Universal Canister System (UCS) for Advanced Reactor Waste Forms in Mined and Borehole Repositories – 24189

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ABSTRACT

Advanced nuclear reactors are intended to provide dispatchable, zero-carbon energy to address the global challenge of climate change. Yet their adoption may be hindered by a lack of an integrated waste management system. To address this gap, Deep Isolation is leading a project to develop a Universal Canister System (UCS) for the safe storage, transport, and disposal of spent nuclear fuel or high-level waste from advanced reactors. Once encapsulated in a UCS canister, the material will be ready for any option the waste owner may choose in the future, such as long-term storage, disposal in a mined geologic disposal facility, or disposal in a deep borehole repository.

This paper reviews the progress of the project team's efforts to date, the risks identified throughout the course of execution, and a forecast of the work yet to be performed prior to completion of the project in July 2025. Once complete, the UCS will provide a disposal option for a multitude of advanced reactor waste streams before some of the reactor designs have progressed to construction and commissioning for use. Furthermore, UCS design efforts include requirements for compatibility with existing licensed systems for transport and storage, eliminating the need for any repackaging once waste streams are loaded and sealed within the UCS, and providing a truly universal solution for the disposal of advanced reactor waste streams.

INTRODUCTION

Although advanced nuclear reactors aim to provide dispatchable, low-carbon energy to address the global challenge of climate change, their adoption may be hindered by a lack of an integrated waste management system. An ongoing concern in the nuclear industry is that development of novel closed fuel cycles without standardization may lead to diverse recycling and disposal requirements and challenges.

To address this challenge, Deep Isolation is developing a universal canister system (UCS) for advanced reactor waste forms with associated waste acceptance criteria (WAC). The UCS is intended to be compatible with both mined and borehole repositories to support cost-effective nuclear waste disposal options and to provide flexibility for a broad range of advanced fuel forms and waste products.

Funded by a grant from the U.S. Department of Energy's Advanced Research Projects Agency – Energy (ARPA-E), this three-year project – Universal Performance Criteria and Canister for Advanced Reactor Waste Form Acceptance in Borehole and Mined Repositories Considering Design Safety (UPWARDS) – will map existing and emerging waste forms in the advanced nuclear space against a matrix of geologic disposal options, all while providing a disposal solution that allows for direct loading into waste canisters and the elimination of the need to repackage prior to disposal.

The UPWARDS project comprises four parallel work streams:

1. University of California Berkeley (UCB) is leading research efforts to characterize applicable advanced reactor high-level waste streams as suitable representative candidates for consideration in UCS design parameters. To date, the project is focusing on waste forms that are trending nearest to market implementation including vitrified waste streams, as well as those that provide the most direct path to disposal, such as tri-structural isotropic (TRISO) particles, compacts, and assemblies from high-temperature reactors and frozen halide salts from molten salt reactors [1]–[3]. The project team is also coordinating with other ARPA-E-supported performers to ensure applicability and compatibility of the UCS design to other novel waste forms. These waste characterization efforts will include laboratory experimentation to fill in knowledge gaps for those novel waste forms, with an effort to identify key data to be used in repository performance modeling efforts.
2. NAC International (NAC) is leading the canister design effort through the establishment of functional requirements, design specifications, and a preliminary canister design. The preliminary design takes into account the most limiting aspects of structural, thermal, criticality, and radiological shielding performance. The design effort will culminate with the fabrication of a UCS prototype by the end of the project.
3. Deep Isolation and Lawrence Berkeley National Laboratory (LBNL) are developing source term and repository models for a screening performance assessment of loaded UCS canisters in a conventional mined repository, a vertical borehole repository, and a horizontal borehole repository.
4. With consideration of the workstreams identified above, Deep Isolation is developing generic waste acceptance criteria for the UCS, to be paired against a matrix of repository configurations.

The goal of the UPWARDS project is to deliver a canister design – coupled with generic waste acceptance criteria – that has universal compatibility across the range of potential back-end operations: storage, transport, and ultimate disposal of spent fuel and high-level waste. By directly loading waste into the UCS, the need to repackage waste streams for ultimate disposal will be eliminated, thereby reducing disposal costs.

The UPWARDS project makes use of the expertise of each partner to deliver joint, interdependent components of an integrated waste management system tailored to the needs of the advanced reactor industry in the U.S. and in international export markets. This interconnectivity between partner work scopes is shown below in Figure 1.

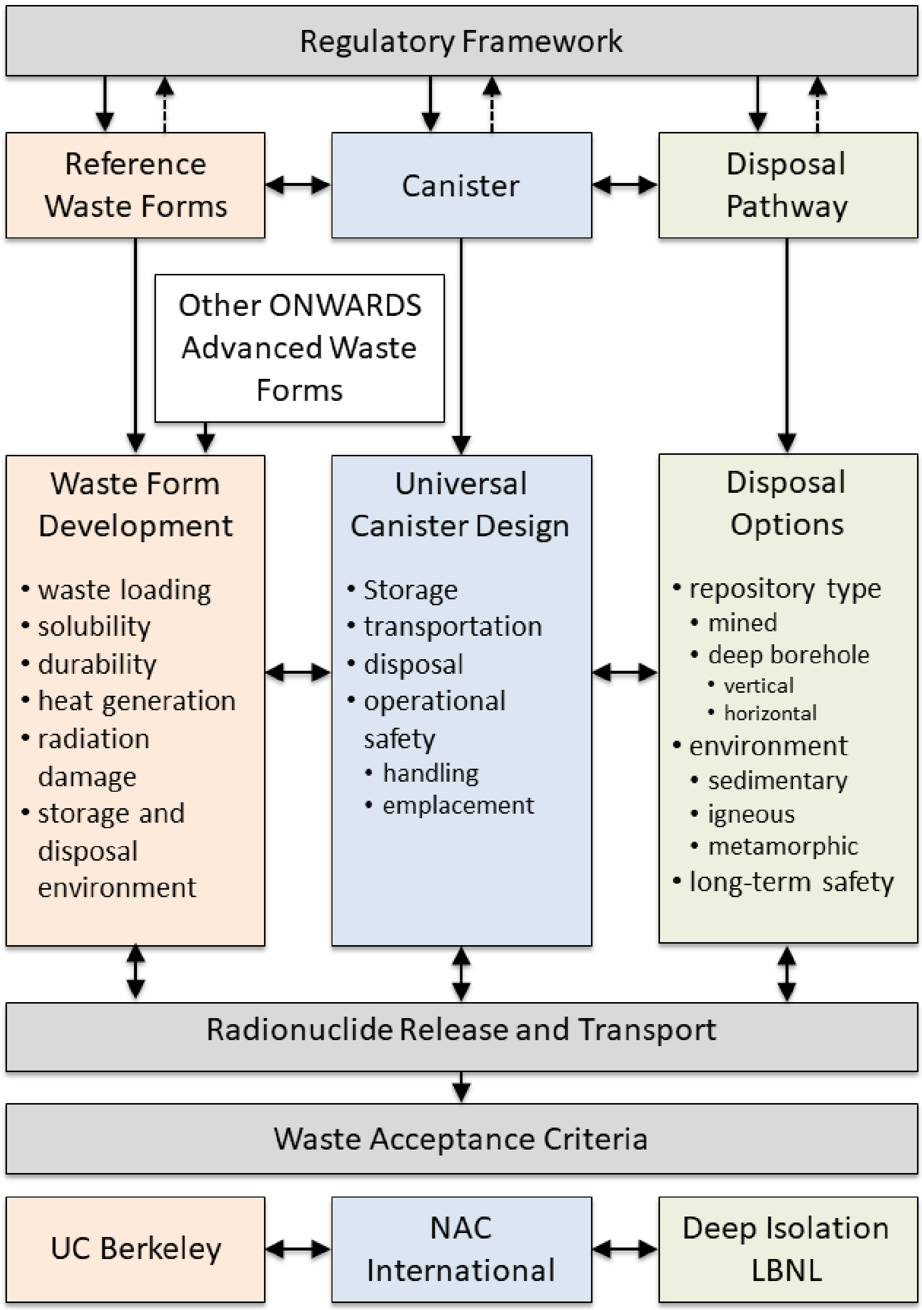

Figure 1. An Integrated Waste Management System.

DISCUSSION

Waste Form Development and Characterization

Advanced reactors may utilize three major fuel forms produced from mining and processing ore, or recycling of spent nuclear fuel from existing reactors.

These include *metallic fuels* and *molten salt fuels* that are produced by pyro-processing, and *solgel-based fuels* (TRISO and vibropac) that are produced by aqueous processing or by pyro-processing with a final aqueous polishing step. These fuel forms may be loaded into pins or directly utilized depending on the advanced reactor design.

While the range of next generation reactor fuel and waste forms is broad, our project team strives to accommodate a meaningful representation of potentially limiting cases by investigating at least three categories of waste forms. Some of these waste forms are novel, while others are developed from existing and well-understood processes or correspond with the direct disposal of a reactor’s fuel form. UCB is leading initial efforts to identify and research knowledge gaps in waste-canister interactions and repository performance, with an emphasis on identification of specific waste form categories and representative selections to direct further research. Inclusion criteria are focused on loading capacity, dissolution rates, thermal stability, limited swelling upon irradiation, mechanical stability, and resistance to fire. To date, UCB has identified vitrified (glass) waste, TRISO particles, compacts, and assemblies, and frozen halide salts from molten salt reactors as likely candidates for further research and inclusion [4]. In some instances, pressurized water reactor (PWR) fuel assemblies serve as an existing and well-understood waste form to benchmark certain waste form properties and behaviors.

Each category of waste form has a combination of common and unique technical issues for packaging and for performance in UCS canisters. For example, design tradeoffs occur when considering the addition of inert materials in the waste form matrix to increase the leach resistance and stability of the waste forms at the expense of increased waste volume and number of canisters. In some cases, as with TRISO fuels, the diluting material is already present, and the major questions involve whether some type of volume reduction may eventually be warranted. In other cases, as with molten salt wastes, the major questions involve whether to process wastes even further, adding diluting materials to change the waste form properties and heat loads, or to directly dispose of the material without further processing and dilution. Because all processing and dilution steps involve the generation of low-level radioactive waste streams and potential hazards for workers, a key goal for the UPWARDS project is to assess and understand tradeoffs inherent in making these decisions.

Vitrified Waste

Vitrification is the conversion of liquid or solid waste into a glass. The process involves co-melting glass-forming additives, known as frit, with the waste to form a homogeneous molten liquid and allowing the mixture to solidify as a glass in a suitable container. This process results in a waste form that contains radionuclides homogeneously distributed throughout an amorphous glass network that are released at rates equal to or less than the rate at which the glass matrix elements are dissolved. These glasses are traditionally comprised of borosilicate or aluminoborosilicate matrix elements, but may also be extended to other formulations such as lanthanide borosilicates (LaBS) and phosphate-based matrices. The process is currently implemented at the US DOE’s Defense Waste Processing Facility (DWPF) and Orano la Hague in France. A general process overview can be seen in Figure 2 below [5].

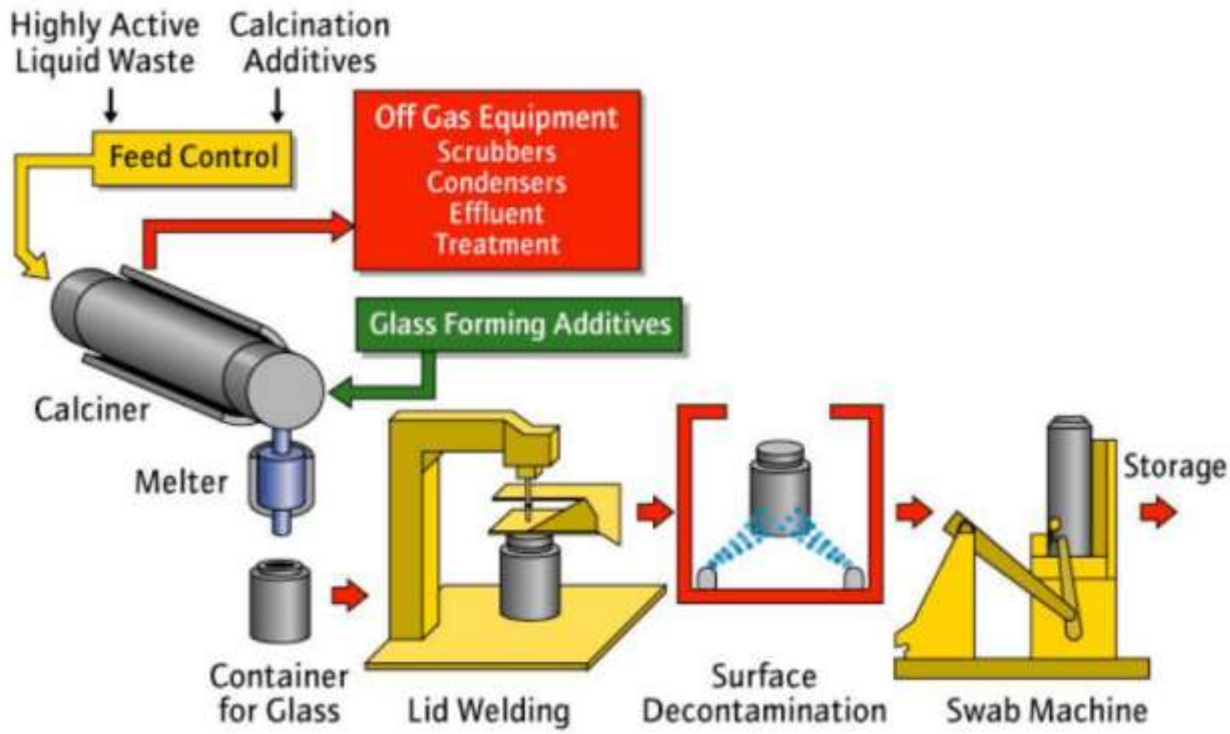

Figure 2. Vitrification Process Overview.

The UPWARDS project team visited the DWPF in March 2023 to gain a better understanding of the vitrification process and any potential constraints to compatibility with the proposed UCS. An early-identified concern is the temperature of the heated mixture, which may reach 1200°C before being poured into a canister and air-cooled over the next 48 hours. At that temperature profile, a direct pour of molten glass would incur deleterious effects on the performance properties of the UCS based on current material selection assumptions. To mitigate this concern, the project is considering a “can-in-can” approach, where molten waste forms will be directly poured into an inner container capable of withstanding the thermal gradients of molten waste forms prior to loading into a UCS.

Although traditional borosilicate waste forms have been thoroughly studied over the past half century, there exists uncertainty in the material’s degradation behavior in a range of repository-relevant conditions over extended time periods. This uncertainty has been reflected in several repository performance assessments (by Nagra in Switzerland [6] and the US DOE for Yucca Mountain [7]) for which significantly pessimistic functions and dissolution rates have been selected to conservatively model waste form durability. More recently proposed compositions of vitrified waste forms (i.e., LaBS), which show promise for use with advanced reactor waste streams, are significantly less studied and require further examination in repository-relevant conditions.

TRISO

TRISO is a highly engineered fuel form with interest from multiple companies and institutions due to its substantial safety benefits. At the smallest level, TRISO fuel resides in spherical particles ~1 mm in diameter.

These particles consist of uranium-based ceramic or alloy surrounded by a graphite buffer, which is in turn surrounded by pyrolytic carbon and structural silicon carbide (SiC). The particles are then imbedded in a graphite matrix to form either pebbles (~6 cm diameter) or cylindrical compacts (12 mm diameter, 25 mm long), as shown in Figure 3 and Figure 4, respectively [8], [9]. TRISO pebbles form blanket layers while compacts can be arranged into large prismatic assemblies.

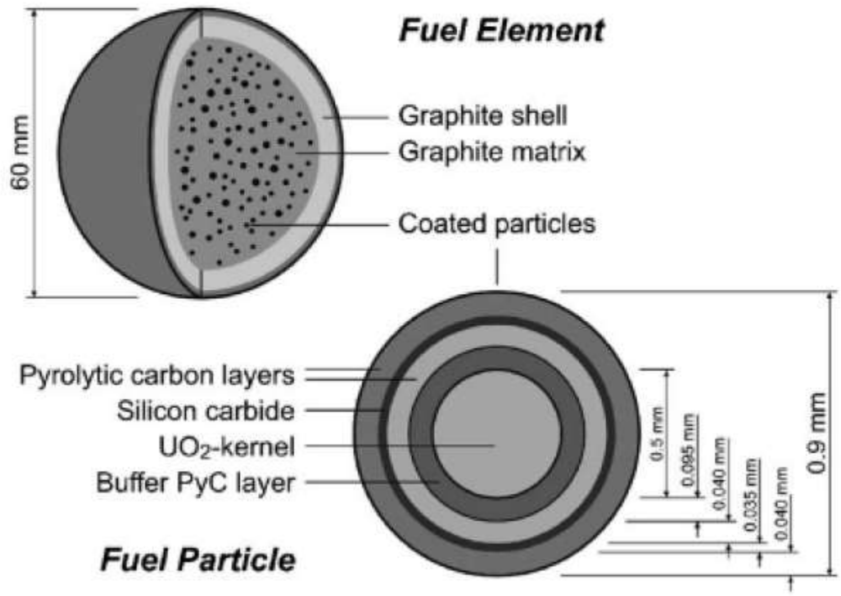

Figure 3. TRISO Fuel Particle and Element Graphic

Figure 4. TRISO Fuel Compact.

Due to the durability TRISO fuels exhibit under operating reactor conditions, this fuel form can be considered as a potential candidate for direct disposal, in particular when used with recycled material as in deep-burn plutonium-based TRISO.

Unlike vitrified waste forms, radionuclides are not homogeneously distributed throughout the TRISO particle, and there are several layers with different dissolution behaviors to consider. Processes such as radionuclide diffusion through the layers of a TRISO particle during reactor operation, multiple failure mechanisms, and variations in the reactions ground water species have with each layer of the particle further increase ambiguity in accurately modeling the durability of TRISO particles in pebbles or compacts, or as individual particles if volume reduction is performed, under repository-relevant conditions. It is necessary to better understand the corrosion mechanisms that govern the degradation of TRISO fuel in all of these forms to facilitate the creation of more accurate source term models for this waste form.

Molten Salt

A third waste form could come from molten salt reactors or from pyro-reprocessing of spent nuclear fuel, which have gained interest in recent years. In a thermal-spectrum, liquid-fuel molten salt reactor, a fuel salt consisting of Li and Be fluorides and either U, Pu, or Th fluorides would run through a primary loop; in a fast-spectrum, liquid-fuel molten salt reactor, chloride salts would be used. A coolant salt would take heat from the primary loop in a heat exchanger. A schematic of a molten salt reactor system is shown in Figure 5 [10]. Likewise, for pyrochemical reprocessing, chloride salts are used, and fission product wastes would be in the form of chloride salts.

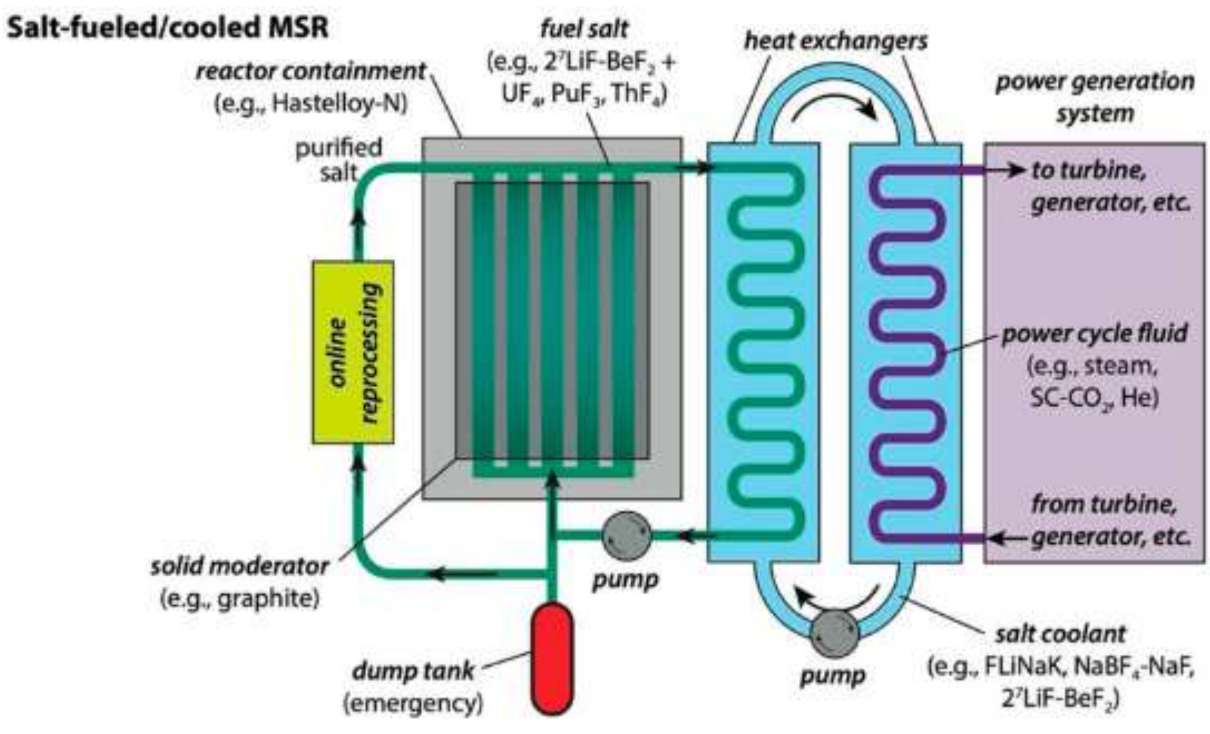

Figure 5. Molten Salt Reactor Concept.

These waste streams may be applicable for direct disposal, but likely lack significant durability in repository-relevant conditions. If this is the case, stabilization of molten salt waste forms may be possible without constituent separation (whole salt), or they could undergo separations such as dehalogenation (separated salt). In both stabilization processes, waste would be converted to some form of mineral, ceramic, glass, and/or ceramic-metal composite.

Similar to vitrified waste, stabilized salt waste is likely to result from melting material above 300 to 500°C, necessitating a “can-in-can” approach. These waste forms represent the least studied of all waste forms and require significant experimentation to identify degradation mechanisms and to determine their durability in repository-relevant conditions.

UCB is completing its initial gap analysis research of the above waste forms and will utilize the results of that research to develop laboratory experimentation plans, with an aim to fill the identified knowledge gaps with analytical data. The compiled research- and experiment-based data sets will be used as inputs to a performance assessment screening modeling for disposal in a variety of repository configurations. Current expectations are that UCB will examine the durability of LaBS, TRISO particles, and frozen halide waste forms in a range of repository conditions to provide data for source term modeling. Various material characterization and isotopic labeling techniques may be utilized to identify the mechanisms involved with TRISO particle and frozen halide salt dissolution upon exposure to various leaching conditions.

Preliminary Design of a Universal Canister

NAC began the design effort for the UCS through the development of a Functional Requirements Specification (FRS) and a UCS Design Specification. The FRS established limiting conditions and requirements for the UCS based on reviews of applicable NRC regulations for storage, transportation, and disposal, as well as reviews of geological, chemical, and radiological conditions anticipated for the canister design life. The FRS provided initial limits for the canister design on depth, pressure, temperature, decay heat, and canister design lifespan (sum of maximum storage time and anticipated maximum retrievability period[1]). These limits, documented in the FRS, informed the UCS Design Specification, in which the Project team aligned on canister material selection (duplex stainless steel 2205) and assumptions for initial canister dimensions such as shell and lid thicknesses necessary for compliance with regulatory requirements and industry standards (e.g., ASME Boiling and Pressure Vessel Code). Additionally, the specification documents establish the project team’s intentions that the UCS will be compatible with existing licensed systems for storage and transport of spent nuclear fuel.

Using the limits established in the FRS and UCS Design Specification, the project team developed a conceptual design for the UCS in three (3) sizes, or classes. The initial conceptual design of the UCS is based on the drillhole canister (DHC) developed by NAC and Deep Isolation (Figure 6), which is designed to contain a single PWR fuel assembly and was discussed previously at Waste Management Symposia 2023 [11].

[1] Transportation duration is considered negligible and not considered in the canister design lifespan.

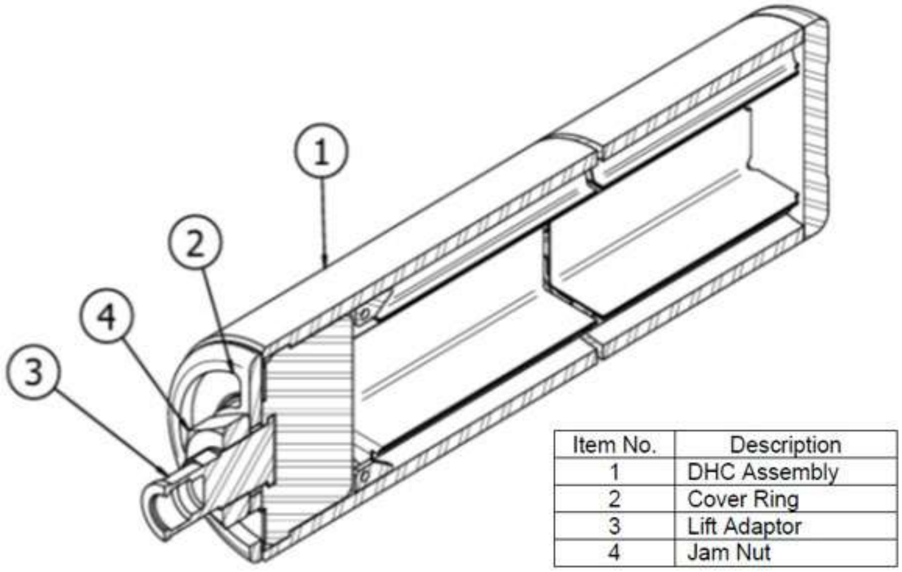

Figure 6. Drillhole Canister Preliminary Design (Disposal Configuration, Rev. D).

All three UCS canister classes are slightly shorter in length than the DHC to accommodate anticipated quantities of waste forms and provide margin to fit in existing storage and transportation cask volumes. Canister classes are differentiated mainly by the inner diameter (ID) and shell thickness. The smallest canister, Class 1, has a similar ID (12.75 in) and thickness (1 in) to the DHC, and can accommodate discharged TRISO compacts, some forms of molten salt waste, and potentially smaller containers of vitrified waste. The Class 2 canister has a larger ID (14.5 in) and thickness (1.125 in), can accommodate the same waste forms as Class 1 canisters, but is specifically designed to fit multiple TRISO graphite blocks. The Class 3 canister has the largest ID (17.5 in) and thickness (1.375 in), can fit all waste forms which fit in the other classes, but is specifically designed to house commercial vitrified waste containers common in the nuclear industry. The compatibility matrix shown in Figure 7 below illustrates which waste forms are intended to fit within each UCS canister class.

UCS Class	Governing Waste Form	UCS Cavity Dia. (in)	TRISO Graphite Block	Discharged TRISO Compacts	Vitrified Waste Canisters	Molten Salt Waste
1	PWR Fuel (☐8.54" Max.)	12.75				
2	TRISO Graphite Block (~∅14.12")	14.50				
3	Std. Vitrified Waste Canister (~∅18")	17.50				

Compatibility Legend

= compatible; = potentially compatible; = not compatible

Figure 7. Matrix of Waste Form Accommodation by Canister Class.

Preliminary analysis efforts are underway to validate the UCS design for limiting (i.e., bounding) scenarios in structural, thermal, criticality, and shielding evaluations. The results of these analyses may prompt changes to some or all canister designs.

To validate the design from a structural standpoint, NAC is performing three specific analyses:

1. *UCS Shell External Pressure and Stacking Load Analysis* – this calculation evaluates the UCS shell for stresses and buckling stability due to combined external pressure and stacking loads (hydrostatic and lithostatic) resulting from borehole emplacement.
2. *UCS Handling Stress Analysis* – this calculation analyzes stresses on the UCS shell and lift adapter at a repository and borehole surface handling facility. It also evaluates the UCS for a borehole retrieval and the accident scenario of a stuck canister retrieval.
3. *UCS Transfer Bottom End Drop Evaluation* – this calculation simulates a 20-foot drop of the UCS onto an unyielding surface and addresses a UCS drop in a borehole.

To ensure the UCS external temperature did not exceed material limits establishing in the project's Design Specification, a bounding thermal performance was determined for a UCS in both vertical and horizontal borehole disposal configurations loaded with TRISO fuel blocks and vitrified waste. To validate the UCS design in these conditions, a pre-closure transient thermal analysis is being performed. Once complete, the thermal analysis will validate or inform changes to the UCS design based upon the criteria in the UCS Design Specification.

To determine if the design contains sufficient shielding, NAC is preparing a *UCS Source Term and Shielding Evaluation for TRISO Fuel*. This analysis uses a three-dimensional Monte Carlo N-Particle (MCNP) model of the UCS loaded with consolidated TRISO fuel compacts to determine the amount of shielding required to satisfy a predetermined occupational dose rate limit.

The UCS MCNP model shall also be used to demonstrate subcriticality. NAC is preparing a *UCS Criticality Evaluation for TRISO Fuel* to validate the design against the most limiting conditions. For this calculation, NAC expects either transportation or disposal (mined geological repository) may be the most limiting condition. The goal is to demonstrate that a single transportation cask loaded with multiple UCS canisters containing TRISO fuel compacts or pebbles, subjected to variations in internal and external flooding, and surrounded by a cylindrical water reflector remains subcritical under normal conditions of transport (NCT) and hypothetical accident conditions (HAC), as defined by NRC regulations [12]. Using the most reactive configuration determined for transport, the UCS is then analyzed for both borehole disposal and mined geological repository disposal. Both configurations are analyzed for optimum moderation in-leakage to the UCS.

The current UCS designs (Classes 1, 2, and 3) have been 3-D printed for design confirmation and display purposes.

As analyses of the initial design concludes and our team gains a better understanding of the chemical, thermal, structural, and radiological attributes of different advanced reactor waste forms, the team will revisit UCS drawings, the FRS, and the UCS Design Specification and consider whether design changes are warranted. Additionally, we will explore manufacturing options and commence with first article prototype manufacturing to provide proof of principle on fabricability, identify manufacturing requirements, and recommend fabrication methods and tooling to establish a commercially optimum cost basis for the UCS.

Integrated Safety and Performance Assessment Screening Model

We are developing a modeling framework which can be applied to a range of waste forms and generic disposal pathways. Throughout the project, interdependent source-term and repository performance models will be developed using TOUGHREACT and iTOUGH2 modeling tools [13], [14]. These models will be based, in part, on previous work performed by Deep Isolation and LBNL related to the performance of commercial spent PWR fuel assemblies in deep boreholes [15]–[18]. The UPWARDS project expands on these models through the analysis of novel waste forms from advanced fuel cycles, while also evaluating disposal in a mined repository scenario in an effort to provide true universality to the application of the UCS.

Deep Isolation has conducted reviews of repository and disposal concepts to formulate scenarios to be examined by the numerical model and to delineate the performance envelope. A list of potentially safety-relevant features, events, and processes (FEPs) was compiled, ranges for continuous parameters were specified, and discrete simulation cases were identified. These scenarios were then refined, prioritized, and preliminary performance metrics were formulated. Preliminary sensitivity analyses with respect to heat loading and disruptive scenarios were performed to identify thermal bounding cases and waste form- and canister-related attributes that impact total system performance.

Use of a Screening Model

The universality of the waste disposal concept and the formulation of WAC requires the evaluation of a performance envelope for a sufficiently wide variety of waste forms and disposal options. This evaluation will be performed using a screening model that integrates relevant aspects affecting repository performance. This includes a source-term model that calculates the release of radionuclides from the UCS, accounting for variability in initial inventory, waste form degradation rates, and release mechanisms. It also includes assumptions about the longevity of the UCS and the thermal output from radioactive decay. This source term is then used to calculate the migration of radionuclides through components of the engineered barrier system (EBS) and the geosphere towards the accessible environment, accounting for diffusive and advective transport processes, as well as retardation effects. The parameters characterizing flow and transport will be varied (using stochastic sampling methods) to delineate the universality of the proposed solution, which reflects different repository concepts, types of host rocks, and uncertainties and variabilities in hydrogeological and geochemical properties of these formations. Note that the model does not explicitly simulate mechanical processes, but accounts for their potential effects by evaluating appropriate ranges in waste degradation, canister, and formation properties. Finally, to obtain a single, relevant metric that allows for quantification of the overall performance of the disposal system for advanced reactor waste streams, the maximum exposure dose to an individual living near the repository will be calculated and compared to a stringent reference dose criterion. In addition to examining the performance of the integrated waste management system for a wide range of conditions, its robustness to discrete disruptive events of low probability will also be examined. Some of the key submodels comprising this integrated safety assessment model are described below.

Inventory

The inventory of radionuclides contained in the waste form depends on the advanced reactor type, its fuel, and its operational parameters. Moreover, potential waste processing (recycling and conditioning) and the elapsed time until the waste is ready for disposal determine the activity and thermal energy of the radionuclides that are encapsulated in the waste form and contained in an individual UCS canister. It is expected that only a relatively small number of the isotopes comprising advanced reactor waste forms will eventually affect the exposure dose (i.e., become safety-relevant). This means that isotopes that need to be tracked in the integrated safety assessment model comprise only a subset of the total number of isotopes that make up the initial inventory. In general, safety-relevant isotopes are those with a considerable initial inventory, combined with a relatively long half-life, high toxicity, and – most relevant – high mobility in the EBS and geosphere (i.e., high solubility in groundwater and weak retardation properties such as their propensity to sorb onto the solid grains of the host rock). Note that these characteristics depend on the geochemical conditions in the repository's environment. The screening of the inventory to arrive at a set of safety-relevant radionuclides can be based on previous safety analyses which consistently show ^{129}I, ^{79}Se, ^{36}Cl, and ^{99}Tc as the predominant safety-relevant isotopes, as well as – under certain conditions and for conservative assumptions – some actinides and decay products. However, should the inventory of advanced reactor waste streams be considerably different from that of conventional reactors, the safety-relevance of these radionuclides will be determined using the screening performance assessment (PA) model developed as part of this project.

Waste Degradation

At the time of disposal, potentially harmful radionuclides are initially encapsulated in the waste form and contained in the canister. Once the canister has been breached, the waste form will be exposed to native pore water and will start degrading. The degradation rate depends on temperature, fluid chemistry, and the physical and chemical conditions of the waste form itself. A number of (generally conservative) waste degradation models have been developed and are implemented in the PA screening model. Moreover, a detailed source-term model has been developed that accounts for the coupled thermal-hydrological-chemical degradation processes under repository conditions. This model also examines the potentially delayed dissolution and release of radionuclides from the degrading waste form. This source-term model will either be directly linked to the PA model used for screening purposes, or suitably abstracted to provide the time-dependent radionuclide release rates needed by the screening model.

Thermal Performance

The decay heat emanating from radioactive waste may have an impact on repository performance by affecting canister integrity, accelerating waste degradation, increasing chemical reactions and diffusion rates, leading to thermal expansion of the fluid and the rock, or potentially generating steam (if exceeding the in-situ boiling temperature), which are conditions that induce advective flow or affect the properties of the buffer or host rock in response to chemical alterations or thermal stresses. These and other secondary thermal effects make it essential to include heat generation (and the natural geothermal gradient) in the PA screening model as a time-dependent heat release rate, which is calculated based on the total initial inventory. Thermal properties (and their uncertainties) will also need to be accounted for to properly calculate heat dissipation mechanisms in the repository.

Model Integration

As discussed above, a key aspect of our approach is to combine and couple all relevant submodels into an integrated PA screening model capable of efficiently, but with sufficient accuracy, calculating repository performance (reflected by the peak dose value) for a wide variety of waste forms, disposal concepts, and repository conditions in an attempt to delineate the performance envelope of the proposed waste management system for advanced reactor waste streams. The integration of a sufficiently detailed source-term model into the large-scale PA screening model is a key objective of this work, as it provides the necessary link between the new waste inventories and waste forms from advanced reactors, the newly developed UCS, and the performance of both a conventional mined repository and innovative disposal concepts using horizontal and vertical boreholes.

Development of Waste Acceptance Criteria

A key deliverable for the UPWARDS project will be the development of generic WAC that are applicable to a wide range of advanced reactor waste forms, the UCS, and different repository types and host formations. This will represent the culmination and total integration of all project work streams noted above.

To begin this effort, we have conducted a review of the US regulatory landscape for application of the UCS in both mined and borehole repositories. This included a review of regulations promulgated by the US Nuclear Regulatory Commission and Environmental Protection Agency related to the storage, transport, and disposal of spent nuclear fuel and high-level waste. The resultant report identified uncertainties and opportunities around topics such as reliance on engineering versus natural barrier systems, retrievability timing and logistics, repository configuration and design features, plug and backfill, closure timing, reference dose rate to the public, assumed groundwater conditions for modeling, and inadvertent human intrusion [19]. Additionally, we conducted a review of WAC in general as well as specific WAC in existing US and international waste streams. The report made related recommendations that form the basis of key project assumptions as the design and analysis of the disposal system move forward. Additional key conclusions include the decision to assume a generic mined repository concept as part of the PA screening analyses as opposed to a specific mined repository location and consideration of the applicability of existing US and international WAC for vitrified waste during the development of the UCS WAC.

In 2024, we will begin development of the generic WAC for UCS applications. The WAC will address, as applicable, fuel-form classification, waste form physical dimensions, fuel consolidation, inventory, waste-loading capacity, thermal output, waste degradation rate, canister design requirements, and associated inspection and documentation requirements already in existence for 10 CFR Part 71 and 10 CFR Part 71 license holders. In parallel with and following the WAC development, we will develop WAC pairing matrices, identifying the applicability of individual criteria for different waste forms with suitable canister classes and repository configurations. These pairings will factor in technical limitations that are identified through the completion of the design and screening performance assessment analyses.

CONCLUSION

A key challenge to the widespread deployment of advanced reactors is a lack of assured disposal solutions for these novel waste forms. To address this challenge, the UPWARDS Project is developing a Universal Canister System capable of facilitating the safe transportation, storage, and disposal of advanced reactor waste in both mined and deep borehole repository configurations. To date, we have defined initial design parameters, designed a canister, conducted canister design analyses (structural, thermal, criticality, and shielding), researched multiple advanced reactor waste forms of interest (i.e., vitrified waste, TRISO, and molten salt), conducted a review of the regulatory landscape for disposal, and began developing models to assess the long-term safety performance in generic repository configurations. Many of these efforts, such as canister design and model development, are iterative and will be revisited throughout the remainder of this three-year project. By project completion in July 2025, the UPWARDS team will also develop waste acceptance criteria and a fabricated prototype canister for subsequent demonstration and testing.

ACRONYMS

ARPA-E	Advanced Research Projects Agency - Energy
ASME	American Society of Mechanical Engineers
CFR	Code of Federal Regulations
DHC	Drillhole canister
DOE	Department of Energy
DWPF	Defense Waste Processing Facility
EBS	Engineered barrier system
FEPs	Features, events, and processes
FRS	Functional Requirements Specification
HAC	Hypothetical accident conditions
ID	Inner diameter
LaBS	Lanthanide borosilicates
LBNL	Lawrence Berkeley National Laboratory
MCNP	Monte Carlo N-Particle
NAC	NAC International, Inc.
NCT	Normal conditions of transport

NRC	Nuclear Regulatory Commission
PA	Performance assessment
PWR	Pressurized water reactor
TRISO	Tri-structural isotropic
UCB	University of California, Berkeley
UCS	Universal Canister System
UPWARDS	Universal Performance Criteria and Canister for Advanced Reactor Waste Form Acceptance in Borehole and Mined Repositories Considering Design Safety
WAC	Waste Acceptance Criteria

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