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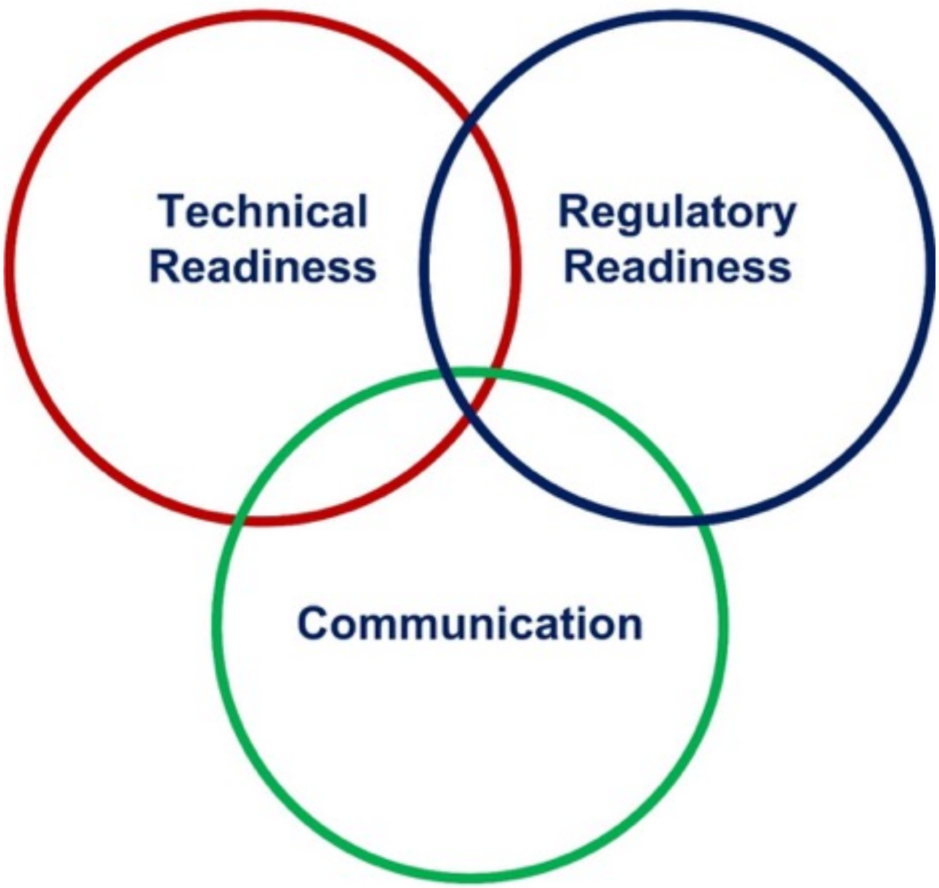
NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy, Volume 5 – *Radionuclide Characterization, Criticality, Shielding, and Transport in the Nuclear Fuel Cycle*



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EXECUTIVE SUMMARY

This report describes the plan to assess changes to the existing fuel cycle from non-Light Water Reactor (non-LWR) designs. Specifically:

a) The ability of the United States Nuclear Regulatory Commission (U.S. NRC) to perform independent fuel cycle safety analyses and consequence assessments,
b) Analytical capability gaps to ensure the NRC can perform independent non-LWR fuel cycle safety evaluations, and
c) Analytical capability readiness of the NRC to perform independent non-LWR fuel cycle safety evaluations.

As a plan, this report serves as an initial approach and will we be updated as we implement the plan. The performance of this work will leverage a number of reports. Specifically NUREG/CR-6410 (1) as a basis to establish the types of analyses required, and analytical development and analysis capability activities planned to support the NRC non-LWR Vision and Strategy, Volume 3 (2) and Volume 4 (3). Other work that will be leveraged includes work under NRCs Accident Tolerant Fuel, High Burnup and High Assay Low Enriched Uranium program (4), and ongoing work assessing existing nuclear data for application with advanced reactors.

The NRC's mission is to license and regulate the Nation's civilian use of radioactive materials to provide reasonable assurance of adequate protection of public health and safety and to promote the common defense and security and to protect the environment (5). For this work an aspect of the mission is to understand, control, and predict the behavior of systems that contain radioactive material. Here, this is accomplished through the use of neutronics and radionuclide characterization tools that are fast, portable, well assessed, understood, and easy to use computer codes.

The existing NRC computer code packages of SCALE, MELCOR, and consequence tools such as MACCS/RASCAL are utilized to establish NRC non-LWR fuel cycle safety analysis capabilities. SCALE is typically used for decay heat estimation, criticality safety and radiation shielding, time dependent radionuclide inventory prediction or depletion, reactor physics, and sensitivity and uncertainty analyses to support code benchmark similarity assessments and bias and uncertainty quantification. MELCOR is typically used for evaluating transport of radioactive and non-radioactive hazardous material within the enclosure of a facility under a range of normal and off-normal conditions. Together, the codes enable the estimation of radiological and non-radiological hazardous material release to the environment (i.e., the radioactive and non-radioactive source term to the environment). MACCS and RASCAL use the source term to the environment to perform dispersion and deposition calculations that support estimation of public health and safety consequences. These codes can be applied as part of a risk-informed, performance-based approach.

The overall strategy in this plan is to develop ten (10) reports, each focusing on different aspects of the reference non-LWR fuel cycles. The 10 reports are as follows:

- Enrichment and UF_6 Handling up to 20-wt%
- TRISO Fuel Kernel Fabrication
- Uranium Metallic Fuel Fabrication
- Fast Reactor Metallic Fuel Assembly Fabrication
- Pebble TRISO Fuel Fabrication

- FHR Fuel Cycle Analysis
- HPR Fuel Cycle Analysis
- SFR Fuel Cycle Analysis
- HTGR Fuel Cycle Analysis
- MSR Fuel Cycle Analysis

Each report will describe scenarios, identify strategies to close capability gaps, and demonstrate through analysis the readiness of the NRC to review non-LWR fuel cycle activities. The effort described in this plan focuses on demonstration of NRC's capability to perform independent analyses in the areas of:

- Criticality safety
- Radionuclide inventory characterization
- Decay heat development
- Radiation shielding
- Radiological and non-radiological hazardous material and energy release and transport
- Characterization of consequences from radiological and non-radiological hazardous material and energy release into a facility and the environment

These analysis capabilities are relevant to assessment of hazards at all stages of the fuel cycle that could evolve into challenges to health and safety.

NRC non-LWR Vision and Strategy Volume 3 (2) and Volume 4 (3) are expected to cover fuel cycle needs. The primary need identified during preparation of this report is establishing NRC experience and analytical approaches for non-LWR fuel cycle safety analysis, and developing experience with the application of the SCALE and MELCOR computer code packages, and consequence analysis computer codes (see Volume 3 and Volume 4). Thus it is expected that the application of this work will lead to performance or usability updates to these codes to improve NRC efficiency in performing independent non-LWR fuel cycle safety analysis, along with publicly available input decks.

Lastly, it should also be understood that this work is being performed ahead of more information being provided by the DOE and industry. As new reactor design information becomes available it will be incorporated into the reports as appropriate. That said, sufficient information exists now to develop a reference plant for each reactor class, and Volume 3 will be leveraged for this purpose.

ABBREVIATIONS

Abbreviation	Definition
ATF	Accident Tolerant Fuels
BE	Back End
CFR	Code of Federal Regulations
DOE	Department of Energy
FE	Front End
FHR	Fluoride-salt-cooled High-temperature Reactor
FLiBe	Lithium fluoride-beryllium fluoride salt (take from ORNL/SPR-2018/987)
HALEU	High Assay Low Enriched Uranium ($^{235}U < 19.75$wt%)
HBU	High Burnup Up
HLW	High Level Waste
HTGR	Hight Temperature Gas Reactor
HVAC	Heating, Ventilation, and Air Conditioning
ICSBEP	International Criticality Safety Benchmark Evaluation Project
INL	Idaho National Laboratory
IRPhe	International Reactor Physics Benchmark Experiments
ISA	Independent Safety Analysis
LANL	Los Alamos National Laboratory
LPF	Leak Path Factor
LWR	Light Water Reactor
MSR	Molten Salt Reactor
non-LWR	Non-Light Water Reactor
PARCS	Purdue Advanced Reactor Core Simulator
SNF	Spent Nuclear Fuel
TRACE	TRAC/RELAP Advanced Computational Engine
TRISO	TRI-structural ISOtropic particle fuel

GLOSSARY

The following terms are defined for the purposes of this effort to reduce ambiguity in how they are used.

Term	Definitions as Used in This Report	See Also
Back End	Part of the fuel cycle after fuel utilization step	
Benchmark	A standard against which comparisons can be made*. *For example, a set of peer reviewed integral or separate effects experiments, such as those available in the IRPhe and ICSBEP	LA-13511 (July 1999)
Benchmarking	Establishing a predictable relationship between calculated results and reality. The main goal of benchmarking is to gain a quantitative understanding of the difference, or "bias," between calculated and expected results and the uncertainty in this difference (bias uncertainty). Also known as code or method "validation.".* * Implied in this process is the same use of computer code, hardware, cross section library, etc between the application case and similar benchmark(s)	NUREG-2215 (April 2020)
Bias	Gain a quantitative understanding of the difference between calculated and expected results	NUREG-2215 (April 2020)
Bias uncertainty	uncertainty in the difference when calculating bias	NUREG-2215 (April 2020)
Burnup Credit	For criticality safety analysis this is the allowance for the decrease in fuel reactivity resulting from irradiation. Accounts for the reduction of fissile materials and the accumulation of actinides and fission products that absorb neutrons according to approved methods provided in NRC-approved technical guidance documents.	NUREG-2215 (April 2020) 2012 ACRS Letter (ADAMS No: ML12261A186)
Cladding	The thin-walled metal tube that forms the outer jacket of a nuclear fuel rod. It prevents corrosion of the fuel by the coolant and the release of fission products into the coolant. Aluminum, stainless steel, and zirconium alloys are common cladding materials.	NRC Online Glossary

Term	Definitions as Used in This Report	See Also
Confirmatory analysis or calculations	Independent calculations performed by the NRC reviewer to confirm the adequacy of the applicant's analyses. These calculations do not replace, nor do they endorse, the applicant's design calculations.	NUREG-2215 (April 2020)
Defense-in-Depth	Defense-in-depth is defined as the application of successive compensatory measures to prevent accidents or to mitigate damage	ACRSR-1887 (ADAMS No: ML003705564)
Dry storage	The storage of SNF in a DSS, which typically involves drying the DSS cavity and backfilling with an inert gas.	NUREG-2215 (April 2020)
Front End	Part of the fuel cycle up to the fuel utilization step	
High Enriched Uranium or HALEU	High-enriched uranium means uranium enriched to 20 percent or greater in the isotope uranium-235	10 CFR Part 110.2 "Definitions" (December 2018)
High Level Waste	The highly radioactive materials produced as byproducts of fuel reprocessing or of the reactions that occur inside nuclear reactors. HLW includes: Irradiated spent nuclear fuel discharged from commercial nuclear power reactors The highly radioactive liquid and solid materials resulting from the reprocessing of spent nuclear fuel, which contain fission products in concentration (this includes some reprocessed HLW from defense activities and a small quantity of reprocessed commercial HLW) Other highly radioactive materials that the Commission may determine require permanent isolation	NRC Online Glossary (June 2020)
Light Water Reactor	A reactor design that uses light water as the heat transfer medium.	
Non Light Water Reactor	A reactor design that does not use light water as the heat transfer medium	
Package	When not "code packages," package means the packaging together with its radioactive contents as presented for transport	10 CFR 71.4 (December 2017)

Term	Definitions as Used in This Report	See Also
Packaging	Packaging means the assembly of components necessary to ensure compliance with the packaging requirements of this part. It may consist of one or more receptacles, absorbent materials, spacing structures, thermal insulation, radiation shielding, and devices for cooling or absorbing mechanical shocks. The vehicle, tie-down system, and auxiliary equipment may be designated as part of the packaging	10 CFR 71.4 (December 2017)
Spent nuclear fuel	Spent Nuclear Fuel or Spent Fuel means fuel that has been withdrawn from a nuclear reactor following irradiation, has undergone at least one year's decay since being used as a source of energy in a power reactor, and has not been chemically separated into its constituent elements by reprocessing. Spent fuel includes the special nuclear material, byproduct material, source material, and other radioactive materials associated with fuel assemblies.	10 CFR 72.3 (August 2017)
Storage	Storage means the temporary holding of radioactive material	10 CFR Part 110.2 "Definitions" (December 2018)
Validation	Validation is the process of comparing the software tool to experimental data that is similar to the application problem, which includes development of uncertainty information. Validation process uses the same software tool system (e.g.: code version, hardware/OS, and cross section library) between application and validation; this allows the use of uncertainty quantification developed from experimental data to be applied to the application problem. If sufficient validation cases are run, typically understood by the level of uncertainty that can be tolerated by the designer, the software tool may be considered benchmarked	LA-14167 (October 2004) DoD 5000.61 (October 2018) NUREG/CR-6361 (March 1997) NUREG/CR-6698 (January 2001) ANSI/ANS 10.4-2008 ANSI/ANS 8.1-2014 ANSI/ANS 8.24-2017

Term	Definitions as Used in This Report	See Also
Verification	verification is the process of checking that the software tool is functioning as intended, or is coded to faithfully represent the phenomelogical models. It includes the concept of unit testing along with integral testing of the tool. NOTE: As the codes become more complicated the verified pieces do not together mean a verified code	LA-14167 (October 2004) DoD 5000.61 (October 2018) NUREG/CR-6698 (January 2001) NSE 168, 128-137 (2011) ANSI/ANS 8.24-2017
VVUQ	See Verification and Validation	n/a

1. INTRODUCTION

1.1. Background and Purpose

This work has been initiated under the NRC's Implementation Action Plan as found in the 2017 report, "NRC Non-Light Water Reactor Near-Term Implementation Action Plans" (ADAMS ML No.: ML1765A069). In turn, the 2017 report is governed by the 2016 report "NRC Vision and Strategy: Safety Achieving Effective and Efficient Non-Light Water Reactor Mission Readiness (ADAMS ML No.: ML16356A670).

The 2017 plan divides into six strategies, of which strategy 2 is of focus here. Strategy 2 is intended to assess and develop Nuclear Regulatory Commission (NRC) tools so that they are available should a licensing evaluation or scoping study be needed.

This document is intended to provide a review of the existing nuclear fuel cycle, and a proposal to understand and prepare for anticipated impacts of non-Light Water Reactor (non-LWR) design requirements on the fuel cycle(s). Specifically the associated neutronics, radionuclide, and non-radionuclide characterization needs, and will include discussion of complimentary impacts from existing reactor designs which may use High Assay Low Enriched Uranium (HALEU) in the future.

Reviewing Non-LWRs is not a new activity for the Agency, with the regulatory capabilities stretching back to the Atomic Energy Commission. More information on the history of Non-LWRs from 1950 through 2019 may be found in a Brookhaven National Laboratory report (6). Further, the NRC has been involved in computer code development and assessment activities in this area and this can be seen from Volume 3 (2).

This plan will lead to "living" computer code assessments in the form of 10 reports that will cover the advanced reactor or non-LWR landscape. Currently reactor designs, as drivers for the fuel cycle, are continuing to develop their designs and neither the Department Of Energy (DOE) nor its industrial partners have provided detailed information non-LWR fuel cycle implications. Hence the content of these 10 reports may be updated as more information is provided by the DOE/industry. This is considered a low risk approach due to the flexibility of the computer codes used.

The NRC regulates the fuel cycle as encompassed in Title 10 of the Code of Federal Regulations Parts 1 through to 199. Licensees must meet, for example, criteria set forth in the U.S. Code of Federal Regulations (CFR) as:

- 10 CFR 36 "Licenses and Radiation Safety Requirements for Irradiators"
- 10 CFR 40 "Domestic Licensing of Source Material"
- 10 CFR 50 "Domestic Licensing of Production and Utilization Facilities"
- 10 CFR 52 "Licenses, Certifications, and Approvals for Nuclear Power Plants"
- 10 CFR 60 "Disposal of High-Level Radioactive Wastes in Geologic Repositories"
- 10 CFR 63, "Disposal of High-Level Radioactive Wastes in a Geologic Repository at Yucca Mountain, Nevada"
- 10 CFR 70 "Domestic Licensing of Special Nuclear Material"
- 10 CFR 71 "Packaging and Transportation of Radioactive Material"
- 10 CFR 72 "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste"

- 10 CFR 74 “Material Control and Accounting of Special Nuclear Material”

The essence of these regulations is captured by the NRC’s mission statement: to license and regulate the Nation’s civilian use of radioactive materials to provide reasonable assurance of adequate protection of public health and safety and to promote the common defense and security and to protect the environment. In another way, an aspect to the mission is to understand, control and predict the behavior of systems that contain radioactive or other hazardous material. This is supported through the use of fast, portable, well understood, appropriately assessed, and easy to use computer codes. These codes will be assessed in the application of this plan with regards to neutronics, radionuclide, and non-radionuclide hazard characterization.

The DOE and industry priorities in the nuclear energy sector have experienced significant evolution. Policy makers have recognized this, with Congress recently enacting or introducing legislation to promote the modernization of the nuclear energy sector.

- The Nuclear Energy Innovation Capabilities Act (NEICA), passed in September 2018, is focused on reducing barriers to, and creating infrastructure necessary for, advanced nuclear technology deployment.
- The Nuclear Energy Innovation and Modernization Act (NEIMA), passed in January 2019, promotes the modernization of the U.S. NRC to promote the certification and licensing of advanced reactor technologies.
- The Nuclear Energy Leadership Act (NELA), introduced in the House and Senate at the beginning of 2019, seeks to stimulate development of advanced reactors through demonstration and commercialization of new reactor designs.
- The Nuclear Energy Renewal Act (NERA), introduced in July 2019, seeks to support research and development (R&D) to achieve a more rapid deployment of advanced reactors, and preserve the currently operating domestic nuclear power plants.

In addition to the significant interest and development in the non-LWR space, activities to assess LWR technologies that could impact the nuclear fuel cycle are relevant to the effort in this plan. These include advances in LWR fuel technology, from candidate ATF to proposals for adoption of HALEU/HBU. Also of relevance to the effort established in this plan are evaluations to capture burnup credit in the back end. For example:

- The need for UF_6 feed at greater than 5% enrichment for LWR HALEU and ATF applications has direct overlap with non-LWR front end needs for increased enrichment.
- Although high burnup levels achieved in LWRs are a factor of 2 or 3 less than the intended non-LWR burnups, back end activities such as burnup credit and validation with sparse experimental data are relevant for both LWR HBU and non-LWR back end applications.

Beyond enhancements to LWR operations expected from the above examples of modifications to the nuclear fuel cycle, a range of new advanced reactor system concepts are under active development. As noted earlier, Congress has made it a national priority to deploy advanced reactor technologies to maintain and advance American leadership in nuclear energy. Of the advanced reactor concepts being developed, they all broadly rely on achieving economic and safety benefits through adoption of innovative technologies that include different heat transport fluids and moderators that are different from the traditional light water systems in existence today. These concepts under development are thus termed non-LWR.

These differences are illustrated in Table 1-1, which is intended to provide the reader an understanding of some of the fuel cycle drivers between existing LWR and proposed non-

LWRs. Clearly modifications to the fuel cycle will be necessary. These modifications will introduce potential scenarios not specifically present in LWR fuel cycle operations, which must be considered as part of an Independent Safety Analysis (ISA) for non-LWR fuel cycle activities.

Table 1-1. Comparison Between LWR and Non-LWR

Reactor Type	Enrichment	Fuel Form	Typical Discharge Burnup*	Fuel Residence Time	On-Site Fuel Processing	Fuel Storage / Transport
LWR (Ref.)	<5%	Fuel: U Oxide	Peak Rod Average: <62 GWd/MTU Max Assembly Average: <55 GWd/MTU	Assemblies burned for approximately 3 to 4 cycles	No	Storage: Fresh and spent fuel storage on-site or off-site Transport: FE: UF_6 solid transport in 30B cylinders, fresh fuel assemblies transportation packages; etc. BE: Used fuel transport and dry storage containers
LWR: HALEU /HBU (Ref.)	>5% <10%	Fuel: U Oxide	Peak Rod Average: ~75 Wd/MTU Max Assembly Average: ~60-70 GWd/MTU	Assemblies burned for approximately 3 to 4 cycles	No	
HPR	> 5% < 20%	Fuel: U Oxide U Metal	2-10 GWd/MTU	Up to 7yrs	No	To be evaluated
SFR	> 5% < 20%	U Metal	Up to 300 GWd/MTU	To be evaluated	No	To be evaluated
HTGR	> 5% < 20%	TRISO (UCO or UO2) in pebble bed or prismatic array	100-200 GWd/MTU	To be evaluated	No	To be evaluated
FHR	> 5% < 17%	TRISO (UCO or UO2) in pebble bed	100-200 GWd/MTU	To be evaluated	No	To be evaluated
MSR	> 5% < 20%	Fuel: ^{235}U dissolved in molten salt	To be evaluated	2-3yrs	Yes	To be evaluated

* 1 atom-% burnup is approximately 9.4 GWd/MTU.

As described previously, this document is a plan to demonstrate that NRC codes are ready to evaluate nuclear fuel cycles for non-LWRs. This document identifies research areas where non-LWRs may impact aspects of the fuel cycle not covered in volumes 1-4 of the NRC vision and strategy for non-LWR readiness. Of primary focus in this document is the computer code capabilities necessary to support characterization of the consequences from radiological and non-radiological hazardous material and energy releases. This is focused on the various stages of a modified nuclear fuel cycle, from the front end to the back end of the anticipated non-LWR fuel cycles.

1.2. Scope of Analytical Capabilities for Non-LWR Fuel Cycle Safety Assessment

1.2.1. Calculation Sequence for Non-LWR Fuel Cycle Safety Assessment

Activities are currently underway for each of the computer code packages identified under the NRC non-LWR Vision and Strategy Volume 3 (2) to:

- identify gaps in modeling capabilities for non-LWR concepts,
- close the identified modeling gaps, and
- perform analyses to demonstrate NRC readiness to perform independent non-LWR safety analyses.

Preliminary assessment of the modeling capability gaps indicate that much of the development effort currently being performed under the NRC non-LWR Vision and Strategy Volume 3 (2) will close anticipated modeling gaps for non-LWR fuel cycle safety analysis. Further the SCALE and MELCOR computer codes are already applied to a range of nuclear facility and fuel cycle operation safety assessments by both the NRC and the DOE. As such, they largely have the necessary capabilities for application to non-LWR fuel cycle safety analysis. Importantly, however, they possess the necessary pedigree for application to regulatory decision-making related to non-LWR fuel cycle safety.

The evaluation model in Volume 3, provided in Figure 1-1 for reference, is also useful here for analyses that require data transfer between the computer codes described in this work. The same type of information will be transferred between the codes. For this work, however, some criticality and shielding evaluations may require, as input, the calculated results from MELCOR. For example, radionuclide transport and deposition in HVAC filters could lead to potential radiation hazards to on-site workers.

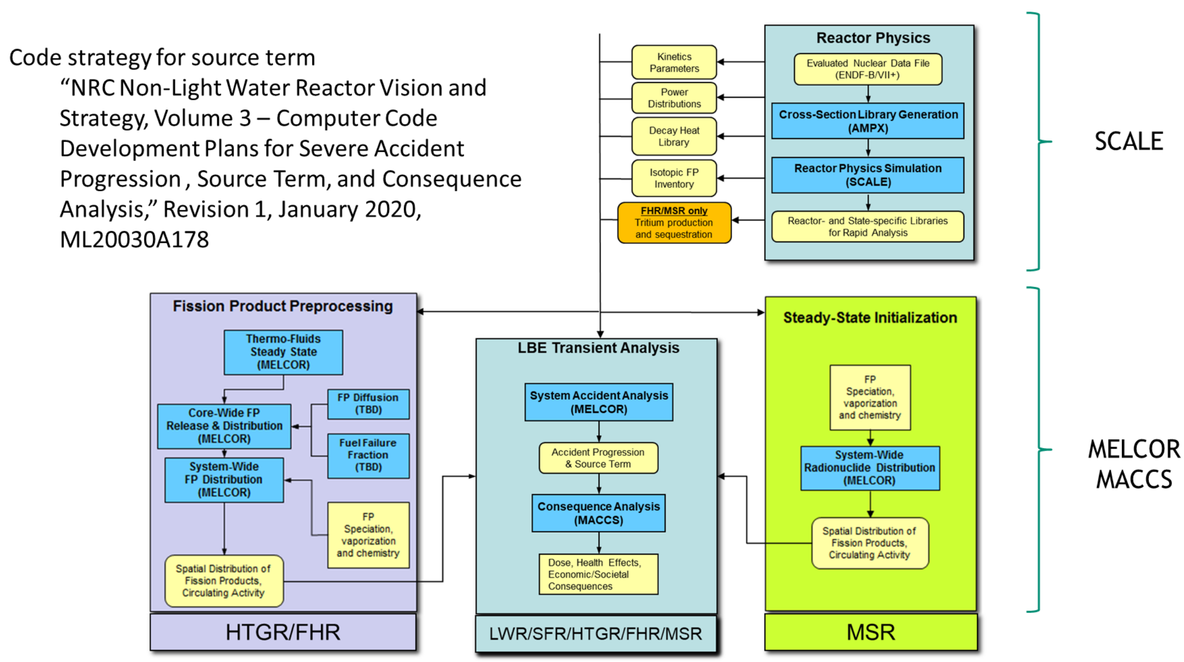


Figure 1-1. Evaluation Model Showing SCALE, MELCOR, and MACCS Calculation Sequence (2)

As illustrated in Figure 1-1, many of the traditional calculations required to assess consequences arising from nuclear reactor accidents are sufficiently captured through one-way flow of information between SCALE, MELCOR, and consequence analyses codes such as MACCS and RASCAL. As indicated earlier however, it is conceivable that radiation hazards may arise for on-site personnel due to suspended or deposited radioactive material that has been released and transported throughout a facility in an accident. In this situation, it is also important to evaluate the efficacy of shielding measures to radiation exposure of on-site personnel. To perform this type of calculation, it is necessary for SCALE to evaluate the transport of radiation using MELCOR estimates of suspended and deposited radioactive material resulting from a radiological release accident at a facility. This evaluation would involve a calculation sequence such as the following:

- SCALE evaluation of radioactive material inventory in sources releasing radioactive material under the postulated accident
- MELCOR evaluation of radioactive material transport and deposition throughout the facility
- SCALE evaluation of radiation or criticality given MELCOR-evaluated suspended and deposited radioactive material distribution in the facility

The output from accident progression and source term analyses conducted with SCALE and MELCOR will be used as input to consequence assessment codes.

1.2.2. Code Packages Applicable to Non-LWR Fuel Cycle Safety Assessment

SCALE is a state-of-the-art, modern, well-validated, user-friendly modeling and simulation suite for nuclear safety analysis and design. It is developed and maintained by Oak Ridge National Laboratory (ORNL) under contract with the NRC, U.S. Department of Energy, and the National Nuclear Security Administration (NNSA) to cover neutronics applications in such areas as reactor physics, criticality safety, radiation shielding, time-dependent inventory calculations or depletion, and sensitivity and uncertainty analyses. SCALE includes nuclear data libraries assessed for use in LWR and non-LWR applications, and nuclear data processing tools should the user find that new continuous energy (CE) and multigroup (MG) data are needed for their application case.

MELCOR is a state-of-the-art, modern, computer code developed by Sandia National Laboratories (SNL) for the NRC to perform accident progression (including reactor, containment, spent fuel pools, etc.), and source term analyses. MELCOR is a flexible, integrated computer code designed to characterize and track the evolution of severe accidents, and the transport of associated radionuclides within a confinement such as a containment or building. It is a knowledge repository comprised of a multitude of experiments and corresponding model development, with particular focus on LWR phenomenology as well as extended capabilities for non-LWR technologies.

For both SCALE and MELCOR, pertinent data needs, development and assessment activities are addressed in Volume 3 of the NRC non-LWR Vision and Strategy (2).

1.2.3. Technical Approach

The effort described in this plan focuses on demonstration of NRC's capability to perform independent analyses, for non-LWRs, in the areas of:

- Criticality safety
- Radionuclide inventory characterization
- Decay heat generation
- Radiation shielding
- Radiological and non-radiological hazardous material and energy release and transport
- Characterization of consequences from radiological and non-radiological hazardous material and energy release into a facility and the environment

Initial activities will be to assess and understand the performance of the SCALE and MELCOR computer codes against a range of scenarios. The scenarios will be developed using NUREG/CR-6410 (7) and DOE hazard and accident analysis handbook (8) as guidance, and documented in the 10 reports described earlier. The reports will include benchmarking of the codes against applicable experiments where possible. The input decks will be made publicly available with an eye to flexibility as possible to accommodate changes.

For each non-LWR fuel cycle considered, a report will utilize the reference fuel cycles and stages described in Section 3. Associated models will be developed to perform a range of analyses using existing NRC codes, relating analyses where possible to the LWR equivalents. Results and methods developed as a result of other non-LWR NRC work will be relied upon heavily for spent fuel isotopic distributions and severe accident analysis. If processes and tools are already adequately described in another NRC document, that document will simply be referenced. For example, with the utilization phase for HTGR, the proposed fuel cycle assessment report for HTGR will summarize work performed during volume 3 activities.

Sensitivity analysis and uncertainty quantification are key components of this work, applied to the specific reference Non-LWR fuel cycle to give NRC staff an idea of uncertainties they may encounter and how to use NRC tools to investigate them. For example, given fuel pebbles passing through a specific HTGR design seven times before discharge with a given probabilistic trajectory through the core, what is the range of burnup and decay heat possible? During the development of each Non-LWR fuel cycle assessment report, preliminary studies will be used to drive investigations of sensitivity/uncertainty, focusing on those which have the most impact from a regulatory point of view.

Due to the focus of this document it will not consider the decommissioning or activation of structural components, lifetime of materials, etc. The reports will also not assess economics or compare systems in terms of safeguards, accountability, or proliferation risk. They will contain the guiding principles to support NRC staff in using existing NRC codes to perform confirmatory analysis of Non-LWR fuel cycle related safety issues.

1.3. Report Structure

Section 2 of this report discusses the application of these codes to safety assessment of the current LWR fuel cycle. This provides the background against which the current fuel cycle analysis approach will be applied. To identify how the analysis capabilities must evolve to support safety analyses of non-LWR fuel cycle operations, it is necessary to specify the current understanding of likely forms of the non-LWR fuel cycles for different non-LWR concepts. The current understanding of the likely types of non-LWR fuel cycles is presented in Section 3, which classifies types of potential non-LWR fuel cycles to develop classes of safety assessment readiness requirements. Section 4 provides the approach for identifying modeling gaps, closing modeling gaps, and performing demonstration analyses to establish independent NRC non-LWR fuel cycle safety analysis readiness.

2. LWR FUEL CYCLE SAFETY ASSESSMENT

This section provides information related to the current NRC capabilities to perform independent LWR fuel cycle safety analyses. It supports the identification of research areas related to how potential non-LWR fuel cycles require evolution of existing fuel cycle analysis capabilities. The document addresses areas of analytical capabilities not explicitly identified in Volumes 1 to 4 of the NRC non-LWR vision and strategy.

2.1. LWR Fuel Cycle

The nuclear fuel cycle is typically categorized as either an open or closed cycle, however there are also modified fuel cycles which are between the open and closed approaches and are applicable to some Non-LWRs, such as those that employ the breed and burn or accelerator driven approaches. All cycles begin with the extraction of ore and ends with disposal. The economics of the commercial fuel cycle is determined with the power generation step. The US operates with a "open" fuel cycle meaning that residual ^{235}U and other fissile isotopes such as ^{239}Pu are not recycled through reprocessing.

Figure 2-1 presents the open nuclear fuel cycle used in the United States. The existing fuel cycle is the result of decades of development and refinement based around a ceramic uranium fuel enclosed in a zirconium-based cladding.

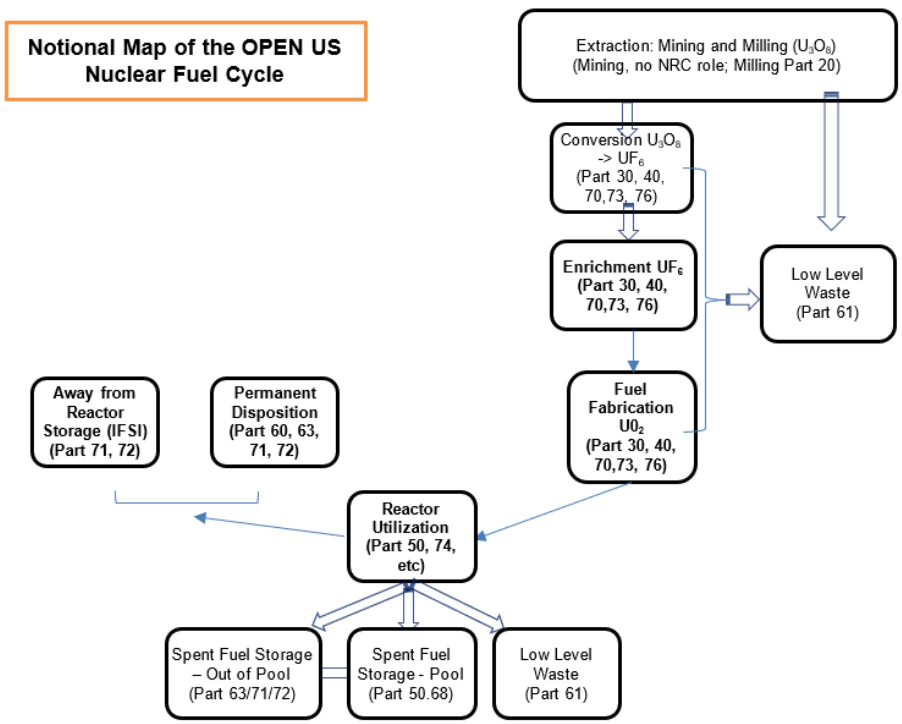


Figure 2-1. Map of US Open Nuclear Fuel Cycle

Prior to and following reactor utilization, various radiological and non-radiological hazards arise as part of fuel cycle operations. Under accident conditions, these hazards can lead to health and safety consequences to on-site workers or to the public off-site. These hazards can broadly be classified into the following classes:

- Radiation hazards that arise due to inadvertent high energy particle generation (i.e., neutron, gamma, alpha and beta particles)
- Radiological hazards that arise due to the radionuclide release into and transport through the atmosphere within a facility enclosure or in the environment
- Non-radiological (chemical) hazards that arise due to non-radioactive, toxic vapor and gas release into and transport through the atmosphere within a facility enclosure or in the environment

Traditionally, the consequences to health and safety from these hazard classes have been quantitatively modeled using a range of methods from:

- Application of sophisticated computer code packages, such as SCALE for criticality safety and shielding
- Analytical methods implemented in special-purpose computer programs or other special-purpose engineering calculations tools

There are many state-of-practice methods available at NRC to characterize these hazards.

- SCALE provides a range of verified and validated capabilities to assess radiation hazards and provide the initial radionuclide content used in assessment of radiological hazards
- MELCOR has a range of verified and validated capabilities for application to assessment of how radiological and non-radiological hazards evolve within a facility and estimate how these hazards propagate into the environment under accident conditions

SCALE has found significant application for a range of fuel cycle assessment needs at the NRC related to radiation hazards. This code package enables NRC to perform assessments related to criticality safety, including burnup credit, and shielding. These assessments reflect the significant need to perform ongoing evaluations of how operational changes impact a range of neutronic safety metrics. Specific examples of the types of calculation sequences supported by SCALE are:

- SCALE/CSAS is the calculation sequence used for criticality safety evaluations
- SCALE/MAVRIC is the calculation sequence used for shielding evaluations
- SCALE/TRITON, SCALE/Polaris, and SCALE/ORIGAMI are the calculation sequences used to evaluate isotopic evolution and decay heat

In-reactor and SFP safety assessments are performed utilizing tools such as SCALE/Polaris or SCALE/TRITON, PARCS, TRACE, and FAST. SCALE/ORIGEN is used internally by SCALE/Polaris and SCALE/TRITON to calculate the isotopic evolution of the fuel during irradiation. Through special methods and libraries generated by SCALE/Polaris or SCALE/TRITON, SCALE/ORIGAMI can re-generate spent fuel isotopics for arbitrary operating histories. This approach with SCALE/ORIGAMI is typically used to generate initial conditions for severe accident analyses with MELCOR, currently used in such areas as reactor core and for the SFP.

Once discharged, the various storage modes such as SFP, on-site and off-site dry-cask storage (e.g., ISFSI) etc. must be modeled in terms of decay heat, shielding, and criticality. This is typically done with ORIGAMI for the spent fuel sources and CSAS or MAVRIC for criticality safety and shielding. Figure 2-2 shows a SCALE model of an ISFSI, presented here to illustrate current SCALE modeling capabilities that can be extended to non-LWR fuel cycle safety analyses.

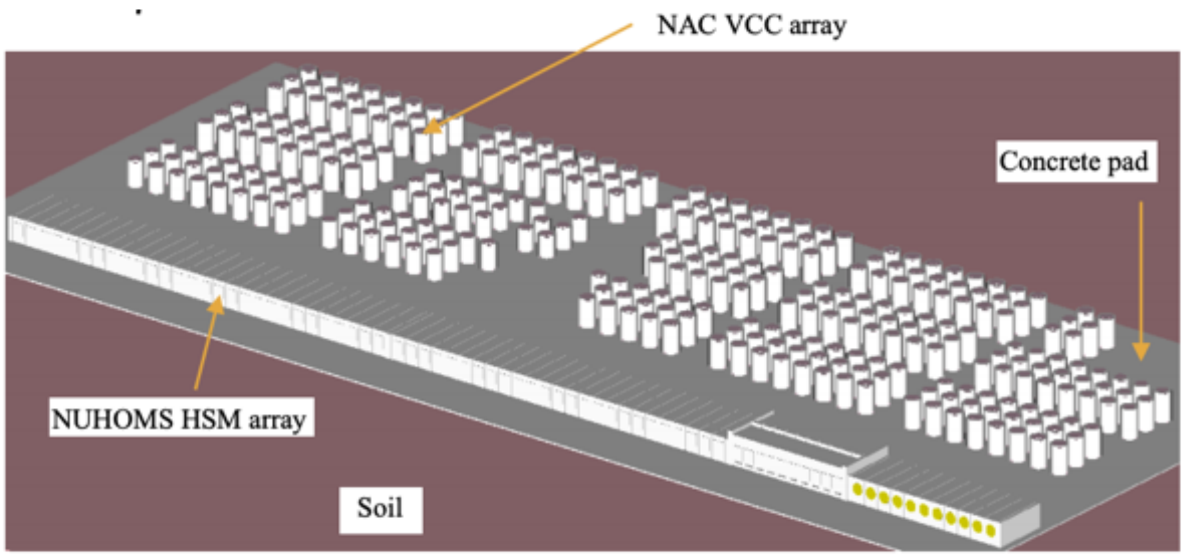


Figure 2-2: SCALE model of portion of an ISFSI[1].

Long-term geologic storage modeling requires similar treatment as the short/mid-term storage listed above, with the addition of criticality consequence modeling, induced by water ingress or cask deformation.

For many of the other operations associated with the LWR nuclear fuel cycle, existing analyses of radiological and non-radiological hazards continue to provide sufficient safety evaluations under any operational change in the LWR fuel cycle. NUREG/CR-6410 (7) provides the traditional NRC methodology for performing safety assessments for LWR fuel cycle facilities.

Many of the methods presented in NUREG/CR-6410 (7) pre-date the development and maturation of NRC computer codes like SCALE and MELCOR. Under circumstances where refined analyses are required, MELCOR has been applied, for example, to better understand the consequences from release and transport of radionuclide or chemically toxic vapors and gases within the atmosphere of a facility enclosure. An illustration of this type of application of MELCOR by the NRC is illustrated in the NRC safety assessment of the Barnwell facility (9).

MELCOR is also used to quantitatively characterize the amount of material that can leave a facility, passing through protective barriers, and enter the environment. MELCOR complements this modeling of hazardous material and energy transport with models that can evaluate the potential for thermal-mechanical loading of structures that serve as barriers to hazardous material and energy release. For example, MELCOR can be applied to the assessment of hydrogen build-up and combustion that causes a tank rupture that leads to hazardous material and energy release, which could also be important for non-LWR safety analyses. This type of calculation is termed leak path factor (LPF) analysis, and is commonly applied to DOE nuclear facilities. A discussion of the overall application of MELCOR to DOE nuclear facility safety assessment is provided in MELCOR leak path factor guidance (10). Included in this guidance is an assessment of the validation basis for MELCOR application to facility safety assessment.

[1] WCS Consolidated Interim Storage Facility Safety Analysis Report, Docket Number 72-1050, Revision 3, Interim Storage Partners, LLC (2020).

MELCOR characterization of radiological and non-radiological hazards relevant to nuclear facilities is supported by an extensive body-of-knowledge. The DOE hazard and accident analysis handbook (8) has compiled the state-of-practice technical basis for identifying hazards and analyzing their implications under accident conditions. This DOE handbook (8) standardizes the extensive body-of-knowledge on release of radionuclide and chemically toxic vapors and aerosols, as well as energy, for the spectrum of accidents to which any nuclear facility operation is subject. This is directly applicable to NRC needs in the area of fuel cycle safety assessment, specifically the non-LWR fuel cycle where an established safety basis does not currently exist. It directly supports the application of MELCOR in performing these safety assessments, providing the necessary technical basis for hazard identification, accident scenario definition, and hazardous material/energy release. This information specifies the initial and boundary conditions necessary for MELCOR to perform an assessment of facility response under accidental conditions.

2.2. Examples of Fuel Cycle Safety Assessments

The aforementioned NRC code packages have been utilized in a number of accident and consequence studies of non-reactor radiological sources. These include assessment of accidents involving radiological release from:

- Spent fuel in on-site spent fuel pools – In light of Fukushima Daiichi, the NRC assessed the consequences from radiological release accidents for an on-site spent fuel pool. This effort aligned with the recommendations of the National Academy of Sciences study of lessons learned from the Fukushima Daiichi accident relevant to improving safety and security at U.S. nuclear power plants (11). In support of this post-Fukushima evaluation of U.S. nuclear power plant safety, the NRC conducted SCALE/MELCOR/MACCS assessments of progression and consequences from loss-of-cooling and loss-of-coolant accidents in an on-site spent fuel pool in NUREG-2161 (12). This effort aligned with the state-of-the-art knowledge on the progression and consequences of spent fuel pool accidents that developed through international collaborations (e.g., the OECD/NEA collaborations under the "Status Report on Spent Fuel Pools under Loss-of-Cooling and Loss-of-Coolant Accident Conditions" (13) and "Phenomena Identification and Ranking Table: R&D Priorities for Loss-of-Cooling and Loss-of-Coolant Accidents in Spent Nuclear Fuel Pools (14)).
- Reprocessing facilities – MELCOR evaluation of accident source terms for the Barnwell spent nuclear fuel reprocessing facility was performed for the NRC by SNL in 2019. This study is documented in Reference (9). It evolved from a previous study to identify the accident phenomena relevant to evaluating accident source terms for facilities involved in spent fuel reprocessing. This review was performed for the NRC by SNL in 2017 and is documented in Reference (15).
- Vogtle full-scope site Level 3 PRA (16) performed by the NRC evaluates risk from sources of radioactivity beyond the reactor using the SCALE/MELCOR/MACCS code packages.
    - Spent fuel pool
    - Dry cask storage
    - Integrated site

As described above, the experience gained and the analyses conducted for LWR fuel cycle safety analyses provide a good context for identifying modeling gaps, closing modeling gaps, and demonstrating through analyses independent NRC non-LWR fuel cycle safety analyses capability. The capability areas for LWR fuel cycle safety analyses are identified above: criticality safety, radionuclide inventory characterization, decay heat development, radiation shielding, radiological and non-radiological hazardous material/energy release and transport, and assessment of consequences from radiological and non-radiological hazardous material release inside a facility or to the environment. These capability areas are the same as required for performing non-LWR fuel cycle safety analyses.

3. NON-LWR FUEL CYCLE CLASSIFICATION

Each Non-LWR reactor design implies a fuel cycle as a result of that particular design. Within this work, “non-LWR fuel cycle” refers to a specific postulated fuel cycle for a specific reference plant designs represented by different classes of non-LWR systems (HTGR, FHR, HPR, SFR, MSR, etc.). These reference plant designs will be developed using publicly available information (i.e., currently being researched in Volume 3). Given that Volume 3 activities precede this work, any changes in reference plant model designs pursued in Volume 3 will lead to updates to this Volume 5 activities. The expectation is that the generic design for a type of technology will be sufficiently representative of the vendor designs that the NRC may see in the future for that class.

This plan provides a framework for achieving NRC readiness to regulate operations of specific non-LWR fuel cycles. To achieve this, classes of potential non-LWR fuel cycles are developed in this report for the purposes of structuring assessments needed to demonstrate overall regulatory readiness. These classes are comprised of distinct stages and operations in a fuel cycle, each with distinct hazards and accident scenarios for which assessment capabilities will be demonstrated.

This work contrasts, but compliments, the approach being taken by the DOE-NE Office of Fuel Cycle Technologies (17). The current DOE efforts are focused on research into categorizing and evaluating a broad range of fuel cycle options, with an overall goal of supporting technology selection. The work conducted by Wigeland et al. (17) assessed 40 fuel cycle evaluation groups in terms of the following criteria:

- Nuclear Waste Management (Study considered waste generation only)
- Proliferation Risk
- Nuclear Material Security Risk
- Worker and public safety
- Environmental Impact
- Resource Utilization
- Development and Deployment Risk
- Institutional Issues
- Financial Risk and Economics

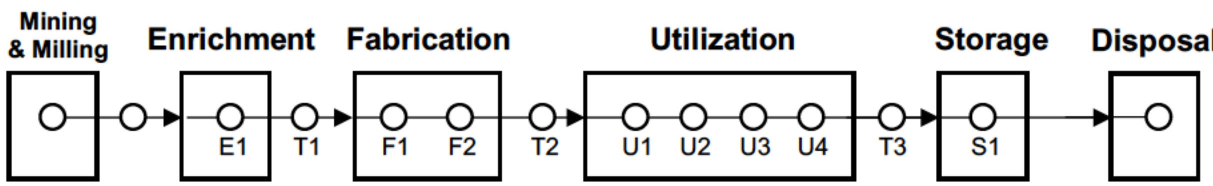
Though the work described in reference (17) is distinct from the strategic considerations of the current report, the work in this report could compliment DOE’s approach by establishing a plan to demonstrate readiness to regulate specific non-LWR fuel cycle operations, thus potentially being a source of collaboration. This section is thus focused on presenting a simplified framework for classifying non-LWR fuel cycle operations into a manageable set of operational conditions to conduct demonstration studies that illustrate criticality safety, shielding, depletion, and accident analysis capabilities.

As a starting point, the non-LWR fuel cycle will be compared against the LWR fuel cycle which is segmented into stages as shown in Figure 3-1. Stages in the fuel cycle are designated using a letter. Sub-stages are designated by a number. For example, the utilization stage of the standard LWR fuel cycle, designated by the letter U, has 4 sub-stages:

1. Fresh fuel staging and loading (U1)
2. Power production (U2)
3. Spent fuel pool/shuffle operations (U3)

4. On-site dry storage (U4)

For each non-LWR type, described in the next section, the sub-stages will be redefined to match anticipated fuel cycle operations specific to that reactor type. Note that mining and milling and final geological disposal are not given labels and are not considered in this plan.



E1 – UF_6 enrichment
T1 – transportation of UF_6 to fuel fabrication facility
F1 – fabrication of UO_2 fuel pellets
F2 – fabrication of LWR fuel assemblies
T2 – transportation of fresh fuel assemblies to the plant
U1 – fresh fuel staging and loading
U2 – power production
U3 – spent fuel pool/shuffle operations
U4 – on-site dry cask storage
T3 – transportation of spent fuel to off-site storage
S1 – off-site storage

Figure 3-1. Stages of the LWR Fuel Cycle with UO_2 Fuel

3.1. Reactor Types

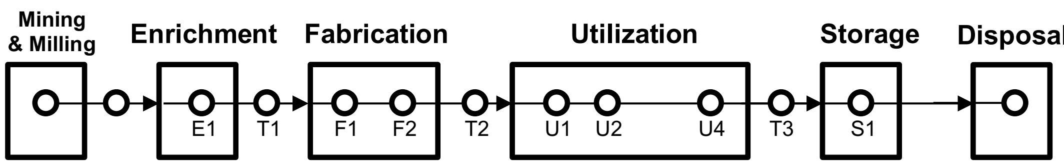
This section identifies the different reactor types considered to identify distinct classes of non-LWR fuel cycle operations. All fuel cycles are assumed to begin with a fresh source of UF_6 feed, with enrichments up to 20 w/o ^{235}U. Further discussion of the assumed form of this UF_6 feed is provided in the approach in Section 4. The following provides a description of the reactor designs that will be used as drivers. Operations that involve recycling of spent fuel are specific to fast non-LWR concepts. Additional considerations are introduced under the SFR type that account for the presence of recycled Pu in the fuel cycle, specifically focusing on how the radionuclide inventory available for release is altered.

3.1.1. High-Temperature Gas-cooled Reactor

The High-Temperature Gas-cooled Reactor (HTGR) fuel is assumed to be in the form of TRISO kernels. These kernels are assumed to be manufactured using a UF_6 feed that is enriched up to 20% ^{235}U. The stages of the fuel cycle, as depicted in Figure 3-2 for a pebble bed core and Figure 3-3 for a prismatic core, are assumed to be the following. In general the fuel cycle steps will consist of the following.

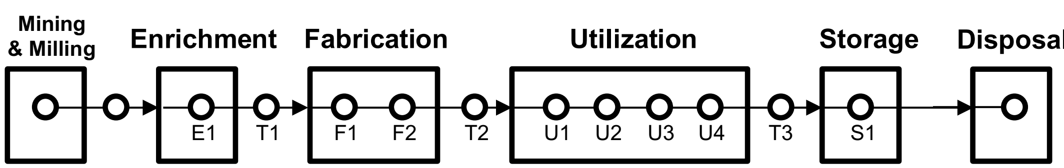
1. UF_6 feed is used for production of either UCO or UO_2 (TRISO) kernels (F1).
2. Either compacts (prismatic reactor) or pebbles (pebble reactor) are fabricated from graphite and TRISO kernels (F2).
3. Fuel compacts or pebbles are loaded in the reactor system and irradiated (U1 and U2).
4. For the prismatic reactor, core shuffle operations and discharge to temporary storage are performed (equivalent of LWR spent fuel pool) (U3).
5. Fuel is discharged to on-site spent fuel storage (U4).
6. Fuel is transported (T3) to off-site storage (S1).

The primary difference between the pebble bed core fuel cycle (Figure 3-2) and the prismatic core fuel cycle (Figure 3-3) is at the utilization stage. For the pebble bed core, in-use fuel handling, inspection, and recycling operations are performed which eliminates the core shuffle and spent fuel storage operations (U3) present in the prismatic core (and in the equivalent LWR operations).



E1 – UF_6 enrichment
T1 – transportation of UF_6 to fuel fabrication facility
F1 – fabrication of TRISO fuel kernels
F2 – fabrication of fuel pebbles
T2 – transportation of fresh fuel pebbles to the plant
U1 – fresh fuel staging and loading
U2 – power production including online refueling
U4 – on-site discharged pebble storage
T3 – transportation of spent fuel to off-site storage
S1 – off-site storage

Figure 3-2. Fuel Cycle Stages for a Pebble Bed HTGR utilizing TRISO Fuel



E1 – UF_6 enrichment
T1 – transportation of UF_6 to fuel fabrication facility
F1 – fabrication of TRISO fuel kernels
F2 – fabrication of prismatic fuel assemblies
T2 – transportation of fresh fuel assemblies to the plant
U1 – fresh fuel staging and loading
U2 – power production
U3 – spent fuel zone/shuffle operations
U4 – on-site spent fuel storage
T3 – transportation of spent fuel to off-site storage
S1 – off-site storage

Figure 3-3. Fuel Cycle Stages for a Prismatic HTGR utilizing TRISO Fuel

For this work, the demonstration system for the HTGR is the PBMR-400 (400 MWth) pebble bed reactor as used in Volume 3. The pebble bed reactor is a more challenging case in terms of fuel management: there are a large number of pebbles in the core (>400,000) each with 10s of thousands of TRISO particles; and the reactor as designed must refuel on-line. The prismatic HTGR is more similar to an LWR (or other assembly-based systems) where fuel movement is an infrequent, operator-driven action.

The capabilities to characterize HTGR neutronic performance, radionuclide inventory, and decay heat have been established through activities identified under the NRC non-LWR vision and strategy, Volume 3 (2). Figure 3-4 illustrates a SCALE model of the demonstration PBMR-400 developed under one of the activities planned in Reference (2). Under the activities planned in Reference (2), complementary MELCOR models to assess accident progression and source terms due to radionuclide release from a reactor have also been developed.

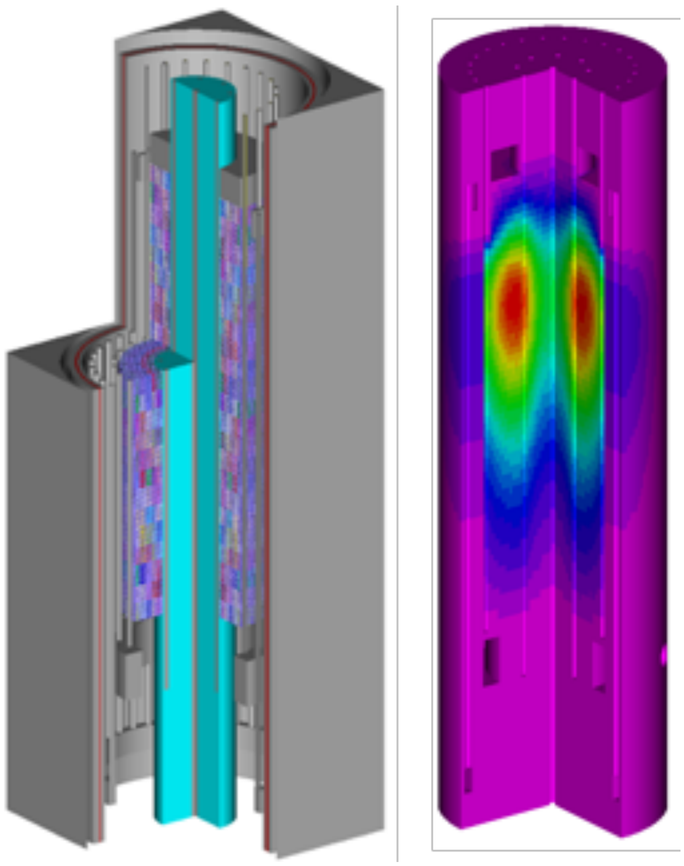


Figure 3-4. SCALE model of PBMR-400 HTGR (left) with neutron flux (right).

Figure 3-4 shows an equilibrium PBMR-400 core with inner reflector (blue), outer reflector (grey), and multiple pebble fuel depletion zones in the annular region (multi-colored). The flux is higher near the top of the core where there is more fresh fuel. Given a discharge burnup of 90 GWd/MTU, with fuel pebbles achieving roughly 15 GWd/MTU burnup per pass in this core, the top of the equilibrium core is at an average of 30 GWd/MTU and the bottom 45 GWd/MTU.

3.1.2. Fluoride-salt-cooled High-temperature Reactor

The Fluoride-salt-cooled High-temperature Reactor (FHR) fuel forms are assumed to use fuel fabricated from TRISO kernels in a manner similar to the HTGR. The fabrication process is assumed to be similar to that adopted for the HTGR demonstration system. Unlike the HTGR however, the FHR uses a pebble with a graphite core surrounded by TRISO kernels, and a fluoride molten salt instead of an inert gas (helium) as the reactor heat transport fluid. The fuel cycle stages for an FHR, as depicted in Figure 3-2, are identical to the pebble bed HTGR. In general the fuel cycle steps will consist of the following.

1. UF_6 feed is used for production of either UCO or UO_2 (TRISO) kernels (F1).

2. Either compacts (prismatic reactor) or pebbles (pebble reactor) are fabricated from graphite and TRISO kernels (F2).
3. Fuel compacts or pebbles are loaded within the reactor system and irradiated (U1 and U2).
4. For the prismatic reactor, core shuffle operations and discharge to temporary storage are performed (equivalent of LWR spent fuel pool) (U3).
5. Fuel is discharged to on-site spent fuel storage (U4).
6. Fuel is transported to off-site storage or final repository (T3).

The demonstration system for the FHR is the Berkeley Mk. 1 pebble bed reactor (236 MWth). The FHR should consider additional modes of fission product inventory migration within the coolant compared to the HTGR, as well as activation of the FLiBe coolant which produces tritium.

The capabilities to characterize FHR neutronic performance, radionuclide inventory, and decay heat have been established through activities identified under the NRC non-LWR vision and strategy, Volume 3 (2). Figure 3-5 illustrates the SCALE model of the demonstration FHR system. On the left is shown a side-view of the annular FHR core, with multi-colored regions assigned to track fuel reaction rates in the various axial and radial zones. Under the activities planned in Reference (2), complementary MELCOR models to assess accident progression and source terms due to radionuclide release from a reactor have also been developed. Note that this particular FHR uses annular pebbles although it is by no means a requirement for FHRs. In addition there is no currently proposed prismatic FHR, although there is nothing by definition which says FHR must be pebble-based.

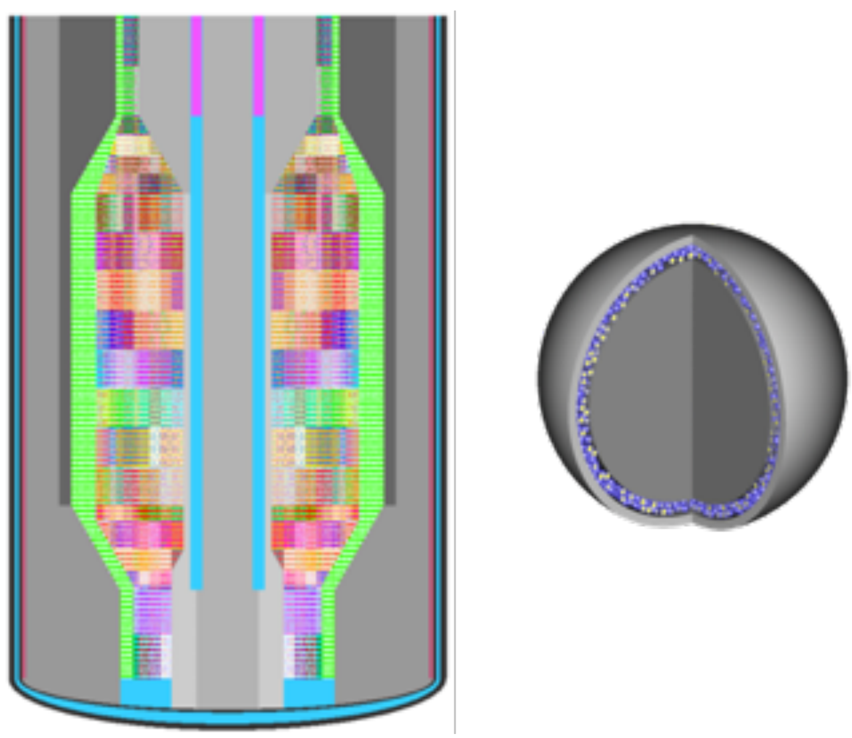


Figure 3-5. SCALE model of Berkeley Mk. 1 FHR and annular fuel pebble

3.1.3. Sodium Fast Reactor

The Sodium Fast Reactor (SFR), and other fast-spectrum reactors, have been proposed to use a wide variety of fuel forms. Examples of the range of proposed fuel form include

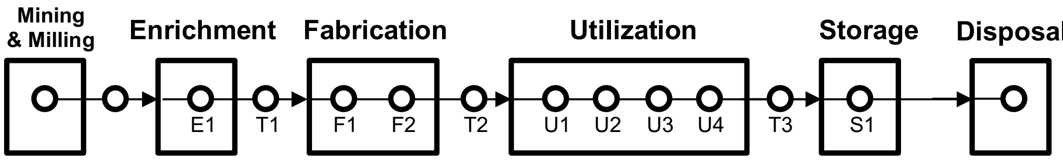
- Oxides
- Carbides
- Nitrides
- Metals using fissile U and Pu (or a mixture of the two)

The demonstration systems for this class are either the OECD/NEA benchmark of the MET-1000 (1000 MWth) concept (18) or a simplified version of the Versatile Test Reactor (VTR). With the MET-1000, this system was specified to use transuranic (TRU)-Zr-Mo metallic fuel. For assessing fuel cycle capability readiness, however, a pure U metallic fuel is considered. The fuel is assumed to have the same level of enrichment as in the MET-1000 benchmark specification. The limitation for this type of fuel is primarily based on the perspective that no fuel cycle will have a front-end that supports recycling in the near-term.

The VTR is to be a smaller (about 300 MWth) version of the GE Hitachi PRISM power reactor, which builds on the EBR-II, an integral sodium-cooled fast reactor prototype that operated at Argonne National Laboratory from 1963 to 1994. VTR, like PRISM, would use metallic alloy fuels. The VTR will provide irradiation services important for science and technology areas including testing and qualification of advanced reactor fuels, innovative structural materials, and instrumentation. Experimental data produced from the VTR will also support validation of advanced modeling and simulation tools.

The fuel cycle stages to be analyzed for the metallic SFR, as depicted in Figure 3-6, are as follows.

1. UF_6 feed is used for production of Uranium metallic fuel slugs (F1).
2. Fuel rods are fabricated from multiple slugs, with the typical process using a sodium bond between slug and inner clad wall (F2).
3. Fuel elements are fabricated from multiple rods, with wire wraps for rod spacing and an outer wrap called a duct which creates an independent cooling channel for each assembly similar to a BWR channel box (F2).
4. Fuel elements are loaded in the reactor and irradiated (U1, U2, and U3).
5. Fuel is discharged to on-site spent fuel storage (U4).
6. Fuel is transported to off-site storage or final repository (T3).



E1 – UF_6 enrichment
T1 – transportation of UF_6 to fuel fabrication facility
F1 – fabrication of metallic fuel
F2 – fabrication of SFR fuel assemblies
T2 – transportation of fresh fuel assemblies to the plant
U1 – fresh fuel staging and loading into core
U2 – power production
U3 – spent fuel/shuffle operations
U4 – on-site storage
T4 – transportation of spent fuel to off-site storage
S1 – off-site storage

Figure 3-6. Stages of the Fuel Cycle for an SFR with Metallic Fuel

Figure 3-7 and Figure 3-8 show the radial fueling layout of the OECD/NEA MET-1000 and VTR, respectively. The sodium bond is an interesting design feature in metallic fast reactors, with the main purpose to ensure good contact between metallic fuel and clad which increases thermal conductivity and decreases the maximum fuel temperature. The fuel rods with internal sodium bond are hermetically sealed and so there is no opportunity for sodium to come into contact with air or moisture and catch fire, under normal operational conditions. However, some accident analysis may be required for accidental conditions causing cladding failure during fabrication or transportation. New processes are being investigated as part of the VTR program that would extrude fuel and cladding together which ensures good thermal conduct without need for a sodium bond (19).

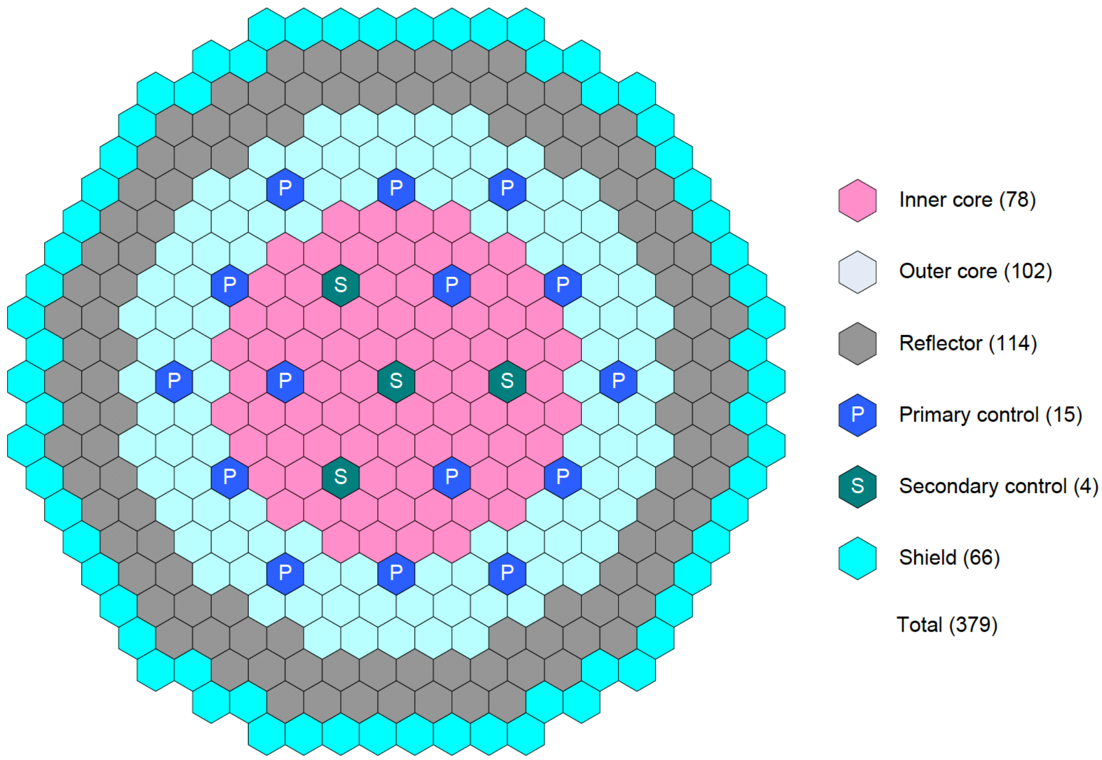


Figure 3-7. Radial layout of the MET-1000 SFR (18)

Parameters	Value
Core Power (MWth)	300
Peak Fast Neutron Flux (n/cm2 s)	>4.0×10^{15}
Number of Fuel Assemblies	66
Number of Radial Reflector Assemblies	114
Number of Shield Reflector Assemblies	114
Assembly Length (m)	3.53
Control Rods (Control + Safety)	6 + 3
Assembly pitch (cm)	12
Fuel Height (cm)	80
Plenum Height (cm)	80

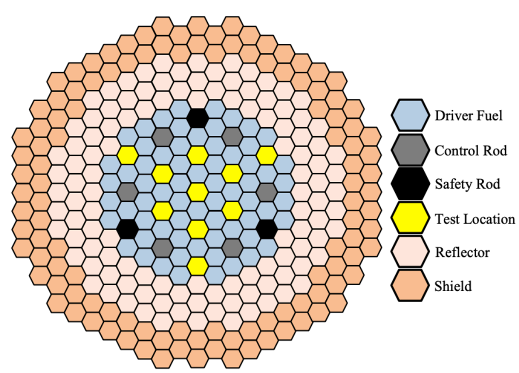


Figure 3-8. Radial layout of the VTR (20)

3.1.4. Heat Pipe Reactor

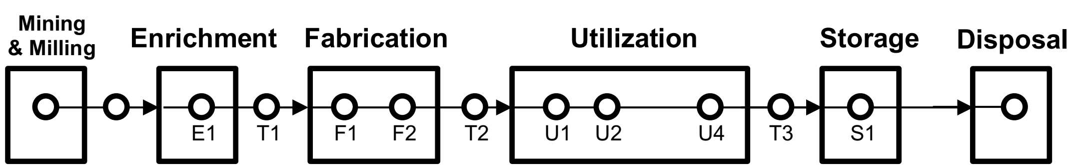
The Heat Pipe Reactor (HPR), like the SFR, is a fast-spectrum system. The HPR fuel cycle shares significant overall similarities to the SFR, but it would operate at a smaller scale due to the reduced size of these proposed reactor concepts. The fuel cycle stages for a uranium metal-fueled HPR, as depicted in Figure 3-9, are as follows.

1. UF_6 feed is used for fabrication of uranium metallic fuel rods (F1).
2. Fuel rods are fabricated from multiple slugs, with the typical process using a sodium bond between slug and inner clad wall (F1).
3. Fuel rods are assembled into the reactor configuration and irradiated (F2, T2, U1, and U2).

4. Fuel is discharged to on-site spent fuel storage (U4).
5. Fuel is transported to off-site storage or final repository (T3).

The SFR and HPR fuel cycles are essentially the same in the front-end except for with the SFR, assemblies are fabricated for traditional power reactor batch loading schemes whereas with the smaller HPR the entire core is assembled. However, these fuel cycles will have different back-ends due to the large difference in discharge burnups. The target HPR is a modification of the INL design A (21), which is based on the LANL Megapower concept (22). The INL design A concept (21) was originally developed utilizing UO_2 fuel with a discharge burnup of roughly 2 GWd/MTU. However, the demonstration system considered for fuel cycle capability readiness assessments will deviate from the INL design A concept to better represent current proprietary HPR designs (e.g., Oklo and eVinci). The demonstration system for these fuel cycle readiness assessments will use metallic fuel up to a discharge burnup of 10 GWd/MTU. This demonstration system will be termed the “INL design A-MET” variant.

The capabilities to characterize HPR neutronic performance, radionuclide inventory, and decay heat have been established through activities identified under the NRC non-LWR vision and strategy, Volume 3 (2). Figure 3-10 shows the SCALE model of the INL design A demonstration system. Under the activities planned in Reference (2), complementary MELCOR models to assess accident progression and source terms due to radionuclide release from a reactor have also been developed.



E1 – UF_6 enrichment
T1 – transportation of UF_6 to fuel fabrication facility
F1 – fabrication of metallic fuel
F2 – fabrication of HPR core
T2 – transportation of core to the site
U1 – fresh core staging/installation
U2 – power production
U4 – on-site storage
T3 – transportation of spent core to off-site storage
S1 – off-site storage

Figure 3-9. Fuel Cycle Stages for an HPR with Metallic Fuel

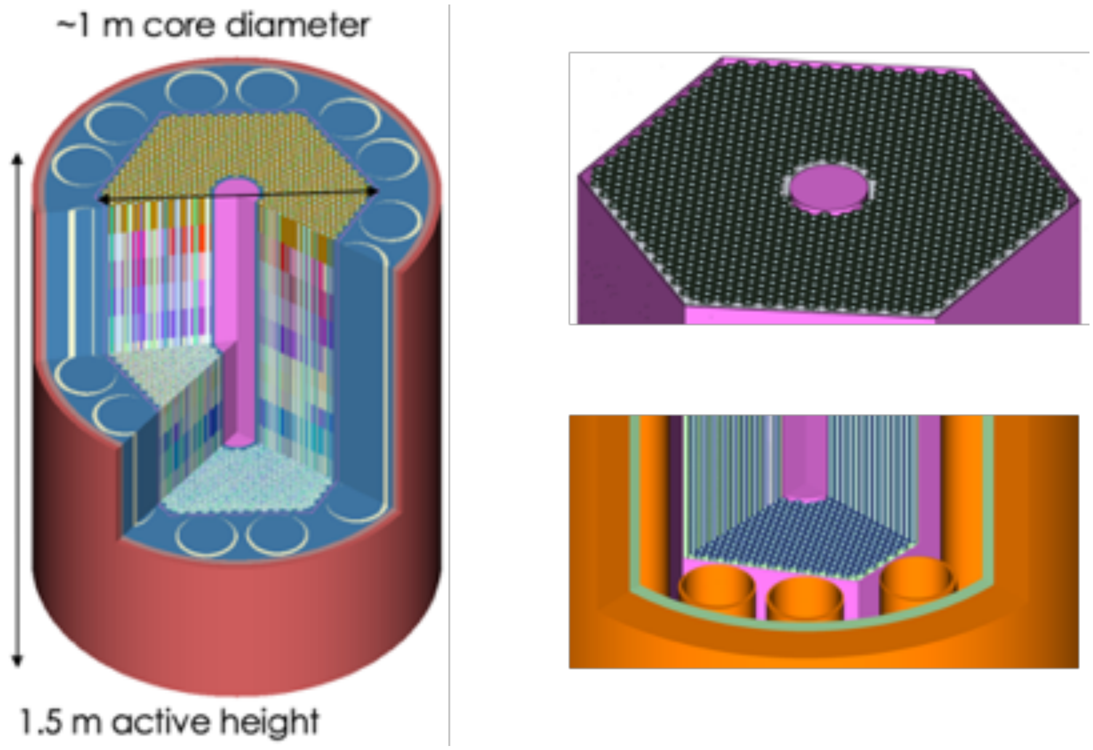


Figure 3-10. SCALE model of INL Design A HPR core

Figure 3-10 shows the HPR core layout on the left, with cutaways exposing the multi-colored fuel zones in the model which will result in rod-by-rod axially-dependent isotopics for accident scenario and back-end analysis. The right images show the heat pipe/fuel rod and crescent control drum geometric detail.

3.1.5. Molten Salt Reactor

The Molten Salt Reactor (MSR) has a neutron spectrum which depends on the molten salt and moderating properties of the system. The overall MSR fuel cycle is assumed to have the following stages.

1. UF_6 feed is used for production of UF_4 (fluoride salt) or UCl_3 (chloride salt) (23) (F1).
2. Basic fuel salt is conditioned, impurities removed, and blended with reactor salt (U1).
3. On-line extraction of fission products and other impurities, e.g. gaseous fission products (U2).
4. Waste product stream discharged to on-site spent fuel storage (U4).
5. Waste fuel is packaged and transported to off-site storage (T3).

The demonstration reactor is anticipated to be the MSRE with FLiBe salt. Figure 3-11 shows a picture of the actual MSRE with workers (left) and the equivalent SCALE model. This model has already been developed and initial capabilities to characterize MSRE neutronic performance, radionuclide inventory, and decay heat are being established through activities identified under the NRC non-LWR vision and strategy, Volume 3 (2). Under the activities planned in Reference (2), complementary MELCOR models to assess accident progression and source terms due to radionuclide release from a reactor have also been developed.

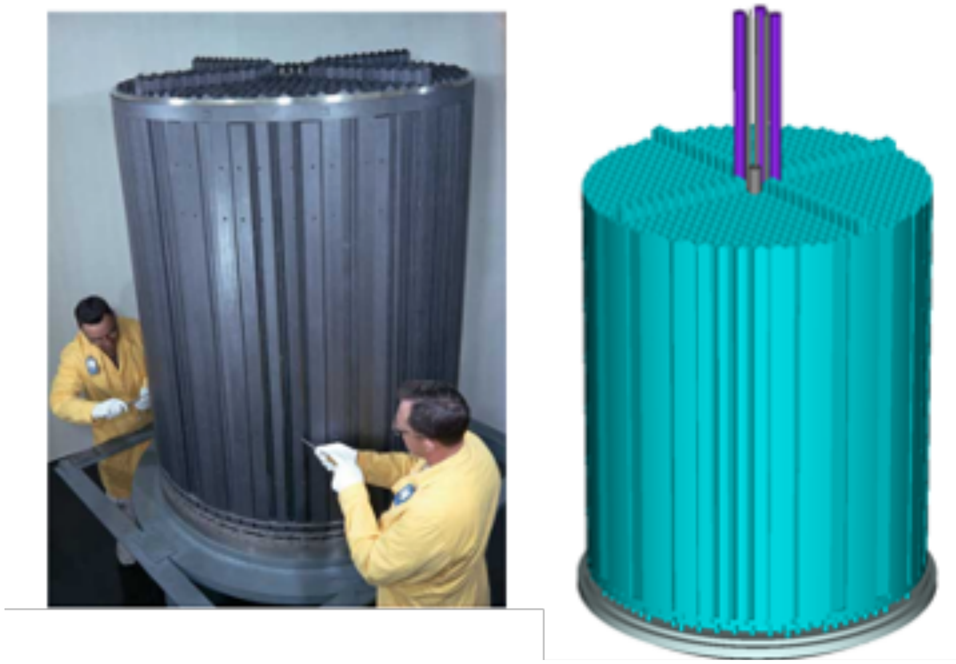
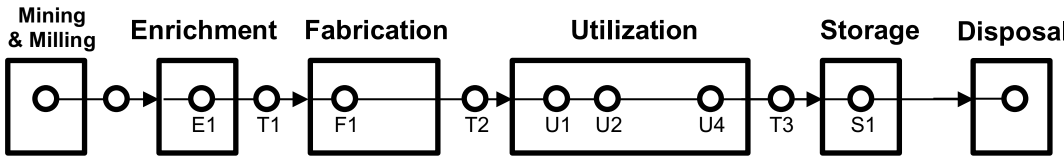


Figure 3-11: MSRE (left) and SCALE model (right)

The recent review of MSR fuel processing hazards (23) provides a key reference for fuel cycle considerations at an MSR plant. The overall stages of the molten salt fueling operation identified in (23) are as follows:

- Enriched fuel salt arrives at the reactor site in solid form in standardized containers (F1)
- Fuel salt is melted and then introduced into the circuit (U1)
- It is likely that multiple containers worth of fuel salt will be stored in a salt maintenance/storage vessel inside containment (U1)
- Fuel salt will be hydraulically transferred from this storage vessel into the primary circuit (U1)
- Refueling will most likely be performed by adding and/or removing pre-defined quantities of fuel salt into the primary circuit at periodicities established to ensure that breeding and burning are adequately compensated (U2 and U4)
- It is anticipated that an MSR plant will receive a large quantity of fuel at one time in order to minimize the frequency of fuel shipments, and minimize the necessity to break the containment boundary to introduce additional fuel material (U1, U2, and U4)

As part of the MSR operation, fission product gases will be generated. This requires an MSR system to provide an off-gas system to remove and process these fission gases. A general summary of the technologies for an MSR off-gas system was recently developed (24), with the MSRE operational experience providing a crucial basis.



E1 – UF_6 enrichment
T1 – transportation of UF_6 to fuel fabrication facility
F1 – fabrication of fuel salt from UF_6
T2 – transportation of fuel salt to the plant
U1 – fuel salt conditioning, mixing, and loading
U2 – power production with online chemical processes
U4 – on-site discharged fuel waste forms
T3 – transportation of fuel waste forms to off-site storage
S1 – off-site storage

Figure 3-12. Fuel Cycle Stages for an MSR

The species that are expected to be found in the headspace of an MSR include:

- Salt aerosols
- Noble gases
- Reactive gases
- Tritium
- Volatile and semi-volatile halides
- Activation products

These species will be swept through a set of traps for decay and storage using a helium sparging gas. Various methods for trapping and immobilizing the gaseous species were described as follows:

1. Molten hydroxide scrubber for particulates, aerosols, reactive gases, and halides
2. Immobilized zeolite for capture of tritium (and hydrogen), which can be recombined to form HTO (and H_2O).
3. Silver-functionalized packed beds to capture residual iodine and other halides
4. Cryogenic capture and release of non-radioactive gases such as N2 and O2 generated by radiolysis
5. Capture and separation of noble gas FPs through cryogenic distillation, activated carbon, metal-organic frameworks

A summary of the range of hazards anticipated to be present as part of MSR fuel processing operations is provided in the work of McFarlane et. al. (23). These hazards were identified during

- Initial criticality
- Reactor operation
- Refueling
- Maintenance
- Waste preparation

Table 3-1 summarizes hazards associated with operations conducted at the site during initial criticality. This is extracted from the work of McFarlane et. al. (23).

Although the MSRE as an experimental system did not include many of these processes, it is recommended to develop a simple set of analogous systems around the MSRE for the purposes of developing demonstration SCALE and MELCOR models.

Table 3-1. Operations Conducted on-site prior to Reactor Operation (initial criticality)

Physical or Chemical Process	Salt Type and Process Objective	Key Hazards	Mitigation Strategies
Receipt of materials at reactor site	Actinides and non-fissile components	• Contamination • Air sensitive • Be (if present as BeF_2)	Double barrier container
On-site storage prior to loading	Actinides and non-fissile components	• Contamination • Air sensitive • Be (if present as BeF_2)	Double barrier container
• Preparation for use • Preparation of quantities to load	Actinide fluoride and chloride salts	• Contamination • Air sensitive • Criticality	• Inert gas flush when transferring salts • Double barrier
Core load prior to criticality		• Air sensitive • Be (if present as BeF_2)	• Inert gas flush when transferring salts • Double barrier
Initial core load		• Contamination • Air sensitive • Criticality	• Inert gas flush • Double barrier
Online sampling of salt during loading	• Fluorides • Chlorides • Fuel salt • Coolant salt • Flush salt composition • Mixing by density measurement	• Contamination • Air sensitive • Criticality • Be (if present as BeF_2)	• Gamma, densitometer, control rod measurements • Inert gas flush • Double barrier containment

3.2. Fuel Forms

In the previous sections describing the various reactor systems considered for demonstration purposes, a number of common fuel forms were identified. As a result of this commonality, additional front-end assessments can be performed to demonstrate capability readiness to a much broader range of possible reactor systems. This section further classifies aspects of the non-LWR fuel cycle based on generic features of the fuel shared across candidate reactor systems.

3.2.1. TRISO

Figure 3-13 shows the TRISO fuel kernel manufacturing process which is the basis for many non-LWR fuel forms. In particular, TRISO kernels are used in the manufacturing of fuel pebbles and fuel compacts, as shown in Figure 3-14. Figure 3-15 shows the utilization of pebbles and compacts in pebble bed and prismatic reactors, respectively. The major concern during TRISO fuel manufacture is likely with respect to criticality at all stages, including packing limits for TRISO in standard packages. However, there are additional chemical hazards associated with various processes which should be addressed as well.

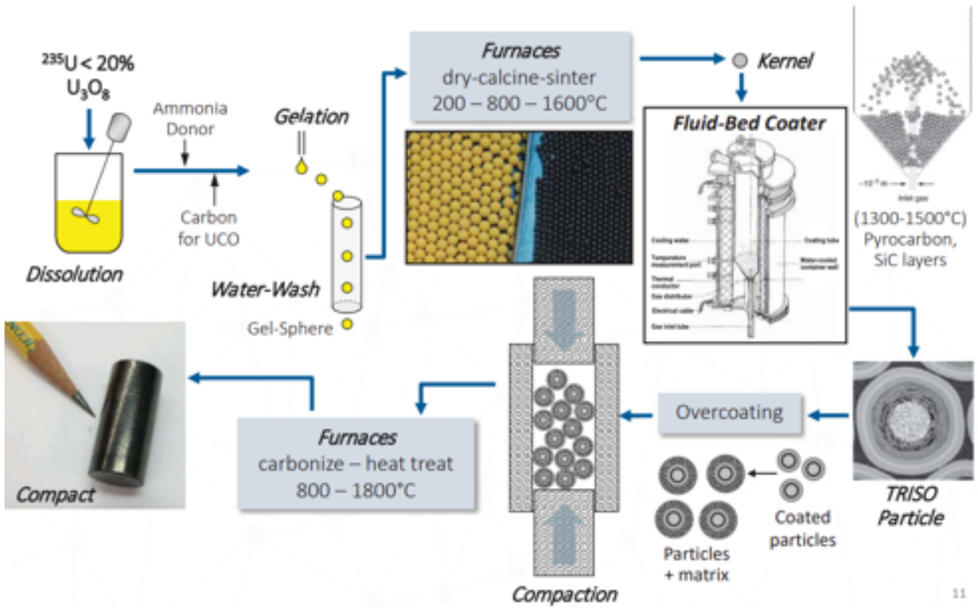


Figure 3-13: Fabrication of TRISO fuel particles (25)

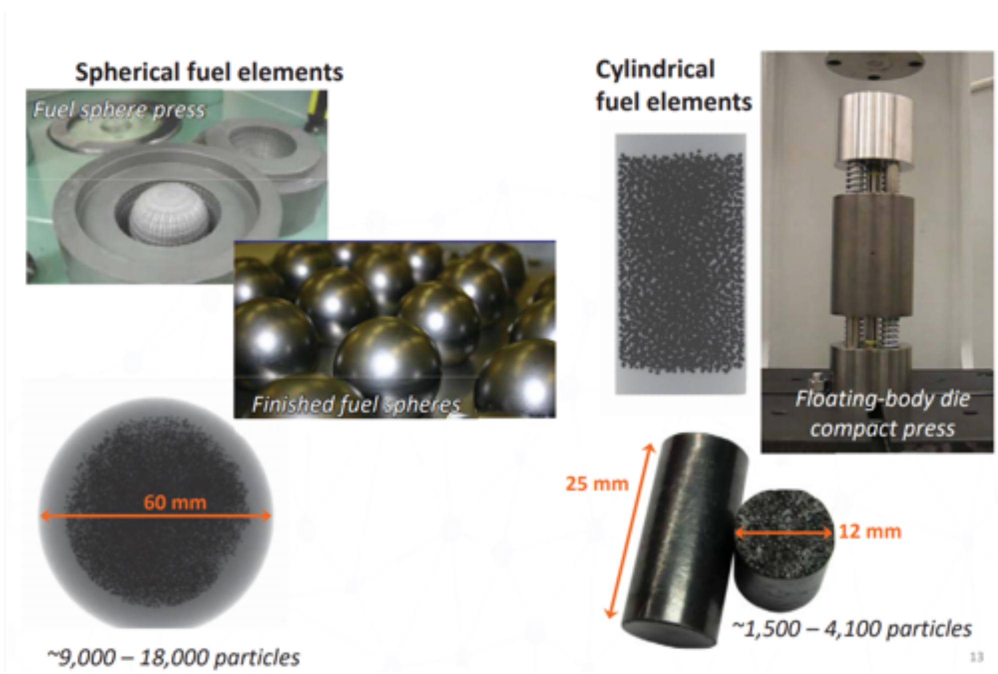


Figure 3-14: Fabrication of pebbles and compacts (25)

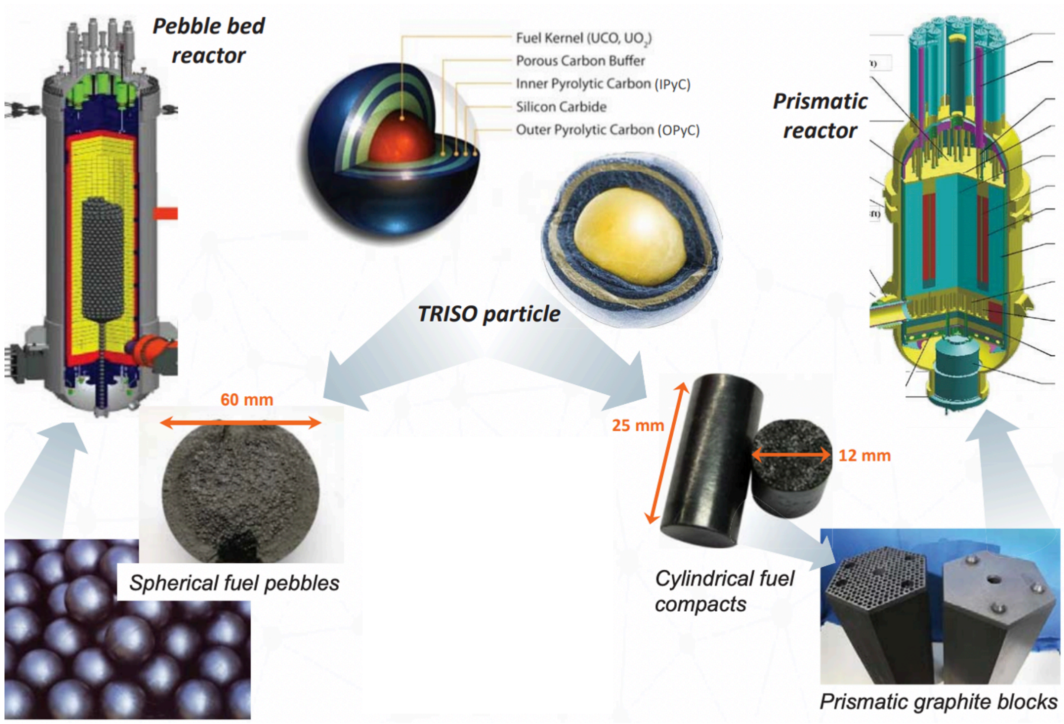


Figure 3-15. TRISO fuel particles used in pebble bed and prismatic systems (25)

3.2.2. Metallic Fast Reactor Fuel

The traditional method for fabricating metallic fuel is injection casting, as shown in Figure 3-16. This fabrication process involves the following steps:

- Fuel is melted and stirred in Y_2O_3 wash-coated graphite crucible
- The furnace is evacuated and ZrO_2 wash-coated SiO_2 molds are submerged
- Pulse pressurization of the vessel rapidly injects and freezes fuel in molds
- The molds are removed and shattered to release fuel slugs
- The crucible is cleaned by wire brush and recoated

Injection casting has the following limitations, under the assumption that the fuel includes minor actinides:

- Residual fuel heel and slug end crops result in only ~33% utilization of melted charge
- Fuel losses (e.g. volatile constituents such as Am)
- High level waste (graphite crucible and Y_2O_3 coating, SiO_2 mold pieces)
- Crucible cleaning and coating

Nevertheless, this process was used for fabricating 36,000 metallic fuel pins for EBR-II. A more efficient casting process is being developed as shown in Figure 3-17. Note that although transuranic fast reactor fuel is not considered as part of these demonstration studies, one of the values of fast reactors in the fuel cycle is their ability to use plutonium and minor actinide as feed.

The major concern during uranium-only metallic fuel manufacturing is likely with respect to criticality safety at all stages, including packing limits in standard packages. However, there may be additional chemical hazards associated with the various processes which should be addressed as well. A preliminary assessment of hazards applicable to metallic fuel fabrication was performed as part of the preparation of this plan. This assessment relied on information in the study recently performed by LeHaye and Burkes (26).

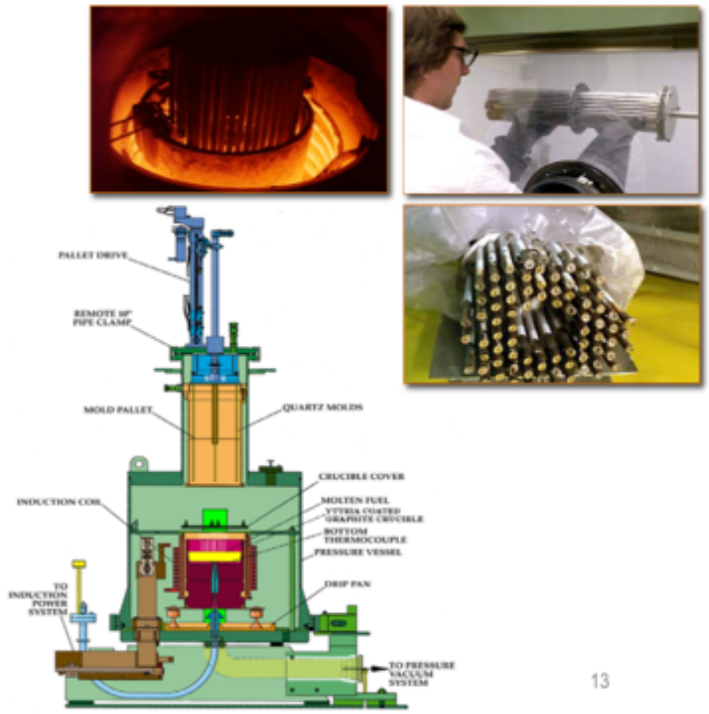


Figure 3-16: Injection Casting Method used for fabricating metallic fuel for EBR-II.

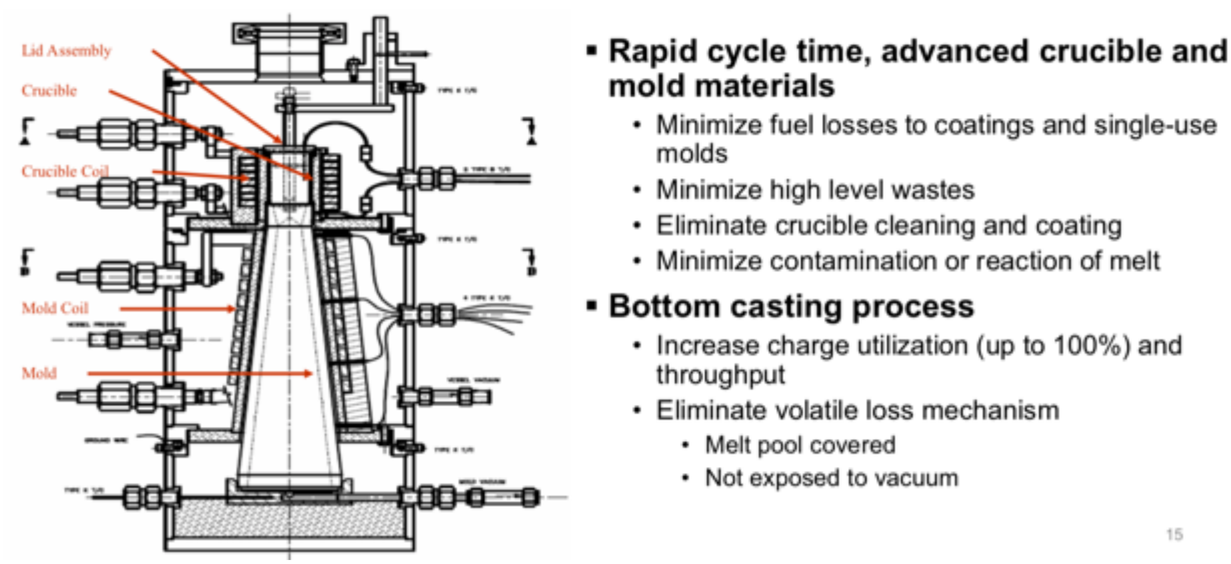


Figure 3-17: New casting process proposed by DOE advanced fuels campaign.

3.2.3. Molten Salt Fuel

One of the benefits of the MSR compared to other solid-fueled designs is the reduction in front-end infrastructure. Fast spectrum molten salt systems typically use a chloride salt with UCl_3 fresh fuel feed. Thermal spectrum molten salt systems typically use a fluoride salt with UF_4 fresh fuel (23). It is not clear yet in the commercial MSR landscape how front-end fuel fabrication processes will be performed. It may be possible that an MSR facility directly accepts UF_6 feed and performs the conversion to UCl_3 or UF_4 and related salt conditioning operations on-site, thus limiting the need for transportation of solid UCl_3 or UF_4 salts and packaging and transportation analyses (27).

4. NON-LWR FUEL CYCLE SAFETY ASSESSMENT READINESS DEVELOPMENT AND DEMONSTRATION

This section first presents the key reports which characterize code readiness and identify models that need to be developed for important stages in the non-LWR fuel cycle. The SCALE and MELCOR assessment approach are described, relying on information in the open literature used to identify scenarios capturing non-LWR fuel cycle risk as well as previous experience with LWR analyses.

4.1. SCALE and MELCOR Generic Evaluation Approach

Each of the reports will comprise a generic evaluation approach which documents the following:

- analytical needs during specific fuel cycle stages,
- identification of analytical modeling gaps,
- closure of analytical modeling gaps, and
- demonstration through analysis of NRC readiness to non-LWR fuel cycle safety analyses.

Each report will utilize the assessment matrix for code capability presented in Table 4-1 to structure the above capability readiness evaluation and development steps. The general evaluation approach is comprised of eight major steps as shown in Table 4-1. Example activities are provided for each step for both SCALE and MELCOR as applied to an HTGR system during development of our predictive capabilities during Volume 3 development (2).

Table 4-1. Assessment Matrix for Code Capability

General Evaluation Approach	HTGR Example	
	SCALE	MELCOR
	With respect to understanding radiological impacts to the health and safety of the public and environment:	
1. Identify safety related items of interest	Predict the inventory and system sub-critical margin of intact fuel pebbles in storage scenario. This inventory is important for back end criticality and shielding scenarios as well as initializing MELCOR	Predict the evolution of fission product gases from burned, intact fuel pebbles in storage scenario. This includes release of fission products from the fuel and the subsequent transport through the facility and potentially into the environment. This enables estimation of potential consequences to both worker and public.
2. Ask the right safety questions / Phenomena of interest / Understand the dominant features	What are the reaction rates and nuclide transmutation behavior of interest	What are the important nuclides relevant to safety and what is their behavior over long times.

Table 4-1. Assessment Matrix for Code Capability

<table>
<tr><th rowspan="2">General Evaluation Approach</th><th colspan="2">HTGR Example</th></tr>
<tr><th>SCALE</th><th>MELCOR</th></tr>
<tr><td>3. Survey experiments available that provide fundamental information</td><td>Basic experiments for the isotopes of interest that captures the reaction rates.

Critical experiment measurements exist for systems that rely ^{235}U enriched <5w/o, and over >93w/o, but little in-between. However, methods have been developed at ORNL and deployed through SCALE to understand the additional bias uncertainty from reliance on a small validation basis.

There is little radio-chemical assay data for TRISO and TRISO in high-burnup pebble-based depletions. However depletion will be similar to that in thermal-spectrum LWRs and some data may become available from the Advanced Gas Reactor (AGR) campaign.</td><td>A significant amount of data exists that is relevant to MELCOR modeling of the key phenomena influencing radiological and non-radiological hazardous material/transport in a facility and potential into the environment. The range of relevant experiments are discussed in more detail in the MELCOR leak path factor guidance (10).</td></tr>
<tr><td>4. Develop physics models to capture dominant feature and allow prediction</td><td rowspan="2">This is captured in the SCALE/TRITON and SCALE/CSAS code packages.

Further SCALE incorporates detailed CE nuclear data libraries as well as MG libraries both suitable for use with LWRs and Non-LWRS. These are based on ENDF/B-VII.1 (impacts of ENDF/B-VIII.0 is being evaluated), JEFF 3.1/A, as well as other sources.</td><td rowspan="2">For applications to nuclear fuel cycle analysis, the following MELCOR packages are relevant (as noted in Table 4-4).

• Control Volume Hydrodynamics (CVH) package
• Flow path (FL) package
• RadioNuclide (RN) modeling package
• Control Function (CF) package</td></tr>
<tr><td>5. Translate physics models into computer code</td></tr>
<tr><td>6. Perform verification testing (unit testing; and integrated testing as code complexity increases)</td><td>SCALE has thousands of unit tests and hundreds of integrated tests that test the majority of features.</td><td>See Volume 3 of the MELCOR Computer Code Manuals (28)</td></tr>
<tr><td>7. Perform validation with experiments. Capture the integrated codes</td><td>SCALE is assessed out of the box through the following and more details are available in the</td><td>MELCOR is assessed out of the box through the following and more details are available in the MELCOR</td></tr>
</table>

Table 4-1. Assessment Matrix for Code Capability

General Evaluation Approach	HTGR Example	
	SCALE	MELCOR
performance (with uncertainty analysis)	SCALE documentation provided with the computer code: - ~100 PWR and BWR decay heat data comparisons - ~100 radiochemical assay comparisons versus measurement for PWR and BWR - ~400 criticality safety validation cases from the ICSBEP - Burst fission experiment data	documentation (see Volume 3 of the MELCOR Computer Code Manuals (28))
8. Document findings	Will be captured in the reports	Will be captured in the reports

4.2. Non-LWR Fuel Cycle Readiness Development and Demonstration Reports

The approach, as previously discussed, is to deliver a set of reports which include descriptions of each of the stages in a postulated fuel cycle and the hazards and code capabilities. Priority will be placed on developing reports for the non-LWR designs and technologies that have the most available design information as mirrored in Volume 3 of the NRC non-LWR vision and strategy. Each report will focus on the code development and modeling of criticality, shielding, and radionuclide and non-radionuclide management and tracking during the postulated fuel cycle. Because of the anticipated reliance of all non-LWRs on the same initial fuel cycle stages from mining to enrichment, a single report on the enrichment phase and UF_6 handling will be produced. TRISO fuel kernels are used by many designs and fuel cycle safety issues will be documented in a single report. Fast reactor fuel fabrication will also be documented in a stand-alone report, as well as pebble TRISO fuel fabrication, respectively. This work will therefore result in the following 10 reports, with one enrichment report, four fabrication reports, and five reactor fuel cycle analysis reports. This type of organization will reduce duplication of effort across all reports and enable the most work to be performed in parallel across different reactor types. Table 4-2 provides an initial estimation of the project schedule and how the reports relate to each other.

Table 4-2. non-LWR Fuel Cycle Analysis Deliverables

Report No.	Report Topic	Assumed Deliverable Schedule[2]	Depends on Report No.
1	Enrichment and UF6 Handling up to 20 wt%	Initiation + 3 months	N/A
2	TRISO Fuel Kernel Fabrication	Initiation + 6 months	1
3	Uranium Metallic Fuel Fabrication[3]	Initiation + 6 months	1
4	Fast Reactor Fuel Assembly Fabrication[4]	Initiation + 6 months	3
5	Pebble TRISO Fuel Fabrication	Initiation + 6 months	2
6	FHR Fuel Cycle Analysis (Berkeley Mk. 1)	Initiation + 12 months	5
7	HPR Fuel Cycle Analysis (INL Design A-MET)	Initiation + 12 months	3
8	SFR Fuel Cycle Analysis (MET-1000/VTR)	Initiation + 12 months	4
9	HTGR Fuel Cycle Analysis (PBMR-400)	Initiation + 12 months	5
10	MSR Fuel Cycle Analysis (MSRE)	Initiation + 12 months	1

Note that while the enrichment and fabrication reports are somewhat general, the reactor system reports are for specific, idealized non-LWRs which are intended to represent the field of designs currently proposed in the US. Given the wide range of possible fuel cycles, higher priority is placed on the more probably fuel cycles to require NRC review in the near term.

Significant experience with radionuclide tracking and transport using SCALE and MELCOR is being accumulated for non-LWR designs as part of work performed in Volume 3 of the NRC Non-LWR Vision and Strategy, with necessary code development integrated into the assessment and demonstration process. Using the same codes for these assessments, little additional development work is anticipated, except for the MSR where consideration of additional chemical processes and storage in the front end and back end may require minor

[2] This is a preliminary prioritization is based on the overall stages in the fuel cycle for ease of analysis based on the current capabilities of SCALE and MELCOR.

[3] The assessment of metallic fuel fabrication will span multiple reactor concepts (e.g., HPR and SFR).

[4] Fast reactor fuel assembly fabrication introduces additional pyrophoric materials (e.g., sodium) that could lead to unique scenarios at this stage of the fuel cycle. For convenience, this is treated separately at present.

code developments to support those analyses. Note that of the documents surveyed to create this plan, the report by D.A. Reed (29) is the best example of the content that should be included in this type of fuel cycle report. The following discussion provides a summary of the content for each of the reports planned for each activity.

4.2.1. Enrichment and UF_6 Handling up to 20 wt%

This report focuses on assessing code readiness for a limited number of stages of the fuel cycle associated with enrichment and UF_6 handling, including transportation. The stages that are the specific focus for this assessment report are highlighted in Figure 4-1.

The report will document the models and analyses to characterize the readiness for evaluating the safety of the enrichment and handling stages and will mainly require criticality modeling assessments with SCALE/CSAS for enrichment, storage, and transportation of UF_6 with U-235 enrichment up to 20% as required by many non-LWR designs. The HTGR front-end report by Reed (29) demonstrates models and analysis that need to be considered in this report.

Figure 4-2 and Figure 4-3 provide examples of how existing capabilities can be applied to assess the criticality in enrichment and UF_6 handling. Figure 4-2 shows k-eff as a function of moderator density fraction for various enrichments in a 30B cylinder that could be used for transportation of UF_6 for all non-LWR fuel cycles. Figure 4-3 provides an example of how k-eff in a 48X cylinder can vary depending on the environment. A 48X cylinder may be used for storing enriched UF_6 for non-LWR fuel cycles at an enrichment facility (48X limited to 4.5 weight percent enriched UF_6 per ANSI N14.1).

Radiological and non-radiological hazards can arise at fuel fabrication facilities. A complementary analysis of the potential for consequences from these hazards will also be presented as part of the work documented in this report. This assessment of consequences will use MELCOR to assess, for different identified hazards, the transport of radionuclide and toxic non-radionuclide vapors and aerosols throughout the facility and potentially out of the facility.

These hazards could occur as a result of a criticality event, which would result in the generation of radionuclides and energy available to be released into the atmosphere of a facility. Dispersal of fission products into the atmosphere will lead to transport to different regions of the facility and potentially the environment. Thus radiological consequences for workers or the public could arise in this scenario. The identification of other hazards and accident scenarios will be performed following the approach identified above.

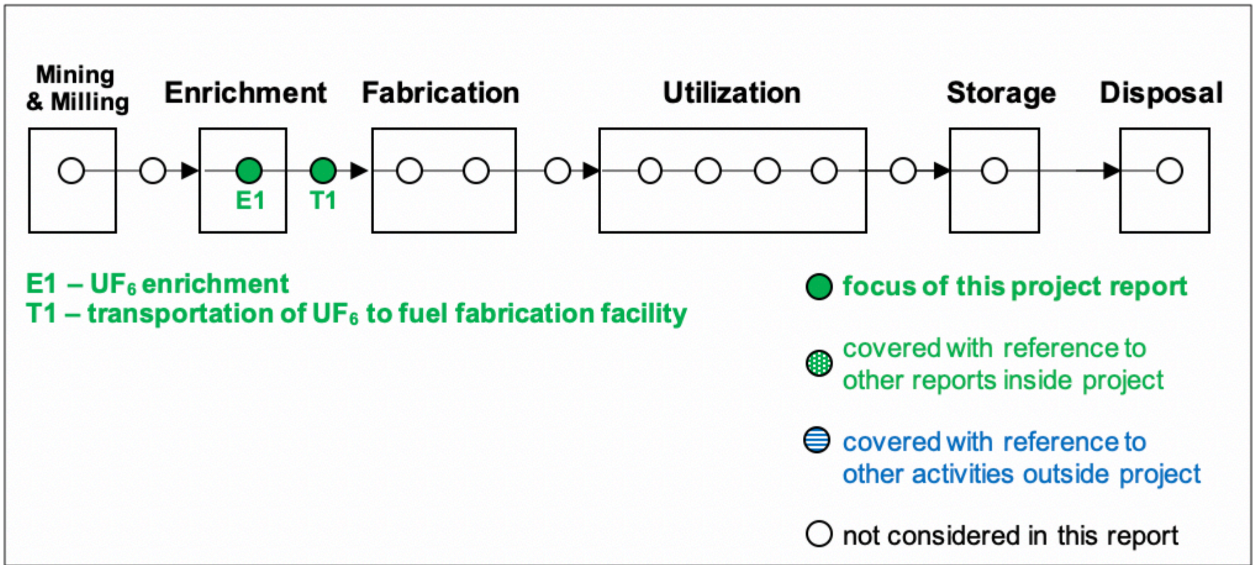


Figure 4-1. Focus Areas for Enrichment and UF_6 Handling Analysis Assessment Report.

Consideration of current activities involving the use of HALEU in LWR will provide a strong basis for efforts in this area. This report will relate these LWR activities to non-LWR ones. For example, with discussion of the volume of UF_6 feed required for the various systems and the compatibility with common storage containers at the specific enrichments required for the reactors in the assessment. There is also a well-known issue with the 30B UF_6 cylinder used for large volumes of material in LWR infrastructure. The subcriticality requirement of 10 CFR §71.55(b), which requires consideration of water in-leakage to the most reactive credible extent, is challenged by 30 in UF_6 cylinders but there exists an exception in 10 CFR §71.55(g) for enrichments 5% or less. The goal of this report is not to design new cylinders or influence rulemaking, but provide the demonstrations of the models and tools, applied to the new non-LWR fuel cycle scenarios postulated. Any available design changes to cylinders to support higher enrichments can be incorporated into the assessment report. This report will include sensitivity and uncertainty quantification (nuclear data and otherwise) as well as validation basis analysis.

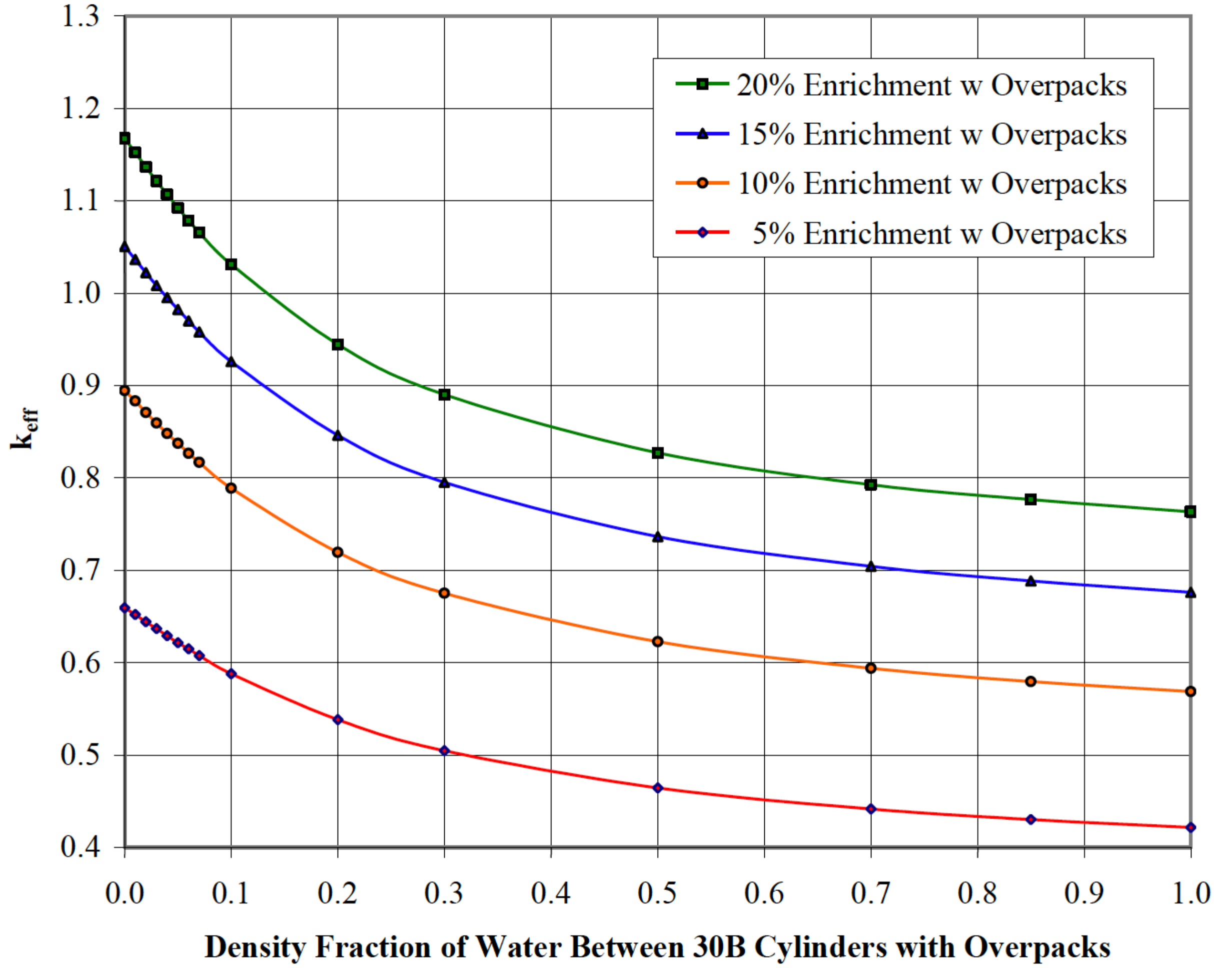
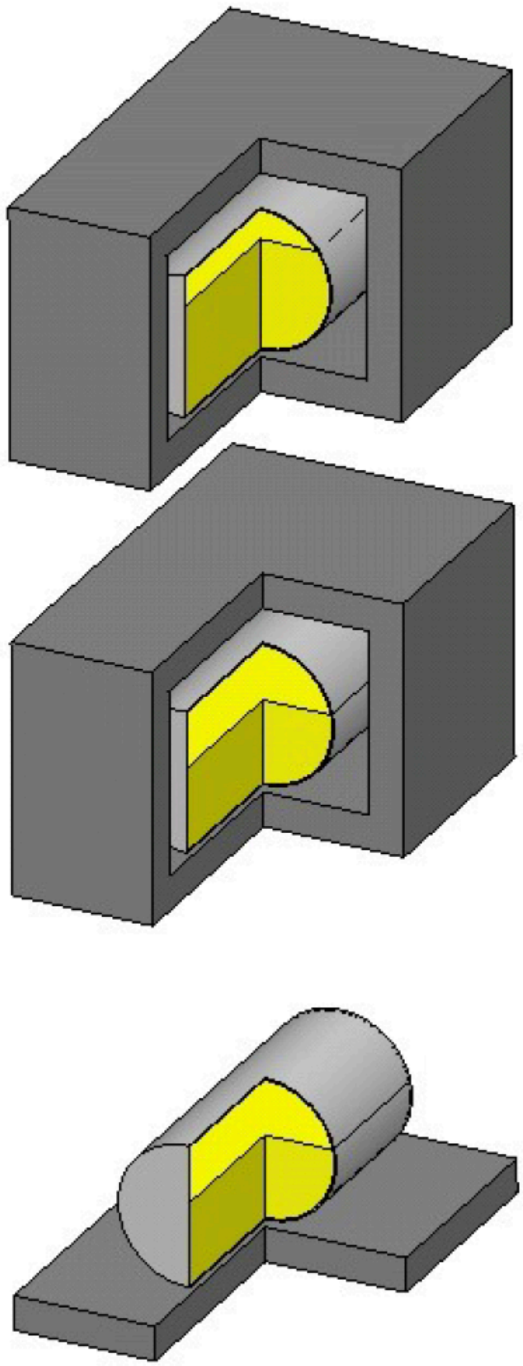


Figure 4-2: k-eff for UF6 cylinder as a function of enrichment and moderator density via SCALE/CSAS (29).



Case 1
48X Cylinder with liquid UF_6
Offset concrete reflector on all sides

k_{eff} = 0.9586 ± 0.0002

Case 2
48X Cylinder with solid UF_6
Offset concrete reflector on all sides

k_{eff} = 1.0209 ± 0.0002

Case 3
48X Cylinder with solid UF_6
Bottom concrete reflector only

k_{eff} = 1.0006 ± 0.0002

Figure 4-3: k-effective for various 48X UF6 cylinder models with 20% enrichment via SCALE/CSAS (29).

4.2.2. TRISO Fuel Kernel Fabrication

This report focuses on assessing code readiness for performing safety evaluations of TRISO fuel kernel fabrication. Figure 4-4 illustrates the areas of the fuel cycle that are the focus of this readiness assessment report.

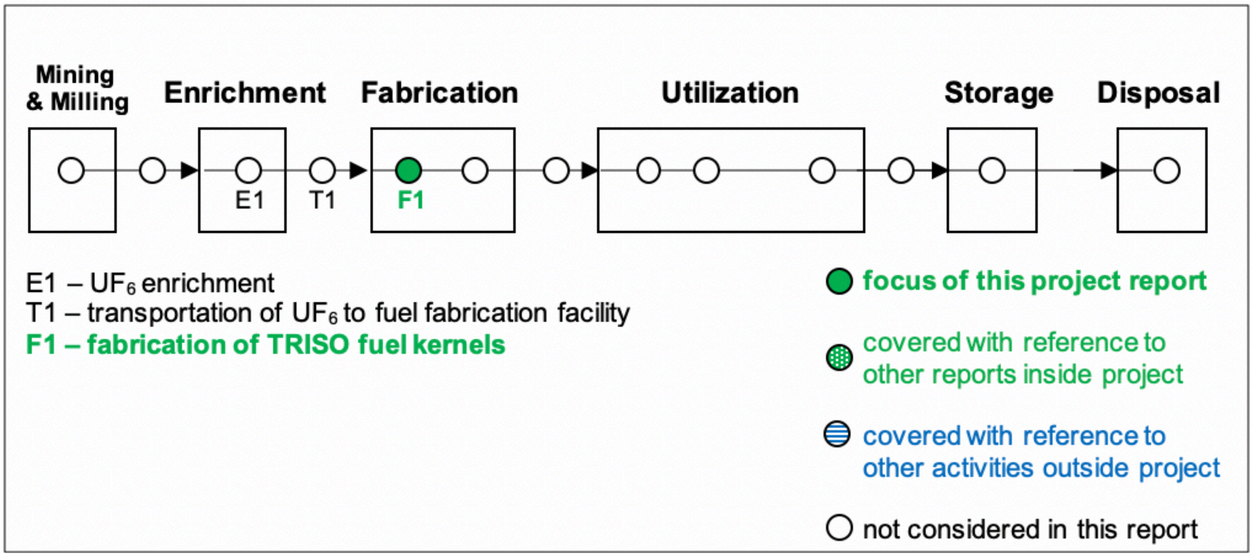


Figure 4-4. Focus Areas for TRISO Fuel Fabrication Readiness Assessment Report.

The fabrication of TRISO fuel kernels assumes fresh fuel only. This report will contain mainly criticality modeling assessments with SCALE/CSAS for the TRISO fabrication (F1) with the following coarse stages.

1. UF_6 is converted to the feed solution for the fuel kernel (e.g., UO_2)
2. The feed is then fabricated into kernels, e.g. via initial processes involving liquid forms and final processes involving high-temperature sintering to create the final, uncoated fuel particles.
3. The kernels are coated by continuous vapor deposition (CVD) and become (TRISO).

Simple SCALE criticality models will be developed which take into account the chemical forms at each stage with best practices for criticality models of systems with many thousands of TRISO particles. It will be assumed that TRISO must be transported from one facility to another to create the final prismatic or pebble fuel. A criticality model for shipping TRISO will be included as part of this work.

Radiological and non-radiological hazards can arise at fuel fabrication facilities. A complementary analysis of the potential for consequences from these hazards will also be presented as part of the work documented in this report. MELCOR will be used to assess the consequences for different radiological and non-radiological hazardous material release scenarios, which are transported throughout the facility and potentially out of the facility.

Releases of radiological material and energy could occur as a result of a criticality event. Dispersal of fission products into the atmosphere will lead to transport to different regions of the facility and potentially the environment. Thus radiological consequences for workers or the public could arise in this scenario. The identification of other hazards and accident scenarios will be performed following the approach identified above.

4.2.3. Uranium Metallic Fuel Fabrication

This report focuses on assessing code readiness for performing safety evaluations of the uranium metallic fuel fabrication process (F1). Figure 4-5 highlights the areas of the fuel cycle focused on in this assessment report.

This report assumes current state-of-the-art metallic fuel fabrication at the time of the report, including processes for both solid metallic fuel and annular metallic fuel as would be used by the SFR and HPR, respectively. The report will describe simple SCALE criticality models representative of the fuel fabrication process assuming uranium-based fuel up to 20% enrichment as well as models and discussion of the likely transportation packages for the fabricated fuel.

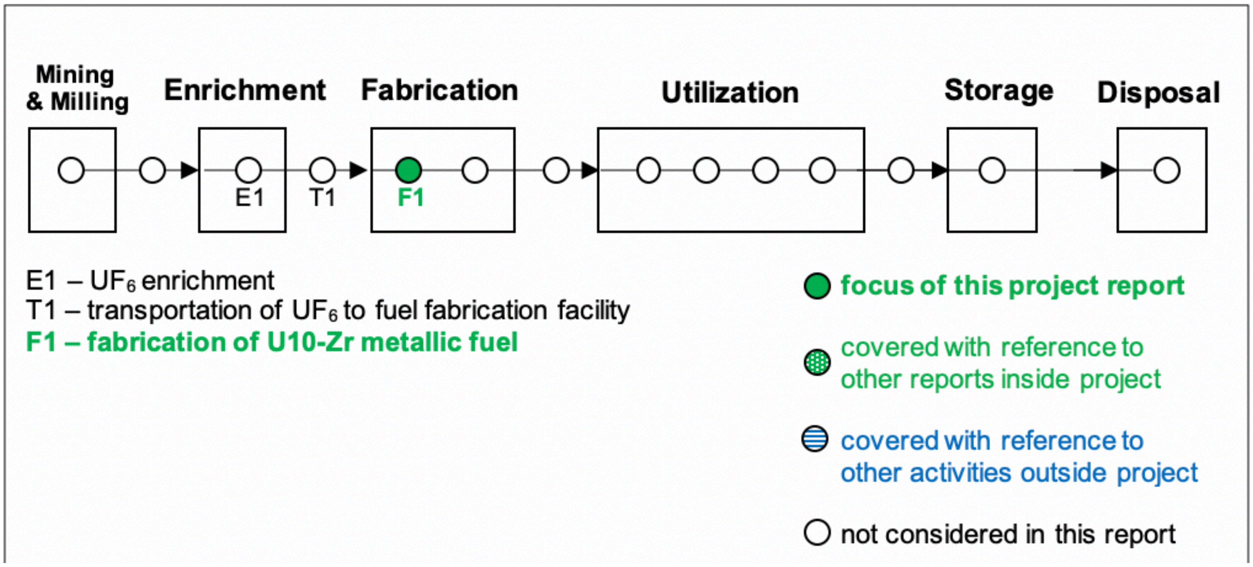


Figure 4-5. Focus Areas for Uranium Metallic Fuel Fabrication Readiness Assessment Report.

Radiological and non-radiological hazards can arise at fuel fabrication facilities. A complementary analysis of the potential for consequences from these hazards will also be presented as part of the work documented in this report. This assessment of consequences will use MELCOR to assess, for different identified hazards, the transport of radionuclide and toxic non-radionuclide vapors and aerosols throughout the facility and potentially out of the facility.

Releases of radiological material and energy could occur as a result of a criticality event. Dispersal of fission products into the atmosphere will lead to transport to different regions of the facility and potentially the environment. Thus radiological consequences for workers or the public could arise in this scenario. The identification of other hazards and accident scenarios will be performed following the approach identified above.

Since metallic fuel fabrication introduces pyrophoric material into the process, additional effort is planned to evaluate scenarios initiated from the chemical reaction of these materials with dry or moist air. Additional hazards that could serve to initiate an accident may be present.

This report will include sensitivity and uncertainty quantification (nuclear data and otherwise) as well as validation basis analysis.

4.2.4. Fast Reactor Fuel Assembly Fabrication

This report focuses on assessing code readiness for performing safety evaluations of the fast reactor fuel fabrication process. Figure 4-6 highlights the areas of the fuel cycle focused on in this assessment report.

This report assumes a uranium feed and the current state-of-the-art processes for metallic fuel fabrication at the time of the report, ideally the same techniques as used for the VTR. Simple SCALE criticality models will be described for relevant stages during fabrication and including likely transportation packages of the fabricated fuel.

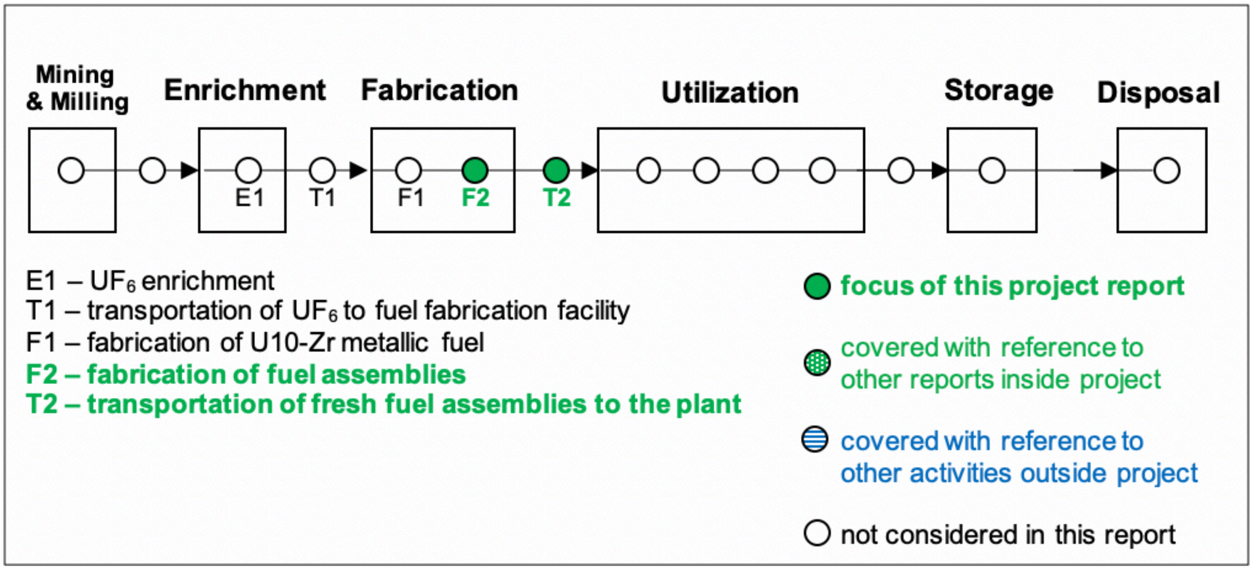


Figure 4-6. Focus Areas for Fast Reactor Fuel Fabrication Readiness Assessment Report.

Radiological and non-radiological hazards can arise at fuel fabrication facilities. A complementary analysis of the potential for consequences from these hazards will also be presented as part of the work documented in this report. This assessment of consequences will use MELCOR to assess, for different identified hazards, the transport of radionuclide and toxic non-radionuclide vapors and aerosols throughout the facility and potentially out of the facility. MELCOR will be used to assess the consequences for different identified hazards, the transport of radionuclide and toxic non-radionuclide vapors and aerosols throughout the facility and potentially out of the facility.

Releases of radiological material and energy could occur as a result of a criticality event. Dispersal of fission products into the atmosphere will lead to transport to different regions of the facility and potentially the environment. Thus radiological consequences for workers or the public could arise in this scenario. The identification of other hazards and accident scenarios will be performed following the approach identified above.

This report will include sensitivity and uncertainty quantification (nuclear data and otherwise) as well as validation basis analysis.

4.2.5. Pebble TRISO Fuel Fabrication

This report focuses on assessing code readiness for performing safety evaluations of the pebble TRISO fuel fabrication process. Figure 4-7 highlights the areas of the fuel cycle focused on in this assessment report.

This report assumes current state-of-the-art pebble fuel fabrication at the time of the report. Assuming a TRISO feed, this report will describe simple criticality models for the fabrication of the pebbles for pebble bed reactors such as the HTGR or FHR, assuming TRISO up to 20%. The report will also describe criticality models for the likely transportation packages of the fabricated fuel.

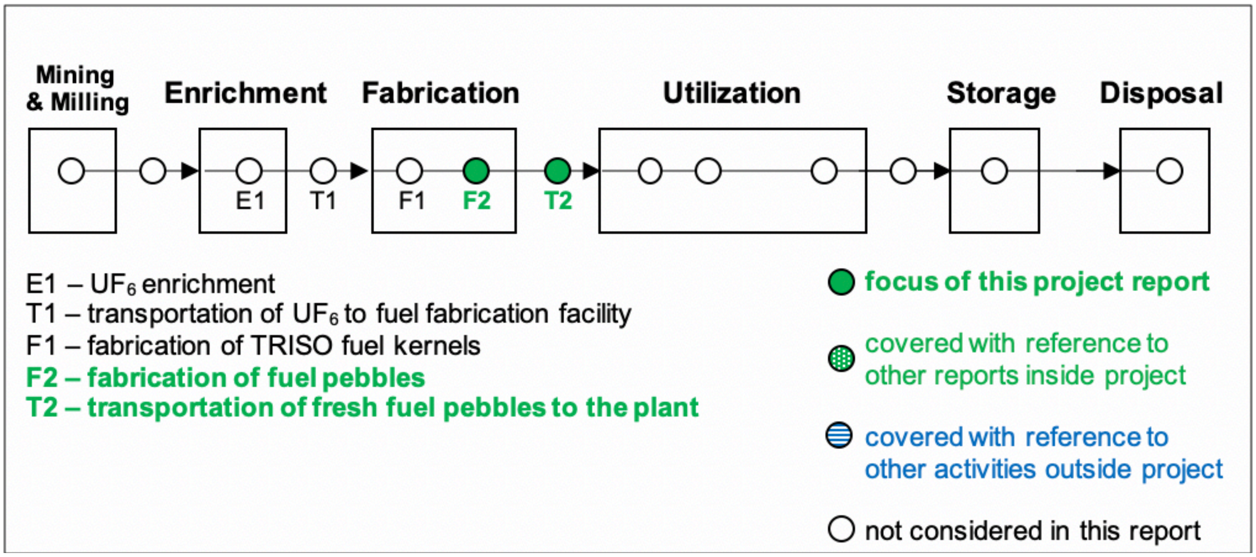


Figure 4-7. Focus Areas for Pebble TRISO Fuel Fabrication Readiness Assessment Report.

Radiological and non-radiological hazards can arise at fuel fabrication facilities. A complementary analysis of the potential for consequences from these hazards will also be presented as part of the work documented in this report. This assessment of consequences will use MELCOR to assess, for different identified hazards, the transport of radionuclide and toxic non-radionuclide vapors and aerosols throughout the facility and potentially out of the facility. MELCOR will be used to assess the consequences for different identified hazards, the transport of radionuclide and toxic non-radionuclide vapors and aerosols throughout the facility and potentially out of the facility.

Releases of radiological material and energy could occur as a result of a criticality event. Dispersal of fission products into the atmosphere will lead to transport to different regions of the facility and potentially the environment. Thus radiological consequences for workers or the public could arise in this scenario. The identification of other hazards and accident scenarios will be performed following the approach identified above.

This report will include sensitivity and uncertainty quantification (nuclear data and otherwise) as well as validation basis analysis.

4.2.6. FHR Fuel Cycle Analysis

This report focuses on assessing code readiness for performing safety evaluations of the FHR fuel cycle. Figure 4-8 highlights the areas of the fuel cycle focused on in this assessment report.

A simple fuel cycle will be assumed for the Berkeley Mk. 1 FHR with stages as described previously in Section 3.1.2, from enrichment through irradiation in the core, discharge, and transport off-site of spent fuel. For the front end of the fuel cycle, the reports on pebble TRISO fuel compact fabrication (F2), TRISO kernel fabrication (F1), and enrichment (E1) and UF6 handling will be referenced extensively. This report will only contain the necessary material to link this specific FHR design to the more general front end studies (e.g., the volume of UF_6

needed each cycle). The majority of the in-reactor studies will be completed as part of other non-LWR activities (e.g., Volume 3 of the NRC non-LWR Vision and Strategy (2)) on this specific design and also will only need be referenced. The main modeling activities undertaken for this new work fuel cycle are:

1. Criticality in the fresh fuel staging and loading operations (SCALE) for the U1 stage.
2. Radionuclide transport within radiological systems such as the Tritium Control System (SCALE, MELCOR) for the U2 stage.
3. Criticality, decay heat, activity, shielding, and accident analysis of on-site spent fuel storage (SCALE, MELCOR) for the U4 stage.

Note also that transportation off-site and off-site storage (T3 or S1) is not currently planned in this set of activities due to lack of information on the fuel forms, packages, and off-site facilities.

This report will include sensitivity and uncertainty quantification (nuclear data and otherwise) as well as validation basis analysis.

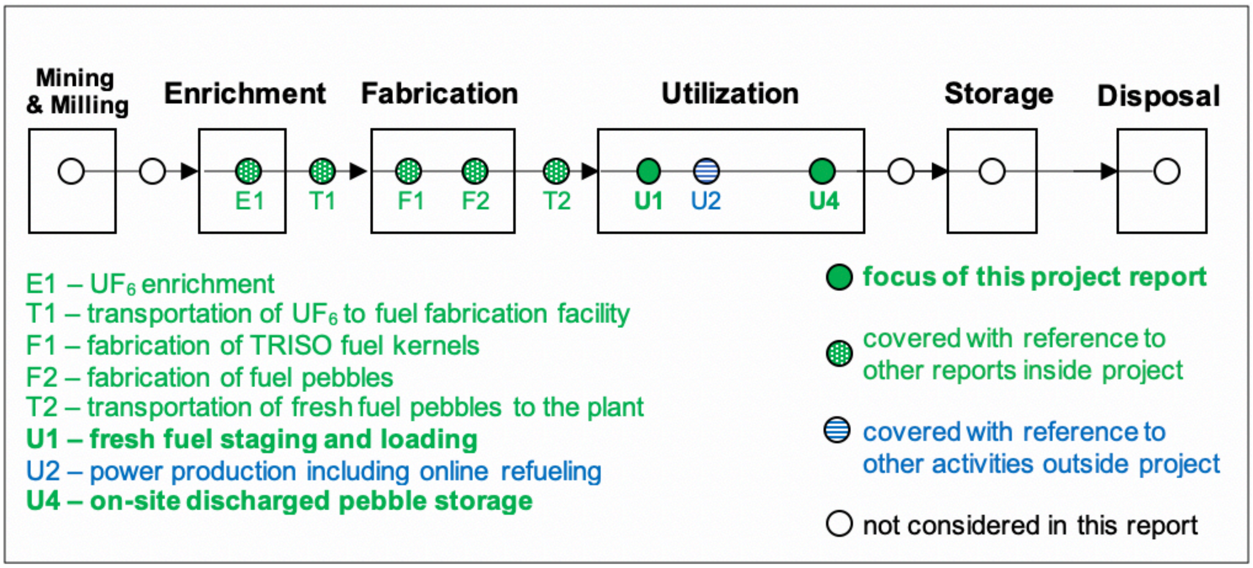


Figure 4-8. Focus Areas for FHR Fuel Cycle Readiness Assessment Report.

4.2.7. HPR Fuel Cycle Analysis

This report focuses on assessing code readiness for performing safety evaluations of the HPR fuel cycle. Figure 4-9 highlights the areas of the fuel cycle focused on in this assessment report.

This report assumes a simple fuel cycle for the INL Design A HPR modified with metallic fuel and 10 GWd/MTU discharge burnup, herein referred to as “INL Design A-MET”. The fuel cycle stages considered are from enrichment through irradiation in the core, discharge, and storage on-site. Due to the low burnups and low volumes of fuel for a single unit, the additional complexity of considering transportation of spent fuel off-site need not be considered in this report because those considerations will be captured in separate assessments expected to be performed by vendors or DOE-NE.

For the front end of the fuel cycle, the reports on metallic fuel manufacture and enrichment and UF_6 handling will be referenced (E1 and T1), extensively. This report will only contain the necessary information to link this specific HPR design to the more general front end studies, (e.g., the volume of UF6 needed for each core). The majority of the in-reactor studies (U2) will have been completed as part of other non-LWR activities (e.g. Volume 3 (2)) for the original INL Design A. However, for this specific variant, the in-reactor studies will need to be repeated due to the use of metallic fuel.

Therefore the main modeling activities undertaken for this work are:

1. Criticality safety during fabrication of the reactor core from metallic fuel feed (SCALE) for the F2 stage.
2. Criticality safety for transportation of that core to the site (SCALE) for the T2 stage.
3. Criticality safety during fresh core staging at the site (SCALE) for the U1 stage.
4. Severe accident analysis for the INL Design A-MET variant (SCALE, MELCOR) which was already performed in Volume 3 for an oxide variant for the U2 stage.
5. Criticality, decay heat, activity, shielding, and accident analysis of on-site spent fuel storage (SCALE, MELCOR) for the U4 stage.

Note that MELCOR is not envisioned to be needed for activity 2 in this specific case with the HPR because of the low burnup and lack of fuel shuffling and storage. However, as a demonstration, in case a site would replace a core, models to simulate the longer term decay heat and activity of the core with SCALE will be described. Note also that unlike the other utilization-focused reports, the fabrication and core transport stages (F2 and T2) are combined with utilization stages in this report because the processes are unique to the HPR and thus there is no advantage to having a stand-alone HPR fuel fabrication report. Note also that transportation off-site and off-site storage (T3 or S1) is not currently planned in this set of activities due to lack of information on the fuel forms, packages, and off-site facilities.

This report will include sensitivity and uncertainty quantification (nuclear data and otherwise) as well as validation basis analysis.

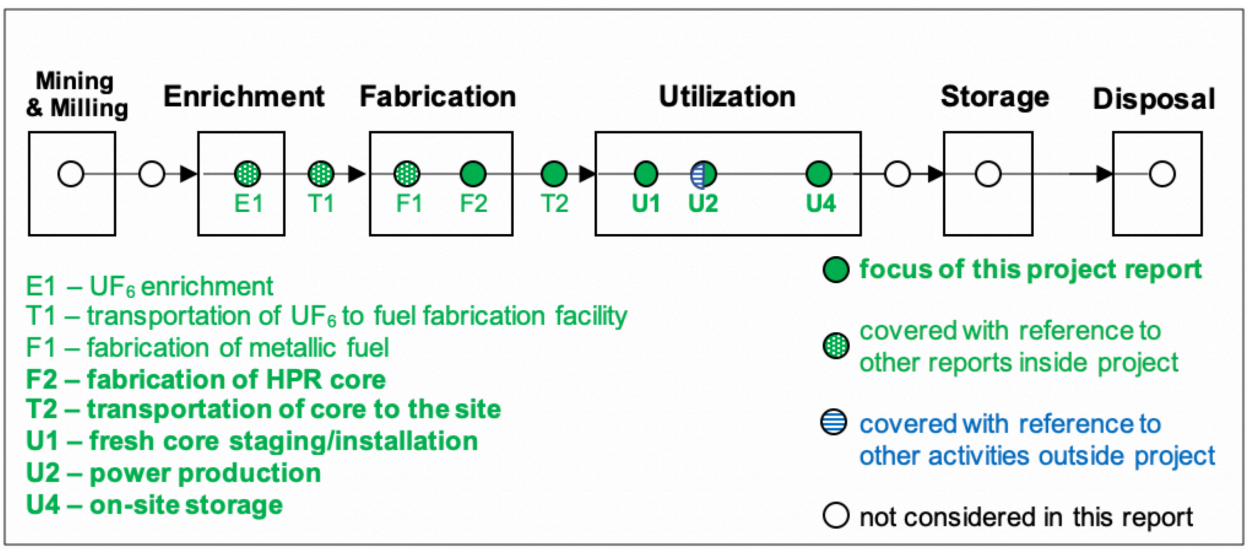


Figure 4-9. Focus Areas for HPR Fuel Cycle Readiness Assessment Report.

4.2.8. SFR Fuel Cycle Analysis

This report focuses on assessing code readiness for performing safety evaluations of the SFR fuel cycle. Figure 4-10 highlights the additional areas of the fuel cycle focused on in the remainder of this assessment report.

This report assumes a simple fuel cycle defined by the VTR if available, with fallback on the MET-1000 SFR, defined in an OECD/NEA benchmark for fast reactor systems if needed. A batch refueling strategy will need to be defined for this assessment, with discharge burnup approximately 200 GWd/MTU. The VTR is currently defined as U-Pu-10Zr, however more detail on this fuel and the fabrication process must become available to consider this mixed U/Pu fuel for the studies here. For now, the fast reactor fuel cycles focus on a pure uranium metallic fuel (U-10Zr). The fuel cycle stages considered are from enrichment through irradiation in the core, discharge, and storage on-site, and transport off-site.

For the front end of the fuel cycle, the reports on metallic fuel manufacture and enrichment and UF_6 handling (E1, T1, and F1) will be referenced extensively from previous reports. This report will only contain the necessary material to link this specific SFR design to the more general front end studies (e.g., the volume of UF_6 needed for each core). Due to the prioritization of assessment of HPR capability readiness under Volume 3 of the NRC non-LWR Vision and Strategy (2), development of demonstration plant models for SCALE and MELCOR have not been performed yet for the SFR. This assessment report will leverage planned work in FY21 for Volume 3 capability readiness demonstration (2). The main modeling activities undertaken for this work are:

1. Criticality of fresh fuel staging areas at the site (SCALE) for the U1 stage.
2. Severe accident analysis for the MET-1000 Uranium metallic variant (SCALE+MELCOR) for the U3 stage.
3. Criticality, decay heat, activity, shielding, and accident analysis of on-site spent fuel storage (SCALE, MELCOR) for the U4 stage.

Note also that transportation off-site and off-site storage (T3 or S1) is not currently planned in this set of activities due to lack of information on the fuel forms, packages, and off-site facilities.

This report will include sensitivity and uncertainty quantification (nuclear data and otherwise) as well as validation basis analysis.

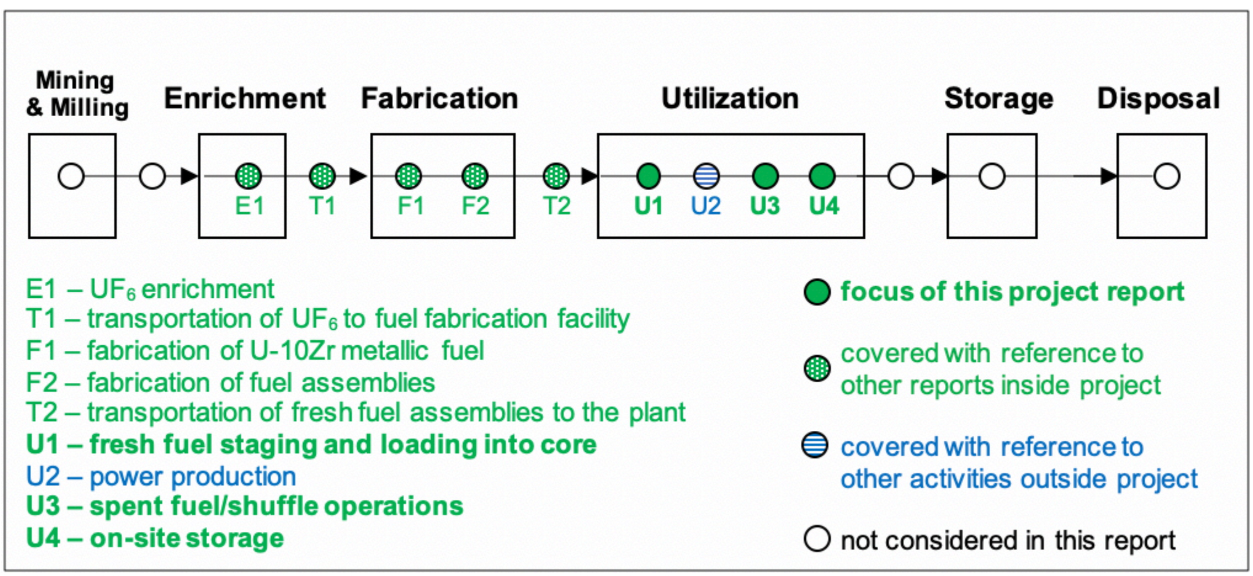


Figure 4-10. Focus Areas for SFR Fuel Cycle Readiness Assessment Report.

4.2.9. HTGR Fuel Cycle Analysis

This report focuses on assessing code readiness for performing safety evaluations of the HTGR fuel cycle. Figure 4-11 highlights the areas of the fuel cycle focused on in this assessment report.

This report assumes a simple fuel cycle for the PBMR-400 HTGR with stages as described previously, from enrichment through irradiation in the core, discharge, and transport off-site of spent fuel. For the front end of the fuel cycle, the reports on pebble TRISO fuel fabrication (F1), TRISO kernel fabrication (F1), and enrichment and UF6 handling (E1 and T1) will be referenced extensively. This report will only contain the necessary material to link this specific HTGR design to the more general front end studies (e.g., the volume of UF_6 needed for each cycle). The majority of the in-reactor studies will have been completed as part of other non-LWR activities (e.g., Volume 3 of the NRC non-LWR Vision and Strategy (2)) on this specific design and also will only need be referenced. The main modeling activities undertaken for this new work are:

1. Criticality of fresh fuel staging areas at the site (SCALE) for the U1 stage.
2. Criticality, decay heat, activity, shielding, and accident analysis of on-site spent fuel storage (SCALE, MELCOR) for the U4 stage.

Note also that transportation off-site and off-site storage (T3 or S1) is not currently planned in this set of activities due to lack of information on the fuel forms, packages, and off-site facilities.

This report will include sensitivity and uncertainty quantification (nuclear data and otherwise) as well as validation basis analysis.

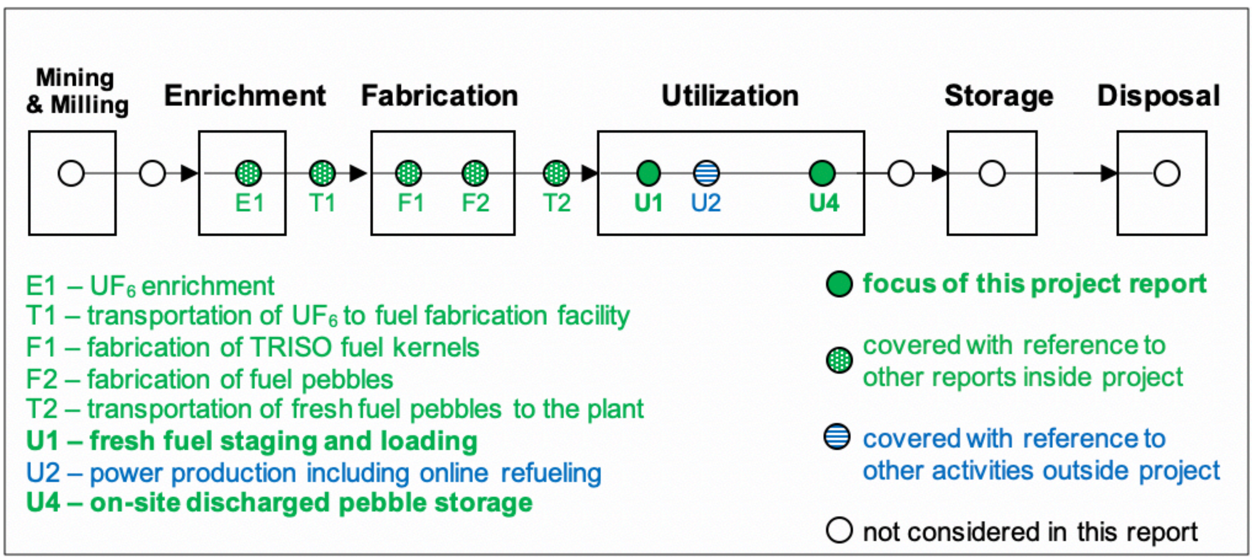


Figure 4-11. Focus Areas for HTGR Fuel Cycle Readiness Assessment Report.

4.2.10. MSR Fuel Cycle Analysis

This report focuses on assessing code readiness for performing safety evaluations of the MSR fuel cycle. Figure 4-12 highlights the areas of the fuel cycle focused on in this assessment report. This report assumes a simple fuel cycle for the MSRE MSR with front end assuming to begin with enriched UF6 and proceeding to the following stages. Note that we consider fabrication stages and transportation (F1 and T2) combined with utilization stages in this report because all processes are unique to the MSR and thus there is no advantage to having a stand-alone MSR fuel fabrication report. Due to the potential for on-line refueling and chemical reprocessing at the plant, off-site storage is not considered in this demonstration MSR fuel cycle.

1. UF_6 feed is used for production of UF_4 (fluoride salt) or UCl_3 (chloride salt) (23)
2. Basic fuel salt is conditioned, impurities removed, and blended with reactor salt
3. On-line extraction of fission products and other impurities, e.g. gaseous fission products
4. Waste product stream discharged to on-site spent fuel storage

The majority of the in-reactor studies will have been completed as part of other non-LWR activities (e.g., Volume 3 of the NRC non-LWR Vision and Strategy (2)) on this specific design and also will only need to be referenced. The main modeling activities undertaken for this new work are as follows.

1. Criticality and chemical transport analysis for salt production (SCALE, MELCOR) in the F1 stage and transportation to the site in the T2 stage.
2. Fuel salt conditioning, blending, and initial criticality operations (SCALE, MELCOR) in the U1 stage,
3. Radionuclide transport from non-reactor components (e.g. fission off-gas system) during normal operation (SCALE, MELCOR) in the U2 stage.
4. Shielding, criticality, and accident analysis from on-site spent fuel storage (SCALE, MELCOR) in the U4 stage.

Note that due to lack of overlap with other designs, F1 through U4 stages will be pursued in this report. Note also that transportation off-site and off-site storage (T3 or S1) is not currently planned in this set of activities due to lack of information on the fuel forms, packages, and off-site facilities.

This report will include sensitivity and uncertainty quantification (nuclear data and otherwise) as well as validation basis analysis.

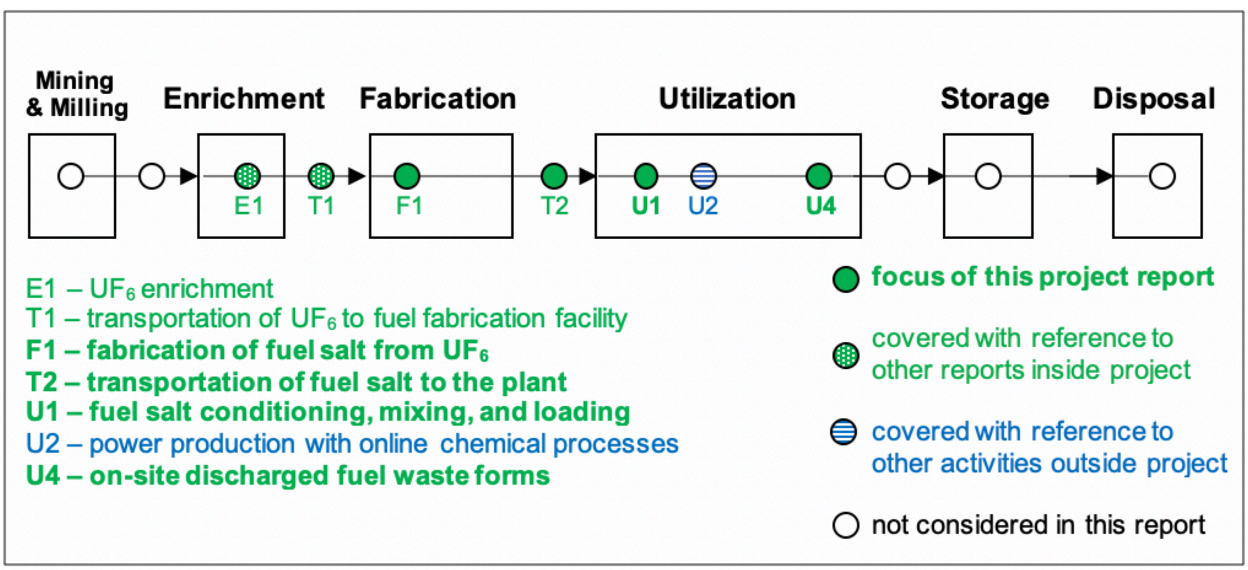


Figure 4-12. Focus Areas for MSR Fuel Cycle Readiness Assessment Report.

4.3. Applicability of SCALE and MELCOR for non-LWR Fuel Cycle Safety Assessment

In this section, the basis for applicability of SCALE and MELCOR tools to non-LWR fuel cycle assessment is discussed. Nuclear facility safety has evolved to generically consider a number of common hazards and accident scenarios. The SCALE and MELCOR code packages model a breadth of physical and chemical phenomena, including capabilities relevant to nuclear (i.e., fuel cycle) facility safety assessments. Since many of the original safety assessments for LWR fuel cycle facilities were performed a number of years ago, the analytical methods may pre-date the development of modern codes. These analytical methods, however, may need to be renewed for application to new or expanded facilities, specifically non-LWR fuel cycle facilities. Since both SCALE and MELCOR have an established validation and regulatory application pedigree in reactor and facility safety, they provide the immediately available analytical tools for performing independent safety assessments of non-LWR fuel cycle facilities.

The scope of typical LWR fuel cycle safety assessments is described in NUREG/CR-6410 (7), which provides the scope of methods required for assessing safety at traditional fuel cycle facilities. Fuel cycle facility safety assessments focus on the evaluation of the impact of different hazards that cause release of radionuclides into an enclosure atmosphere or the environment. The transport of released radionuclides to individuals on- or off-site has the potential to cause health effects. Thus, a facility safety assessment evaluates the impact of a number of different postulated radionuclide release scenarios on ultimate consequence to public health and safety.

The content of NUREG/CR-6410 (7) is also characteristic of the range of safety assessments required for other types of nuclear facilities operated by the DOE (8), owing to shared physical and chemical processes for different nuclear facilities. The analysis of hazards has been DOE nuclear facility guidance has been developed for the analysis of hazards (8). Differences between facilities arise in the:

- Form of the radioactive material being handled, processed or stored
- Distribution of radionuclides in the radioactive material that could be available to release to an enclosure atmosphere or environment
- Nature of dispersal (or release) of radionuclides from the radioactive material into an atmosphere (i.e., in the form of vapors or aerosolized particulates)
- Energy content of radionuclide release into an atmosphere (e.g., rapid due to a nuclear criticality event)
- Environmental conditions in the facility that affect transport of radionuclides within the facility or to the environment (e.g., whether or not a fire is occurring coincident with the radiological release)

Independent of these differences, the fundamental neutronic, thermal hydraulic, and radionuclide transport processes are common across nuclear facilities. A facility safety assessment thus leads to specification of:

- Nuclear criticality and radiation hazards
- Release of radionuclide material (mass and distribution) into an atmosphere
- Release of hazardous non-radiological material into an atmosphere
- Release of energy into the atmosphere

The modeling of neutronic, thermal hydraulic, and radionuclide transport processes to assess the potential for public health and safety consequences is then shared across facilities.

NUREG/CR-6410 (7) and the DOE accident analysis handbook (8) provide examples of the range of modeling that may be needed for fuel cycle safety assessments. The NRC non-LWR vision and strategy, Volume 3 (2), provides a summary of the SCALE capabilities for application to assessment of criticality safety, shielding, depletion, activation and spent fuel source term studies. The MELCOR leak path factor guidance (10) describes the application of the MELCOR code package to facility accident scenario modeling. Fuel cycle assessment can broadly be categorized into a set of distinct classes, which are summarized in Table 4-3.

Owing to the more limited scope of physical and chemical phenomena relevant to fuel cycle facility safety assessments, the existing verification and validation for the SCALE and MELCOR code packages cover application to fuel cycle safety analysis. The MELCOR leak path factor guidance (10) presents the validation, in many cases developed for reactor applications, relevant to representation of phenomena occurring in fuel cycle facility accidents. In addition to the validation exercises originally developed for reactor applications, a number of specific validation exercises were developed to further demonstrate applicability of MELCOR to facility accident modeling. The MELCOR leak path factor guidance (10) summarizes these additional validation cases.

SCALE and MELCOR have been used in nuclear facility safety analyses. For example, Reference (9) summarizes a recent application of the SCALE and MELCOR code packages to a Barnwell facility safety analysis. This highlights the utility and applicability of these code packages for assessing facility safety.

Table 4-3. Nuclear Fuel Cycle Assessment Classes

Assessment Class	Assessment Characteristics	Validation Basis
Materials Accounting	• Destructive assays of used fuel is not practical so a method to develop sufficiently accurate masses of isotopes of interest is required • SCALE can provide isotopic information for this purpose as well as calculation of various common “signatures” such as decay heat, gamma emission, and neutron emission which can be used in non-destructive assay	• Destructive assay and decay heat calorimeter measurements of LWR systems has been used to validate SCALE/ORIGEN depletion and decay physics which use general methods and the best available nuclear data • Safeguards activities have been used to validate gamma and neutron emission capabilities, such as the U.S. DOE-EURATOM SCALE/ORIGEN integration with RADAR (Remote Acquisition of Data and Review) and CRISP (Central RADAR Inspection Support Package) which compares measured gamma and neutron count rates to expected based on fuel declarations
Fires	• Source of radioactive and non-radioactive particulates to enclosure atmosphere • Source of energy to enclosure atmosphere and structures	• Sources can be established per existing facility safety guidance • SCALE depletion analysis benchmarking • MELCOR containment thermal hydraulic benchmarking • MELCOR radionuclide modeling benchmarking

Table 4-3. Nuclear Fuel Cycle Assessment Classes

Assessment Class	Assessment Characteristics	Validation Basis
	• SCALE provides radionuclide inventory to be released to atmosphere • MELCOR determines thermal hydraulic response of facility atmosphere • MELCOR determines radionuclide transport within facility and amount released to environment	• Additional benchmark evaluations performed against DOE-HDBK-3010 experiments
Explosions	• Source of radioactive and non-radioactive particulates to enclosure atmosphere • SCALE provides radionuclide inventory to be released to atmosphere • Explosions classified as either detonations or deflagrations • Explosions defined for MELCOR analysis as a transient energy and by-product gas source term (10) • Deflagrations can be modeled using the MELCOR BURN package • Detonations happen too quickly to require MELCOR modeling of hydrodynamic feedback	• Sources of mass and energy associated with explosive event defined per • Structural response to the mechanical loading induced by an explosive event assessed per DOE facility safety guidance (10) • MELCOR containment thermal hydraulic benchmarking • MELCOR radionuclide modeling benchmarking • Additional benchmark evaluations performed against DOE-HDBK-3010 experiments
Spills or Material Drops	• Source of radioactive and non-radioactive particulates to enclosure atmosphere • SCALE provides radionuclide inventory to be released to atmosphere • Modeling of powders or liquids under gravity • Aerosol source introduced for MELCOR modeling of particulate transport in enclosure atmosphere	• MELCOR containment thermal hydraulic benchmarking • MELCOR radionuclide modeling benchmarking • Additional benchmark evaluations performed against DOE-HDBK-3010 experiments

Table 4-3. Nuclear Fuel Cycle Assessment Classes

Assessment Class	Assessment Characteristics	Validation Basis
Nuclear Criticality Events	• Source of radioactive material as well as energy to enclosure atmosphere • SCALE provides radionuclide inventory as well as energy to be released to atmosphere • Unlike a chemical explosion, this type of scenario does not introduce by-product gases into the facility atmosphere • MELCOR treats radionuclides and energy as transient source to enclosure atmosphere	• SCALE criticality evaluation validation basis • MELCOR containment thermal hydraulic benchmarking • MELCOR radionuclide modeling benchmarking • Additional benchmark evaluations performed against DOE-HDBK-3010 experiments
High Radiation Fields (i.e., shielding)	• Radioactive sources emit radiation that with insufficient shielding could be harmful to on-site personnel • Suspend or deposited radioactive materials will generate radiation fields that could be harmful to on-site personnel without appropriate shielding	• SCALE shielding calculation validation basis • MELCOR containment thermal hydraulic benchmarking • MELCOR radionuclide modeling benchmarking

4.3.1. Evolution of Fuel Cycle Safety Assessment Analytical Capabilities

NRC fuel cycle safety assessment activities have been guided by the technical bases and methods established in NUREG/CR-6410 (7). This accident analysis handbook, published in 1998, established analytical methods for performing an Integrated Safety Analysis (ISA) for fuel cycle facilities.

Table 4-4 presents an assessment of how analytical methods provided in NUREG/CR-6410 (7) map to the range of capabilities available in the NRC code packages of SCALE, MELCOR, and consequence analysis tools such as MACCS, and RASCAL.

Table 4-4. Evolution of Fuel Cycle Safety Assessment Capabilities relative to NUREG/CR-6410 (7)

NUREG/CR-6410 Analytical Capability	Applicable Codes for Safety Assessment Analytical Capability
Nuclear Criticality and Radiation Shielding	
Inadvertent nuclear criticality events • Solution systems • Fully moderated and reflected solids • Powder systems	SCALE *Criticality Safety Analysis Sequences (CSAS) with KENO V.a (CSAS5) or KENO-VI (CSAS6)*

Table 4-4. Evolution of Fuel Cycle Safety Assessment Capabilities relative to NUREG/CR-6410 (7)

NUREG/CR-6410 Analytical Capability	Applicable Codes for Safety Assessment Analytical Capability
• Large storage arrays	These calculation sequences rely on the SCALE KENO Monte Carlo code for performing eigenvalue neutronics calculations. KENO V.a utilizes a simplified geometry package applicable to most systems of interest in criticality safety. KENO-VI relies on the SCALE Generalized Geometry Package. Both variants perform neutron transport eigenvalue calculations that provide the multiplication factor (k_{eff}) and neutron flux distributions. The Monte Carlo calculations can be performed in either continuous energy or multigroup modes. The outputs from these calculations will be processed to develop estimates of • Radionuclide inventory released from the fissile material achieving criticality • Energy release into the enclosure atmosphere These two outputs will be utilized in subsequent MELCOR calculations to determine the transport of radiological material throughout a facility and potentially into the environment.
Radiation shielding	SCALE *Monaco with Automatic Variance Reduction using Importance Calculations (MAVRIC) fixed source radiation transport calculation sequence* SCALE shielding analysis capabilities are provided by the (MAVRIC) calculation sequence. This calculation sequence performs a fixed-source radiation transport calculation utilizing the Monaco Monte Carlo code. This code performs multi-group and continuous energy fixed-source Monte Carlo transport calculations with unbiased Monte Carlo methods. MAVRIC is based on the Consistent Adjoint Driven Importance Sampling (CADIS) methodology. With this modeling capability, fluxes and dose rates

Table 4-4. Evolution of Fuel Cycle Safety Assessment Capabilities relative to NUREG/CR-6410 (7)

NUREG/CR-6410 Analytical Capability	Applicable Codes for Safety Assessment Analytical Capability
	can be calculated with low uncertainties for deep penetration problems. Estimates of the transport and deposition of radiological material throughout a facility provided by MELCOR will be relevant for assessing radiation hazards under certain sequences of interest. For example, radionuclide transport and deposition in HVAC filters could lead to appreciable radiation hazards to on-site workers.
Depletion, Activation, and Decay	SCALE *Oak Ridge Isotope Generation (ORIGEN) code to provide a depletion/irradiation/decay solver* ORIGEN determines time-dependent concentrations, activities, and radiation source terms for an array of isotopes that are simultaneously generated or depleted by neutron transmutation, fission, and radioactive decay. SCALE utilizes ORIGEN to provide a number of modules utilized in depletion, activation, and decay calculations. Estimation of radionuclide inventory at risk (or available to be released) is an important input for subsequence MELCOR calculations of radiological material/energy transport within a facility.
Source Term Determination	
Characterization of particle size and aerosol physics	MELCOR *RadioNuclide package* This package implements a generalized sectional model that allows tracking of aerosol particle transport, accounting for the distribution of particle sizes. This sectional model discretizes the aerosol particle size distribution into a series of sections. Agglomeration processes are represented in the sectional model as leading to growth of aerosols and transition to sections representing larger particle sizes. The section model implemented in MELCOR is a generalized approximation to the full aerosol

Table 4-4. Evolution of Fuel Cycle Safety Assessment Capabilities relative to NUREG/CR-6410 (7)

NUREG/CR-6410 Analytical Capability	Applicable Codes for Safety Assessment Analytical Capability
	transport equation. It is able to treat a range of different problems involving different aerosol particle size distributions. It is not restricted to modeling aerosol physics for the LWR severe accident applications to which it has been applied most often.
Determination of characteristics of radioactive material airborne releases - Gases - Volatile materials - Low-volatility liquids - Solids	MELCOR *Control Volume Hydrodynamics package* *Flow Path package* *RadioNuclide package*
Characterization of mitigation measures - HEPA filters	MELCOR *Control Volume Hydrodynamics package* *Flow Path package* *RadioNuclide package*
Transport within Building Enclosures	
Definition of walls/corridors/ventilation systems	MELCOR *Control Volume Hydrodynamics package* *Flow Path package* The compartmentalization of an enclosure into various rooms has a significant impact on flow of hazardous material from a source to an ultimate release point to the environment. The nodalization of an enclosure is performed through user input with the MELCOR code, which allows users to flexibly define control volumes, flow paths between control volumes (to represent openings like doors, ducts or fans), heat structures to represent walls, etc.
Engineered mitigative systems	MELCOR *Control Volume Hydrodynamics package* *Flow Path package* *RadioNuclide package* *Control Function package*
Survivability of barriers and mitigative systems	MELCOR *Control Volume Hydrodynamics package* *Flow Path package* *Control Function package*
HVAC	MELCOR *Control Volume Hydrodynamics package* *Flow Path package*
Flow disturbances due to fires or uncontrolled chemical reactions	MELCOR *Control Volume Hydrodynamics package*

Table 4-4. Evolution of Fuel Cycle Safety Assessment Capabilities relative to NUREG/CR-6410 (7)

NUREG/CR-6410 Analytical Capability	Applicable Codes for Safety Assessment Analytical Capability
	Flow Path package *Control Function package*
Explosions	MELCOR *Control Volume Hydrodynamics package* *Flow Path package* *Burn package (deflagrations)* *Control Function package*
Flow disturbances induced by exterior wind pressure	MELCOR *Control Volume Hydrodynamics package* *Flow Path package* *Control Function package*
Definition of flow path flow rates	MELCOR *Control Volume Hydrodynamics package* *Flow Path package*
Attenuation of airborne radioactive and hazardous chemical materials along flow path	MELCOR *Control Volume Hydrodynamics package* *Flow Path package* *RadioNuclide package*
Overall leak path factor calculation	MELCOR *Control Volume Hydrodynamics package* *Flow Path package* *RadioNuclide package*
Characterization of release from facility	MELCOR *Control Volume Hydrodynamics package* *Flow Path package* *RadioNuclide package*
Atmospheric Dispersion and Consequence Modeling	
Gaussian atmospheric dispersion	Using consequence assessment code packages described in NRC non-LWR vision and strategy Volume 3 and Volume 4
Puff atmospheric dispersion	
Single-particle Lagrangian atmospheric dispersion	
Health effects	

Initial review indicate that both SCALE and MELCOR code packages have the ability to perform a broad range of analyses necessary to perform safety analyses necessary to characterize risk associated with each of the assessment classes identified in Table 4-1 and modeling capabilities presented in Table 4-4.

4.3.2. Fuel Cycle Hazard Identification and Accident Analysis Approach

The assessment of the potential and consequences from various operations conducted throughout the different stages of a specific nuclear fuel cycle involves a set of steps shown in Figure 4-13. This type of assessment is typically conducted for the different stages of a nuclear fuel cycle, ranging from material extraction and fuel fabrication to reprocessing and disposal. The scope of this particular document, however, excludes detailed assessment of the potential for health and safety consequences from long-term disposal of radiological and non-radiological hazardous materials generated during the different stages of a nuclear fuel cycle.

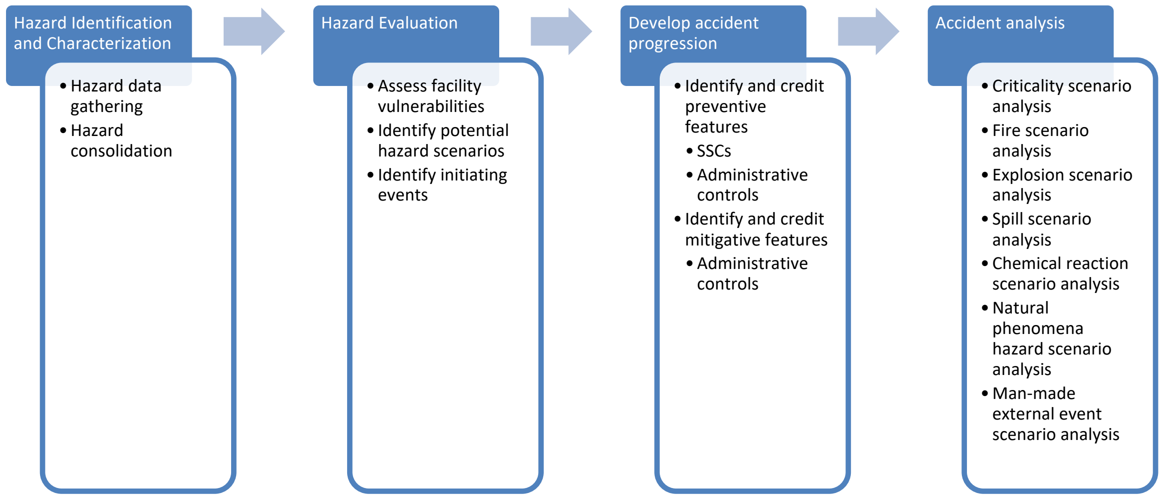


Figure 4-13. Hazard and Accident Analysis Process

The four stages in the assessment represent the standard approach followed in the assessment of hazardous material consequences introduced through operation of DOE facilities, as further discussed in NUREG/CR-6410 (1) and the DOE accident analysis handbook (8). The steps of such a safety assessment have the following goals:

- Hazard identification and characterization represents a systematic process intended to identify all the possible hazards to worker and public health and safety that could arise
- Hazard evaluation is the step at which the identified hazards are identified in light of the facility vulnerability and operations to identify specific initiating events
- With initiating events defined, evaluation of measures to prevent or mitigate accidental conditions are identified in order to develop how an accident would progress
- Finally, the analysis of accident scenarios is conducted in order to evaluate the evolution of an accident to support assessment of worker and public health and safety consequences

The broad sources of consequence that can arise at different stages of a nuclear fuel cycle can be grouped broadly as:

- Radiation hazards arising from *direct* interaction of high energy particles emitted due to nuclear fission (neutrons as well as gamma, alpha and beta particles) with humans.
- Radiological hazards arising from radioactive nuclides released into the atmosphere of the facility or the environment.
- Chemical hazards arising from toxic chemical released into the atmosphere of the facility or the environment.

Radiation hazards arise due to the fission of fissile material that has been arranged into a critical configuration. Radiological hazards are those that arise due to the release into the atmosphere of radionuclides from radioactive material. The transport of radionuclides through the atmosphere of a facility or the environment is influenced by features of the accident scenario, (e.g., the release of other vapors or gases as well as energy), as well as characteristics of the facility (e.g., the presence of filtration systems that could scrub some radionuclides from the atmosphere). Similarly chemical hazards arise when toxic vapors or aerosols are released into and transported through the atmosphere of a facility or the environment. Chemical hazards can have public health and safety consequences due to the toxicity of a number of different chemicals used as part of the nuclear fuel cycle. As in the case of radiological hazards, assessing these consequences requires determination of transport of toxic vapors and aerosols throughout the atmosphere of the facility or the environment.

Each of the four steps identified in Figure 4-13 are described in more detail.

Hazard Evaluation and Scenario Development

As part of the assessments discussed below, a hazard and accident initiator identification may be required for some stages of a fuel cycle should existing studies not be available. For the assessments defined below, this identification process will be a scoping effort to support subsequent steps analyzing accident scenarios when such information has not already been developed[5].

Hazards, consistent with NUREG/CR-6410 (1) and DOE facility safety assessment methodology (8), are categorized into the following classes:

- Electrical
- Thermal
- Pyrophoric Material
- Spontaneous Combustion
- Open Flame
- Flammables
- Combustibles
- Chemical Reactions
- Explosive Material
- Kinetic (linear and rotational)
- Potential (pressure)
- Potential (height/mass)
- Internal Flooding
- Radioactive Material
- Hazardous Material (toxicological/chemical/biological)
- Direct Radiation Exposures
- Non-ionizing Radiation
- Criticality
- External Man-made Events
- Vehicles in Motion
- Natural Phenomena

These hazards are the basis for identifying initiating events that can result in accident scenarios generating conditions that release radiological and non-radiological hazardous material into the atmosphere of a facility or the environment.

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[5] The molten salt fuel processing operation has been the subject of a recent study to identify hazards. This work is presented in a recent report (24).

A systematic approach is followed as part of DOE facility safety assessments to evaluate the potential for these hazard classes to be present. Such an evaluation can involve, for example, physical walk-downs of a facility and/or review of a range of facility operations and design documentation. The process followed establishes a systematic framework for the identification of hazards that could lead to accident initiators for a facility or process in a fuel cycle.

The DOE accident analysis handbook presents the process for nuclear facility hazard and accident analysis (8). The development of accident initiators is structured around hazards identified from a hazard checklist.

Accident Progression Development

The initiating events identified in the first step of an assessment form the basis for accident progression development. As in the case of reactor event scenario development, the approach for a facility or process in a fuel cycle focuses on identifying:

- SSCs that can be credited to prevent or mitigate the consequences from an initiating event where potential for hazardous material release exists.
- Administrative controls (e.g., procedures) that can be credited to prevent or mitigate the consequences from an initiating event where potential for hazardous material release exists.

The various credits lead to changes in evolution of an accident that must be represented in analysis tools capturing the:

- Magnitude and rate of hazardous material and energy release into the atmosphere of a facility or the environment.
- Transport of hazardous material and energy through the atmosphere of a facility or the environment.

Accident Scenario Analysis

The accident scenarios identified in the previous step are typically grouped into the following classes based on the phenomena that must be represented.

- Criticality scenarios
- Fire scenarios
- Explosion scenarios
- Spill scenarios
- Chemical reaction scenarios
- Natural phenomena scenarios
- Man-made external event scenarios

More detailed discussion of how these scenarios are represented is presented in the DOE accident analysis handbook (8). Guidance for accident analysis is provided in the leak path factor analysis guidance report (10). The SCALE and MELCOR packages will be used to perform the assessment of how radiation and radionuclide hazards develop and evolve. The evolution of chemical hazards is simulated using the MELCOR code, with the magnitude and rate of introduction of chemical hazards into the atmosphere of a facility are established based on procedures specified for DOE facility safety assessments (8).

4.3.3. Related Activities

There are three main activities underway at the NRC which supports development of analytical capabilities to perform independent non-LWR fuel cycle safety analysis with SCALE and MELCOR.

1. Volume 3 of the NRC non-LWR vision and strategy (2)
2. ATF/HBU/HALEU activities in RES and NMSS
3. Nuclear data gap analysis for non-LWRs in NRR

The majority of code development and non-LWR assessment efforts for SCALE and MELCOR are expected be contained within already scheduled activities in the first two projects. In the first project for non-LWR severe accident analysis (e.g., Volume 3 of the NRC non-LWR Vision and Strategy (2)), development of demonstration plant models with SCALE and MELCOR for representative non-LWRs is already included. For the second project on ATF/HBU/HALEU, SCALE front end studies of 5-20% enriched fuel are included, including criticality validation extensions for SCALE into intermediate spectra systems. In the last project, using the reference non-LWRs from volume 3, the nuclear data gaps and validation gaps will be assessed and additional validation cases created to extend SCALE's validation basis, e.g. with more graphite or lithium-moderated critical experiments.

5. CONCLUSIONS

A plan has been proposed to assess code readiness for performing safety evaluations of non-LWR fuel cycle stages. Compared to existing LWR fuel cycle analysis, non-LWR fuel cycles involve different fuel fabrication processes as well as spent fuel management systems. In addition, some designs involve very different concepts for fuel and fission product retention (e.g., fluid-fueled MSR do not involve solid, contained fuel and some fission products are continuously being distributed throughout the reactor system as a result of fission production deposition and fuel processing). All of the differences result in revisiting the NRC's analytic capabilities for radionuclide inventory, tracking, criticality, decay heat, shielding and other radionuclide and non-radionuclide hazards during the various stages of non-LWR fuel cycles.

This Volume proposes to develop reports that will characterize any code development, modelling needs and assessments for the following 10 advanced reactor fuel cycle topical areas:

1. Enrichment and UF6 Handling up to 20 wt%
2. TRISO Fuel Kernel Fabrication
3. Uranium Metallic Fuel Fabrication
4. Fast Reactor Fuel Fabrication
5. Pebble TRISO Fuel Fabrication
6. FHR Fuel Cycle Analysis (Berkeley Mk. 1)
7. HPR Fuel Cycle Analysis (INL Design A-MET)
8. SFR Fuel Cycle Analysis (MET-1000/VTR)
9. HTGR Fuel Cycle Analysis (PBMR-400)
10. MSR Fuel Cycle Analysis (MSRE)

Information in item 1 on enrichment and UF_6 handling is valid for all of the non-LWR fuel cycles. Reports 2-5 will describe the code development and modelling needs to assess the safety of various fabrication processes, which will provide a new non-LWR fleet with fuel. For these first 5 reports, criticality and chemical hazards are expected to be the main concerns, and modelling needs for SCALE and MELCOR codes, respectively, will be described. For the 5 specific non-LWR fuel cycle analysis reports, non-proprietary reference plant models for which we have already gained experience will be considered along with fuel cycle specific details including fuels (e.g., metallic fuel for INL Design A) to more closely represent those designs likely to be submitted to NRC in the near-term.

In addition to demonstrating computer code readiness for non-LWR fuel cycle safety analysis, another important outcome of this work will be the additional SCALE and MELCOR models covering the fuel cycle (e.g., simple criticality modeling for TRISO coating machines or radionuclide inventory and tracking models for MSR spent fuel tanks). These will be made publicly available along with the reports and documentation for the input decks.

It is important to note that this effort is focused around code assessments to primarily support the demonstration of NRC readiness with respect to regulation of non-LWR fuel cycle operations. The existing SCALE and MELCOR code development effort (under Volume 3 of the NRC Non-LWR Vision and Strategy (2)) is expected to provide sufficient capabilities to carry out both reactor and non-reactor safety analyses. Any additional items identified as part of these first-of-a-kind code assessments are expected to be minor.

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