose per GWe-yr (person-mSv/ GWe-yr)
1994	44283	95740	2.16	52.3976	1.83E+03
1995	49985	117620	2.35	54.1382	2.17E+03
1996	46852	94170	2.01	55.3378	1.70E+03
1997	50690	95460	1.88	48.9853	1.95E+03
1998	38586	63581	1.65	53.2887	1.19E+03
1999	43938	72313	1.65	56.235	1.29E+03
2000	42922	65620	1.53	57.5299	1.14E+03
2001	38773	62732	1.62	58.8224	1.07E+03
2002	42264	60184	1.42	59.3697	1.01E+03
2003	44054	62961	1.43	57.9206	1.09E+03
2004	35901	49169	1.37	60.3987	8.14E+02
2005	44583	54598	1.22	59.7909	9.13E+02
2006	46106	60314	1.31	59.7513	1.01E+03
2007	42015	47316	1.12	61.9556	7.64E+02
2008	44808	46735	1.04	60.586	7.71E+02
2009	45547	47419	1.04	60.4679	7.84E+02
2010	37796	38237	1.01	60.8594	6.28E+02
				10-year average (2001-2010)	**8.8E+02**
				3-year average (2008-2010)	**7.3E+02**
Sources: [C-5.50] (Table 4.2: Summary of Information Reported by Commercial Pressurized Water Reactors 1994-2010)					

Heavy water Reactors

There are multiple HWR designs in the world. For this effort occupational dose data on CANDU reactors in Canada formed the basis for metric quantification. The collective occupational dose for Canadian CANDU reactors was obtained from the NEA Information System on Occupational Exposure (ISOE) 2011.[C-5.51] Reactor electricity generation capacities were taken from Ref. C-5.52. A capacity factor of 0.8 was adopted based on CANDU nuclear station reliability web page from April 2009, Ref. C-5.53. The data from these sources for the Canadian CANDU fleet in 2009 are summarized in Table C-5.26 leading to an electricity-normalized collective worker dose of 1830 person-mSv/GWe-yr.

CANDU collective worker doses are higher than PWR collective worker doses even though both are two-loop pressurized water reactors. The reasons for this are not clear. However, most worker doses at U.S.

LWRs are incurred during refueling/maintenance outages and the situation is similar in CANDUs (see Table C-5.27). The NEA ISOE [C-5.51] notes that Canadian CANDUs experience an estimated 2-3 planned and forced outages each year as compared to U.S. PWRs which typically shut down for refueling and maintenance once every 18 months. That CANDUs have more outages than PWRs is consistent with the lower capacity factor of 0.8 for the CANDUs compared to over 0.9 in U.S. LWRs. It is possible that doses associated with more frequent maintenance outages at CANDUs accounts for the higher collective dose to workers. Another possibility is that workers received additional doses during on-line refueling.

Table C-5.26. Occupational Dose and Energy Production Data for Canadian CANDU HWRs from 2009.

CANDU Reactor - Unit	Capacity (MWe)	Year of Exposure	Reactor Collective Dose (person-mSv/yr)	Electrical Energy Produced (GWe-yr/yr)	Reactor Normalized Collective Dose (person-mSv/GWe-yr)
Bruce-A-3	750	2009	1371.5	0.6	2.29E+03
Bruce-A-4	740	2009	1371.5	0.592	2.32E+03
Bruce-B-1	822	2009	1076.75	0.6576	1.64E+03
Bruce-B-2	822	2009	1076.75	0.6576	1.64E+03
Bruce-B-3	822	2009	1076.75	0.6576	1.64E+03
Bruce-B-4	822	2009	1076.75	0.6576	1.64E+03
Darlington-1	881	2009	798.25	0.7048	1.13E+03
Darlington-2	881	2009	798.25	0.7048	1.13E+03
Darlington-3	881	2009	798.25	0.7048	1.13E+03
Darlington-4	881	2009	798.25	0.7048	1.13E+03
Gentilly-2	638	2009	677	0.5104	1.33E+03
Pickering-A-1	515	2009	1220	0.412	2.96E+03
Pickering-A-4	515	2009	1220	0.412	2.96E+03
Pickering-B-5	516	2009	852.5	0.4128	2.07E+03
Pickering-B-6	516	2009	852.5	0.4128	2.07E+03
Pickering-B-7	516	2009	852.5	0.4128	2.07E+03
Pickering-B-8	516	2009	852.5	0.4128	2.07E+03
				Average	**1.83E+03**
Sources: Worker exposure data comes is reported for the year 2009 from [C-5.51]; Generation capacities of reactors were taken from [C-5.52] Reactor capacity factor of 0.80 taken from [C-5.53];					
* After multiplying by capacity factor of 0.8 and converting to GWe-yr/yr					

Table C-5.27. Occupational Dose Data from Outages and During Electricity Generation from 2009 for Canadian CANDUs (HWRs).

CANDU Reactor - Unit	Dose while Generating Power (person-mSv/yr)	Outage Dose (person-mSv/yr)	Total Collective Occupational Dose (person-mSv/yr)
Bruce-A-3	170.5	1201.0	1371.5
Bruce-A-4	170.5	1201.0	1371.5
Bruce-B-1	142.5	934.3	1076.8
Bruce-B-2	142.5	934.3	1076.8
Bruce-B-3	142.5	934.3	1076.8
Bruce-B-4	142.5	934.3	1076.8
Darlington-1	64.0	734.3	798.3
Darlington-2	64.0	734.3	798.3
Darlington-3	64.0	734.3	798.3
Darlington-4	64.0	734.3	798.3
Gentilly-2	156.0	521.0	677.0
Pickering-A-1	235.0	985.0	1220.0
Pickering-A-4	235.0	985.0	1220.0
Pickering-B-5	143.3	709.0	852.3
Pickering-B-6	143.3	709.0	852.3
Pickering-B-7	143.3	709.0	852.3
Pickering-B-8	143.3	709.0	852.3
Average	**139**	**847**	**986**
Source: [C-5.51]			

Notes:

- Darlington Units 1-4: Outages were extensive due to vacuum building outage that required all units to shutdown.
- Bruce-A Units 3 & 4: Two planned outages were performed during 2009 that required the two units to shutdown.
- Bruce-B Units 1-4: An unknown number and cause of outages occurred during 2009
- Gentilly Unit 2: A decrease in outage-dose occurred in 2009 vs. 2008 due to less schedule times of maintenance
- Pickering-A Units 1 & 4: Reported planned and forced outages occurred in 2009 that resulted in a outages-dose. There was a reduction in routine operations compared to the previous year's operations
- Pickering-B Units 5-8: A lesser number of outages were required for the year 2009 vs. 2008 that lead to a lower collective worker dose compared to 2008. Internal doses were a record low from implementing several airborne exposure reduction initiatives (e.g., improved drier performance, decreased tritium curie content in moderator and heat transport of D_2O)

Sodium-Cooled Fast Reactors

Defensible data concerning worker dose at SFRs is limited, primarily because: (a) most SFRs have been prototype, demonstration, or test reactors, where sustained high power operation was not the main objective, and (b) most SFRs had low capacity factors due to reliability issues. An exception is the Russian BN-600 reactor that has been operating relatively reliably (75% - 80% capacity factors) for 20 years. For the 5 years ending in 2010, the average annual collective dose was 540 person-mSv at an average capacity factor of 78% [C-5.54]; therefore, this data has been used to estimate dose from use of SFRs in an equilibrium fuel cycle. Using a BN-600 electricity generating capacity of 600 MW_e yields the normalized collective worker dose given in Table C-5.23 and calculated in Table C-5.28 below.

Some SFR designs call for deployment of multiple small modular reactors (SMRs) at a site so as to constitute a virtual large (1000 MW_e-class) reactor. It is likely that the average individual annual dose would be about the same for large reactors and aggregate SMRs. However, there is no basis for concluding that the number of workers required in a large reactor would be the same as the sum of workers for equivalent SMR capacity. To the extent that the number of workers per unit of electrical output differs, so too will the normalized worker dose. Regulations for SMRs are still evolving and there is no evident way to quantify any differences at present. This same thought is equally applicable to using other reactor technologies such as PWRs and HTGRs for SMRs.

Table C-5.28. Occupational Dose and Energy Production Data for the Russian BN-600 from 2005-2010.

Parameter Description (Unit)	Parameter Value
Average Annual Worker Collective Dose (person-mSv/yr)	540
Electricity Generating Capacity (MW_e)	600
Capacity Factor (unitless)	0.78
Annual Electrical Energy Production (GWe-yr/yr)	.468
Normalized Worker Collective Dose Metric for SFR system (person-mSv/GWe-yr)	1.2E+03
Source: 5-year average from 2005-2010 (annual breakdown is not available} was adopted from [C-5.54] describing the Russian BN-600 fast reactor.	

High-Temperature Gas Reactors

A helium-cooled, graphite-moderated HTGR using prismatic block fuel was taken as the representative technology for an HTGR for estimating occupational dose. Data is limited because experience has mostly involved testing of fuel in non-HTGR reactors, and small prototype or test HTGRs dating to the 1980s. For example, Peach Bottom Unit 1 only generated 15 MWy(e) with graphite fuel and accessible NRC documents containing occupational dose do not go back this far.

The single exception to the non-HTGR and small prototype experience is operation of the Fort St. Vrain reactor (FSV, 342 MWe) between 1974 and 1991. Unfortunately, this reactor had reliability issues and, in its best two years, achieved a capacity factor of only 28% [C-5.55] which does not provide a defensible basis for estimating worker dose for a future, commercial HTGR–that must be assumed to operate reliably. In 1981 and 1983, the FSV reactor produced 94 GWe-yr out of a possible 330 and worker collective dose was 10 person-mSv leading to an imputed/inferred 0.11 person-mSv per MWeyr. This value is likely low because of the many people doing maintenance.

On balance, for an HTGR, like a PWR, the coolant is not significantly activated (although both coolants would contain trace activation and fission products) so it was assumed that an HTGR would have the same normalized radiological dose to workers as a PWR: 730 person-mSv/MW_eyr. This value is considered more realistic than paper studies that estimated collective worker doses ranging from 0.7 to 2.0 person-mSv/MW_eyr [C-5.56] but the disparity leads to confidence in the result being low.

Molten Salt Reactors

A representative MSR is taken to be graphite-moderated using thorium/U-233 fuel dissolved in a circulating molten fluoride coolant. It would be a two-loop design similar to PWRs, CANDUs, and SFRs. It would have a fully integrated fuel processing plant to remove fission products, remove and/or feed fissile material, isolate Pa-233 for decay, and feed thorium. There is no reactor operating experience on which to base a collective worker dose estimate. The only MSR to operate was the Molten Salt Reactor Experiment (MSRE) at ORNL, which was a 7.4 MW_t reactor designed to test the reactor concept and materials. The MSRE operated for about 1.5 full-power years during its 5-year life, which is not sufficiently representative of a potential future MSR so as to provide a basis for estimating collective worker doses. Additionally, finding worker dose information for the MSRE has been unsuccessful. An MSR worker collective dose estimate was developed by recognizing that: (a) most worker dose at reactors where the secondary loop is not radioactive results from maintenance performed during maintenance/refueling outages, and (b) both the MSR primary and reprocessing loops will have to be designed as a hot-cell- (or canyon-) type facility with remote maintenance because these loops contain what is essentially spent fuel. Thus, in concept, a MSR is similar to a nuclear fuel reprocessing plant and, thus, the MSR would be expected to have individual worker dose rates similar to a reprocessing plant, i.e., 14 person-mSv/GWe-yr, where conversion assumed fuel burnup from Ref. C-5.48 of 30 GWtd/MTIHM and thermal efficiency of 0.33, resulting in 27.1 MWeyr/MTIHM. However, the normalization basis for a reactor is different from that of a reprocessing plant. Thus, in concept, a MSR is similar to a nuclear fuel reprocessing plant in that it would be remotely operated and involves a substantial number of flowing nuclear materials including those in the integral fuel reprocessing plant. As a consequence, the annual individual dose to MSR workers was assumed to be the same as in a fuel reprocessing plant. Additionally, absent detailed information about the number of exposed workers at a MSR, the same number was assumed as for a fuel reprocessing plant. This means that the un-normalized collective dose to MSR workers (in person-mSv) is the same as that for fuel reprocessing plant workers. However, a typical aqueous fuel reprocessing plant can support about 35 reactors each generating ~1 GWe-yr annually for the collective dose it imparts to workers whereas a single MSR would be producing ~1 GWe-yr annually. Thus, the electricity-normalized collective dose from a reprocessing plant (14 person-mSv/GWe-yr) needs to be multiplied by 35 to yield the electricity-normalized collective dose for the MSR which is 490 person-mSv/GWe-yr.

Accelerator Driven Systems

This system is composed of a high-energy, high-current proton accelerator and target, which produce spallation neutrons. These neutrons drive a closely coupled subcritical assembly generating fission energy that is converted to electricity by conventional means. In essence, the sub-critical assembly is a stand-alone nuclear reactor. Transport of the accelerator-produced neutrons evenly through the core and efficient production of fissile material that will be used to support other reactors and/or generate power or transmute selective elements/isotopes generally favors use of a fast spectrum. On the basis that the reactor portion will effectively share many of the same characteristics as a SFR, we represent this portion of the system with an SFR and adopt the associated radiological dose to workers. The accelerator portion of the system leads to the need to add a contribution from maintaining radioactive components of the accelerator: beam tubes (especially the target), target cooling system, and possibly target material cleanup operations. There is no experience with high-availability, high-current accelerator systems, and target materials vary widely: solid or liquid, various metals such as lead, mercury, and tungsten. For this draft, we assume that the target involves a single loop that is similar to a single-loop reactor (BWR). This combination leads to a worker radiological dose of 1200 + 1570 = 2770 person-mSv/GWe-yr and is shown below in Table C-5.29.

Table C-5.29. Analogous ADS System Components and Associated Worker Dose Estimates.

ADS System Component	Analogous Reactor System	Collective Dose for Analogous Reactor System (person-mSv/GWe-yr)
Proton-accelerator with worker doses from high-energy protons	SFR	1200
Metal targets	BWR	1570
	Total:	2.8E+03

Fission-Fusion Hybrid Reactors

This concept is sufficiently immature so that it is possible to postulate very different representative technologies. One possibility is a fusion reactor using excess neutrons to produce U-233 (from Th-232), which is then recovered and used to make fuel for fission reactors, and within this various types of fusion and fission reactors are possible. Another possibility is a fusion reactor that produces U-233 in a blanket of Th-232 that is nearly critical and in which most of the power is produced (much like an ADS concept with MSR-like components). For the purpose of this effort we assume the latter. In particular, we assume the FFH is composed of a torus fusion reactor having one or more molten salt fluoride blankets that combine fusion heat removal, a subcritical assembly generating substantial fission power, and tritium production to continue to fuel the fusion reactor.

The blanket portion of the system, in which most of the power is generated, is similar to a MSR because it transports heat to generate electricity, contains nuclear materials that are dissolved in the subcritical blankets, and produces tritium, all of which are comparable to an MSR. Thus, the MSR worker radiological dose is adopted here. The fusion portion of the system entails the need to maintain highly activated components of the fusion device in the presence of blanket material after the blankets are assumed to be drained and to maintain tritium storage and feed systems. There is no experience with maintaining such systems and the design philosophy (e.g., remote, semi-remote) that will be used has not been established. Because the primary fusion system components are radioactive and there is no radioactive secondary loop, we assume that worker doses will be similar to that for a single-loop BWR. This combination leads to a worker radiological dose of 490 + 1570 = 2060 person-mSv/GWe-yr and is shown below in Table C-5.30.

Table C-5.30. Analogous FFH System Components and Associated Worker Dose Estimates.

FFH System Component	Analogous Reactor System	Collective Dose for Analogous Reactor System (person-mSv/GWe-yr)
Torus Fusion Reactor Blanket	MSR	490
Primary coolant loop	BWR	1570
	Total:	2.1E+03

Reprocessing and Waste Conditioning

This section addresses radiological dose to workers from SNF reprocessing and, to the extent that it is an integral part of reprocessing (e.g. for MSRs), recycle fuel fabrication. To avoid double-counting for MSRs, MSR fuel fabrication is assigned zero occupational impact since the reprocessing scheme was accounted for in the reactor dose. Any makeup fuel using enriched U or Pu can be made by simply mixing and melting the component chemicals (e.g., UF_4, LiF, BeF) in a chemical lab with hoods. ^{233}U makeup fuel would be made similarly but inside a hot cell with little additional dose. Values for the normalized collective radiological dose to workers are given in Table C-5.31 and discussion of the basis for these values follows.

Table C-5.31. Radiological Worker Impacts for Recycling and Reprocessing Operations.

Fuel Reprocessing/ Recycling Technology	**High Temp/Dry**					**Aqueous**			
	E-Chem	Melt Refining	Halide Slagging	OREOX	In-line MSR (F, Bi extraction)	PUREX	Co-decontamination, UREX-NPEX	THOREX	TRISO (crush, crack, dissolve)
Occupational Radiological Dose (person-mSv/MTIHM)	0.38					0.38			
Confidence Level	Low					High	Med	Med	Low

Aqueous Technologies

There is significant experience with commercial reprocessing of LWR fuels using the standard PUREX process and the occupational impacts of reprocessing as a result of the operation of the La Hague reprocessing plants in France and THORP in the U.K. The value for the collective worker radiological dose of 14 person-mSv/GWe-yr was based on experience at La Hague.[C-5.48] Normalization is based on the 30 GWd/MT burnup assumed in Ref. C-5.48 which, assuming a thermal efficiency of 33%, leads to a conversion factor of 0.027 GWe-yr/MTIHM and the mass-normalized occupational impact of 0.38 person-mSv/MTIHM shown in Table C-5.31.

Currently envisioned co-decontamination and fuel fractionation processes such as the UREX variants include many processes that are essentially identical to PUREX plus additional processes needed to accomplish the fractionation. Fractionation processes might be used, for example, to separate minor actinides and lanthanides from the raffinate, minor actinides from the lanthanides, and cesium and strontium from the raffinate. While any additional processes would be performed in a hot cell or canyon environment, imparting the same low dose rate to individual workers as the PUREX process per se, it is likely that some number of additional workers would be needed to conduct the additional processes leading to an increase in collective dose. However, the extent of the increase is unknown because it would depend on the number of additional processes and their design, which has not yet been determined. On balance, the increase is expected to be small in comparison to the scope of the entire reprocessing operation which supports adoption of the same value as for PUREX: 0.38 person-mSv/MTIHM.

There is a limited (and dated) basis for estimating how occupational doses from THOREX processing of thorium-based fuels might differ from PUREX experience.[C-5.57] UREX is expected to be a good first approximation for THOREX for thorium-based oxide fuels because a future deployment of THOREX would presumably require additional separations processes functionally similar to what was added to PUREX to yield UREX, and because THOREX and PUREX are basically the same process. However, the additional processes have not yet been conceived and there is presently no basis for differentiating the occupational impacts of THOREX from those of UREX or, as discussed previously, from those for PUREX. Thus, we adopt the same value as for PUREX: 0.38 person-mSv/MTIHM.

There is also no data for estimating impacts from reprocessing graphite fuels containing TRISO particles. Separations for HTGR fuel would presumably involve the additional processes and attendant occupational impacts discussed above for a UREX variant or THOREX depending on the nature of the fuel matrix. Further, graphite-based HTGR fuel reprocessing would or could entail additional occupational impacts because of the need for additional head-end processes to either crush graphite blocks, requiring rotating dust-generating equipment inside cells leading to additional ventilation system complexity, or to remove the fuel-bearing material from the bulk graphite moderator. Either of these steps would be followed by burning some or all of the graphite leading to major expansion of the off-gas system to remove contaminants from the large volume of carbon dioxide. Occupational impacts from handling a substantial volume of stabilized waste containing C-14 would also be increased. However, the extent of the additional worker radiological impacts is presently unknowable because of the lack of experience or even an integrated design for such a facility. Again, we assume the same value as for PUREX: 0.38 person-mSv/MTIHM.

Confidence in the PUREX and UREX co-decontamination values for collective worker radiological dose is high because it is based on experience in an industrial-scale facility for PUREX and the relatively modest differences between PUREX and UREX. Confidence in using the same value for THOREX is medium because, while this process has many conceptual similarities to PUREX, there are many differences in detail—coupled with the potential additions to fractionate UF constituents and no experience with these differences. Confidence in using the same value for graphite-based fuels is low because of the additional uncertainties resulting from dealing with the graphite in addition to the uncertainties associated with THOREX and UREX.

High Temperature/Dry Technologies

Dry reprocessing technologies (e.g., electro-chemical) have been operated at laboratory and engineering scale at national laboratories in the U.S. and elsewhere to process nuclear materials as a part of ongoing R&D under non-commercial regulations. As a consequence, available information is not adequate to differentiate collective occupational doses among the dry technologies or between dry and aqueous technologies. Because any of these technologies would have to be implemented in hot-cell or canyon-type facilities conceptually similar to those used for PUREX reprocessing and in accordance with the same regulations and standards, the collective occupational dose value based on operating the La Hague aqueous reprocessing plant from Ref. C-5.48 are also adopted for dry reprocessing: 0.38 person-mSv/MTIHM.

It should be noted that dry reprocessing is typically proposed for deployment at a size adequate to support a single reactor site containing the equivalent of 1 or 2 large reactors, while aqueous reprocessing is typically deployed at a scale so that one plant supports 30-40 large reactors. It is likely that the average individual dose would be about the same for the two facilities because this is driven by regulatory requirements. However, there is no basis for concluding that the number of workers per unit of throughput and, thus, the collective worker radiological dose at a reprocessing plant serving one site scales linearly with that of a large centralized facility. To the extent that the number of dry reprocessing workers required to achieve a given throughput is different than the number of workers for aqueous reprocessing, so too will be the collective worker radiological dose; the lack of a detailed analysis precludes estimating this impact, however.

Storage, Transport, and Disposal

This section quantifies normalized collective radiological dose to workers from SNF and HLW storage, transportation, and disposal (ST&D) operations. The ST&D operations deal with "packages" of nuclear material and the dose rate from a package is regulated. Thus, if there are more packages, there is correspondingly an increase in ST&D handling operations and collective dose to workers. Lower burnup fuels, for example, may require more packages per $MW_e yr$ or per MTIHM. Additionally, even for a given burnup, limits on package sizes may require the use of more packages as might be the case for bulkier SNF such as that from HTGRs. However, radioactivity or decay heat (i.e. higher burnup and/or less cooling time) can also decrease the capacity per package to meet regulatory limits, which would result in additional packages. These factors should be taken into account if additional precision is needed in the ST&D dose estimate.

Estimating these doses was complicated by several other factors. For example, dose data for at-reactor storage facilities are not typically segregated from the dose associated with reactor operations and thus is likely already counted in the dose associated with reactor operations. Further, one must know the duration of the storage period in order to estimate the associated dose. Similarly, doses associated with transportation and storage operations are directly related to the number of handling and transport operations, the distance traveled, and the population density along the route. Lastly, it is not known what, if any repackaging or other preparations may be needed prior to disposal.

Despite potentially large variability resulting from the factors discussed in the previous paragraph, values for normalized collective doses to workers were developed and are presented in Table C-5.32 which is followed by discussion of the bases for the values. The remainder of this subsection provides additional qualitative insights regarding potential differences between these values and collective dose values for thorium-based fuel cycles.

Table C-5.32. Radiological Worker Impacts for ST&D Operations for Repository Wastes.[1]

Process	Storage[2]		Transportation	Disposal	
Technology	Wet	Dry		Near-Surface	Deep
Once-through LWR cycle (person-mSv/MTIHM)	3.8E+00	1.2E+00	5.9E-01	1.32E-01	1.06E+00
LWR fuel reprocessing cycle	NA	NA	5.9E-01	1.7E-01	1.27
LWR MOX fabrication cycle	NA	NA	5.9E-01	9.7E-02	3.07

1. Dose data given per unit electricity produced was converted to dose per MTIHM using 45.2 MWeyr/MTIHM, based a typical LWR fuel cycle (i.e. 50 GWtd/MTIHM and 33% thermal efficiency)
2. The storage doses are based on a 10-year period for wet storage and an 88-year period for dry storage

There is relatively high uncertainty in the estimates shown above. For wet storage, it is unclear how much of the dose is already counted in reactor operations. For transportation, the data in Ref. C-5.48 is based on European data and there is considerable scatter. Additionally, transportation distances in Europe are much less than in the U.S. and adjusting for this is likely to increase these values. There also appears to be inconsistent accounting for transportation impacts across the various phases of the fuel cycle. For disposal, Greater-than-Class-C (GTCC) wastes and DU are not included.

Wet Interim Storage

Obtaining worker doses for wet interim storage is complicated by the fact that individual or collective dose data for at-reactor storage facilities is not typically reported separately from the total dose associated with reactor operations. This complication reduced the available database to that from standalone wet storage facilities of which there are few. For this report, collective doses to workers for the CLAB facility in Sweden were used because it is such an 'away-from-reactor' wet storage facility. A range of normalized occupational doses are provided (50 to 140 person-mSv/GWe-yr) for years between 1986 to 1996, along with the normalized mass throughput of 25 MT/GWe-yr.[C-5.48] The average of the two extremes of the electricity-normalized collective doses was used for this report (95 person-mSv/GWe-yr). Using these two sets of data, the mass-normalized radiological metric is calculated in Table C-5.33. Confidence that the value is representative of the actual impacts is medium because of the limited amount of data available.

Table C-5.33. Occupational Dose and Production Data for Wet Interim Storage Facility from 1986-1996.

Parameter Description (Unit)	Parameter Value
Collective Worker Dose Normalized by Energy Production (person-mSv/GWe-yr)	95
Mass Throughput for 1 unit of GWe-yr Produced (MT SNF/ GWe-yr)	25
Radiological Occupational Metric (person-mSv/MTIHM)	3.8

Dry Interim Storage

A study of the impacts of moving SNF from wet pools to dry storage after various cooling times provides collective doses to workers for representative PWRs and BWRs from dry storage including loading, annual maintenance and inspection, and construction during ISFSI operation.[C-5.58] Using the EPRI results to quantify a radiologic metric for worker impacts faces two major complications. The first complication is that the worker dose is composed of three components: cask loading and unloading, construction of new storage pads or vaults adjacent to existing ISFSIs, and ongoing inspection, surveillance, operations, and maintenance (ISOM). Based on industry experience, EPRI assumes that loading, unloading, and construction are one-time events for each cask. Based on the EPRI's assessment of industry experience, the collective dose from cask loading and unloading is 8 person-mSv/cask and the dose from construction is 1.7 person-mSv/cask. EPRI did not account for cask unloading so it was conservatively assumed that worker dose from unloading was equal to the dose from loading. In the EPRI baseline scenario a total of 10,822 casks contain 136,600 MTIHM of SNF or an average of 12.6 MTIHM/cask. However, the collective worker dose from ISOM activities is assumed to be incurred annually at a rate of 16.2 person-mSv/cask-yr. The relatively large worker dose from ISOM and its being time-dependent means that it dominates the collective dose to workers from dry storage, and that the value of a mass-normalized metric increases as the assumed storage time increases. The second complication is that EPRI's baseline scenario assumes that the inventory of SNF in dry storage increases approximately linearly from 1400 casks to 10,822 casks between 2011 and 2050, and remains constant thereafter until 2099. As a consequence, the amount of SNF subject to ISOM activities (i.e., the normalization basis) and the worker dose therefrom is not constant until 2050. There are also other complications such as variations in SNF burnups and cask capacities but the magnitude of the variation is small compared to the impacts of collective doses to workers having multiple components, one of which is time dependent, and the time- and scenario-dependence of the SNF inventory in dry storage.

A simplified approach to obtaining a mass-normalized value for collective radiological impacts to workers from dry storage involves dividing the cumulative worker dose from the EPRI baseline scenario over 88 years (158,000 person-mSv) by the total amount of SNF in dry storage (136,600 MTIHM) to yield 1.16 person-mSv/MTIHM. This value is unique to the scenario analyzed and would change depending on the duration of the scenario, and assumptions concerning the rate at which the inventory changes. This result is summarized in Table C-5.34. Confidence that the value in Table C-5.34 is representative of the expected radiological impact to workers is medium because the value is scenario-specific, but based on industry analysis and exposure data.

Table C-5.34. Occupational Dose and Production Data for Dry SNF Interim Storage Based on Baseline Scenario in [C-5.58].

Parameter Description (Unit)	Parameter Value
Storage time assumed in scenario(years)	88
Cumulative Collective dose in scenario (person-mSv)	158,000
Steady-state SNF mass storage in Dry Interim Facility (MTIHM)	136,600
Radiological Occupational Metric (person-mSv/MTIHM)	1.16

Transport

Nuclear materials are transported between operations from uranium and thorium recovery through waste disposal. Noticeable doses (on the order of 20 mrem/yr) are received by drivers – who can be exposed for long times – and workers loading and unloading the nuclear materials because of their proximity to the packages. Additionally, the doses from transportation in the front end of the fuel cycle are not negligible [C-5.59] because of the relatively large amount of nuclear material involved and because these materials are not transported with the amount of radiation shielding used in the backend of the fuel cycle. The types of material transported may include natural uranium, natural thorium, uranium hexafluoride, and LLW in the front end of the fuel cycle; fresh nuclear fuel and LLW in the middle of the fuel cycle; and SNF, HLW, LLW, and recycled nuclear material in the backend of the fuel cycle.

Data on nuclear material transportation is sparse, and the data sets that exist are not complete, contain ranges of values, and at times combine occupational and public dose. Data from Ref. C-5.48 was used as a basis for the collective dose to workers from transportation because it attempts to cover the whole once-through fuel cycle although the data are not complete, and there appear to be errors in some tables (e.g., UK dose “All” value should be 3.6, not 36 or the values comprising it do not add up), and transportation of uranium ore and yellowcake in the front end of the fuel cycle, and SNF or HLW to the repository was not included. The consensus range given in Table 16 of Ref. C-5.48, which is for a once-through fuel cycle was used in what follows: 0.005 – 0.022 person-mSv/MWe-yr. Taking the average yields 0.013 person-mSv/MW_eyr. Applying a conversion factor of 45.2 MWe-yr/MTIHM based on a typical LWR fuel cycle (50GW_d/MTIHM and 33% thermal efficiency) yields 5.9E-01 person-mSv/MTIHM (as shown in Table C-5.35).

Table C-5.35. Occupational Dose and Production Data of Fuel Cycle Material Transportation from 1986-1996.

Parameter Description (Unit)	Parameter Value
Collective Worker Dose Normalized by Energy Production (person-mSv/MW_eyr)	1.3E-02
Electricity per unit fuel mass (MW_eyr/MTIHM)	45.2
Radiological Occupational Metric (person-mSv/MTIHM)	5.9E-01

Transportation impacts for fuel cycles involving SNF reprocessing and MOX fuel fabrication were assumed to be the same as for the once-through cycle because the limited data base for transportation is not sufficient to support differentiation. There is relatively high uncertainty in the estimates shown above. For transportation, the data in Ref. C-5.48 is based on European data, and there is considerable scatter. Additionally, transportation distances in Europe are shorter than in the U.S. and adjusting for these distances is likely to increase these values. There also appears to be inconsistent accounting for transportation impacts across the various phases of the fuel cycle, e.g., for disposal, GTCC wastes and DU are not included.

Near-Surface/Shallow Land Disposal

Near-surface burial is assumed to be used for Class A, B, and C low level wastes (LLW). Collective radiological doses to workers values are based on the 2004 EIS of the Richland LLW Disposal Facility owned and operated by U.S. Ecology [C-5.60]. Operational capacity in units of volume of LLW disposed per year for the Richland facility was obtained from the NRC. During 2005-2008, the average annual LLW volumes disposed were 1234 m^3/year (850, 704, 2738, and 645 m^3/year). Sealed sources used for industrial and medical purposes are disposed at LLW facilities, but only comprise less than 1% by volume and activity of all LLW. Due to the low percent of volume and activity of sealed sources, and the fact that sealed sources are used in many industrial applications outside of nuclear power generations, the worker doses attributed to disposing of this type of material is excluded from the quantified metric presented here. The volume-normalized collective occupational dose at the LLW facility is estimated below in Table C-5.36.

Table C-5.36. Radiological Worker Impacts for LLW Shallow Land Burial Operations at U.S. Ecology.

Number of workers	28
Average worker dose (mSv)	0.96
Collective Dose (person-mSv)	26.9
Volume of LLW Disposed of (m^3)	1234
LLW Radiological Worker Metric (person-mSv/m^3 LLW disposed)	0.022

Converting the volume-normalized impact to a mass-normalized impact from specific fuel cycle facilities involves a multi-step process. First, the volume-normalized value of collective worker dose from LLW disposal in Table C-5.36 was used to calculate a volume-normalized value for collective worker dose from disposal of LLW from LWRs. The LWR volume-normalized value was then converted to a mass-normalized value. Then, the normalized volume of waste from front-end and back-end fuel cycle operations from a variety of sources was combined with representative assumptions and the LLW worker data in Table C-5.36 to yield mass-normalized values for collective worker dose from disposal of LLW from these facilities. These steps are elaborated below.

First, a breakdown of types of dry and wet waste and the production of LLW by BWRs and PWRs was obtained from Ref. C-5.61 and is shown in Table C-5.37.

Table C-5.37. Annual LLW Volumes Produced by LWRs.

LLW Type	1978-1981		1982-1985		1985-1986	
	PWR	BWR	PWR	BWR	PWR	BWR
Average LLW Volume (m^3 per year)						
Dry waste						
Compacted	136	405	180	296	122	222
Noncompacted	156	228	105	228	59	139
Filters	1	1	7	3	6	1
Subtotal	299	634	292	527	187	362
Wet Waste						
Resins	30	59	42	62	31	68
Sludges	0	157	7	170	11	123
Concentrates	113	130	35	50	23	48
Oils	--	--	8	25	8	31
Miscellaneous	--	--	3	1	4	6
Subtotal	143	347	96	309	78	276
Total Average Annual LLW Volume, m^3/yr	442	981	388	835	265	639
Source: [C-5.61] Table 1.2 LLW from Nuclear Energy Plants (pg 8); Originally taken from Radioactive Waste						

Next, data in Table C-5.36 are combined with net electricity generation from reactors in the respective time periods, obtained from Ref. C-5.62, and with average discharge burn-up of reactors in the respective time periods, obtained from Ref. C-5.63. The calculated electricity-normalized LLW volume is calculated to be 0.82 m^3/MTIHM as shown in Table C-5.38.

Table C-5.38. Electricity-Normalized LLW Volumes Generated by LWRs.

Parameter	1978-1981		1982-1985		1985-1986	
	PWR	BWR	PWR	BWR	PWR	BWR
LLW Produced (m^3/yr) [from Table C-5.37]	442	981	388	835	265	639
Electricity Production GW_ed/yr)[a]	10994		13415		16619	
Electricity-Normalized LLW Production from LWRs [m^3 LLW/ GW_ed]	0.129		0.091		0.054	
Average Burn-up from LWRs [GW_ed/MTIHM][b]	8.54		9.41		9.32	
Initial fuel content normalized LLW production from LWRs [m^3 LLW/MTIHM]	1.102		0.856		0.503	
Average initial fuel content normalized LLW production from LWRs [m^3 LLW/MTIHM]	0.82					
[a] Derived from [C-5.62]						
[b] Derived from [C-5.62]. Assumes 33% thermal-to-electrical efficiency. Weighted based on initial uranium content for PWRs and BWRs.						

For non-reactor fuel cycle operations, the volumes of LLW produced by uranium enrichment and uranium fuel fabrication were taken from Ref. C-5.61. The waste disposal doses from mining/milling are not included because these doses are included in the worker doses we already have for the FEFC. These results are summarized in Table C-5.39 which also includes the electricity-normalized volume of LLW from LWRs taken from Table C-5.38. These values were converted to mass-normalized collective dose to workers from LLW disposal as described in Table C-5.39.

Ref. C-5.1 gives a production of 7.57 MTDU per MTIHM for uranium enrichment at 4.2% U-235 product enrichment, 0.25% U-235 tails assay, and 0.711% U-235 feedstock enrichment. Collective doses to workers from disposition of depleted uranium from enrichment assume that the uranium is de-converted to uranium oxide which is disposed by near-surface burial. The bulk density of the uranium oxide is assume to be 3 g/cc which leads to a DU oxide volume of 2.52 m^3 DU per MTIHM and, when multiplied by 0.022 person-mSv/m^3 LLW, a collective worker dose of 0.056 person-mSv/MTIHM.

Table C-5.39. Mass-Normalized Radiological Worker Impacts for Near-Surface LLW Disposal from Individual Fuel Cycle Operations.[1]

Operation	Value	Native Unit	Value	Normalized Unit
Enrichment:	3.11E-05	[m^3 / kg SWU][2]	4.3E-03	[person-mSv/ MTIHM]
Uranium Fuel Fabrication:	2.5	[m^3 / MTIHM] of 2-3% enriched LEU[3]	5.4E-02	[person-mSv/ MTIHM]
LWR Reactor Operation:	0.82	[m^3/ MTIHM]	1.8E-02	[person-mSv/MTIHM]
Depleted uranium disposal	2.52	[m^3 / MTIHM]	5.6E-02	[person-mSv/MTIHM]
Once-through fuel cycle total			1.32E-01	[person-mSv/MTIHM]
LWR Fuel Reprocessing	7.56	[m^3/MTIHM of SNF]	1.7E-01	[person-mSv/MTIHM]
LWR MOX Fuel Fabrication	0.44	[m^3/MTIHM as MOX]	9.7E-02	[person-mSv/MTIHM]

1. The normalized radiological impact incurred from a unit volume of LLW disposed is 0.022 person-mSv/ m^3 LLW. The value of 0.022 was multiplied by the values in the second column to produce the values found in 4th column.
2. To calculate the enrichment impacts: 1 MT SWU = 1 kiloSWU = 1000 kg SWU = 1.60E-01 MTIHM at LEU U235 wt% as 4.2%
3. Volume of LLW generated from fuel fabrication is expected to relatively be independent of LEU product enrichment.

The sources and methods described in the section below on deep geologic disposal were used to estimate that an LWR SNF fuel reprocessing plant would produce 37.8 LLW packages per MTIHM of SNF processed by the plant. Each LLW package is a 55-gallon (200 L) drum leading to a volume of 7.56 m^3 per MTIHM which, when multiplied by 0.022 person-mSv/m^3 LLW disposed of yields 0.17 person-mSv/MTIHM. Similarly, the radiological impact to workers from LLW from LWR MOX fuel fabrication that is disposed of in the near-surface is estimated to be 0.097 person-mSv/MTIHM as MOX.

Deep Geological Disposal

Deep geologic disposal metrics for direct disposal of PWR SNF were quantified based on Ref. C-5.11 and constitute estimates because a SNF/HLW repository has not yet operated. The metric value is based on the Supplemental EIS for Yucca Mountain, Table D-12 [C-5.11] which gives a collective worker dose of 7,400,000 person-mrem (74,000 person-mSv) for the nominal 70,000 MTIHM that would have been disposed of in YM, or 1.06 person-mSv/MTIHM (as shown in Table C-5.40).

Table C-5.40. Radiological Worker Impacts for Deep Geological Disposal (SNF).

Parameter (unit)	Value
Collective Dose (person-mSv)	74,000
Mass Disposed in the Disposal Facility (MT SNF)	70,000
SNF Disposal Rad Worker Metric (person-mSv/MT SNF disposed)	1.06
Reprocessing Waste Disposal Rad Worker Metric (person-mSv/MT SNF reprocessed	1.27
MOX Fuel Fabrication Waste Disposal Rad Worker Metric (person-mSv/MT MOX fuel fabricated	3.07

Deep geologic disposal metrics for fuel cycle options involving reprocessing of PWR SNF and fabrication of PWR MOX fuel were quantified based on information provided in recent estimates of the amount of various process and secondary wastes produced by a 800 MTIHM/yr reprocessing plant coupled with a MOX fabrication facility producing about 80 MT/yr of MOX fuel.[C-5.64, C-5.65] It was assumed that the radiological impact to repository workers is proportional to the number of waste packages that have to be disposed. Plans for operating Yucca Mountain called for the SNF to be disposed of in a large sealed transportation and disposal (TAD) packages that would hold 21 PWR fuel assemblies which means that 0.10 TAD is required per metric ton of heavy metal in the PWR SNF.

To provide a comparable value for reprocessing and MOX fuel fabrication wastes that require repository disposal, i.e., vitrified HLW, metal wastes (e.g., cladding), I-129 waste, and greater-than-Class C (transuranic) secondary wastes (deep geological disposal is one of the options under consideration by DOE for this class of wastes), it was assumed that the HLW and metal wastes would be in 2 ft diameter by 15 ft long cylindrical containers of which five fit into a TAD, and the remaining wastes would be contained in 55-gallon drums of which 25 fit into a TAD. The radiological impact to workers is obtained by dividing the fraction of a TAD required for reprocessing or MOX fabrication wastes requiring repository disposal by the fraction of a TAD required for disposal of SNF and multiplying the worker impact for disposal of SNF (1.06 person-mSv/MTIHM) in Table C-5.40 by the result. Based on estimates from the sources cited in the previous paragraph, reprocessing wastes requiring repository disposal would require 0.12 TAD per MTIHM of fuel reprocessed which leads to a worker impact of 1.27 person-mSv per MTIHM of SNF reprocessed. PWR MOX fuel fabrication wastes would require 0.29 TAD per MTIHM as MOX leading to a worker impact of 3.07 person-mSv/MT of MOX produced.

Summary

Table C-5.41 provides a summary of the multipliers that are used to describe the radiologic exposure total estimated worker dose for each element of the nuclear fuel cycle for this metric.

Table C-5.41. Summary of Radiologic Exposure - Total Estimated Worker Dose Impacts.

Portion of Nuclear Fuel Cycle	**Multiplier**	**Units**
Front End of Fuel Cycle		
Mining – Uranium	5.2E-01	Person-mSv/MTNU
Milling – Uranium	9.0E-01	Person-mSv/MTNU
Conversion – Uranium	8.8E-02	Person-mSv/MTNU
Enrichment – Uranium	3.1E-05	Person-mSv/SWU
Deconversion – Uranium	2.9E-02	Person-mSv/MTDU
Extraction and Refining - Thorium	3.8E+00	Person-mSv/MTNTh
Fuel Fabrication		
UOX	1.43E+00	Person-mSv/MTIHM
MOX	1.17E+01	Person-mSv/MTIHM
Reactor		
Reactor Construction	Not Applicable	Person-mSv/GWe-yr
Reactor Operation	7.30E+02	Person-mSv/GWe-yr
Reprocessing and waste conditioning	3.80E-01	Person-mSv/MTIHM
Disposal and Transportation		
Shallow Land Burial	1.32E-01 (see note 1)	Person-mSv/MT waste
Geologic Repository	1.32E-01 (see note 2)	Person-mSv/MTIHM
Interim Storage	1.16E+00	Person-mSv/MTIHM

1. Units of dose for shallowland burial are person-mSv/MTIHM. Based on once-through LWR fuel cycle. This value increases by a factor of ~3 if reprocessing and MOX fuel fabrication is included.
2. This value is estimated based on disposal of typical LWR spent fuel. Worker dose impacts from disposal of other high level wastes may be slightly higher.

Example Calculation of Radiological Dose to Workers for Basis of Comparison

Table C-5.42 contains an example of the radiological dose to workers calculations for the Analysis Example for Evaluation Group #1 (EG01), a once-through LWR fuel cycle, which serves as the Basis of Comparison for this report. In this case the fuel cycle requires that 188.628 tons of natural uranium must be mined for each 1 GWe-y produced. Uranium mining was calculated to result in a worker dose of 97 person-mSv /GWe-yr. Milling resulted in a worker dose of 170 person-mSv /GWe-yr. Conversion was calculated to result in 17 person-mSv /GWe-yr. The reactor required 21.915 tons of uranium enriched to 4.21%. The tails were 0.25%. The enrichment operation required 137.6 kSWU. The appropriate multipliers are shown in the table and the resulting radiological dose to workers is calculated by multiplying the kSWU by the multiplier to arrive at the appropriate radiological dose to workers of 4.3 person-mSv /GWe-yr. In a like manner, the worker dose for each of the functions of the nuclear fuel cycle was calculated as shown in the table. The total worker dose was estimated to be 1100 person-mSv /GWe-yr. The reactor radiological dose to workers accounts for the major portion of the worker dose (730 person-mSv /GWe-yr) of the total radiological dose to workers in this example. Similar calculations were completed for the Analysis Example of each Evaluation Group.

Table C-5.42. Example Calculation of Radiological Dose to Workers for EG01.

	Worker Radiological Dose Impact Factor	units	EG01 per GWe-yr	units	EG01 Worker Dose (person-mSv/ GWe-yr)	Notes
Front-end Activities						
Mining	5.2E-01	Person-mSv/MTNU	188.6	MTNU/GWe-yr	9.7E+01	
Milling	9.0E-01	Person-mSv/MTNU	188.6	MTNU/GWe-yr	1.7E+02	
Conversion	8.8E-02		188.6	MTNU/GWe-yr	1.7E+01	
Enrichment	3.1E-05	Person-mSv/SWU	137613.5	SWU/GWe-yr	4.3E+00	
Deconversion	2.9E-02	Person-mSv/MTDU	166.7	MTDU/GWe-yr	4.8E+00	Negative due to energy embodied in HF by-product
Fuel Fabrication						
Fuel Fabrication	1.4E+00	Person-mSv/MTIHM	21.9	MTIHM/GWe-yr	3.1E+01	Impact factor for hand-on fabrication of oxide fuels
Reactor Construction and Operations						
Reactor Construction and Operations	7.3E+02	Person-mSv/GWe-yr	1.0	GWe-yr/GWe-yr	7.3E+02	Impact factor for PWR
Recycling/Reprocessing						
Recycling/ Reprocessing	3.8E-01	Person-mSv/MTHM	0.0	MTHM/GWe-yr	0.0E+00	Based on aqueous reprocessing
Storage, Transport, and Disposal						
Wet Storage	N/A					Wet storage impacts included as part of reactor operations
Dry Storage	N/A					Assumed no dry storage needed for steady state system
Transport	5.9E-01	Person-mSv/MTHM LLW disposed	0.0	MTHM/GWe-yr	0.0E+00	
LLW Disposal	1.6E-01	Person-mSv/MTHM LLW disposed	166.7	MTHM/GWe-yr	2.6E+01	
HLW Disposal	1.1E+00	Person-mSv/MTHM HLW disposed	21.9	MTHM/GWe-yr	2.3E+01	
Collective Worker Dose (person-mSv/GWe-yr)					**1.1E+03**	

Binning of the Metric Data

The calculated data derived for the Analysis Examples of the 40 Evaluation Groups are then used as the basis for binning the Metric Data for each Evaluation Group into a metric bin. An Evaluation Group was placed in a different bin than was indicated by the calculated data for the Analysis Example if assumptions for the associated Analysis Example were considered as not being representative of the fuel cycles within the Evaluation Group. By using bins, it is expected that the Metric Data to be used for evaluating an Evaluation Group relative to the Basis of Comparison is sufficiently representative of the best capabilities of fuel cycles within the group. Details on the radiological exposure – total estimated worker dose per energy generated metric calculation approach, the binning process, and the metric bins for the 40 evaluation groups, are described in Appendix D-2.13.

References for C-5.

C-5.1 Carlsen et al., “Environmental Impacts, Health and Safety Impacts, and Financial Costs of the Front End of the Nuclear Fuel Cycle,” US Department of Energy, FCRD-FCO-2012-000124, Rev. 1. July 2013.

C-5.2 US Atomic Energy Commission (AEC) (1974). Environmental Survey of the Uranium Fuel Cycle. Other Information: Orig. Receipt Date: 31-DEC-74, US Department of Energy (US DOE), WASH-1248: Medium: ED; Size: Pages: 277.

C-5.3 Louisiana Energy Services, “Environmental Impact Statement for the Proposed National Enrichment Facility in Lea County, New Mexico,” also issued as NRC Technical Report NUREG-1790, 2005.

C-5.4 US Nuclear Regulatory Commission, “Environmental Impact Statement for the Proposed GE Hitachi Global Laser Enrichment, LLC Facility in Wilmington, North Carolina,” Technical Report NUREG-1938, 2012.

C-5.5 Westinghouse Electric Co., Overview of Columbia Fuel Fabrication Facility, 2010. Available: http://www.erh.noaa.gov/cae/Hazmat2008/Steve%20Carver%20-%20Westinghouse.pdf

C-5.6 Bailly, F., “French Operational Experience in Recycling Used Nuclear Fuel,” *Proceedings of FCIX Conference*, Rockville, MD, June 2009.

C-5.7 U.S. Nuclear Regulatory Commission, “Generic Environmental Impact Statement for License Renewal of Nuclear Plants: Supplement 32, Crystal River Unit 3 Nuclear Generating Plant – Draft Report for Comment,” NUREG 1437, Supplement 44, 2012.

C-5.8 U.S. Nuclear Regulatory Commission, “Generic Environmental Impact Statement for License Renewal of Nuclear Plants,” NUREG 1437, 2012. Available from: http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1437/

C-5.9 U.S. Nuclear Regulatory Commission, “Generic Environmental Impact Statement for License Renewal of Nuclear Plants: Supplement 34, Regarding Wolf Creek Generating Station,” NUREG 1437, Supplement 32, 2012.

C-5.10 US Nuclear Regulatory Commission (NRC). Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities (2000) Thirty-Third Annual Report. NUREG-0713:

Volume 22 (available at http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr0713/v22/nureg0713.pdf).

C-5.11 US Department of Energy (US DOE). Final Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada. EIS-0250 (February 2002).

C-5.12 Hardin, E., et al. *Repository Reference Disposal Concepts and Thermal Load Management Analysis*. FCRD-UFD-2012-00219 Rev. 2, Sandia National Laboratories, 2012.

C-5.13 Peterson, P. F. (2003). Will the United States need a second geologic repository?. *BRIDGE-WASHINGTON-NATIONAL ACADEMY OF ENGINEERING-*, *33*(3), 26-32.

C-5.14 E. Schneider, B.W. Carlsen, and E. Tavrides, "Measures of the Environmental Footprint of the Front End of the Nuclear Fuel Cycle," US Department of Energy, FCRD-SYSA-2010-000104, August 2010.

C-5.15 Areva, FBFC Etablissement de Romans, "Rapport environmental, social, societal, de surete nucleaire et de radioprotection 2008," 2009.

C-5.16 Rapport d'information sur la surete nucleaire et la radioprotection de MELOX, 2010.

C-5.17 Shropshire, D., Schneider, E., "Techno-Economics Summary for Nuclear Power Plants," INL/MIS-09-16406, 2009.

C-5.18 Areva NC, Etablissement De La Hague, "Rapport environnemental, social et societal 2009", 2009.

C-5.19 US Nuclear Regulatory Commission (NRC). Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities (2000) Thirty-Third Annual Report. NUREG-0713: Volume 22 (available at http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr0713/v22/nureg0713.pdf).

C-5.20 Rotty, R. M., Net Energy from Nuclear Power, IEA-75-3 IEA Report, 1975

C-5.21 Areva, "Cezus, Dedicated to Zirconium, Key Component of the Nuclear Fuel," 2013. Available from: http://www.areva.com/EN/operations-942/cezus-specializing-in-zirconium.html

C-5.22 Dixon et al., "Advanced Fuel Cycle Cost Basis (2012 Addendum)," U.S. Department of Energy, FCRD-FUEL-2012-000099, 2013.

C-5.23 U.S. Energy Information Agency, "Annual Energy Outlook 2012," DOE/EIA-0383(2012), 2012.

C-5.24 Hammond, G., Jones, C., "Inventory of Carbon and Energy (ICE) Version 2.0," Sustainable Energy Research Team (SERT), University of Bath, 2011.

C-5.25 J. Kessler et al., "Cost Estimate for an Away-From-Reactor Generic Interim Storage Facility (GISF) for Spent Nuclear Fuel," EPRI Report 1018722, May 2009.

C-5.26 Pidwirny, M., Heins, W., Hussein, G.H., "Structure of the Earth," In: Encyclopedia of Earth, March 2013. Available from: http://eoearth.org/article/Structure_of_the_Earth

C-5.27 U.S. Department of Energy, Energy Efficiency & Renewable Energy, "Energy and Environmental Profile of the U.S. Mining Industry," 2002.

C-5.28 U.S. Environmental Protection Agency, "Emissions Factors & AP 42, Compilation of Air Pollutant Emission Factors, Chapter 11.9 Western Surface Coal Mining," available from: www.epa.gov/ttnchie1/ap42/

C-5.29 U.S. Department of Energy, Office of Civilian Radioactive Waste Management, "Overall Ventilation System Flow Network Calculation for Site Recommendation," CAL-SVS-HV-000002 REV 00, 2001.

C-5.30 U.S. Department of Energy, Office of Civilian Radioactive Waste Management, "Analysis of the Total System Life Cycle Cost of the Civilian Radioactive Waste Management Program, Fiscal Year 2007," DOE/RW-0591, 2008.

C-5.31 U.S. Department of Energy, Office of Civilian Radioactive Waste Management, "Yucca Mountain Science and Engineering Report: Technical Information Supporting Site Recommendation Consideration, Revision 1," DOE/RW-0539-1, 2002.

C-5.32 Armijo, J. S., Kar, P., and Misra, M., "Second generation waste package design and storage concept for the Yucca Mountain Repository," Nuclear Engineering and Design, **236**, pp. 2589-2598, 2006.

C-5.33 US Nuclear Regulatory Commission (NRC). Nuclear Fuel Cycle Facility Accident Analysis Handbook, Science Applications International Corporation (SAIC), US Nuclear Regulatory Commission (NRC). NUREG/CR-6410.

C-5.34 US Nuclear Regulatory Commission (NRC). Fuel Cycle Processes Directed Self-Study Course (Module 5: Fuel Fabrication). (available at http://pbadupws.nrc.gov/docs/ML1204/ML12045A009.pdf).

C-5.35 US Department of Energy (US DOE). "Nuclear Fuel Cycle." Retrieved March, 2013, from http://energy.gov/ne/nuclear-fuel-cycle.

C-5.36 Argonne National Laboratory (ANL). Evaluation of UF_6-to-UO_2 Conversion Capability at Commercial Nuclear Fuel Fabrication Facilities ANL-EAD-TM-110 (available at http://web.ead.anl.gov/uranium/pdf/ANL-EAD-TM-110.pdf).

C-5.37 Areva. "Fuel Production and Areva's Richland Facility." Retrieved March, 2013, from http://us.areva.com/EN/home-427/fuel-design-and-production.html. (Last Revised March 2013).

C-5.38 Global Nuclear Fuel - Americas LLC. (GNF-A). License Renewal Application For Global Nuclear Fuel – Americas LLC NRC License SNM-1097, Docket 70-1113 (available at http://pbadupws.nrc.gov/docs/ML0710/ML071000128.pdf).

C-5.39 US Nuclear Regulatory Commission (NRC). Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities (2003) Thirty-Fourth Annual Report. NUREG-0713:

Volume 23 (available at http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr0713/v23/sr0713v23.pdf).

C-5.40 US Nuclear Regulatory Commission (NRC). Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities (2003) Thirty-Sixth Annual Report. NUREG-0713: Volume 25 (available at http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr0713/v25/sr0713v25.pdf).

C-5.41 US Nuclear Regulatory Commission (NRC). Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities (2004) Thirty-Seventh Annual Report. NUREG-0713: Volume 26 (available at http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr0713/v26/sr0713v26.pdf).

C-5.42 US Nuclear Regulatory Commission (NRC). Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities (2005) Thirty-Eighth Annual Report. NUREG-0713: Volume 27 (available at http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr0713/v27/sr0713.pdf).

C-5.43 US Nuclear Regulatory Commission (NRC). Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities (2006) Thirty-Ninth Annual Report. NUREG-0713: Volume 28 (available at http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr0713/v28/sr0713.pdf).

C-5.44 US Nuclear Regulatory Commission (NRC). Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities (2007) Fortieth Annual Report. NUREG-0713: Volume 29 (available at http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr0713/v29/sr0713v29.pdf).

C-5.45 US Nuclear Regulatory Commission (NRC). Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities (2008): Forty-First Annual Report. NUREG-0713, Volume 30 (available at http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr0713/v30/sr0713v30.pdf).

C-5.46 US Nuclear Regulatory Commission (NRC). Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities (2009) Forty-Second Annual Report. Volume 31: NUREG-0713 (available at http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr0713/v31/).

C-5.47 US Nuclear Regulatory Commission (NRC). Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities (2010). Forty-Third Annual Report (NUREG-0713, Volume 32) (available at http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr0713/v32/).

C-5.48 Nuclear Energy Agency (NEA). Radiological Impacts of Spent Nuclear Fuel Management Options. ISBN 92-64-17657-8 -No. 50203.

C-5.49 Krahn, S.L., Croff, A.G., et al, “Evaluating the Collective Radiation Dose to Workers from the U.S. Once-Through Nuclear Fuel Cycle,” (*Accepted June 2013)* Nuclear Technology

C-5.50 U.S. Nuclear Regulatory Commission, “Operating Nuclear Power Reactors,” 2012. Available from: http://www.nrc.gov/info-finder/reactor/

C-5.51 Nuclear Energy Agency (NEA). ISOE Information Sheet North American Technical Center Information System on Occupational Exposure North American Boiling Water Reactor (BWR) 2006 Occupational Dose Benchmarking Charts. NATC ISOE Information Sheet No. 06-5.

C-5.52 World Nuclear Association (WNA). (2013, Last Revised January 2013). "Nuclear Power in Canada." Retrieved March, 2012, from http://www.world-nuclear.org/info/Country-Profiles/Countries-A-F/Canada--Nuclear-Power/.

C-5.53 Brown, M. "CANDU Nuclear Station Reliability." Retrieved March, 2013, from http://media.cns-snc.ca/media/reliability/reliability.html (2009, Last Revised April 23, 2009).

C-5.54 Vasilyev, B. A., Effectiveness Evaluation for the Fast Sodium-Cooled Reactor Design Solutions and Their Evolution in New Designs. 7th International Scientific and Technical Conference "Nuclear Power Safety, Effectiveness and Economy", MNTK-2010. (available at http://mntk.rosenergoatom.ru/mediafiles/u/files/Archive/Plennarnye/Vasiliev.ppt).

C-5.55 US Nuclear Regulatory Commission (NRC). Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities (1991) Twenty-Fourth Annual Report. NUREG-0713: Volume 13.

C-5.56 Su, S. and B. A. Engholm. “Personnel radiation exposure in HTGR plants,” GA-A-15994, paper presented at the IAEA specialists meeting on gas cooled reactor safety and licensing aspects, Lausanne, Switzerland.

C-5.57 Wymer, R. G., editor, “Thorium Fuel Preparation and Irradiation Testing” pp377-508 and “Thorium Fuel Processing Studies and Demonstrations,” pp 509-670, International Thorium Fuel Cycle Symposium, U.S. Atomic Energy Commission.

C-5.58 Electric Power Research Institute (EPRI). Impacts Associated with Transfer of Spent Nuclear Fuel from Spent Fuel Storage Pools to Dry Storage After Five Years of Cooling. Palo Alto, CA. 000000000001021049 (available at: http://www.epri.com/abstracts/Pages/ProductAbstract.aspx?ProductId=000000000001021049).

C-5.59 World Nuclear Transport Institute (WNTI) (2006). Radiation Dose Assessment for the Transport of Nuclear Fuel Cycle Materials. WNTI Review Series No. 2 (Revised July 2006).

C-5.60 Washington State Department of Health (2004). Final Environmental Impact Statement Commercial Low-Level Radioactive Waste Disposal Site Richland, Washington. DOH Publication 320-031 (available at http://www.doh.wa.gov/Portals/1/Documents/Pubs/320-031_vol1_w.pdf).

C-5.61 Saling, J. H. and A. W. Fentiman (2002). Radioactive Waste Management. New York, NY, USA, Taylor & Francis.

C-5.62 U.S. Energy Information Administration, “Annual Energy Review: Nuclear Power Plant Operations, 1957-2011,” available from: www.eia.gov/totalenergy/data/annual/showtext.cfm?t=ptb0902 webpage accessed: 7/24/2013

C-5.63 U.S. Energy Information Administration, “Nuclear and Uranium: Detailed U.S. Spent Nuclear Fuel Data,” www.eia.gov/cneaf/nuclear/spent_fuel/ussnftab3.html webpage accessed: 7/24/2013

C-5.64 Foare, G. et al., “Waste Estimates for a Future Recycling Plant in the US Based Upon AREVA Operating Experience,” paper 13206, Waste Management 2013 Conference (February 24-28).

C-5.65 Phillips, C. et al., “Improving the Estimates of Waste from the Recycling of Used Nuclear Fuel, paper 13410, Waste Management 2013 Conference (February 24-28).

C-6. Resource Utilization Criterion

For the purposes of the fuel cycle Evaluation and Screening, the Resource Utilization Criterion is defined as follows:

***Resource Utilization** - A broad definition of resource utilization may include any natural resources that are required for a system. For this fuel cycle evaluation and screening, the assessment of resource utilization is focused on the natural fuel resources required for the mature deployed commercial nuclear energy system.*

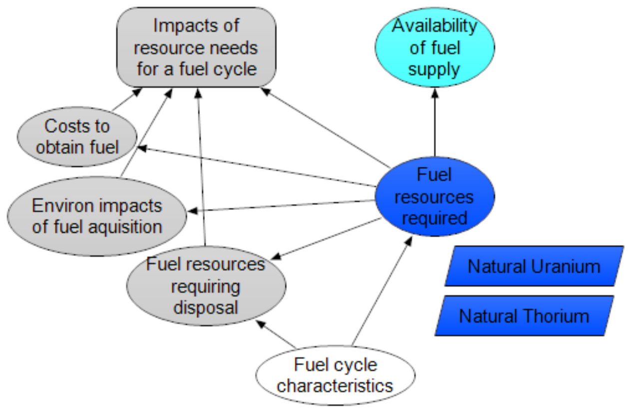
C-6.1 Background on Resource Utilization

Nuclear power uses a number of natural resources, but for the purposes of the fuel cycle evaluation and screening, the focus is on the unique natural resource needs for nuclear power generation: the natural resources required for fuel. Natural fuel resources can be uranium, thorium, or both, depending on the fuel cycle. These are the only resources for fission energy fuel that occur in nature. The efficiency with which a nuclear fuel cycle utilizes the fuel resource is reflected in the amount of fuel resource required to produce a unit of energy.

Influence Diagram

The influence diagram for the resource utilization criterion is shown in Figure C-6.1. Many of the reasons for being interested in resource utilization, such as the costs and environmental impacts of fuel acquisition, and the challenges of disposing of used fuel, are addressed with other criteria and other metrics. Some of these factors are shown in the grey nodes in the figure.

The main concern about resources required that is not addressed by other criteria is the availability of the supply. By calculating the metrics proposed for this criterion and having estimates of the uranium and thorium resources expected in nature, an indication of the amount of time over which the fuel cycle is assured fuel can be made. Options can be made in the fuel cycle design to prolong the use of natural resources, such as using breeding designs and recycle of fuel material.



Note: Each oval represents a factor, element, or question related to resource utilization. Blue indicates factors for which Evaluation Metrics were defined, white indicates factors related to resource utilization that are considered in development of Metric Data. Grey indicates factors not considered but which are recognized as relevant to resource utilization.

Figure C-6.1. Influence Diagram Showing Some of the Relationships Between Resource Utilization and Other Factors.

C-6.2 Metric Development for the Resource Utilization Criterion

For uranium fuel, enrichment of natural uranium concentrates U-235 for use as fuel, typically around 4-5% for current LWRs. The enrichment process leaves depleted uranium as a waste product. As a result, not all of the uranium provided for fuel is used, and this lowers the utilization (in percent) of the resource. However, enrichment enables using the uranium fuel for a longer time in the reactor or other irradiation system, offsetting some of the non-productive "loss" of uranium in the enrichment process. A significant amount of uranium is discharged from the reactor, along with new fissile isotopes that were created during irradiation. For a once-through fuel cycle, the discharged uranium oxide fuel is considered as waste and disposed. Overall, the uranium utilization for such systems is less than 1%, i.e., over 99% of the uranium supplied to the enrichment plant is ultimately discarded.

Reactors and externally-driven systems can be designed to use uranium fuel more efficiently, usually by increasing the production rate of new fissile isotopes during irradiation, called the internal conversion ratio. The higher the internal conversion ratio, then the higher the uranium utilization in percent and the lower the natural fuel resource required. Such systems can in principle achieve up to potentially/possibly 30% uranium utilization without recycle.

To achieve very high utilization, recycle must be used to recover the usable fuel materials from the used fuel, which must then be reused in new fuel. When combined with high internal conversion ratio, and if all of the potentially useful fuel materials are recycled (including all actinides), utilization can approach 100%, depending on the process losses during reprocessing and recycle fuel fabrication.

The same considerations apply for thorium. Thorium itself, however, is not usable directly in reactors to produce power in a critical system, but must first capture neutrons to create fissile U-233. If the internal conversion ratio is high enough, at equilibrium the thorium can be converted to U-233 at a sufficient rate to allow refueling with thorium. Otherwise, other fissile material must be added, usually enriched uranium, or an external source of neutrons must be provided.

There are significant differences in the uranium and thorium fuel cycles that could impact fuel cycle performance. Therefore the two are considered separately for this evaluation. Fuel cycles that use both uranium and thorium have characteristics of both fuel sources.

In summary two metrics for evaluating resource utilization are provided:

- Natural uranium required per energy generated
- Natural thorium required per energy generated

C-6.3 Natural Uranium Required per Energy Generated

The natural uranium required per energy generated is the focus of this section.

Definition of Metric – The metric is defined in terms of the amount of elemental uranium required, not in terms of ore grade or other similar quantity. Such effects are included in the environmental impact for the generic fuel resource of the fuel cycle. The information on natural uranium required per unit of energy produced is generated as part of the detailed reactor physics-based calculations for the Analysis Example for each Evaluation Group. The EST used this information as the basis for developing the Metric Data, as described in Appendix D.

Given that the natural uranium required per unit of energy generated is calculated as part of the reactor physics-based analysis of the fuel cycle, which has low uncertainty, the nominal value of fuel usage is appropriate for the Evaluation and Screening. The calculated data derived for the Analysis Examples of the 40 Evaluation Groups are then used as the basis for binning each Evaluation Group into a metric bin. An Evaluation Group was placed in a different bin than was indicated by the calculated data for the Analysis Example if assumptions for the associated Analysis Example were considered as not being representative of the fuel cycles within the Evaluation Group. By using bins, it is expected that the Metric Data to be used for evaluating an Evaluation Group relative to the Basis of Comparison is sufficiently representative of the best capabilities of fuel cycles within the group. Details on the *natural uranium required* metric calculation approach, the binning process, and the metric bins for the 40 Evaluation Groups, are in Appendix D-2.14. The approach for metric binning and the developed bins are also contained in that appendix. Table C-6.1 provides the bins developed for the Natural Uranium Required per Energy Generated metric.

Table C-6.1. Metric Bins for Natural Uranium Required per Energy Generated.

Bin ID	Data Range (t/GWe-yr)	Bin Description
A	< 3.8	Natural uranium mass required < 3.8 t/GWe-yr; includes fuel cycle options with uranium utilization ≥ 30% and thorium-only options
B	3.8 to < 35.0	Natural uranium required mass from 3.8 t/GWe-yr to < 35.0 t/GWe-yr; includes options with uranium utilization ≥ 3% and < 30%; bounded by performance of advanced approaches constrained by physics performance <u>without fuel reprocessing</u>
C	35.0 to < 145.0	Natural uranium required mass from 35.0 t/GWe-yr to < 145.0 t/GWe-yr; includes options with uranium utilization ≥ 0.8% and < 3%; bounded by performance of more traditional proposals for increasing utilization
D	≥ 145.0	Natural uranium required mass equals or greater than 145.0 t/GWe-yr; contains options with uranium utilization similar to or lower than those of currently operating thermal reactors (LWRs and CANDU); <u>contains the basis of comparison</u>

C-6.4 Natural Thorium Required per Energy Generated

The natural thorium required per energy generated is the focus of this section.

Definition of Metric – The metric is defined in terms of the amount of elemental thorium required, not in terms of ore grade or other similar quantity. Such effects are included in the environmental impact for the generic fuel resource of the fuel cycle. The information on natural thorium required per unit of energy produced is generated as part of the detailed reactor physics-based calculations for the Analysis Example for each Evaluation Group. The EST used this information as the basis for developing the Metric Data.

Given that the natural thorium required per unit of energy generated is calculated as part of the reactor physics-based analysis of the fuel cycle, which has low uncertainty, the nominal value of fuel usage is appropriate for the evaluation and screening. The calculated data derived for the Analysis Examples of the 40 Evaluation Groups are then used as the basis for binning each evaluation group into a metric bin. An Evaluation Group was placed in a different bin than was indicated by the calculated data for the Analysis Example if assumptions for the associated Analysis Example were considered as not being representative of the fuel cycles within the Evaluation Group. By using bins, it is expected that the Metric Data to be used for evaluating an Evaluation Group relative to the Basis of Comparison is sufficiently representative of the best capabilities of fuel cycles within the group. Details on the *natural thorium required* metric calculation approach, the binning process, and the metric bins for the 40 evaluation groups, are in Appendix D-2.15. The approach for metric binning and the developed bins are also contained in that appendix. Table C-6.2 provides the bins developed for the Natural Thorium Required per Energy Generated metric.

Table C-6.2. Metric Bins for Natural Thorium Required per Energy Generated.

Bin ID	Data Range (t/GWe-yr)	Bin Description
A	< 3.8	Natural thorium mass required < 3.8 t/GWe-yr; includes fuel cycle options with thorium utilization ≥ 30% or uranium-only options
B	3.8 to < 35.0	Natural thorium required mass from 3.8 t/GWe-yr to < 35.0 t/GWe-yr; includes options with thorium utilization ≥ 3% and < 30%
C	35.0 to < 145.0	Natural thorium required mass from 35.0 t/GWe-yr to < 145.0 t/GWe-yr; includes options with thorium utilization ≥ 0.8% and < 3%
D	≥ 145.0	Natural thorium required mass equals or greater than 145.0 t/GWe-yr.

C-7. Development and Deployment Risk Criterion

For the purposes of the fuel cycle Evaluation and Screening, the Development and Deployment Risk Criterion is defined as follows:

Development and Deployment Risk - *A broad definition of development and deployment risk may include all the financial, technical, industrial, and institutional challenges to bringing a system to commercial viability. For the purpose of this fuel cycle evaluation and screening, the assessment of development and deployment risk is focused on the challenge of bringing to maturity and integrating any new technologies required for a fuel cycle, including the time and cost required for successful research development and deployment starting from the current level of technical maturity.*

C-7.1 Background on Development and Deployment Risk

The E&S specifically assumes that the fuel cycle options included in the Evaluation Groups can be successfully developed and deployed given sufficient resources (i.e. all technical issues can be overcome in the development stage), and therefore the risks associated with development and deployment represent those required to execute the necessary activities to complete deployment. There are two important considerations in creating appropriate metrics for assessing the relative development and deployment risks for comparing fuel cycle options: First, the risks of development and deployment are inherently dependent on the technology choices for implementing a fuel cycle as well as on the characteristics the fuel cycle itself; and second, this criterion refers to the challenges of getting from the current levels of maturity to a commercially deployed system, so the assumption used for most of the other criteria (evaluation of a fully deployed fuel cycle at equilibrium) do not apply to this criterion. Almost any fuel cycle other than the ones that have been implemented to date are by definition at a lower level of maturity.

The development and deployment of fuel cycles potentially involves four distinct phases, as shown in Figure C-7.1. The first phase is associated with the research and development required to develop the necessary technology through the engineering prototype stage. This is followed by the deployment phase that results in the deployment of the first-of-a-kind system. The other phases include transitioning from a current technology to the equilibrium fuel cycle deployment. Metrics for each of the first three phases are included in the Development and Deployment Risk Criterion.

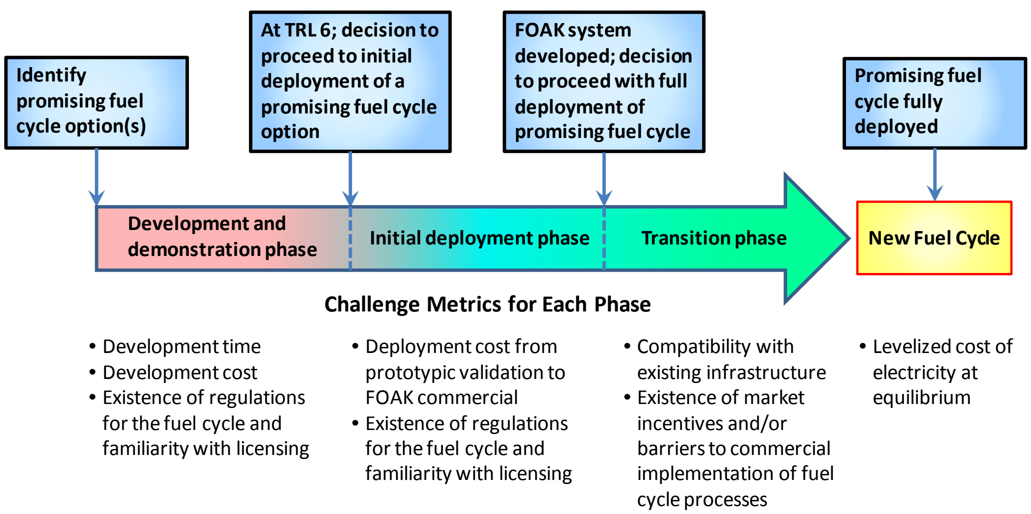


Figure C-7.1. Fuel Cycle Development and Deployment Phases.

Development and Deployment Risk is composed of development risk, deployment risk and institutional issues as shown in Figure C-7.2; each of the lowest level factors in this hierarchy represents individual metrics. The relationships between these metrics and other factors that influence the development and deployment risk are illustrated in the influence diagram presented in Figure C-7.3.

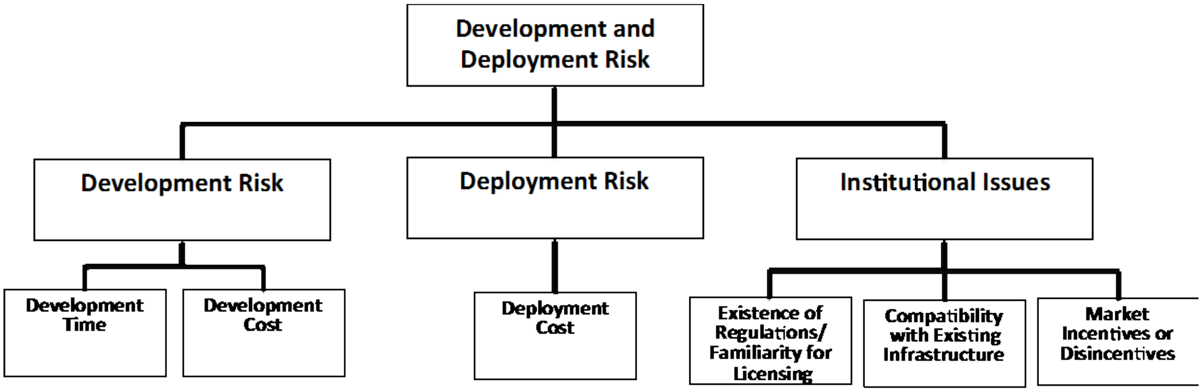
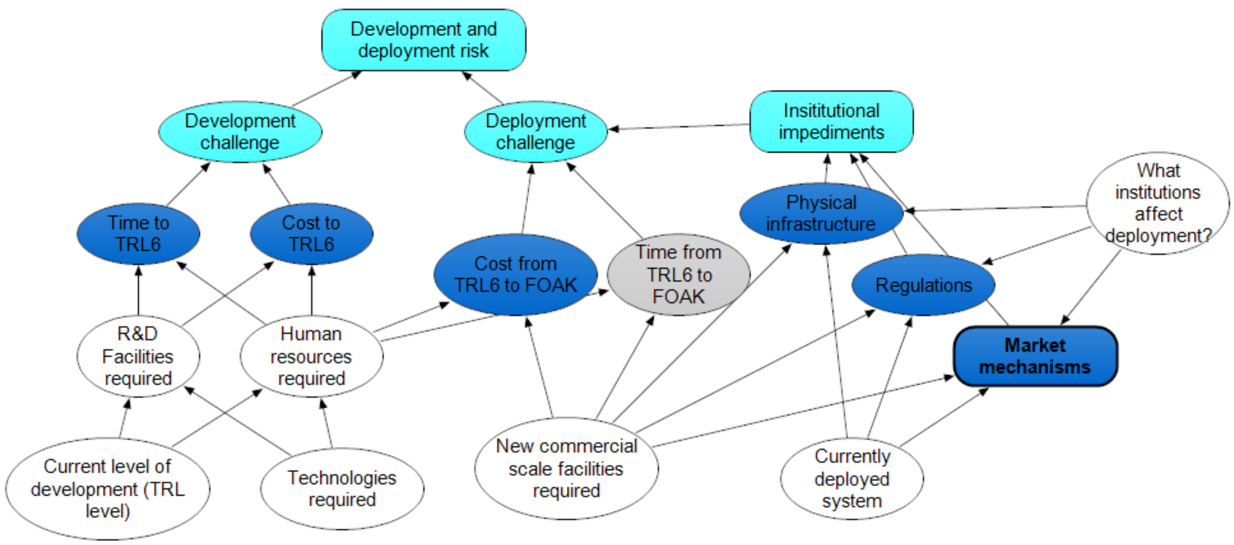


Figure C-7.2. Development and Deployment Risk Relationships.



Note: Each oval represents a factor, element, or question related to the challenges of developing and deploying a fuel cycle. Rounded rectangles represent the two Evaluation Criteria related to these questions. Blue indicates factors for which Evaluation Metrics were defined, white indicates factors related to development and deployment challenges that are considered in development of Metric Data. Grey indicates factors not considered but which are recognized as relevant to the development and deployment challenges.

Figure C-7.3. Influence Diagram for Development and Deployment Risk.

C-7.2 Metric Development for the Development and Deployment Risk Criterion

Differentiating Development and Deployment

While development and deployment risk refers to all the steps and stages necessary to go from the current level of technical maturity to a deployed commercial system, it is useful in considering appropriate metrics to characterize the steps along that path either as "development" or as "deployment" steps. One widely-used measure for assessing or describing the status of a technology is the Technology Readiness Level (TRL). Initially developed by the National Aeronautics and Space Administration (NASA) and adopted by DOE [DOE2011], the TRL scale ranges from 1 (basic principles observed) through 9 (total system used successfully in project operations). Figure C-7.4 provides a schematic of the development and deployment timeline, including TRLs in the context of DOE/EM waste processing projects.[C-7.1] Table C-7.1 describes each TRL.[C-7.2]

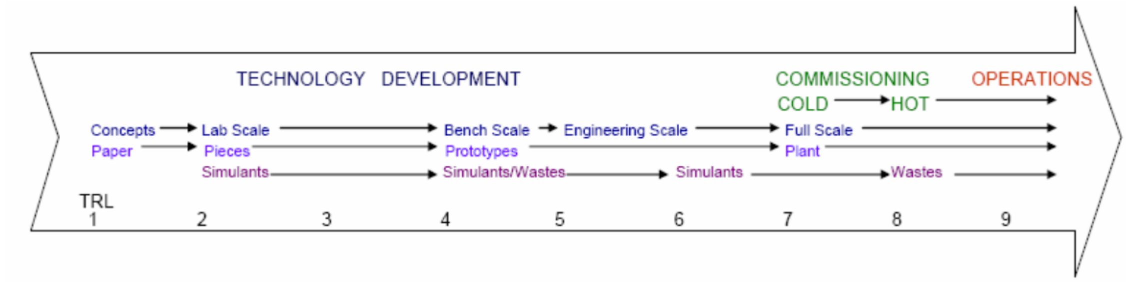


Figure C-7.4. Technology Readiness Levels. [C-7.1]

For purposes of fuel cycle evaluation and screening, "development" is considered all the steps necessary to reach a TRL 6 for all the technologies required to implement a fuel cycle, and for the integration of those technologies. That is, development includes all activities up through developing an integrated, engineering-scale prototype of the fuel cycle. At this point, the government and industry's role in pursuing the necessary R&D would be completed, and subsequent development is related to industrialization and commercialization. "Deployment" is considered to be the steps necessary to go from this integrated engineering-scale prototype to a first-of-a-kind commercial system (FOAK, TRL 9). The deployment stages typically involve significant industry and utility investments, and may or may not involve government investments in the form of tax incentives, credit instruments and direct investment.

Table C-7.1. Technology Readiness Levels. [C-7.2]

Relative Level of Technology Development	Technology Readiness Level	TRL Definition	Description
System Operations	**TRL 9**	Actual system operated over the full range of expected mission conditions.	The technology is in its final form and operated under the full range of operating mission conditions. Examples include using the actual system with the full range of wastes in hot operations.
System Commissioning	**TRL 8**	Actual system completed and qualified through test and demonstration.	The technology has been proven to work in its final form and under expected conditions. In almost all cases, this TRL represents the end of true system development. Examples include developmental testing and evaluation of the system with actual waste in hot commissioning. Supporting information includes operational procedures that are virtually complete. An Operational Readiness Review (ORR) has been successfully completed prior to the start of hot testing.
	TRL 7	Full-scale, similar (prototypical) system demonstrated in relevant environment	This represents a major step up from TRL 6, requiring demonstration of an actual system prototype in a relevant environment. Examples include testing full-scale prototype in the field with a range of simulants in cold commissioning1. Supporting information includes results from the full-scale testing and analysis of the differences between the test environment, and analysis of what the experimental results mean for the eventual operating system/environment. Final design is virtually complete.
Technology Demonstration	**TRL 6**	Engineering / pilot-scale, similar (prototypical) system validation in relevant	Engineering-scale models or prototypes are tested in a relevant environment. This represents a major step up in a technology's demonstrated readiness. Examples include testing an engineering scale prototypical system with a range of simulants.1 Supporting information includes results from the engineering scale testing and analysis of the differences between the engineering scale,

		environment	prototypical system/environment, and analysis of what the experimental results mean for the eventual operating system/environment. TRL 6 begins true engineering development of the technology as an operational system. The major difference between TRL 5 and 6 is the step up from laboratory scale to engineering scale and the determination of scaling factors that will enable design of the operating system. The prototype should be capable of performing all the functions that will be required of the operational system. The operating environment for the testing should closely represent the actual operating environment.
Technology Development	**TRL 5**	Laboratory scale, similar system validation in relevant environment	The basic technological components are integrated so that the system configuration is similar to (matches) the final application in almost all respects. Examples include testing a high-fidelity, laboratory scale system in a simulated environment with a range of simulants1 and actual waste2. Supporting information includes results from the laboratory scale testing, analysis of the differences between the laboratory and eventual operating system or environment, and analysis of what the experimental results mean for the eventual operating system/environment. The major difference between TRL 4 and 5 is the increase in the fidelity of the system and environment to the actual application. The system tested is almost prototypical.
Technology Development	**TRL 4**	Component and/or system validation in laboratory environment	The basic technological components are integrated to establish that the pieces will work together. This is relatively "low fidelity" compared with the eventual system. Examples include integration of ad hoc hardware in a laboratory and testing with a range of simulants and small scale tests on actual waste.[2] Supporting information includes the results of the integrated experiments and estimates of how the experimental components and experimental test results differ from the expected system performance goals. TRL 4-6 represent the bridge from scientific research to engineering. TRL 4 is the first step in determining whether the individual components will work together as a system. The laboratory system will probably be a mix of on hand equipment and a few special purpose components that may require special handling, calibration, or alignment to get them to function.
Research to Prove Feasibility	**TRL 3**	Analytical and experimental critical function and/or characteristic proof of concept	Active research and development (R&D) is initiated. This includes analytical studies and laboratory-scale studies to physically validate the analytical predictions of separate elements of the technology. Examples include components that are not yet integrated or representative tested with simulants.[1] Supporting information includes results of laboratory tests performed to measure parameters of interest and comparison to analytical predictions for critical subsystems. At TRL 3 the work has moved beyond the paper phase to experimental work that verifies that the concept works as expected on simulants. Components of the technology are validated, but there is no attempt to integrate the components into a complete system. Modeling and simulation may be used to complement physical experiments.
	TRL 2	Technology concept and/or application formulated	Once basic principles are observed, practical applications can be invented. Applications are speculative, and there may be no proof or detailed analysis to support the assumptions. Examples are still limited to analytic studies. Supporting information includes publications or other references that outline the application being considered and that provide

			analysis to support the concept. The step up from TRL 1 to TRL 2 moves the ideas from pure to applied research. Most of the work is analytical or paper studies with the emphasis on understanding the science better. Experimental work is designed to corroborate the basic scientific observations made during TRL 1 work.
Basic Technology Research	**TRL 1**	Basic principles observed and reported	This is the lowest level of technology readiness. Scientific research begins to be translated into applied R&D. Examples might include paper studies of a technology's basic properties or experimental work that consists mainly of observations of the physical world. Supporting Information includes published research or other references that identify the principles that underlie the technology.

[1] Simulants should match relevant chemical and physical properties.

[2] Testing with as wide a range of actual waste as practicable and consistent with waste availability, safety, ALARA, cost and project risk is highly desirable.

Development Risk

Focusing on development risk first, the development risk can be summarized as the likelihood that technologies required to implement a fuel cycle would be successfully developed given sufficient time and resources. Given the assumption, discussed above, that every Evaluation Group includes fuel cycle options that can be successfully developed and deployed, given sufficient resources, there is no technical development "risk." There are, however, non-technical risks in development: specifically, development pathways with very long development times or very high costs may be less likely to be pursued to completion. When considering nuclear fuel cycle options and the evaluation groups, the level of R&D required for a particular fuel cycle is a function of the technologies required and the current status of the potential supporting technologies. These factors are shown as nodes in the bottom left portion of the influence diagram in Figure C-7.2. The factors also inform on the level of R&D effort, both time and cost, that would be needed to produce a specific example of the complete fuel cycle demonstrated at the engineering or pilot scale level corresponding to a TRL of 6. The development would therefore represent the R&D needed to bring the fuel cycle technology to a development level at which a determination can be made to pursue commercialization.

Therefore the Development Risk is represented by two metrics:

- **Development time -** Time necessary to develop the required technologies to a level of maturity required for an integrated, engineering-scale prototype (e.g., to TRL6)
- **Development cost -** Costs to develop the required technologies through to an integrated, engineering-scale prototype (e.g., to TRL6)

Since more than one part of the fuel cycle may require technology development, e.g., fuel fabrication technology, separations, reactor, etc., the total development cost for the entire fuel cycle is the sum of the individual development cost estimates, and the development time is the longest development time (and assuming that R&D can be performed in parallel for each of the processes), again accounting for the need for a successful demonstration of the integrated fuel cycle These parts of the fuel cycle consist of primary fuel cycle processes that make up that fuel cycle option. For the purpose of the evaluation and screening a common list of fuel cycle processes was presented in Table C-4.1. Evaluation of the development time and cost is conducted at the process level, and then assessed at the Evaluation Group level based on a mapping for processes to Evaluation Groups.

Deployment Risk

To address issues beyond the R&D phase requires consideration of additional items, such as potential costs, incentives or barriers to initial commercial deployment. Whether the government continues to be involved in these initial deployment stages is an open question, but it is clear that significant industrial and utility involvement will be necessary. While these deployment activities are beyond the R&D stage that is the specific interest for the fuel cycle evaluation and screening, understanding of the relative deployment challenges for differing fuel cycles may be relevant to the overall attractiveness of an R&D investment, and so specific metrics related to deployment risk were developed.

From a technical viewpoint, deployment risk is the risk of being able to successfully deploy the fuel cycle, which would include all of the required facilities and supporting industrial and skills infrastructure, as illustrated by the "new commercial infrastructure required" and "human resources required" items in the influence diagram. The issues center on the risk that the required industrial infrastructure would not be developed, or that the skills needed to support the fuel cycle would not be developed, and the costs to develop those components if the challenges to deployment can be overcome. From a societal or market perspective, deployment risk may include various non-technical incentives or barriers (disincentives) to deployment of the fuel cycle or specific technologies. Such incentives or barriers could be in the form of government policies, laws, financial practices, or commercial practices that favor or hinder deployment of the processes, technologies and facilities needed for a fuel cycle.

Four metrics are used to characterize the deployment risks for a fuel cycle, including three that are also used to characterize Institutional Issues, as discussed in Section C-8. There can be significant costs to take the technologies required for a fuel cycle from the TRL-6 level discussed above to commercial deployment. These costs are highly uncertain and difficult to estimate until specific technologies are identified and they are developed to the TRL 6 level, but they may play an important role in comparing alternative R&D investments, so an Evaluation Metric of "deployment costs" accompanies the development challenge oriented metrics above.

- **Deployment cost from prototypic validation to FOAK commercial -** Estimated cost for the step from TRL-6 to 'first of a kind' commercial system

Institutional Issues

In principle and practice, the remaining three metrics for deployment risk cover the same fundamental concerns as the metrics proposed for the Institutional Issues, which are elaborated in Sections C-7.6 to C-7.8, and repeated in Section C-8. Each of these metrics identifies and was used to estimate the severity of challenges to the commercial deployment of a fuel cycle:

- **Compatibility with existing infrastructure -** The amount of existing versus new technology required (for both development and deployment), including the number of new industrial scale facilities required
- **Existence of regulations for the fuel cycle and familiarity with licensing -** Regulatory experience with each part of the fuel cycle, e.g., reprocessing, reactors, subcritical systems, extended storage, etc., and familiarity with reviewing and licensing such facilities
- **Existence of market incentives and/or barriers to commercial implementation of fuel cycle processes -** Market considerations, such as the existence of market incentives and/or disincentives to commercial implementation of fuel cycle processes.

Approach for Determining Development and Deployment Risk Metric Data

The overall approach for determining the metric data for the Development and Deployment Risk Metrics is based on determining the metric data for fuel cycle processes that make up the Evaluation Groups and then combining the metric data for the relevant fuel cycle processes to obtain the data for the overall Evaluation Group. The manner for combining the data is defined separately from each Evaluation Metric.

The list of fuel cycle processes considered for Development and Deployment Risk is the same as that used for determining the metric data for the Safety Criterion, which is discussed in detail in Section C-4.

C-7.3 Development Cost

Definition of Metric – This metric is the total development cost to bring the fuel cycle technologies required for a particular Evaluation Group from their current state of development to the state required for deployment as an engineering-scale prototype (TRL 6). This cost assumes that the development approach balances time and cost. All research and development is assumed to be funded such that it is neither constrained by too-limited funding (that leads to extended overall development time and potentially increased cost), nor performed on an overly accelerated basis (e.g. a "crash project") that may also require increased cost in trade for a shorter development time.

Development costs for nuclear technologies are known to be highly variable and have significant uncertainty. For the purposes of this evaluation and screening precise estimates were not required, and cost estimated were based on bin definitions that represent an increasing level of development required. Each bin is defined by a cost range that may be indicative of the investment in R&D facilities in addition to the R&D effort. The bin structure for development cost is provided in Table C-7.2.

Table C-7.2. Development Cost Bin Description.

Bin	Bin Descriptions for Development Cost
Bin A: No Development Needed (Already at TRL6 or beyond)	No R&D needed. Technology is already at TRL 6 or beyond. Development cost is $0.
Bin B: Development costs of < $200M	R&D required but is limited in scope and can be supported with existing facilities with little or no modifications. Development costs expected to be less than $200 million
Bin C: Development cost of $200M- $2B	R&D required, but can primarily be performed without significant investment in new major nuclear facilities for engineering/pilot scale demonstration. May, for example, just require modification of existing facilities. Development costs expected to be between $200 million and $2 billion.
Bin D: Development cost of $2B - $10B	R&D required including construction of a major nuclear facility to provide an engineering/pilot scale demonstration of one component of a fuel cycle. Development costs expected to be between $2 billion and $10 billion
Bin E: Development cost of $10B - $25B	Significant R&D required including construction of several nuclear facilities to provide an engineering/pilot scale demonstration of several components of a fuel cycle. Alternatively, the scale of the facilities required for engineering/pilot scale demonstration is large and results in significantly increased cost. Development costs expected to be between $10 billion and $25 billion
Bin F: Development cost of >$25B	Very significant R&D required including construction of many new facilities to provide an engineering/pilot scale demonstration of several components of the fuel cycle. May require more than one scale of facility development for particular fuel cycle components. Development cost expected to be greater than $25 billion

The bin structure considers that some technologies are already deployed at beyond an engineering prototype and therefore have a zero development cost. The cost increases with the amount of R&D required and with the development that requires the establishment of new R&D facilities. The overall development cost for an evaluation group is the sum of the development costs for the fuel cycle processes.

Development cost metric data for each fuel cycle process was developed based on judgment by the EST and input received through discussions with industry and experts. The process data is presented in Table

C-7.4 along with data for the Development Time Metric. This data is used to develop the metric data presented in Appendix D.

C-7.4 Development Time

Definition of Metric – This metric is the total development time associated with a fuel cycle option to bring the associated fuel cycle technology from their current state of development to the state required for deployment as an engineering-scale prototype (TRL 6). The same assumptions about the development approach described as the basis for the development cost estimates apply to the development of time estimates. Development time and development cost are related as longer development times are generally associated with larger development costs. These two metrics are estimated separately to provide data to inform on potential R&D, they require the same underlying assumptions and are considered together when informing on the Development and Deployment Risk Criterion.

Also similar to the Development Cost metric, development time for nuclear fuel cycle R&D is inherently uncertain. A set of bins have been chosen to represent this uncertainty while maintaining the ability to inform on any development time differences between Evaluation Groups. The bin structure has been developed based on ranges in development time from no development needed, nearer-term development, longer-term development, and much-longer term development that represents both the state of the current technology maturity as well as the need to build R&D facilities and perhaps involving several different scales of R&D facilities before achieving an engineering-scale prototype. The development time for the fuel cycle option is the longest of the development times for the individual fuel cycle processes, based on an assumption that R&D can be performed in parallel for each of the processes. The Development Time metric bin structure is provided in Table C-7.3.

Table C-7.3. Development Time Bin Description.

Bin	Bin Descriptions for Development Time
Bin A: No Development Needed (Already at TRL6 or beyond)	No R&D needed. Technology is already at TRL 6 or beyond. Development time is 0 years.
Bin B: < 5 Years of development needed	R&D required, but most of the required capabilities already demonstrated and any additional R&D is limited in scope and can be supported with existing facilities with little or no modifications. Estimated development time is less than 5 years
Bin C: 5 – 10 years of development needed	R&D required, but many of the required capabilities are either already demonstrated or nearly demonstrated. Additional engineering/pilot scale in the near term likely using existing facilities or based on historical experience. Example may be the qualification of a well-established fuel. Estimated development time is 5-10 years
Bin D: 10 – 25 years of development needed	R&D required that requires extended development time to arrive at a workable capability and demonstration at the engineering/pilot scale. Estimated development time is 10-25 years
Bin E: 25 – 50 years of development needed	Significant R&D required that may require more fundamental development at laboratory scale before developing capabilities that can be demonstrated at engineering/pilot scale. Estimated development time is 25-50 years
Bin F: > 50 years of development needed	Very significant R&D required that may require significant technical breakthroughs, new discoveries or extended research, development of long-lead time laboratory experiments before engineering/pilot demonstration. Estimated development time is greater than 50 years

Fuel cycle process data was developed by estimating the development time bin for each process based on judgment and input received through discussions with industry and experts. Since this development time is closely related to development cost, the development time process data is presented along with the Development Cost data in Table C-7.4. This data is used to develop the metric data presented in Appendix D.

Table C-7.4. Development Cost and Development Time Fuel Cycle Process Data.

Process ID	Process	Development Cost Bin	Development Time Bin	Bin Justification
FS-1	Fuel supply - Mined uranium	A. No Development Needed	A. No Development Needed	Uranium mining is fully commercially deployed and therefore exceeds a TRL 6 level of development.
FS-2	Fuel supply - Mined thorium	A. No Development Needed	A. No Development Needed	Although thorium is not mined on a large scale today for direct nuclear fuel cycle use, it is mined as a by-product of other industrial processes including from uranium and more especially rare-earth mining. The thorium recovery and refining process involves a chemical extraction processes that developed beyond a TRL 6 level of development.
UE-1	Uranium enrichment < 5 wt. %	A. No Development Needed	A. No Development Needed	Enrichment to 5 wt. % is fully commercially deployed and therefore exceeds a TRL 6 level of development.
UE-2	Uranium enrichment >5 wt. %	A. No Development Needed	A. No Development Needed	Enrichment technology for >5 wt. % is the same as for < 5 wt% and therefore exceeds a TRL 6 level of development.
FF-1	Fuel fabrication with unirradiated uranium (Contact handled)	A. No Development Needed	A. No Development Needed	Fabrication of fuel with unirradiated uranium is fully commercially deployed and therefore exceeds a TRL 6 level of development.
FF-2	Fuel fabrication with unirradiated thorium or uranium/thorium (Contact handled)	B. < $200M	B. < 5 years	Fabrication of thorium fuels have been performed in the past in support of R&D programs and may require a low level of development to achieve a TRL 6 level of development to support deployment to support deployment in the U.S.
FF-3	Recycle fuel fabrication with RU/Pu (Glove Box handled)	B. < $200M	C. 5 – 10 years	Fabrication of fuels with Pu (such as MOX) is fully commercially deployed in Europe and Japan. A low level of development may be needed to support deployment in the U.S.
FF-4	Recycle fuel fabrication with RU/TRU (Remote handled)	D. $2B - $10B	D. 10 – 25 years	Development of RU/TRU fuels to a TRL 6 (engineering prototype) level of development requires investment in engineering-scale facilities that allow remote fabrication to produce fuels for irradiation testing.
FF-5	Recycle fuel fabrication with U3/Th/TRU (Remote handled)	D. $2B - $10B	D. $2B - $10B	Development of U3/Th/TRU fuels to a TRL 6 (engineering prototype) level of development requires investment in engineering-scale facilities that allow remote fabrication to produce fuels for irradiation testing.
RX-0	Reactor: Thermal-critical (no development required)	A. No Development Needed	A. No Development Needed	This process represents thermal reactors, such as LWRs, that require no additional development to achieve a TRL 6 level of development.

RX-1	Reactor: Thermal-critical (fuel development required)	B. < $200M	B. < 5 years	This process represents thermal reactors, such as LWRs, that require only fuels development (such as higher burnup fuels) to achieve a TRL 6 level of development.
RX-2	Reactor: Thermal-critical (all other thermal reactors)	D. $2B - $10B	D. 10 – 25 years	This process represents the development of a thermal reactor, such as HTGR, that requires building an engineering scale prototype to achieve a TRL 6 level of development.
RX-3	Reactor: Fast-critical	D. $2B - $10B	D. 10 – 25 years	This process represents the development of a fast reactor, such as an LMR, that requires building an engineering scale prototype to achieve a TRL 6 level of development.
RX-4	Reactor: Sub-critical	E. $10B - $25B	D. 10 – 25 years	This process represents the development of a sub-critical reactor (EDS), such as an ADS or FFH, that requires building an engineering scale prototype to achieve a TRL 6 level of development. For EDS' this involves developing and building the neutron source as well as developing and building the fission blanket system.
RP-1	Reprocessing with RU/Pu product	C. $200M - $2B	C. 5 – 10 years	There are currently deployed industrial scale facilities that perform reprocessing of standard fuels, however, process improvements (such as off-gas capture and treatment) will need to be demonstrated before deployment. Reprocessing of advanced fuels require maturing of technology before deployment.
RP-2	Reprocessing with RU/TRU product	D. $2B - $10B	D. 10 – 25 years	Development of additional separations approaches for RU/TRU products can be currently performed at laboratory scale, but development at engineering scale requires additional R&D facility development.
RP-3	Reprocessing with U3/Th/TRU products	D. $2B - $10B	D. 10 – 25 years	Development of additional separations approaches for U3/Th/TRU products can be currently performed at laboratory scale, but development at engineering scale requires additional R&D facility development
ST-1	Storage of fuel cycle materials	A. No Development Needed	A. No Development Needed	Fuel cycle materials storage has been deployed and no development is required.
TR-1	Transport of fuel cycle materials	A. No Development Needed	A. No Development Needed	Fuel cycle materials transport has been deployed and no development is required.
DS-1	Management and packaging of DU, RU, RTh	A. No Development Needed	A. No Development Needed	Management and Packaging of DU, RU, and RTh requires no development before deployment.
DS-2	Management and packaging of Discharged Fuel	A. No Development Needed	A. No Development Needed	Management and Packaging of discharged fuel requires no development before deployment.
DS-3	Preparation and packaging of High Level Waste	B. < $200M	B. < 5 years	High level waste preparation and packaging is currently performed commercially (vitrified waste) and requires minimal development to support process improvements for commercial deployment.

C-7.5 Deployment Cost from Prototypic Validation to FOAK Commercial

Definition of Metric – This metric represents the additional costs required to take a fuel cycle or fuel cycle process from an engineering-scale prototype (TRL 6) and deployment as a first of a kind commercial facility

Again, the deployment cost for nuclear technologies is known to be highly variable and have significant uncertainty and therefore, the cost estimate was based on bin definitions. Several options were considering for estimating deployment costs, including the use of "rules of thumb" based on industry experience in the deployment of fuel cycle facilities that may relate FOAK and "nth-of-a-kind" cost, which may be more readily available. Industry feedback indicated that based on the best available information, only rough deployment costs can be specified for fuel cycle facilities. Further, the costs at the fuel cycle process level, as defined in Table C-4.1, can be roughly estimated based on the cost of the facility for that process without specifying the technology details.

Similar to the approach for development cost, a set of bins were developed that can be used to represent the deployment costs for fuel cycle options based on the sum of the deployment costs for the individual fuel cycle processes that make up that fuel cycle option. The definition of the bins is loosely based on the state of deployment for a fuel cycle process and the investment in facilities required to achieve a FOAK commercial deployment. The deployment cost bins are as shown in Table C-7.5.

Table C-7.5. Deployment Cost Bin Description.

Bin ID	Data Range	Bin Descriptions for Deployment Cost
A	Bin A: Previously deployed as FOAK or beyond	Technology already has a FOAK (or beyond) deployment. FOAK deployment cost is $0.
B	Bin B: < $2B to deploy FOAK	Deployment of FOAK may, for example, require small-scale nuclear facility or modifications to existing nuclear facility. Estimated cost to go from an engineering/pilot scale system to FOAK is less than $2 billion
C	Bin C: $2B - $10B to deploy FOAK	Deployment of FOAK represents a single nuclear facility or a few small-scale nuclear facilities. Estimated costs to go from an engineering/pilot scale system to FOAK is between $2 billion and $10 billion
D	Bin D: $10B - $25B to deploy FOAK	Deployment of FOAK represents a single large-scale nuclear facility or several medium-scale nuclear facilities. Estimated cost to go from an engineering/pilot scale system to FOAK is between $10 billion and $25 billion
E	Bin E: $25B - $50B to deploy FOAK	Deployment of a first-of-a-kind (FOAK) commercial system represents a significant effort to move from engineering/pilot scale to first-of-a-kind commercial system (FOAK) that may include several large-scale nuclear facilities. Estimated cost to go from an engineering/pilot scale system to FOAK is between $25 billion and $50 billion
F	Bin F: >$50B to deploy FOAK	Deployment of a first-of-a-kind (FOAK) commercial system represents a very significant effort to move from engineering/pilot scale to FOAK system that may include several large-scale nuclear facilities that require several stages of deployment representing additional scales of facilities needed to achieve FOAK. Estimated cost to go from an engineering/pilot scale system to FOAK is over $50 billion

Fuel cycle process data was developed by estimating the deployment cost bin for each process based on judgment and input received through discussions with industry and experts. The process data is presented in Table C-7.6. This data is used to develop the metric data presented in Appendix D.

Table C-7.6. Deployment Cost Fuel Cycle Process Data.

Process ID	**Process**	**Deployment Cost Bin**	**Bin Justification**
FS-1	Fuel supply - Mined uranium	A. Previously deployed as FOAK or beyond	Uranium mining is fully commercially deployed and has previously been deployed as a FOAK process.
FS-2	Fuel supply - Mined thorium	A. Previously deployed as FOAK or beyond	Although thorium is not mined on a large scale today for direct nuclear fuel cycle use, it is mined as a by-product of other industrial processes including from uranium and more especially rare-earth mining. Therefore extraction of thorium has previously been deployed at a FOAK process.
UE-1	Uranium enrichment < 5 wt. %	A. Previously deployed as FOAK or beyond	Enrichment to 5 wt. % is fully commercially deployed and has previously been deployed as a FOAK facility.
UE-2	Uranium enrichment >5 wt. %	A. Previously deployed as FOAK or beyond	Enrichment technology for >5 wt. % is the same as for < 5 wt% and therefore is considered as previously been deployed as FOAK.
FF-1	Fuel fabrication with unirradiated uranium (Contact handled)	A. Previously deployed as FOAK or beyond	Fabrication of fuel with unirradiated uranium is fully commercially and has previously been deployed as FOAK.
FF-2	Fuel fabrication with unirradiated thorium or uranium/thorium (Contact handled)	B. < $2B	Industry-scale thorium fuel fabrication plan has not been previously deployed as a FOAK and therefore deployment of the FOAK facility is required.
FF-3	Recycle fuel fabrication with RU/Pu (Glove Box handled)	A. Previously deployed as FOAK or beyond	Fabrication of fuels with Pu (such as MOX) is fully commercially deployed in Europe and Japan and therefore has previously been deployed as a FOAK facility.
FF-4	Recycle fuel fabrication with RU/TRU (Remote handled)	C. $2B - $10B	Industry-scale RU/TRU fuel fabrication plan has not been previously deployed as a FOAK and therefore deployment of the FOAK facility is required.
FF-5	Recycle fuel fabrication with U3/Th/TRU (Remote handled)	C. $2B - $10B	Industry-scale U3/Th/TRU fuel fabrication plan has not been previously deployed as a FOAK and therefore deployment of the FOAK facility is required.
RX-0	Reactor: Thermal-critical (no development required)	A. Previously deployed as FOAK or beyond	This process represents thermal reactors, such as LWRs, that have been previously deployed and therefore a FOAK facility is not required.
RX-1	Reactor: Thermal-critical (fuel development required)	B. < $2B	This process represents thermal reactors, such as LWRs, that require only fuels development (such as higher burnup fuels). Deployment cost represents qualification of fuel.
RX-2	Reactor: Thermal-critical (all other thermal reactors)	C. $2B - $10B	This process represents the development of a thermal reactor, such as HTGR, that requires building FOAK facility.
RX-3	Reactor: Fast-critical	C. $2B - $10B	This process represents the development of a thermal reactor, such as HTGR, that requires building FOAK facility.

RX-4	Reactor: Sub-critical	D. $10B - $25B	This process represents the deployment of a sub-critical reactor (EDS), such as an ADS or FFH, that requires building of a FOAK facility that involves deployment of both the neutron source and the fission blanket, exceeding the cost of a critical reactor alone.
RP-1	Reprocessing with RU/Pu product	D. $10B - $25B	There are currently deployed industrial scale facilities that perform reprocessing of standard fuels in Europe and Japan, however, the deployment of a commercial reprocessing plant in the U.S. will be similar to the deployment of a FOAK facility.
RP-2	Reprocessing with RU/TRU product	D. $10B - $25B	There have been no commercially deployed reprocessing faculties with RU/TRU products and therefore a FOAK facility is required.
RP-3	Reprocessing with U3/Th/TRU products	D. $10B - $25B	There have been no commercially deployed reprocessing faculties with RU/TRU products and therefore a FOAK facility is required.
ST-1	Storage of fuel cycle materials	A. Previously deployed as FOAK or beyond	Fuel cycle materials storage is an established process.
TR-1	Transport of fuel cycle materials	A. Previously deployed as FOAK or beyond	Fuel cycle materials transport is an established process.
DS-1	Management and packaging of DU, RU, RTh	A. Previously deployed as FOAK or beyond	Management and Packaging of DU, RU, and RTh is an established process.
DS-2	Management and packaging of Discharged Fuel	A. Previously deployed as FOAK or beyond	Management and Packaging discharged fuel is an established process.
DS-3	Preparation and packaging of High Level Waste	A. Previously deployed as FOAK or beyond	High level waste preparation and packaging is an established process.

C-7.6 Compatibility with the Existing Infrastructure

Definition of Metric – This metric represents the degree to which a fuel cycle option can utilize existing infrastructure in terms of fuel cycle facility types and the knowledge and expertise to construct and operate them versus the need to develop new infrastructure representing technologies for which little experience exists. More compatibility with existing infrastructure represents a lower deployment risk and lower institutional barrier to deployment.

Compatibility with existing infrastructure can be estimated based on the degree to which the fuel cycle processes for a fuel cycle option overlap with those of the Basis of Comparison, representing existing infrastructure. The primary fuel cycle processes that represent the largest overall investment in infrastructure are the reactors, fuel fabrication facilities, and reprocessing facilities. Therefore, compatibility with existing infrastructure is determined by comparing these processes required for an Evaluation Group to those used in the Basis of Comparison. The fraction of the facilities that represent existing facilities can be determined and the fuel cycle option is placed in a bin based on this fraction. The compatibility with existing infrastructure bins are provided in Table C-7.7.

Table C-7.7. Compatibility with Existing Infrastructure Bin Description.

Bin	Bin Descriptions for Compatibility with Existing Infrastructure
A. Requires Nearly No New Infrastructure	The fuel cycle option fully utilizes the existing infrastructure as represented by the fuel cycle Basis of Comparison, needing very little additional infrastructure to deploy the fuel cycle. Estimate is that 90% or more of the required infrastructure can be based on existing infrastructure
B. Requires Some New Infrastructure	The fuel cycle option utilizes mostly components of the existing infrastructure and may require some additional infrastructure components. Estimate is that more than 50% (but less than 90%) of the required infrastructure can be based on existing infrastructure
C. Requires Mostly New Infrastructure	The fuel cycle option utilizes some components of the existing infrastructure and requires mostly additional infrastructure components. Estimate is that less than 50% (but more than 10%) of the required infrastructure will be based on existing infrastructure
D. Requires Almost Entirely New Infrastructure	The fuel cycle option utilizes few or none of the components of the existing infrastructure and requires mostly new infrastructure components for deployment. Estimate is that less than 10% of the required infrastructure can be based on existing infrastructure

For the purposes of developing metric data, the existing infrastructure was represented by the fuel cycle process for Evaluation Group EG01 (Basis of Comparison) and existing and new infrastructure was determined by the common fuel cycle processes for each Evaluation Group.

C-7.7 Existence of Regulations for the Fuel Cycle and Familiarity with Licensing

Definition of Metric – This metric represents the regulatory maturity for a fuel cycle option based on a determination of whether regulations exist for the fuel cycle and an estimate of what level of experience regulatory organizations have in applying those regulations.

The lowest institutional barriers and lowest deployment risk are for fuel cycles comprised of processes for which U.S. regulations exist and have been demonstrated through the issuance of licenses that have resulted in operating facilities. These regulations would be clearly identifiable by considering the history of the licensing of fuel cycle facilities and are primarily associated with the currently deployed fuel cycle facilities. Lack of regulations for new fuel cycle facilities or technologies represents a significant risk in terms of timely resolution of the licensing process, and lesser familiarity by the regulatory authorities increases the risk further since the regulator will have to build sufficient knowledge and experience to not only develop appropriate regulations but to assess the proposed facilities as well. The bin structure in Table C-7.8 defines several levels of compatibility, each of which poses an increase in the deployment challenge for a fuel cycle.

Table C-7.8. Existence of Regulations and Familiarity with Licensing Bins.

Bin	Bin Descriptions for Existence of Regulations and Familiarity with Licensing
A. Demonstrated U.S. Regulations/Familiarity	U.S. Regulations and regulatory experience exists for fuel cycle facility types that have been demonstrated through issuing operating licenses
B. Limited U.S. Regulations/Familiarity	U.S. Regulations and regulatory experience exists for fuel cycle facility types but have not been demonstrated through issuing operating licenses. Regulatory authorities have some previous experience with key fuel cycle components, but may not have licensed these facility types
C. No U.S. Regulations/Familiarity	No U.S. regulatory experience for fuel cycle facility types and use, but international regulatory experience exists through licensing of operating facilities
D. No Regulations/Familiarity	No regulatory experience exists for key fuel cycle facility types and use

Similar to the approach for the Development Cost and Development Time Metrics, the data was estimated for each fuel cycle process and combined for the evaluation group. The data for the fuel cycle processes is provided in Table C-7.9. The metric data for each evaluation group was the highest bin of all fuel cycle processes that make up that evaluation group.

Table C-7.9. Existence of Regulations for the Fuel Cycle and Familiarity with Licensing Fuel Cycle Process Data.

Process ID	Process	Deployment Cost Bin	Bin Justification
FS-1	Fuel supply - Mined uranium	A. Demonstrated U.S. Regulations/Familiarity	Uranium mining has established regulations and have been licensed.
FS-2	Fuel supply - Mined thorium	B. Limited U.S. Regulations/Familiarity	Regulatory experience exists for rare-earth mining with thorium product has established regulatory infrastructure. Regulations exist, but experience is limited.
UE-1	Uranium enrichment < 5 wt. %	A. Demonstrated U.S. Regulations/Familiarity	Enrichment to 5 wt. % has extensive past and current regulatory experience and currently operating licensed facilities.
UE-2	Uranium enrichment >5 wt. %	A. Demonstrated U.S. Regulations/Familiarity	Enrichment to >5 wt. % has been performed in the past and established regulations can be applied.
FF-1	Fuel fabrication with unirradiated uranium (Contact handled)	A. Demonstrated U.S. Regulations/Familiarity	Fabrication of fuel with unirradiated uranium is currently licensed.
FF-2	Fuel fabrication with unirradiated thorium or uranium/thorium (Contact handled)	D. No Regulations/Familiarity	There has been no experience with the regulation and licensing of thorium fuel fabrication facilities.
FF-3	Recycle fuel fabrication with RU/Pu (Glove Box handled)	B. Limited U.S. Regulations/Familiarity	There has been limited regulatory experience in licensing fuel fabrication with RU/Pu. Current experience is with the MOX fuel fabrication facility, which is yet to be licensed.
FF-4	Recycle fuel fabrication with RU/TRU (Remote handled)	D. No Regulations/Familiarity	There is currently no regulatory experience in licensing of RU/TRU fuel fabrication facilities.
FF-5	Recycle fuel fabrication with U3/Th/TRU (Remote handled)	D. No Regulations/Familiarity	There is currently no regulatory experience in licensing of U3/Th/TRU fuel fabrication facilities.

RX-0	Reactor: Thermal-critical (no development required)	A. Demonstrated U.S. Regulations/Familiarity	Thermal reactors that have been fully developed (e.g. LWRs) have full regulations that have been demonstrated through licensing of operating facilities.
RX-1	Reactor: Thermal-critical (fuel development required)	A. Demonstrated U.S. Regulations/Familiarity	The regulations for operating reactors can be applied to licensing of new fuels.
RX-2	Reactor: Thermal-critical (all other thermal reactors)	D. No Regulations/Familiarity	Advanced thermal critical reactors have operated commercially (e.g., Peach Bottom Unit 1, Fort St. Vrain, and thorium fuel cycles in Shippingport and Indian Point Unit 1) but were licensed by exception to the current LWR regulations. Overall US regulatory experience is limited with advanced thermal critical reactors.
RX-3	Reactor: Fast-critical	C. No U.S. Regulations/Familiarity	Few fast reactors have been operated in the US, and most of these were operated as AEC/DOE research reactors. Some past US regulatory experience exists (e.g., Fermi-1) but it is limited and current regulatory experience is lacking. Fast reactors have been commercially licensed in France and in Russia, which provides some non-U.S. regulatory experience.
RX-4	Reactor: Sub-critical	D. No Regulations/Familiarity	Research and some development has been performed in the US and internationally, but no regulatory experience exists for externally-driven subcritical nuclear systems.
RP-1	Reprocessing with RU/Pu product	C. No U.S. Regulations/Familiarity	While the processes used for RU/Pu reprocessing are well-known, there is limited past regulatory experience in the US with it. Substantial international regulatory experiences exist for reprocessing Pu. Some guidance exists for regulating reprocessing facilities in the US but substantial work would be needed.
RP-2	Reprocessing with RU/TRU product	D. No Regulations/Familiarity	There exists very little or no US regulatory experience for RU/TRU reprocessing. Substantial international regulatory experiences exist for reprocessing Pu but not RU/TRU. Some guidance exists for regulating reprocessing facilities in the US but substantial work would be needed.
RP-3	Reprocessing with U3/Th/TRU products	D. No Regulations/Familiarity	Limited past US regulatory experience for U3/TRU/U/Th exists, mostly based upon work performed at West Valley for thorium fuel reprocessing, and some international regulatory experience exists for reprocessing thorium fuels; however, the processes and facilities used in proposed fuel cycles would likely differ substantially from historical experience.
ST-1	Storage of fuel cycle materials	A. Demonstrated U.S. Regulations/Familiarity	The storage of nuclear materials is an existing commercial/industrial operation with extensive regulatory experience. Some modifications may be needed to govern changes introduced by new technologies or materials.
TR-1	Transport of fuel cycle materials	B. Limited U.S. Regulations/Familiarity	The transportation of nuclear materials is an existing commercial/industrial operation with extensive regulatory experience. Some modifications may be needed to govern changes introduced by new technologies or materials.
DS-1	Management and packaging of DU, RU, RTh	B. Limited U.S. Regulations/Familiarity	Extensive regulatory experience exists for technologies used for management and packaging of DU, which can also be applied to RU and RTh.
DS-2	Management and packaging of Discharged Fuel	B. Limited U.S. Regulations/Familiarity	Extensive regulatory experience exists for technologies used for the management and packaging of discharged

			fuel. however, technologies in some fuel cycles will require significant new regulatory work and actual operations even using existing technologies may require some additional regulatory work.
DS-3	Preparation and packaging of High Level Waste	B. Limited U.S. Regulations/Familiarity	High level waste preparation and packaging is an established process. Other waste forms for alternate reprocessing approaches would require additional regulatory work.

C-7.8 Existence of Market Incentives and/or Barriers to Commercial Implementation of Fuel Cycle Processes

C-7.8.1 Background

The EST sought stakeholder feedback as well as industry input regarding the risks associated with deploying fuel cycles. Based on this feedback, it became apparent that the existence of incentives or disincentives for commercial implementation of a fuel cycle represented a relevant "institutional issue" that may contribute to the development and deployment risk associated with a particular fuel cycle option, and ultimately may help differentiate among alternative fuel cycles. As described more fully below, incentives and disincentives for commercial implementation include investment return considerations, the existence of functioning markets and adequate price signals, and relevant factors driving demand. These considerations, broadly termed "Market Incentives and/or Barriers to Commercial Implementation" or the "Market Metric", seek to address the market and commercial challenges that may be confronted as a fuel cycle is introduced into existing industrial infrastructure and market systems.

C-7.8.2 Approach to Metric Evaluation

A Market Metric Working Group consisting of a subset of EST members and individuals with experience in economic analysis and utility finance (hereafter referred to as the "Market Metric Working Group") focused on determining how to develop a metric to address the incentives and/or barriers to commercial implementation of a fuel cycle. The steps followed by the Market Metric Working Group are described briefly below.

A. **Identification of Factors that May Affect Fuel Cycle Deployment:** This step consisted of identifying market-related factors that might create incentives or barriers to commercial implementation. By design, this process was intended to be creative and the goal was to identify factors for consideration as part of the metric.
B. **Determine Relationship Between Factors and Fuel Cycles:** The next step identified the relationships between factors and the relevance of these factors to fuel cycles. Throughout this step, the Market Metric Working Group was challenged by the need to avoid technology-specific considerations and implementation choices.
C. **Review and Test Factors Against a Subset of Fuel Cycle Options:** In order to inform the process, the Market Metric Working Group tested the identified factors by applying them to several example fuel cycles. This process identified several challenges, particularly given the need to avoid consideration of implementation choices and technologies.
D. **Simplify Structure and Number of Factors:** Based on the practical application of identified factors and the interrelationships among factors, the Market Metric Working Group simplified the evaluation approach by combining factors under two "umbrella" market factors. The combined structure simplified the analytical requirements and reduced the examination of multiple, highly correlated factors.
E. **Develop Scales:** Based on consideration of the "umbrella" market factors, a descriptive set of bins were developed for each factor to facilitate the development of metric data for market incentives and disincentives.

F. **Establish Analytical Parameters and Guidelines (Rules of Thumb):** An important step in the process for developing metric data included establishing limitations to the analysis and guidelines and "rules of thumb" to guide the Market Metric Working Group's efforts. Since the evaluation largely relied on informed qualitative judgments, the establishment of analytical parameters and guidelines facilitated consistency in the review of evaluation groups.
G. **Conduct Process-Level Analysis and Apply Results Against Evaluation Groups:** Similarly to how metric data were developed for several of the other Evaluation Metrics within Development and Deployment Risk criterion, the processes that comprise an Evaluation Group formed the basis for understanding the market incentives and barriers of each evaluation group. Accordingly, the Market Metric Working Group identified market incentives and barriers associated with each process. The evaluation groups were then assessed by "rolling up" the results for each process within the evaluation group.
H. **Document and Summarize Results:** The process for developing the market metric and the results of the analysis are summarized in this appendix and Appendix D.

C-7.8.3 Identification of Factors that May Affect Fuel Cycle Deployment

Market considerations, such as the existence of market incentives and/or barriers to commercial implementation of fuel cycle processes, relate to the forces influencing availability of capital and competition in the marketplace and may include consideration of several factors associated with each fuel cycle. The Market Metric Working Group identified several market-related considerations that might create incentives or disincentives to commercial implementation of fuel cycles. The identified factors exhibit interrelationships and interdependencies but can be broadly classified into the following two factors (the "umbrella factors" described in Step D above):

- Capital at Risk; and
- Market Incentives and Drivers.

The influence diagram shown in Figure C-7.2 illustrates many of the factors and discusses their relationships both to the two umbrella factors and to each other.

A. Capital at Risk

The capital at risk "umbrella" factor recognizes the considerations investors may have in deploying capital to develop new facilities: the magnitude of the investment and the timing for the return of and on capital deployed. For the deployment of complex new fuel cycle technologies, investors will require substantial risk premiums on invested capital. This will affect the economics of deploying fuel cycles, and in this context, higher capital at risk represents a market disincentive to commercial implementation of a fuel cycle. For example, the introduction of a new fuel cycle that requires a series of facilities to handle processes that do not currently exist would require the investment of more upfront capital and potentially subject investors to greater uncertainty regarding financial returns as well as higher risk premiums for borrowed capital. This uncertainty will be heightened with the perceived risks (e.g., technology, market, etc.) associated with such investments. In examining capital at risk, key inputs are the number and types of facilities required, the cost of each facility, the benefits related to past investments in the fuel cycle (e.g., mining investments), and the risk profile associated with investments in the fuel cycle.

To inform on this factor, the Market Metric Working Group identified a number of investment considerations, which when taken together supported the assignment of evaluation groups into one of five qualitative bins. These investment considerations and their supporting rationale are detailed below.

1. Capital Investment

Capital investment relates to the amount of capital that would be required for implementing the fuel cycle. A key question is whether investment of substantial new capital (beyond the cost of replacing existing

/aging facilities) would be required to implement the fuel cycle. An example for capital investment would be a fuel cycle that requires reactors that utilize new technologies. This would introduce significant capital investment to support generation needs and corresponding supply chain investments.

2. Payback Period

Payback period addresses the timing of the return of and on capital invested. For large-scale infrastructure facilities, the Market Metric Working Group assumed that a payback period in excess of 20 years would be considered a disincentive to commercial implementation. An example of an investment with long payback prospects is a reprocessing facility designed to serve multiple reactors, where the reprocessing facility would be built first and the reactors would come on line over time.

3. Scaling/Penetration

Scaling addresses the need to build facilities that implement the fuel cycle at an optimal scale in order to reach full deployment. For example, mining infrastructure, fuel fabrication facilities, reprocessing facilities, storage and disposal would likely need to be designed to service multiple generation facilities and such investments would likely need to be made in advance of demand. The key question is whether a fuel cycle under consideration would require development of one or more facilities in advance of demand. Such a circumstance represents a barrier that would have to be overcome in deployment. Therefore, to inform on the market metric, the Market Metric Working Group considered the plausible deployment scenarios of fuel cycle processes.

4. Existing Infrastructure

For some fuel cycle processes, industrial and physical infrastructure may exist that provides benefits to a given fuel cycle option. For example, an existing fuel fabrication supply chain may be utilized for fuel cycle alternatives. Therefore, the Market Metric Working Group considered whether a fuel cycle would benefit from existing facilities or infrastructure. Information gathered to support the assessment of the "Compatibility of Existing Infrastructure" metric was relevant here.

5. Technical Complexity

Fuel cycle options deployed at commercial scale would vary in complexity, linkages and integration. Complex and tightly integrated systems tend to present greater risk since subsystems must perform flawlessly to avoid diminished performance. Investors and other financial stakeholders would consider such technical uncertainties, and perhaps invest more capital to mitigate attendant risks. The key question related to this investment consideration is whether the level of complexity of the fuel cycle option introduces additional risk or cost and therefore, represents a barrier to implementation. While nuclear power generation already exhibits a high degree of technical complexity and tight linkages, a fuel cycle option considered especially technically complex could include a continuous reprocessing facility where system balancing represents a key design and operational challenge.

6. Flexibility / Forward Compatibility

Given the level of investment required for implementation of fuel cycle options, and the degree of uncertainty and risk involved in early deployment, fuel cycles containing elements, processes, or facilities that could be used or readily adapted for use in other fuel cycles may be more attractive to industry investors. Accordingly, the key question related to this factor is whether elements required for fuel cycle option be used or readily adapted for use in other fuel cycles or other markets. An example of flexibility and forward compatibility would be investments in thorium mining infrastructure, which would benefit all thorium fuel cycles.

The above considerations provided insights to the market incentives and disincentives of the evaluation group under review. The Market Metric Working Group identified five qualitative bins, labeled "A" to

“E”, for characterizing the Capital At Risk associated with an Evaluation Group. These bins are summarized in Table C-7.10.

Table C-7.10. Capital at Risk Bins.

Capital at Risk “Bins”	
A	The fuel cycle option exhibits promise with respect to the capital investment required and benefits significantly from incentive related to capital at risk.
B	The fuel cycle option exhibits promise with respect to the capital investment required. Although disincentives exist, the fuel cycle option, on balance, benefits from incentives related to capital at risk.
C	The fuel cycle option is neutral with respect to the capital investment required, exhibiting off-setting incentives and disincentives.
D	The fuel cycle option exhibits challenges with respect to the capital investment required. While incentives exist, the fuel cycle type on balance, is weakened from disincentives/barriers related to capital at risk.
E	The fuel cycle type exhibits challenges with respect to the capital investment required and is weakened significantly from disincentives related to capital at risk.

B. Market Incentives and Drivers

Industry structure, market mechanisms for cost recovery, market distortions and government participation affect or are affected by market incentives and drivers. Given the level of investment required for implementation of fuel cycle options and the degree of uncertainty and risk involved in early deployment, private investment in fuel cycles would need to be market driven. Market drivers include financial and economic incentives for making investments in a fuel cycle. For example, market mechanisms for generating income from invested capital and consumer demand represent market drivers.

To inform on this factor, the Market Metric Working Group identified a number of market-related considerations, which when taken together, supported the assignment of evaluation groups into one of three qualitative bins. These market-related considerations or drivers and their supporting rationale are detailed below.

1. Industry Structure

Industry structure relates to forces that shape competition over the long-term. This includes consideration of ownership concentration, suppliers, customers, substitute products and the ability of participants to enter and exit the industry. The current industry structure may encourage or discourage the types of investments required to deploy a particular fuel cycle option at scale. Private entities involved in each component of the fuel cycle need to consider the financial risks of their involvement and their risk tolerance given the prevailing industry structure. The key question in reviewing industry structure is whether the fuel cycle option would require significant changes to the existing industry structure for successful implementation.

2. Market Distortions Caused by Law or Regulation

This market consideration looks to existing laws and regulations and how they may distort the market in ways that would encourage or deter investment in a given fuel cycle option. An example is the current nuclear waste fee (1 mill/kWh) charged to nuclear power generators. This fee does not vary with the volume or activity of the waste form and therefore, does not encourage investments in fuel cycle technologies that reduce the amount of waste produced per unit of output.

3. Government Participation

Government investment or mandates are not considered market drivers for the purpose of this analysis. However, it is noted that the absence of market drivers would require government participation in order for the fuel cycle option to be deployed. This is viewed as a challenge to full-scale commercial deployment of a fuel cycle. In reviewing the prospects for government intervention, the key question was how much government intervention would be required in order for the fuel cycle to be deployed. For example, highly significant government participation is characterized by the government having to fully fund the required investments or mandate the use of the process through changes in laws or regulations. A significant level of government participation is characterized as the government having to share the cost through direct investment or to encourage the use of the process through changes in law or regulations. Limited government participation is characterized by the government seeking to induce investment through new financial or regulatory incentives. Finally, a de minimus level of government participation includes routine/ordinary investments, incentives and actions by the federal government (akin to current government incentives such as limited tax incentives or other forms of financial assistance).

4. Market Systems & Regulatory Frameworks Affecting Cost Recovery

This market consideration relates to incentives and cost recovery mechanisms that serve to mitigate the risks associated with investments in new nuclear technologies. Market systems focus on the existence of willing buyers and sellers in a marketplace and adequate pricing signals sufficient to facilitate trade. In the United States, energy market systems and cost recovery mechanisms vary by region. Some states (e.g., Southeast) continue to regulate electric utility rates through public service commissions and public utility commissions. This provides a dependable mechanism for cost recovery for utilities, insulating utility investors from fluctuating energy markets. In other regions, such as the Northeast, wholesale markets have been deregulated and expose electric utilities to competitive forces. Under either approach, market systems exist for the production and sale of electricity. However, the risk profile of each system is different and would affect a utility's investment decisions. A key question in considering market systems is whether such systems exist for cost recovery of investments in a given fuel cycle option.

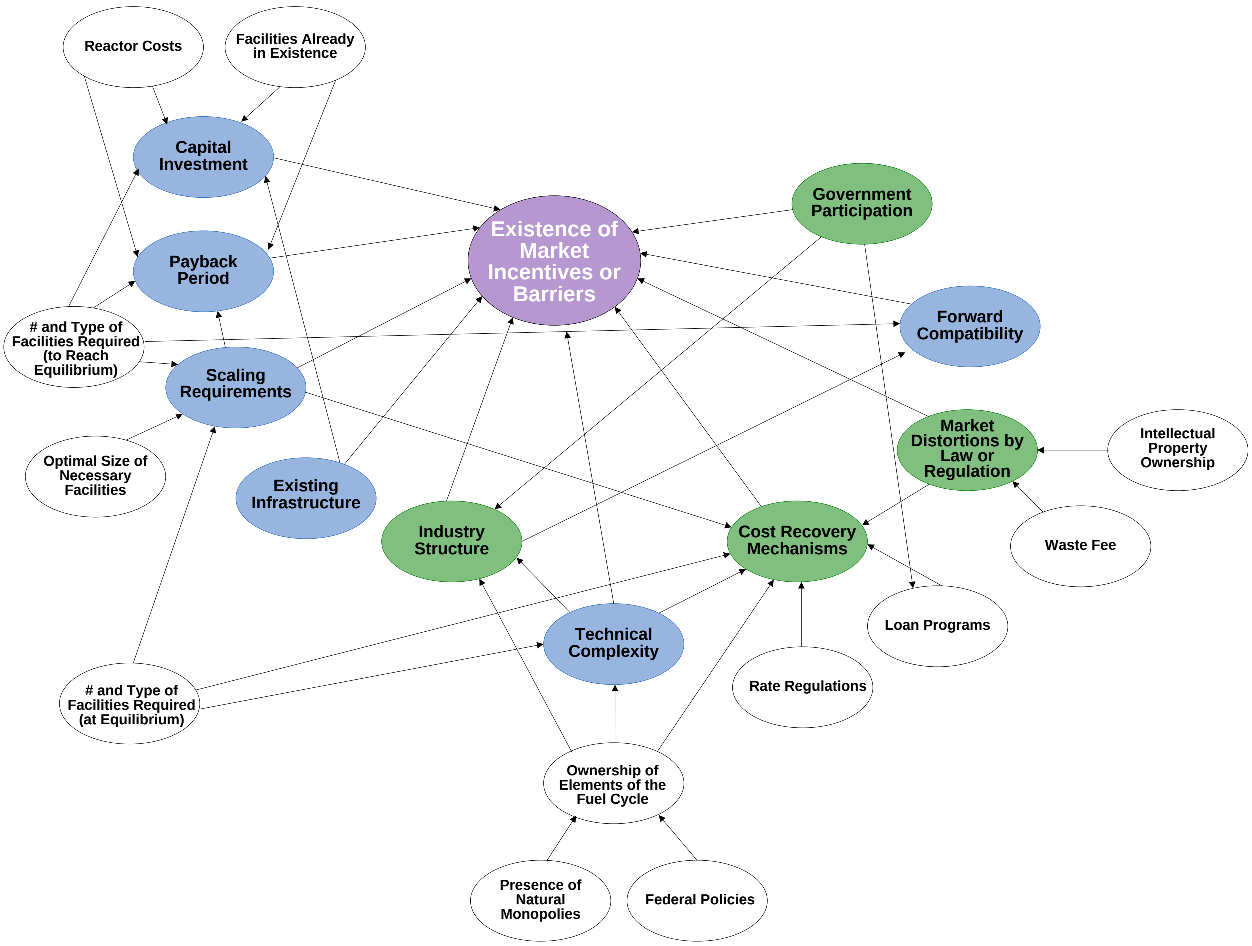
To evaluate the market incentives and drivers, the Market Metric Working Group identified three qualitative bins, labeled "A" to "C" to characterize the market considerations for a given fuel cycle option. These bins are summarized in Table C.7-11.

Table C.7-11. Market Incentives and Drivers Bins.

Market Incentives and Drivers "Bins"	
A	Markets and market mechanisms exist that support private investment for most of the fuel cycle processes/facilities needed or the fuel cycle option and Federal government intervention in the form of direct investment or mandates will not be required for most of the fuel cycle processes.
B	Markets and market mechanisms exist that support private investment for some of the fuel cycle processes/facilities needed or the fuel cycle option and significant or sustained Federal government intervention in the form of direct investment or mandates will not be required to establish market drivers.
C	Markets and market mechanisms are weak or exhibit distortions, requiring significant and sustained Federal government intervention in the form of direct investment or changes in law in order to establish market drivers.

C-7.8.4 Metric Structure

As indicated by the influence diagram, Figure C-7.5, technical considerations, including the numbers, types, and sizing of facilities, tend to influence capital at risk factors. Market incentives and drivers are influenced by regulations, ownership and other economic and market considerations.



Note: Each white oval represents a factor or input to market incentives of barriers. Blue ovals represent Capital-at-Risk considerations and green ovals represent Market Incentives and Drivers. The purple oval in the center of the diagram represents the Market Metric.

Figure C-7.5. Influence Diagram for the Existence of Market Incentives.

The Market Metric Working Group identified five bins that combined its consideration of capital at risk and market incentives and drivers. Table C-7.12 provides descriptions for Bins A through E. The description of each bin is provided below.

Table C-7.12. Final Market Metric Bins.

Bins for the Final Market Metric	
A	Markets and market mechanisms exist that support private investment for most of the fuel cycle processes/facilities needed for the fuel cycle option, and Federal government intervention in the form of direct investment or mandates will not be required for most of the fuel cycle processes. In addition, the fuel cycle option exhibits promise with respect to the magnitude of capital investment required and payback prospects; on balance the fuel cycle option benefits from incentives related to capital at risk.
B	Markets and market mechanisms exist that support private investment for most of the fuel cycle processes/facilities needed for the fuel cycle option and Federal government intervention in the form of direct investment or mandates will not be required for most of the fuel cycle processes. However, the fuel cycle option exhibits challenges with respect to the magnitude of capital required and payback prospects, and on balance the fuel cycle option is weakened by disincentives related to capital at risk.
C	Markets and market mechanisms exist that support private investment for some of the fuel cycle processes/facilities needed for the fuel cycle option and, while some Federal government intervention in the form of direct investment, mandates or incentives may be necessary, significant and sustained Federal government intervention will *not* be required to promote investment.
D	Markets and market mechanisms are weak or exhibit distortions, requiring Federal government intervention in the form of direct investment or changes in law in order to establish and sustain market drivers. However, the fuel cycle option exhibits promise with respect to the magnitude of capital required and payback prospects, and benefits from incentives related to capital at risk.
E	Markets and market mechanisms are weak or exhibit distortions, requiring Federal government intervention in the form of direct investment or changes in law in order to establish and sustain market drivers. In addition, the fuel cycle option exhibits challenges with respect to the magnitude of capital required and payback prospects.

In determining the final metric data for an evaluation group, the Market Metric Working Group considered the bin placement of that group on the two umbrella factors. When both market incentives and drivers and capital at risk factors were considered together, the Market Metric Working Group agreed that market drivers should carry greater influence than capital at risk. The rationale for this viewpoint was that all investment decisions would need to be responsive to market pressures (i.e., such investment would need to be market driven). To the extent a particular fuel cycle option is neither driven by the market or government intervention, the analysis of its investment requirements and returns is of little value. The matrix below in Figure C-7.6 depicts the relationship of the capital at risk bins and the market driver bins and shows how market incentives and drivers are assigned greater influence in the market metric.

		Market Incentives/Drivers		
		A	B	C
Capital at Risk	**A**	A	A	D
	B	A	B	D
	C	A	C	D
	D	B	C	E
	E	B	E	E

Note: In this matrix, boxes shaded green (light and dark) indicated generally positive market influences in the form of incentives of the absence of barriers on the evaluation group under consideration Boxes shaded in pink and red indicate varying degrees of negative influences such as barriers and disincentives for the evaluation group under consideration.

Figure C-7.6. Matrix of Bin Results as a Function of Capital at Risk and Market Incentives.

Figure C-7.6 illustrates how a fuel cycle option exhibiting strong market drivers can overcome capital at risk challenges. Conversely, a fuel cycle option with weak market drivers would face significant challenges in implementation regardless of the financial incentive profile.

As noted in Appendix D, the Market Metric Working Group did not consider disposal within the framework of the market incentives or disincentives. This exclusion recognizes that all fuel cycles face a similar challenge with regard to disposal. While fuel cycle options may reduce the volume of material destined for long-term disposal, they would not eliminate the need for a suitable long-term disposal option. Therefore, it was determined that the consideration of disposal would not provide meaningful differentiation among fuel cycle options for this Metric.

References for C-7.

C-7.1 Technology Readiness Assessment Guide, DOE G 413.3-4A, September 15, 2011

C-7.2 Fernandez2009. Joseph Fernandez, Contextual Role of TRLs and MRLs in technology management, Sandia SAND2010-7595, November 2010. See also Azizian, et al, “A Comprehensive Review….,” presented to WCESC 2009, October 2-22, 2009.

C-8. Institutional Issues Criterion

For the purposes of the fuel cycle Evaluation and Screening, the Institutional Issues Criterion is defined as follows:

Institutional Issues *– A broad definition of institutional issues may include a wide range of societal and infrastructure issues that help or hinder implementation. For the purpose of this fuel cycle evaluation and screening, the assessment of institutional issues is focused on potential challenges for implementation of a nuclear fuel cycle, including existence of industrial infrastructure, market mechanisms, and regulatory framework.*

C-8.1 Background

Institutional issues are those associated with organizations involved in the development and deployment of nuclear fuel cycles representing public, political and industrial considerations, concerns and constraints.

The Institutional Issues metrics consider the ability to deliver the infrastructure elements of a given fuel cycle or associated technology. Other criteria also consider some of these issues, particularly under "Financial Risk and Economics" and "Development and Deployment Risk." In particular, the metrics included for Institutional Issues are also reflected in the Deployment Risk portion of the Development and Deployment Metrics, as there is a significant connection between those organizations that would be engaged in deploying future fuel cycles and the institutions that are discussed below.

Institutional issues are associated with delays and risks from the potential large numbers of organizations, stakeholders, concerns and constraints - and the ability to deliver the infrastructural elements of a given fuel cycle or associated technology. The number of institutions involved in the development and the use of nuclear energy is large and diverse, with the primary ones considered for the fuel cycle evaluation and screening being investors, utilities and industry, government, and regulators. These organizations consider a broader context related to the deployment of nuclear energy involve the public, environmental groups, and the courts.

C-8.2 Metric Development for the Institutional Issues Criterion

Institutional issues are driven by the primary institutions involved in the deployment and use of nuclear energy and are discussed in more detail below.

Investors

Investors not only include those spending money on technology deployment (such as the equity investors, lenders or government agencies), but also the developers of the technology who invest time, intellectual property and resources. A key element is the associated market drivers and/or incentives for investing in nuclear energy development. The major concern to financial investors would be risk associated with the return on invested capital and the cash flow profile during the development and deployment phase (when costs are at their highest). The major constraints are therefore likely to be total capital cost, timescales for development and deployment and the spending profile (cash flow) during those periods.

Utilities and Industry

In addition to the financial and technological constraints outlined above, the commercial institutions such as utilities and vendors also have to consider the likely paradigm shift in technology, moving from a well-known, proven technology of today, to something different not only in the reactor, but in the entire fuel cycle. Many of these commercial institutions are traditionally risk averse, as their task is ensuring a reliable, sustainable electricity supply to their customers. Any change to current practice represents a substantial barrier for any utility, particularly given the regulated market in which many utilities operate. In addition, industrial capability, the pool of available expertise, and existing infrastructure represent important considerations for utilities when pursuing technology alternatives. Examples of how difficult and limiting these changes are to the utility can be seen by considering a potential move by U.S. utilities from a uranium fuel to a Mixed Oxide (MOX) fuel of uranium and plutonium, as in the case of the U.S. weapons plutonium disposition program. Even with significant incentives on the fuel supply price and risk mitigation underwritten by DOE, utility willingness to move to the MOX fuel is uncertain. Key considerations for the utilities and industry are the cost to deploy the fuel cycle technology, the incremental risks associated with the new fuel design, the ability to meet regulatory requirements, and the ability to finance the deployment.

Government

From a government perspective, it is the issues of nuclear waste management, the environment, economics and energy security (fuel resources, energy and facilities) that drives the need for R&D on advanced fuel cycles. Many of these considerations are included in other criteria such as "Waste Management," "Environmental Impact," and "Resource Utilization". Government policies, laws and regulations also form an institutional environment that may benefit or hinder commercial acceptance and deployment of a fuel cycle and the technologies needed.

Regulators

Regulators include NRC for the nuclear material and facilities, the EPA for environmental regulations, state and local regulators controlling land and water use requirements, and public utility commissions regulating the cost that can be borne by rate payers. The major institutional issue or deployment risk relates to the ability and willingness of a regulator to issue a license for a new technology i.e. the difficulty in obtaining a license for required facilities. For example, if a fuel cycle or reactor facility submitted for review was outside the experience base of the NRC, at a minimum this would result in delays and uncertainty in the licensing process. Similarly, if the underlying phenomena important to safety were viewed as not being fully understood, or if there was insufficient experimental evidence, the NRC would be hesitant to proceed with issuing a license without additional supporting data.

C-8.3 Compatibility with Existing Infrastructure

Definition of Metric – This metric represents the degree to which a fuel cycle option can utilize existing infrastructure in terms of fuel cycle facility types and the knowledge and expertise to construct and operate them versus the need to develop new infrastructure representing technologies for which little experience exists. More compatibility with existing infrastructure represents a lower deployment risk and lower institutional barrier to deployment.

The discussion for this metric is in Section C-7.6.

C-8.4 Existence of regulations for the fuel cycle and familiarity with licensing

Definition of Metric – This metric represents the regulatory maturity for a fuel cycle option based on a determination of whether regulations exist for the fuel cycle and an estimate of what level of experience regulatory organizations have in applying those regulations.

The discussion for this metric is in Section C-7.7.

C-8.5 Existence of market incentives and/or barriers to commercial implementation of fuel cycle processes

Definition of Metric – this metric seeks to address the market and commercial challenges that may be confronted as a fuel cycle is introduced into existing industrial infrastructure and market systems.

This discussion for this metric is in Section C-7.8.

C-9. Financial Risk and Economics Criterion

For the purposes of the fuel cycle Evaluation and Screening, the definition of the Financial Risk and Economics criterion is:

Financial Risk and Economics *– A broad definition of financial risk and economics may include a wide range of financial considerations for development and use of a system. For the purpose of this fuel cycle evaluation and screening, the assessment of financial risk and economics is focused on the cost of using* *the mature deployed system, including siting, construction, operation of facilities, and consideration of financial risk.*

C-9.1 Background on Financial Risk and Economics

The contents of this Section are taken from the report describing a financial risk and economics evaluation approach [C-9.1] and further detail about financial risk and economics evaluation can be obtained from that reference. This Section describes the evaluation metrics developed by the EST for the quantification of the Financial Risk and Economics high level criterion. The identified metric is described in detail below, including the definition of the metric, the justification for using the metric, and examples for the calculation of the metric.

In the context of considering nuclear energy system options, financial risk was defined as the perceived risk of investments in nuclear facilities, including both capital requirements and financing costs, while economics addressed the appropriate specific revenue necessary to recover the costs of deploying and operating each nuclear energy system, including all facilities necessary to perform the fuel cycle functions.

The work presented below fills an identified gap of the pilot demonstration of the evaluation and screening process. Since then, the following actions have been accomplished:

1. Identification of a metric;
2. Development of an appropriate methodological framework for the quantification of the metric;
3. Development of a consistent methodology to facilitate the calculation, from a practical perspective, for a large number of fuel cycle options; and
4. Development of a systematic approach for the treatment and interpretation of uncertainty in the results.

The work performed to accomplish each step is described in Sections C-9.2 and C-9.3. The cost of research, development and demonstration of nuclear fuel cycle was not addressed in the Financial Risk and Economics Criterion since that it was included in the Development and Deployment Risk Criterion. (Section C-7).

C-9.2 Metric Development for the Financial Risk and Economics Criterion

The metric developed for informing on the economic and financial risk performance of each fuel cycle is the “Levelized Cost of Electricity at Equilibrium” (or LCAE). The LCAE is the cost of electricity which renders the net present value of the project cash flow equal to zero. For a reactor, the included costs are those associated with capital investment, operation and maintenance, fuel, waste disposal, and decommissioning the plant at the end of life, while the revenue is obtained by the sale of products, (e.g., wholesale electricity for a reactor). Each expense and revenue stream is discounted to an arbitrary point in time, typically the beginning of construction or irradiation, using a discount rate that reflects the financial risk of the project: by means of an opportune discount rate, the LCAE metric addresses both the “financial risk” and “economic” performance high level criterion.

Other metrics to evaluate the economic performance of energy generating assets are infrequently used in other works on the topic [C-9.2], such as “Book rate of return”, “internal rate of return” and “payback time”. Of particular importance is the “total capital at risk” metric, which gauges the magnitude of the overall investment. These metrics all reflect details of financial risk and economics that are all included in the LCAE.

The Nuclear Fuel Cycle Evaluation and Screening was performed for fuel cycles at mass balance equilibrium. Mass balance equilibrium implies that all the mass streams in a given fuel cycle do not change with time, or from one irradiation cycle to the next. This condition is an important assumption that allows the evaluation of performance of fuel cycles independently of the transients required to reach equilibrium situations. As a result, the LCAE was chosen as the metric for the Financial Risk and Economics criterion since it intentionally excludes all the expenditures and revenues associated with each fuel cycle option during transition. The concept and equations are the same as the more familiar Levelized Cost of Electricity (LCOE), but the transients at the beginning and end of each fuel cycle are excluded when calculating the LCAE.

The LCAE is a cost at the plant busbar, thus lower than the ultimate cost to ratepayers, which also includes transmission and distribution. However, differences in busbar costs are a good indicator of ultimate differences in rate cost to ratepayers. Moreover, both transmission and distribution costs are largely independent of the specific nuclear fuel cycle choices. It is possible to envision a transmission cost advantage associated with certain nuclear fuel cycle options, such as for example in the case of deployment of units of smaller generating capacity, that could be located closer to the consumption centers. However, these effects on transmission cost are not included in the present work, since such a study would involve the need to specify specific technologies and would require a substantially more complex model of regional interconnections, geographical distributions of the regulated U.S. utilities, Federal Energy Regulatory Commission (FERC) regulations, demand growth forecast, transmission availability, deployment scenarios, etc.

Transportation costs for operating the fuel cycle are not included in the analysis, since they would largely depend on the geographical location and arrangement of each fuel cycle. The Nuclear Fuel Cycle Evaluation and Screening is intended to evaluate fuel cycles that could be deployed anywhere in the U.S. at any time in the future, and to identify the performance characteristics of each based on its intrinsic properties, which should be largely independent on the geographic location. For this reason, transportation costs are outside the scope of this work.

Development of a Framework for the Quantification of the Metric

The need for a robust framework to be applied consistently throughout a vast set of possible fuel cycle schemes motivated the development of the novel “island” approach described in Section C-9.3. The key concept of this methodology was to divide each fuel cycle in subsets of facilities (here generally called “islands”) that contain one single reactor, and all the related fuel cycle facilities. This approach greatly simplified the economic analysis of multi-reactor fuel cycles, by allowing the user to model the cash flow of each reactor-containing-island separately. The LCAE of the system depended on the time and amplitude of the cash flow of each subsystem, and substantial simplification in the calculation of the LCAE was achieved by modeling each of these cash streams separately. The LCAE of the overall system was calculated by the weighted average of the LCAE of each of the islands, where the weights were the fractional energy generated by each island. It is shown in Section C-9.3 under which conditions this solution is exact versus an approximation.

Development of a Tool Set to Facilitate Calculations

A new code (NE-COST) was developed specifically for the calculation of the LCAE metric for complex fuel cycles, using the “island approach”. One design objective for NE-COST was to allow the calculation of the cost of electricity of arbitrarily complex systems by just changing the input, without the need to alter the code. For this purpose, the general structure of the code was developed while allowing several alternative front-end and back-end paths, which can be selected by the user by using switches in the input. NE-COST was developed with the capability to handle uncertainty as a required functionality. To this end, the NE-COST structure was developed specifically to handle distribution information. A Monte-Carlo sampler as well as a methodology for the propagation of uncertainty between islands was developed to create a system-wide cost of electricity uncertainty distribution. A suite of tools was also created to handle the stochastic combination of distributions and the plotting of the results.

Development of an Approach for the Treatment of Uncertainty

It is essential for a robust economic analysis to have a basis for the input cost data used, while acknowledging the existence of uncertainties and the limitations of the basis used for the generation of such data, together with clearly stated assumptions and limitations. Such uncertainty range should, of course, be higher for less mature concepts. In order to provide such basis for the input cost data to be used for the economic analyses, the FCO campaign has been maintaining and updating a report and associated database of cost references – the Cost Basis Report [C-9.3] – for all of the steps of the nuclear fuel cycle. Acquisition of the cost data was from public reports, the trade press, other fuel cycle studies, discussion with private industry, and also life cycle cost calculations made by the FCO campaign.

Large uncertainties in the input cost data led to large uncertainties in the calculated values of the LCAE metric. This was not a consequence of the choice of the metric, but rather of the uncertainties that were almost inevitably present in the input cost data. The comparison of economic performance based on the LCAE, led to largely overlapping probability distributions.

As reported in Ref. C-9.1, the overall feedback as received by a group of experts asked to review the approach was that the methodology, based on the levelized cost of energy at equilibrium (LCAE) is consistent with the standard practice of the field for systems at equilibrium, was fundamentally sound, and was appropriate for the purpose of the Evaluation and Screening. The methodology implemented to calculate the LCAE for a large number of fuel cycles – as embedded in the NE-COST code – appeared rigorous within the stated approximations and assumptions (most notably the assumption of a system in equilibrium).

There was unanimity among the experts on the following: the discount rate should not be treated as an uncertain variable (with Monte Carlo sampling) but as a parameter that can assume a set of discrete values; and probability distributions of the LCAE should be generated and displayed separately for each selected discount rate. Good agreement among the experts existed also in the suggested range for the discount rates: from 2% (as suggested by OMB for government projects) to 7% and possibly 10%.

The potential importance of "total investment cost" as an additional metric was raised. However, there was disagreement on the need to include it in the analysis since (1) the capital at risk strongly depends on the institutional structure of the utilities and (2) it appears difficult to make a distinction between designs that all appear very capital intensive. As a practical course of action it was decided not to add this concept as an additional metric.

After some discussion, as a consequence of the uncertainties inherent in estimates of future fuel cycles, there appeared to be consensus among the experts that the best and most defensible alternative appears to be "Treat the LCAE as a separate topic in a ranking study". The experts appeared to support a conceptual approach of using the results of the calculated LCAE (and also, possibly, of the total capital at risk) for each system as a "cost" (or "price", under the assumptions of economic equilibrium).

C-9.3 Levelized Cost of Electricity at Equilibrium

The metric for financial risk and economic performance is the Levelized Cost of Electricity at Equilibrium (LCAE). The EST, as well as experts outside of DOE, judge this metric to be adequate for the purpose of informing on the differences in financial risk and economics between fuel cycle options for the following reasons.[C-9.1]

- ***Levelized Cost of Electricity at Equilibrium (LCAE)***
  - The metric addresses both financial risk and economics and takes into account multiple economic factors, including the cost of construction, operation and decommissioning/closure

of each type of facility needed for the nuclear energy system, in proper ratio for the mass flows of the system to be in balance.

- The metric accounts for capital investments and the time value of money over the life cycle of each mine, mill, plant, reactor, storage and disposal facility while considering all of the related activities to be occurring simultaneously as needed for the equilibrium fuel cycle option screening.
- The financial risk involved with a fuel cycle option is included in the LCAE metric through the use of a risk adjusted opportunity cost of capital, or discount rate. Calculations were performed for several discount rates reflecting the range of possible financial conditions that may exist.

The calculations include the uncertainty in cost estimates for the facilities.

Description of the LCAE

Figure C-9.1 shows a schematic representation of the typical cash flow profile for a nuclear facility over its lifetime: it could be a power reactor or any other fuel cycle facility, such as a reprocessing plant. Cash flow is typically large and negative during the construction phase of the plant. After successful startup, the facility begins selling products (e.g., electricity or heat for a nuclear reactor, or other products such as fuel services for other fuel cycle facilities) and receives positive revenue. After shutdown, cash is required in the decommissioning phase. The revenue obtained by selling products (e.g., electricity can be taken as constant in real dollars, or adjusted for inflation. This simplified treatment is suitable for base-load plants, since it is possible to calculate “a priori” the amount of product (e.g., electricity) generated during the lifetime of the asset, and the rate of production would be constant per unit of time. For a plant planned for working at peak demand (such as “peak load” gas turbines) or subject to unpredictable availability (such as wind turbines), more complex considerations is required. These considerations are not of concern for nuclear plants, however, since the assets are expected to operate at a capacity factor that is limited only by the plant downtimes due to refueling and other outages. Of course, risks that may reduce the capacity factor below the expected values are of major concern to the investors, and are addressed in a manner that is explained in this chapter. Predictable down times occurring at more or less regular intervals (e.g., during refueling) are not shown on Figure C-9.1 for simplicity.

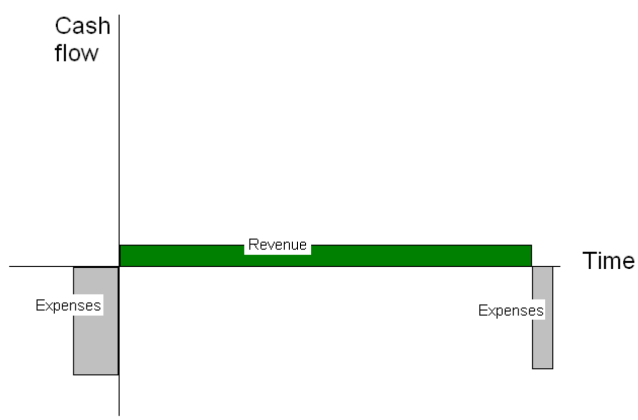


Figure C-9.1. Schematic Representation of a Typical Cash Flow Profile for a Power Plant Over its Lifetime.

The revenue from the sale of electricity is the product of the number of units sold and of the price of each unit: the number of units produced is, as mentioned, reasonably assured under the baseload assumption, and the price per unit of product is uncertain, making the revenue projection also uncertain. For this reason there are two typical ways of evaluating the economic performance of the asset: (1) assign a price of electricity (typically the most recent wholesale price or a time average of past prices) and calculate the "internal rate of return" that would be obtained for the investors at that particular price; and (2) calculate the cost of electricity that would provide an adequate return to both equity and debt investors. Both methods, when applied correctly, are equivalent in the sense that they would likely lead to the same ranking of fuel cycle options by economic performance. The latter, employing the concept of levelization of the expenditures and revenue, is called the Levelized Cost of Electricity (LCOE).

The Nuclear Fuel Cycle Evaluation and Screening was performed for fuel cycles at mass balance equilibrium: in this case the LCOE was called the Levelized Cost of Electricity at Equilibrium (LCAE). Mass balance equilibrium implies that all the mass streams in a given fuel cycle do not change with time, or from one cycle to the next, and that each fuel cycle facility is preceded and followed by a sequence of identical facilities, with identical cash flow profiles in constant dollars. The equilibrium condition is an important assumption that allows the evaluation of performance of fuel cycles independently of the transients required to reach equilibrium situations.

The revenue generated by the sale of the product (e.g., electricity) needs to cover both the fixed and variable operating and maintenance expenses incurred during normal operations including fuel, decommissioning costs and repayments on capital invested during the construction. Capital costs include both overnight costs and financing charges; and returns to compensate both debt and equity investors for the risk taken on the project. The constant electricity price that, in real dollars, covers all these charges is the LCAE. In more formal terms, the LCAE is the net present value of a continuous stream of revenue charged against the sale of electricity, equalized to the sum of the net present value of all the expenditures incurred by the plant owner throughout the physical life of the plant for a system in equilibrium, according to Equation (1),

$$\int_0^{T_{plant}} C_{lev} E(t) e^{-rt} dt = \int_0^{T_{plant}} K(t) e^{-rt} dt \qquad (1)$$

where

C_{lev}	=	the levelized cost of electricity at equilibrium (LCAE)
E(t)	=	represent the time profile of the electricity generated over the life of the plant
K(t)	=	the dollar value of the expenditures sustained at time *t*

The discounting is expressed here as continuously compounded, rather than the perhaps more familiar annual compounding, to represent the fact that the revenue from selling electricity is collected continuously. Since, by definition, C_{lev} is a constant, an explicit solution for C_{lev} can be found under the assumption that *E(t)* is also constant, "*E*", under the "baseload" assumption.

$$C_{lev} = \frac{1}{E} \frac{r}{1-e^{-rT_{plant}}} \int_0^{T_{plant}} K(t) e^{-rt} dt \qquad (2)$$

In this expression, it is noted that the resulting LCAE is simply the integral of the net present value of the expenditures sustained by the plant operator, multiplied by a term called "capital recovery factor", which is a function of the discount rate and of the financial life of the plant.

Discount Rates

As can be seen from Equations (1) and (2), the discount rate is of importance for the estimation of the LCAE. While a detailed discussion of the appropriate discount rates for nuclear project is quite complex, and beyond the scope of this report, the basic methodological framework is highlighted here.

A quantitative measure of the time value of money, the *discount rate* can be theoretically defined as "*The opportunity cost of investing in a certain project rather than in an alternative investment featuring equivalent risk*" [C-9.2]. On this basis, it is clear that a given company's cost of capital is not always the correct discount rate to use since the degree of risk of the particular project may be different from the overall risk of the firm undertaking the project. However, since in practice discount rates are difficult to measure and/or calculate, this is often a reasonable assumption to make. On this basis the weighted average cost of capital (WACC) of publicly traded utilities can be used as a proxy for the opportunity cost of capital, or discount rate, for nuclear energy systems. Importantly, this assumption relies on two principles: (1) the investors in each stage of the fuel cycle face similar risks; and (2) specific development and deployment risks of a given fuel cycle are not present under the assumed equilibrium conditions. With respect to the first principle, investors in various parts of the fuel cycle face different risks and would likely have different discount rates. However, these investors may range from mining companies to the federal government. To address this, a range of discount rates was applied, as detailed below. Regarding the second principle, the analysis specifically excludes the quantification of risks associated with development and deployment. These risks would not be present under equilibrium conditions and were considered under other metrics, including development and deployment risk and institutional issues.

The standard formula for the WACC is shown in Equation (3) (including the effect of taxation) where C_E and C_D are the costs of equity and debt, E and D are total dollar values of equity and debt of the representative firm, and tax_{rate} is the tax rate of the representative firms. Equation (3) is based on the fact that, under current U.S. Internal Revenue Code fiscal laws, interest paid on debt is tax-deductible.

$$WACC = C_E \frac{E}{E+D} + C_D \frac{D}{E+D}(1 - tax_{rate}) \tag{3}$$

The recommended cost of debt and equity is based on 2004 data from the publicly traded utilities in the U.S., from Ref. C-9.4. The costs of debt and equity for U.S. utilities was reported by Bloomberg in 2004 and adjusted from the reported after-tax to the pre-tax rate, is 5.3% for debt and 8.6% for equity. However, it is noted that the Bloomberg data are calculated from the spreads over treasury bonds with 10 years maturity, so there is a need to convert those data to a longer maturity to reflect the longer duration of nuclear projects. For this reason, it is recommended in Ref. C-9.4 to add a 0.5% to 1% extra cost of capital to these values of WACC. Additionally, another 0.5% should be added to the reported rates to account for the abnormally low rates present in 2004, yielding a nominal cost of debt between 6.3 and 6.8% and a nominal cost of equity between 9.6 to 10.1%. In the Ref. C-9.4 economic study and calculations, these values have been rounded to 7% and 10%, giving a final value of WACC of 8.5%, assuming 50% debt and 50% equity financing, also based on the average capital ratios of publicly traded utilities. It is noted that these are nominal rates, (i.e., including the inflation rate). When a standard level of inflation of 3% is excluded to obtain the “real” cost of capital, the resulting discount rate is 5.3%. A rounded value of 5% is therefore suggested as reference for the calculation of the LCAE for the Nuclear Evaluation and Screening. It is recommended that, for the purpose of the present Nuclear Fuel Cycle Evaluation and Screening, the LCAE calculations were repeated for a set of commonly accepted values for discount rates: 5% as reference value, 3% as a low value, and 10% as a high value. The low value of 3% corresponds to the long term average risk free rate: at this rate, investors are not compensated for any risk associated with the nuclear investment.

Computational Framework for the Quantification of the LCAE Metric

The practical calculation of the cost of electricity for any fuel cycle is, in principle, a straightforward application of Equation (1). This involves the identification of the amount and timing of all the expenses sustained during the entire life of the system, and the calculation of the amount of electricity and/or heat available for sale from the system during that time. An additional important parameter is the payback time for each nuclear asset. This is largely a financial consideration, and in reality it would be affected by the regulatory, fiscal and financial situation of the entity owning and operating the facility. For the purpose of the present evaluation and screening, a common set of criteria for this parameter need to be selected across the possible fuel cycle options. The most obvious choice is to allow a financial life identical to the operational life, in turn identical to the licensed life of each asset in each fuel cycle (this for example, would be 60 years for standard Light Water Reactors).

It is noted that these methodological principles and Equation (1), are valid for any facility of each fuel cycle, from reactors to fuel fabrication plants, enrichment and reprocessing facilities, storage and disposal facilities etc. The LCAE would be for "electricity" when referring to nuclear reactors designed to produce electricity, for "heat" when referring to nuclear reactors intended for the production of heat and for "unit of product", when referring to facilities that are intended to process materials. Generally, for material processing facilities, the cost data reported in the Cost Basis Report [C-9.3] are already levelized per unit of output (i.e. in $/kgHM): therefore, in general, it was not necessary to perform off-line calculation of the levelized unit cost of material processing facilities.

An Example Calculation of LCAE for a Once-Through Fuel Cycle

In this section, the process of calculating the LCAE is illustrated with reference cost values (i.e., ignoring uncertainties in the input cost data), for a reference UO_2 Pressurized Water Reactor (PWR) once-through fuel cycle, with geologic disposal for the spent nuclear fuel. For simplicity, in this example a few items have been neglected (e.g. the cost of disposal of deconverted depleted uranium) as compared to the actual calculation using NE-COST (please see Ref C-9.5) for EG01.

The input parameters for the illustrations are obtained from Analysis Example for the Basis of Comparison (EG01) (see Appendix B) and from the relevant modules of the Cost Basis Report.

Steps of this fuel cycle are:

- Mining and milling of natural uranium;
- Conversion of natural uranium to UF_6;
- Enrichment of the natural uranium to the level required for the reference PWR;
- Fabrication of oxide pellets, clad fuel elements and finished assemblies ready for reactor irradiation;
- Irradiation in the reactor;
- Storage of spent fuel in the reactor pools;
- Disposal of depleted uranium; and
- Final disposal of spent fuel after cooling in the on-site pools.

The first step is to obtain the cycle length and the capacity factors: both parameters were calculated, as shown, from basic characteristics of the core.

The thermal power of the core is 3000 MW_{th}, the thermal efficiency is 33% and the power density is 34 MWth/MTiHM. Therefore, the core initial HM mass is 88.23 metric tons. Additionally, a capacity factor of 90% is assigned to this system. Based on this information, the annual energy production can be calculated from Equation (4).

$$Energy_{year} = Core_{Thermal\_pow} \cdot \eta \cdot L \cdot 8766 \qquad (4)$$

where

$Core_{Thermal\_pow}$ = the core thermal power in kW, η is the thermal conversion efficiency

L = the plant capacity factor

8766 = the number of hours in a year.

Additionally, the capital recovery factor can be calculated once the real opportunity cost of capital "r" and the financial life of the plant "T_{plant}" are known. Assuming the suggested reference values of 5% and 60 years, respectively, the capital recovery is obtained from Equation (5).

$$Capital\ recovery\ factor = \frac{r}{1-e^{-rT_{plant}}} = 0.0526 \tag{5}$$

The magnitude of the fuel cycle expenditures at each reactor refueling is calculated next. The timing of the fuel cycle purchases by the reactor operator is dependent on the fuel management scheme employed by the utilities.

The timing of the expenditures has however, a small effect on the overall calculated LCAE, as long as these expenditures happen relatively frequently throughout the life of the plant, as is the case in standard PWR (in this example, there are exactly 40 cycles fitting in the 60 years life of the plant, each lasting 18 months). The reason for the small effect of the timing of the fuel cycle expenditures is that they happen roughly simultaneously with the revenue from the sale of electricity: therefore the effect of discounting on these expenses is small.

In the calculations, the fabrication losses and leads times for the fuel cycle purchases are taken into account.

The following masses and amount of services that need to be purchased was calculated next. In particular, the following quantities need to be calculated:

- Mass of fuel to be fabricated;
- Mass of natural uranium to be purchased;
- Amount of enrichment units [in Separative Work Unit (SWU)] to be purchased;
- Amount of conversion services to be purchased; and
- Amount of de-conversion services to be purchased.

For convenience, the masses and services are normalized to 1 kg of HM of fabricated fuel ready for reactor irradiation.

The mass of fuel to be fabricated is obtained from Equation (6), which includes losses at the fabrication plant, which are 0.1% from the data for the relevant Analysis Example:

$$m_{fabrication} = \frac{1}{1-fabrication_{losses}} \tag{6}$$

The mass of natural uranium that needs to reach the enrichment plant is calculated with Equation (7), where $E_{product}$, E_{feed} and E_{tails} are, respectively, the enrichment of the product (4.2%), that of the feed (0.711%) and that of the tailings (0.25%); no mass losses are experienced at the enrichment plants.

$$m_{enrichment} = m_{fabrication} \frac{E_{product}-E_{tails}}{E_{feed}-E_{tails}} \tag{7}$$

The mass to be converted (or to be purchased from the mine operations) is calculated with Equation (8), which includes losses at the conversion plant. The losses at the conversion plant assumed to be 0 in this example.

$$m_{conversion} = m_{enrichment} \frac{1}{1-conversion_{losses}} \tag{8}$$

The mass of enrichment tailings for which de-conversion services would need to be purchased is the difference between $m_{enrichment}$ and $m_{fabrication}$.

The potential functions [shown in Equation (9)] for products, tailing and feeds are necessary to calculate the amount of enrichment services.

$$V_x = 2E_x \cdot \log\left(\frac{E_x}{1-E_x}\right) \tag{9}$$

where

x = represents each of the product, feed and tail: the calculated potential functions are:

$V_{product}$ = 2.86

V_{tails} = 5.96

V_{feed} = 4.87.

The total amount of enrichment services to be purchased (in SWU/kgHM) is calculated with Equation (10).

$$SWU = V_{product} \cdot m_{fabrication} + V_{tail} \cdot \left(m_{enrichment} - m_{fabrication}\right) - V_{feed} \cdot m_{enrichment} \qquad (10)$$

Subsequently, the net present value of the costs sustained during each fuel cycle step needs to be calculated in reference to a common point in time, in this example the first criticality. A convenient way to do this is to “discount” the fuel services expenses to the lead purchase time to the beginning of each shutdown, and then “discount” all the fuel cycle expenditures including the effect of cost inflation.

For example, the cost of fabrication discounted to the refueling time for which the services were purchased can be obtained from Equation (11).

$$NPV_{fabrication} = Cost_{fabrication} \cdot m_{fabrication} \cdot e^{-rT_{fabrication}} \qquad (11)$$

where

$T_{fabrication}$ = the lead time for the purchases of fabrication services, which in the relevant FCDP has been set at 6 months.

From the Cost Basis Report the average cost for PWR UO_2 fuel fabrication services ($Cost_{fabrication}$ in Equation 11), including the costs of zirconium cladding and hardware, is 350 \$/kgHM of fabricated fuel. Similar computations are performed for the purchase of uranium from the mine, for the conversion, enrichment and the de-conversion of the depleted uranium.

The present values for the front-end services, necessary to purchase 1 kg of HM fabricated and ready for reactor irradiation, is shown in Table C-9.1.

29.4 tons of new fuel need to be purchased at every reloading, resulting in an un-discounted cost of front-end services at each reloading of $66.8 million. This expenditure needs to be repeated at every reloading, resulting in a discounted total front-end fuel costs over the life of the reactor of $880 million.

Table C-9.1. Average Unit Costs and Net Present Values of Fuel Cycle Front End Services for 1kgHM of Fabricated Fuel (Data from the Cost Basis Report [C-9.3]).

Service	Average Unit Cost of Each Material or Service [C-9.3]	Cost of Fabricated Fuel NPV ($/kgHM)
Fabrication	350 $/kgHM	359
Enrichment	97 $/SWU	606
Natural Uranium Ore	135 $/kgU	1158
Conversion	12 $/kgU as UF_6	103
Depleted Uranium De-Conversion	6 $/kgU as UF_6	45
Total		2272

To obtain the fuel cycle front-end component of the LCAE, these costs need to be annualized – using the capital recovery factor of Equation (5) – and normalized by the annual electricity production, according to Equation (12).

$$levelized\ front\ end\ costs = NPV_{front\ end\ costs} \frac{r}{1-e^{-rT_{plant}}} \frac{1}{E_{yearly}} \quad (12)$$

With an annual electricity production E_{yearly} of $7.8 \cdot 10^9$ kWh, and with a capital recovery factor of 0.0526, the resulting LCAE for the fuel cycle front-end services is 5.9 mills/kWh.

A similar calculation with equations similar to Equation (11) was performed for the fuel cycle back-end, which for this fuel cycle includes storage in the reactor pools, conditioning before shipment and final disposal. The wet storage in the pools at the reactor site has no extra cost in the fuel cycle section, since the construction cost of the facility is included in the reactor capital cost (discussed below) and the operation and maintenance is included in the O&M costs of the reactor (discussed below). Long term dry storage was not considered in this example.

It was assumed for the purpose of the example that conditioning and disposal happen one fuel cycle after discharge from the core. The resulting discounted cost for Spent Nuclear Fuel (SNF) conditioning is 86 $/kgHM and the cost of geologic disposal is 501 $/kgHM, as shown in Table C-9.2.

Table C-9.2. Nominal Unit Costs and Net Present Values of Fuel Cycle Back End Services for 1kgHM of Irradiated Fuel (Data from the Cost Basis Report [C-9.3])

Service	Nominal Unit Cost of Each Material or Service from the Cost Basis Report	Cost of Back End Services ($/kgHM)
SNF Conditioning	93 $/kgHM	86
SNF Geologic Disposal	540 $/SWU	501
Total		587

The total discounted back-end fuel costs over the life of the reactor of $ 227 million, over the life of the plant [using Equation (12)]; including the yearly electricity production E_{yearly} of $7.8 \cdot 10^9$ kWh, the LCAE for the fuel cycle back-end services is 1.5 mills/kWh.

The total reactor construction cost is the sum of the overnight cost and of the interest during construction. The timing of the construction costs expenditures can, in practice, often be approximated by Equation (13), plotted in Figure C-9.2 where t_{norm} is the time of construction normalized to $-\pi/2$ and $\pi/2$ for the convenience of the calculation.

$$Cumulative\ expenditures = \frac{1}{2}(\sin\,(t_{normalized} + 1) \quad (13)$$

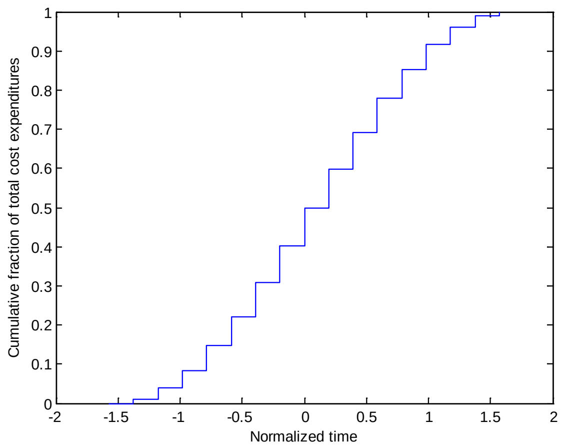


Figure C-9.2. Functional Form of the Timing of the Construction Expenditures.

From the Cost Basis Report, the expected value for the specific overnight capital cost is \$4000/kWe, resulting in total overnight costs of \$3960 million. The undiscounted expenditures, by quarter, during the assumed construction time, are plotted in Figure C-9.3 (in million \$): the sum of the total undiscounted expenditures in Figure C-9.3 sums up to the total the overnight cost of \$3960 million.

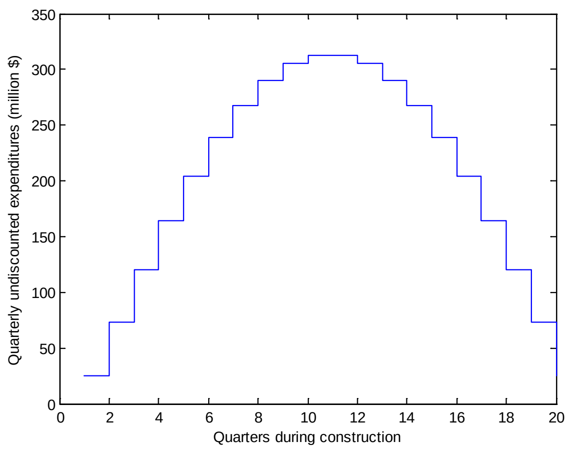


Figure C-9.3. Undiscounted Quarterly Expenditures During Construction.

The total capital cost is obtained by including the cost of capital during construction, according to Equation (14).

$$Capital\ Cost = \sum_{q=1}^{N_q} C_q (1 + r_{quarterly})^q \tag{14}$$

where

N_q = the number of quarter during construction, C_q are the expenditures in each quarter as shown in Figure C-9.3

$r_{quarterly}$ = the interest rate during construction converted from annual to quarterly compounding.

The interest during construction is assumed to be equal to the discount rate, under the assumptions used to assess the discount rate from the utilities cost of capital, as explained earlier.

Therefore, the total construction cost, including interest during construction, has been calculated at 4546 M$, or 15% higher than the overnight construction cost. When levelized over the life of the plant [using Equation (15)] and including the yearly electricity production E_{yearly} of $7.8 \cdot 10^9$ kWh, the LCAE necessary to pay the capital charges is 30.4 mills/kWh.

$$levelized\ capital\ costs = NPV_{capital} \frac{r}{1-e^{-rT_{plant}}} \frac{1}{E_{yearly}} \tag{15}$$

The O&M costs in this example are calculated from the specific O&M costs reported in the Cost Basis Report: a fixed component of $71 kWe-yr including decontamination and decommission (D&D) funding, and a variable component of 1.8 mills/kWh including capital replacement, resulting in a total fixed annual cost of $70.3 million, or 9.0 mills/kWh, and an additional variable cost of 1.8 mills/kWh, for a total levelized O&M cost of 10.8 mills/kWh.

Table C-9.3 summarizes the LCAE , broken down in its individual components as calculated in this section.

Table C-9.3. Summary of the LCAE Components as Calculated in this Section.

Cost Component	LCAE (mills/kWh)
Front End Fuel Cycle	5.9
Back End Fuel Cycle	1.5
Capital	30.4
O&M	10.8
TOTAL	48.6

The Case of Multiple Reactors and More Complex Fuel Cycles: The Island Approach

The computation for a single reactor fuel cycle gets more complex as the complexity of the fuel cycle increases. More complex fuel cycles can involve a large number of material processing facilities and more than one reactor. Each reactor would generally have a different reloading schedule and operational life, and the proper computation of the LCAE for the entire fuel cycle requires the inclusion of the amount and timing of all the expenditures for every facility and reactor in the fuel cycle. Additionally, the facilities that supply materials and services to the reactors are generally interconnected in a manner unique to each particular fuel cycle. For these reasons, it is difficult to devise a code flexible enough to compute the LCAE for each unique fuel cycle without changes in the code itself, and in practice a new code (or a new spreadsheet) would be required for each different fuel cycle (this approach was taken during the GNEP program, see Ref. C-9.3), making the computation of the cost of electricity for a large number of fuel cycle options impractical in a reasonable time frame with limited financial resources.

To alleviate these issues, an approach has been developed called the "island approach" because of its logical structure and computational framework. In the island approach, a generic complex fuel cycle is subdivided into subsets of fuel cycle facilities, called islands, each containing one and only one reactors or blanket type and an arbitrary number of fuel cycle facilities. As an example, Figure C-9.4 shows a complex, three stage fuel cycle scheme involving three reactor types:

1. A fleet of standard PWR using mined and enriched uranium;
2. A fleet of PWR MOX utilizing as primary fissile the plutonium recovered from spent fuel from the fleet of standard PWR; and
3. A fleet of fast reactors with a conversion ratio lower than 1, recycling their own fuel through electrochemical processing and utilizing as makeup fissile TRU recovered from the MOX spent fuel and MA recovered from the spent UO_2 LWR fuel.

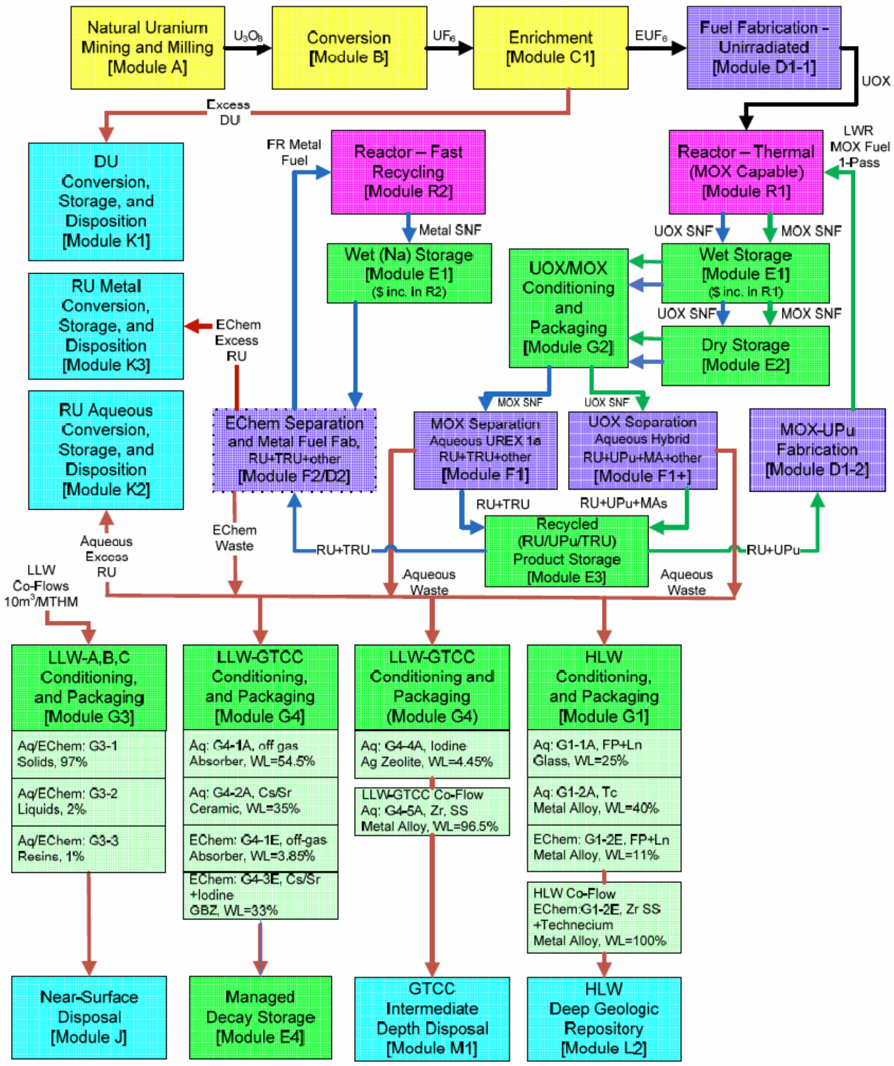


Figure C-9.4. Schematic of a 3-Stage Fuel Cycle.

The calculation of the LCAE for this system can be simplified, as will be shown in the remainder of this section, by splitting the fuel cycle into the three subsections (islands) as shown in Figure C-9.5 each containing only one of the three reactor types and a number of related non-electricity-producing fuel cycle facilities.

It is noted that each fuel cycle facility can belong to only one island, although it is possible to subdivide a facility "logically" into two or more sub-facilities that could be located on different islands, with the caution that economies of scale have to be accounted properly for the size of the single facility before the split-up. Between islands, material is allowed to flow, but there is no cash transfer. Therefore, the cash flow of each sub-section of the system was evaluated separately and the LCAE computed, based on the calculated cash flow, separately for each island. The system-wide cost of electricity was then calculated as the weighted average of the cash flow of each individual subsection. The weights are the fractional energy generated by each system subsection.

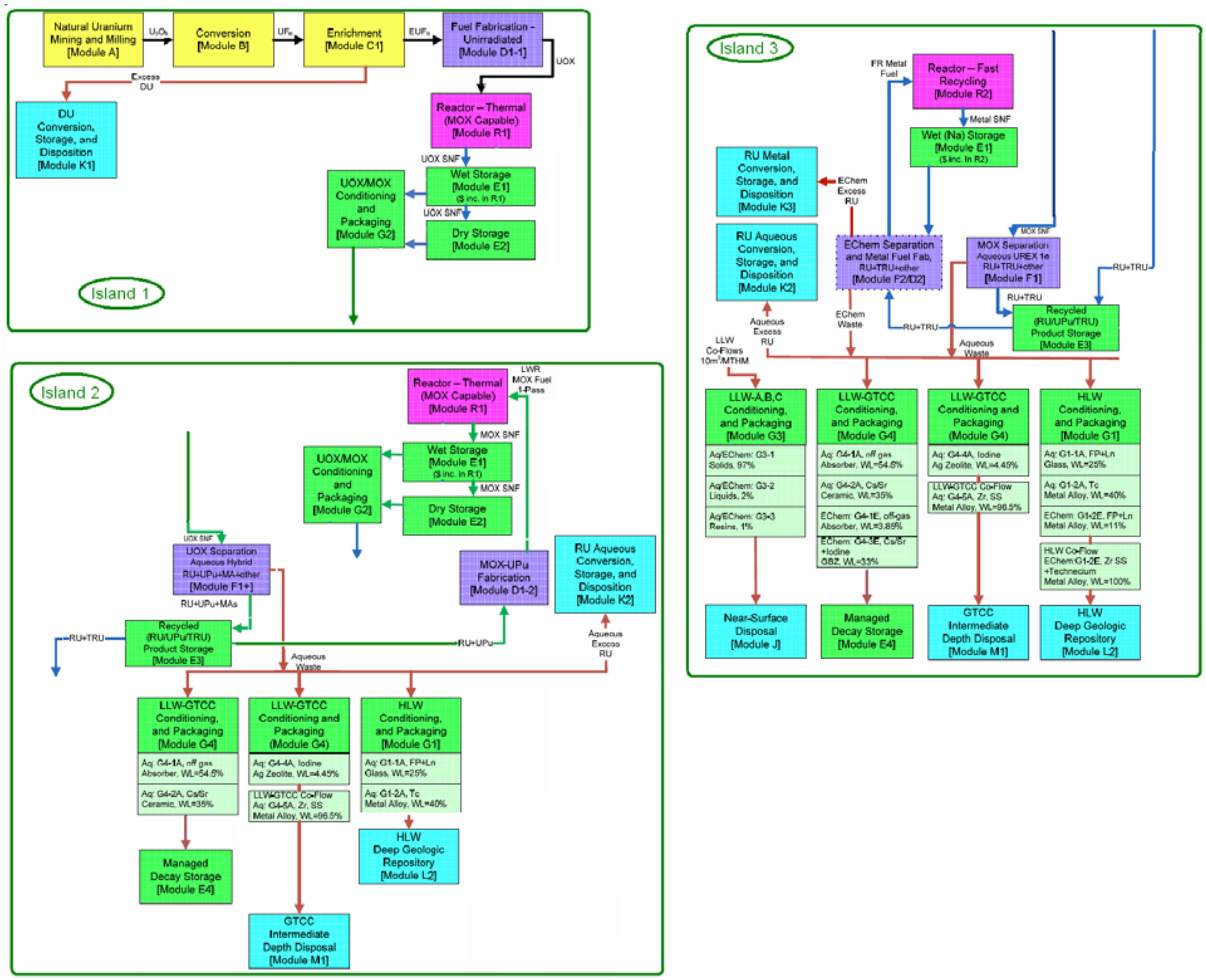


Figure C-9.5. Schematic of the 3-Stage Fuel Cycle of Figure 4 Sub-divided into 3 Islands.

Additionally, the island approach presents the following advantages:

- It allows substantial savings in the complexity of the set-up, thus reducing the time and effort necessary in setting up the input and increasing the robustness and the reliability of the calculated output
- It minimizes new model development

The methodological framework of the island approach produces a close approximation of the theoretically exact LCAE for complex fuel cycle systems. In this section, the appropriate methodology to incorporate the time-offsets of reactor startups in different islands is identified. The equations are developed for a two-islands case as an example, but can easily be extended to situations involving more than two islands. The results obtained in this process would also provide a basis for understanding the conditions under which the approximate solution provided by the island methodology is exact.

In the island approach the LCAE of each island is estimated independently. The cost of electricity C_{lev1} for island 1 is determined with Equation (16).

$$C_{lev1} = \frac{1}{E_1}\frac{r}{1-e^{-rT_{plant1}}}\int_0^{T_{plant1}} K_1(t)e^{-rt}dt \quad (16)$$

For simplicity of notation, the integral of the expenditures K(t) over the life of the plant is indicated as the "net present value of expenditures 1", or NPV1, as in Equation (17).

$$\int_0^{T_{plant1}} K_1(t)e^{-rt}dt = NPV_1 \quad (17)$$

Using the notation of Equation (17), Equation (16) becomes:

$$C_{lev1} = \frac{NPV_1}{E_1}\frac{r}{1-e^{-rT_{plant1}}} \quad (18)$$

Similarly, the cost of electricity C_{lev2} for island 2 is determined with Equation (19), similar to Equation (18), with the same simplified notation for the present value of the expenditures.

$$C_{lev2} = \frac{NPV_2}{E_2}\frac{r}{1-e^{-rT_{plant2}}} \quad (19)$$

Expressing E_2 as a multiple of E_1 as in Equation (20), the overall weighted average of the cost of electricity for the system composed of the two islands is shown in Equation (21), where the weights are the fractional electricity produced by each reactors on each island.

$$E_2 = \alpha E_1 \quad (20)$$

$$C_{average} = C_{lev1}\frac{E_1}{E_1+E_2} + C_{lev2}\frac{E_2}{E_1+E_2} = C_{lev1}\frac{1}{1+\alpha} + C_{lev2}\frac{\alpha}{1+\alpha} \quad (21)$$

When substituting Equations (18) and (19) for C_{lev1} and C_{lev2} respectively in Equation (21), Equation (22) is obtained.

$$C_{average} = \frac{r}{E_1(1+\alpha)}\left(\frac{NPV_1}{1-e^{-rT_{plant1}}} + \frac{NPV_2}{1-e^{-rT_{plant2}}}\right) \quad (22)$$

However, if one of the two reactors feature a start-up time offset T_0 (as shown in Figure C-9.6), evaluating the levelized cost for the combined system $C_{combined}$ yields Equation (23) (in discrete annual compounding).

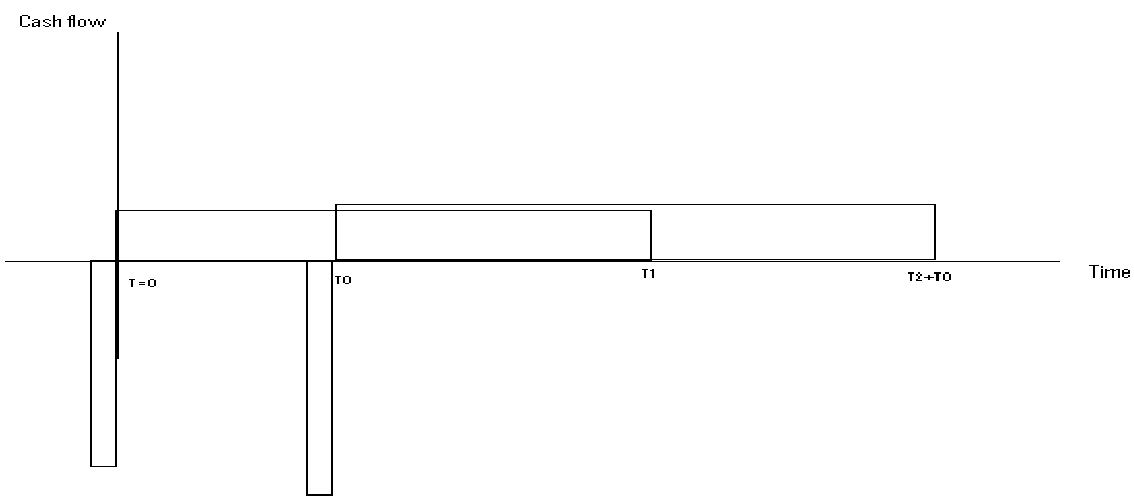


Figure C-9.6. Qualitative Representation of the Cash Flow for Two Reactors on Different Islands Within the Same Fuel Cycle with Time Offset.

$$\mathrm{NPV_1} + \frac{\mathrm{NPV_2}}{(1+r)^{T_0}} = \mathrm{C_{combined}} \mathrm{E_1} \left[\sum_{i=0}^{T_0} \frac{1}{(1+r)^i} + (1+\alpha) \sum_{i=T_0}^{T_{plant1}} \frac{1}{(1+r)^i} + \alpha \sum_{i=T_{plant1}}^{T_0+T_{plant2}} \frac{1}{(1+r)^i} \right] \quad (23)$$

Equation (23) can be easily translated into the equivalent Equation (25) using continuous compounding $r_{continuous}$, which simplifies the calculation. In turn, $r_{continuous}$ can be easily obtained from r_{annual} by using Equation (24).

$$r_{continuous} = \ln(1 + r_{annual}) \quad (24)$$

$$\mathrm{NPV_1} + \mathrm{NPV_2} e^{-rT_0} = \mathrm{C_{combined}} \mathrm{E_1} \left[\int_0^{T_0} e^{-rt} dt + (1+\alpha) \int_{T_0}^{T_{plant1}} e^{-rt} dt + \alpha \int_{T_{plant1}}^{T_0+T_{plant2}} e^{-rt} dt \right] \quad (25)$$

Equation (25) can be solved to yield an explicit expression for $C_{combined}$, as shown in Equation (26).

$$\mathrm{C_{combined}} = \frac{r}{\mathrm{E_1}} \left[\frac{\mathrm{NPV_1} + \mathrm{NPV_2} e^{-rT_0}}{\left(1 - e^{-rT_{plant1}}\right) + \alpha e^{-rT_0}\left(1 - e^{-rT_{plant2}}\right)} \right] \quad (26)$$

By comparing Equations (26) and (22), it is possible to understand under which conditions the island approach approximate solution of Equation (22) ($C_{average}$) is exact, as the $C_{combined}$ in Equation (26): Equation (22) is identical to Equation (19) if T_0=0 (i.e., there is no time offset in the startup of the two islands) and if T_{plant1}=T_{plant2}. Typically, because of the long times involved in the reactor's lifetimes, the "error" incurred because of possibly slightly different lifetimes is normally <1% of the COE, smaller than the uncertainty in LCAE deriving from the uncertainty in the input data.

Uncertainties in the Calculated Cost of Electricity

As discussed above, virtually all the available cost data in nuclear economics feature a degree of uncertainty. Large uncertainties in the input cost data, of course, lead to large uncertainties in the calculated LCAE. As an example, Figure C-9.7 shows the LCAE distribution resulting from a previous study [C-9.3] for once-through, "1 tier" and "2 tier" (2 stage and 3 stage) systems. The uncertainty distribution for each system is shown, as is the substantial overlap between the LCAE distributions for the three systems.

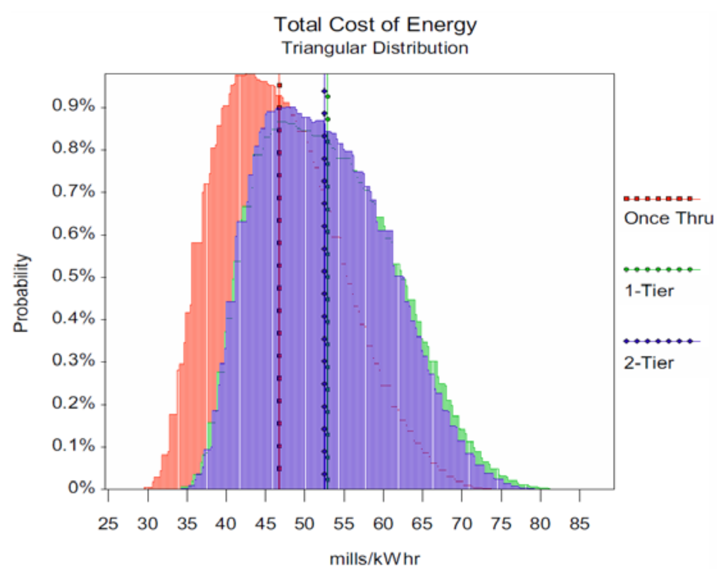


Figure C-9.7. LCAE as Calculated in Ref. C-9.3 for Three Example Fuel Cycles (Once-Through, 1 Tier and 2 Tier Systems).

Use of the LCAE Metric

The comparison of economic performance based on the LCAE often led to largely overlapping probability distributions, since for most fuel cycle options the absolute difference in LCAE is smaller than the uncertainty with which the LCAE can be calculated. It is important to highlight that the issue of probability distribution overlap was not a consequence of using the LCAE metric, but rather of the uncertainty intrinsic in the input cost data.

Given all of the inherent uncertainties in the calculation of an LCAE, mainly from the input data to the analysis process, the approach used by the EST was to focus on the differences in LCAE rather than on the LCAE itself. The uncertainty distributions were used to inform on the extent to which LCAE estimates for one Analysis Example would overlap with the LCAE for the Basis of Comparison. Once all of the LCAE estimates and uncertainties were obtained for each Analysis Example, comparison of all of the results allowed the establishment of bins to place each of the Evaluation Groups. Since this process required the LCAE results, it is discussed in detail in Appendix D-22.

References for C-9

C-9.1 F. Ganda, B. Dixon, and K .Williams, "Nuclear Energy System Evaluation and Screening - Evaluation Approach and Metric for Financial Risk and Economics," FCRD-FCO-2012-000270, August 31, 2012.

C-9.2 R.A. Brealey and S.C. Myers, “Principles of Corporate Finance”, McGraw Hill, New York, 2003.

C-9.3 D. E. Shropshire et al., “Advanced Fuel Cycle Cost Basis – Rev 2”, December 2009, INL/EXT-07-12107.

C-9.4 “The Economic Future of Nuclear Power”, A study conducted at the University of Chicago, Aug, 2004.

C-9.5 F. Ganda, E. A. Hoffman, and K. Williams, “Nuclear Energy System Evaluation and Screening—Levelized Cost of Electricity at Equilibrium”, FCRD-FCO-2013-000196, June 30, 2013.