INL/EXT-09-17427
Rev. 1

Conceptual Safety Design Report for the Remote-Handled Low-Level Waste Disposal Project

May 2010

Idaho National Laboratory

The INL is a U.S. Department of Energy National Laboratory
operated by Battelle Energy Alliance

INL/EXT-09-17427
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http://www.inl.gov

Prepared for the
U.S. Department of Energy
Office of Nuclear Energy
Under DOE Idaho Operations Office
Contract DE-AC07-05ID14517

ABSTRACT

A new onsite, remote-handled low-level waste (LLW) disposal facility has been identified as the highest ranked alternative for providing continued, uninterrupted, remote-handled LLW disposal for remote-handled LLW from the Idaho National Laboratory and for spent nuclear fuel processing activities at the Naval Reactors Facility. Historically, this type of waste has been disposed of at the Radioactive Waste Management Complex. Disposal of remote-handled LLW in concrete disposal vaults at the Radioactive Waste Management Complex will continue until the facility is full or until it must be closed in preparation for final remediation of the Subsurface Disposal Area (approximately at the end of Fiscal Year 2017).

This conceptual safety design report supports the design of a proposed onsite remote-handled LLW disposal facility by providing an initial nuclear facility hazard categorization, identifying potential hazards for processes associated with onsite handling and disposal of remote-handled LLW, evaluating consequences of postulated accidents, and discussing the need for safety features that will become part of the facility design.

NOTE:

This document summarizes the hazards for processes and associated safety strategies to address the hazards associated with onsite handling and disposal of remote-handled low-level waste. A new onsite facility has been identified as an alternative for providing continued remote-handled low-level waste disposal capability in support of ongoing Department of Energy missions at the Idaho site. However, a decision has not been made by the Department of Energy to develop a new onsite disposal facility. The decision, following all required analyses and evaluation of the impacts of all viable alternatives, will be made in accordance with the National Environmental Policy Act of 1969. Use of words indicating requirements or specifying intention, such as “shall” or “will,” are used for the convenience of discussion or to indicate requirements or activities that are conditioned on a decision to develop a new onsite disposal facility. Such usage should not be construed to mean that a final selection of an alternative has been made.

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ACRONYMS

ALARA	as-low-as-reasonably-achievable
ARF	airborne release fraction
ATR	Advanced Test Reactor
CSDR	conceptual safety design report
CVAS	cask-to-vault adapting structure
DBA	design-basis accident
DOE	Department of Energy
DR	damage ratio
HC	hazard category
INL	Idaho National Laboratory
LLW	low-level waste
LPF	leak path factor
MAR	material-at-risk
NPH	natural phenomena hazard
NRF	Naval Reactors Facility
PDSA	preliminary documented safety analysis
PHA	preliminary hazards analysis
RSWF	Radioactive Scrap and Waste Facility
RWMC	Radioactive Waste Management Complex
SDC	seismic design category
SSC	structure, system, and component
TRU	transuranic

Conceptual Safety Design Report for the Remote-Handled Low-Level Waste Disposal Project

1. INTRODUCTION

The Idaho National Laboratory (INL) routinely generates contact-handled (less than 200 mrem/hour on contact) and remote-handled (greater than 200 mrem/hour on contact) low-level waste (LLW) from facility operations. Historically, INL has disposed of its LLW in a disposal facility located at the Radioactive Waste Management Complex (RWMC). This facility includes disposal pits and concrete vaults. As part of ongoing cleanup activities at INL, closure of RWMC is proceeding under the Comprehensive Environmental Response, Compensation, and Liability Act (42 USC 9601 et seq. 1980). Disposal of remote-handled LLW in concrete disposal vaults at RWMC will continue until the facility is full or until it is closed in preparation for final remediation of the Subsurface Disposal Area (approximately at the end of Fiscal Year 2017).

On July 1, 2009, the Department of Energy (DOE) approved a mission need statement for the INL Remote-Handled LLW Disposal Project to develop replacement remote-handled LLW disposal capability in support of INL's nuclear energy mission and the Naval Nuclear Propulsion Program (DOE-ID 2009). The continuing nuclear mission of INL, associated ongoing and planned operations, and Naval spent nuclear fuel activities at the Naval Reactors Facility (NRF) require continued capability to appropriately dispose of remote-handled LLW. Development of a new onsite disposal facility has been identified as the highest ranked alternative for providing continued, uninterrupted remote-handled LLW disposal capability at INL (as documented in INL 2010a).

This conceptual safety design report (CSDR) has been prepared in accordance with DOE-STD-1189-2008, "Integration of Safety into the Design Process," and does the following:

- Documents and establishes a preliminary inventory of hazardous materials, including radioactive materials and chemicals
- Documents and establishes the preliminary hazard categorization of the proposed facility
- Identifies and analyzes primary facility hazards and facility design-basis accidents (DBAs)
- Provides an initial determination, based on the preliminary hazard analysis (PHA; INL 2010b), of safety-class and safety-significant structures, systems, and components (SSCs)
- Evaluates the security hazards that can impact the facility safety basis
- Includes a commitment to the nuclear safety design criteria of DOE Order 420.1B, "Facility Safety."

This CSDR is based on TFR-483, "Technical and Functional Requirements Remote-Handled Low-Level Waste Disposal Project," conceptual design report (INL 2010c), and other associated documents. In addition to this CSDR, other safety analysis documents will be prepared in accordance with the requirements of 10 CFR 830, "Nuclear Safety Management," Subpart B, "Safety Basis Requirements."

1.1 Waste Stream Overview

The proposed Remote-Handled LLW Disposal Project will be designed and constructed to support disposal of remote-handled LLW streams generated at the Idaho site. A summary of these waste streams is provided in Table 1.

Table 1. Waste streams proposed for the Remote-Handled Low-Level Waste Disposal Project.

Waste Stream	Generator	Description
Resins	INL Advanced Test Reactor (ATR) Complex	ATR produces ion-exchange resins from pool and reactor operations.
	NRF	NRF produces ion-exchange resins from pool operations. Currently, waste is disposed of in the RWMC vaults in waste containers transported using a 55-ton cask.
Activated Metals	INL ATR Complex	ATR produces activated metals during reactor core internal changeout operations approximately every 8 years. These components require an approximate 8-year decay time and are in storage at the ATR Complex. Previous disposal has been at RWMC.
	NRF	NRF produces activated metals during routine operations. Currently, waste is disposed of in the RWMC vaults in 55-ton scrap cask liners.
	INL Materials and Fuels Complex	Materials and Fuel Complex will generate activated metals during waste segregation operations for waste removed from storage at the Radioactive Scrap and Waste Facility (RSWF).
Various	INL	ATR and MFC may product a variety of remote-handled LLW streams from new INL programs and waste segregation operations at RSWF.

Ion-exchange resins from pool and reactor operations are generated at the ATR Complex and from pool operations at NRF. ATR ion-exchange resin is generated approximately four to six times annually from reactor loop and reactor ion-exchange systems. The generation rate depends on reactor operations and varies during the years when core internal changeouts are performed. The ion-exchange resin waste stream has typical contact exposure rates up to 15 R/hour, although individual waste containers may have higher contact exposure rates.

ATR also produces activated metals during reactor core internal changeout operations (approximately every eight years). These components require decay time before they can be handled for disposal and are currently in temporary storage at the ATR Complex. NRF produces activated metals from examination of test components and during routine operations removing irradiated non-fuel components from spent nuclear fuel modules. The activated metal waste streams have typical contact exposure rates up to 30,000 R/hour, although individual waste containers may have higher contact exposure rates.

In addition, activated metals and other remote-handled LLW streams are expected from new INL programs and from processing of remote-handled LLW stored at RSWF. These materials can contain a variety of radionuclides and can have contact exposure rates up to 30,000 R/hour, although individual waste containers may have higher contact exposure rates.

1.2 Facility and Mission Overview

INL-generated radioactive waste has been disposed of at RWMC since 1952. RWMC disposal practices have evolved over time, including changes in the disposal facility, waste treatment, and containers. Current disposal operations within the Subsurface Disposal Area are limited to subsurface burial of INL-generated LLW. Waste emplaced in the Subsurface Disposal Area is classified as either remote or contact-handled LLW, depending on radiation levels.

Providing continued disposal capability for remote-handled LLW supports the Office of Nuclear Energy, Science, and Technology's mission "to lead the DOE investment in the development and exploration of advanced nuclear science and technology." Without established, viable remote-handled LLW disposal capability, ongoing and future Office of Nuclear Energy, Science, and Technology programs at INL would be adversely impacted as remote-handled LLW disposal options would need to be considered on a program-by-program basis, resulting in increased costs and schedule delays. The lack of remote-handled LLW disposal capability also may impede DOE's ability to initiate new programs at INL.

Remote-handled LLW disposal capability also is critical to meeting the National Nuclear Security Administration's mission to "provide the United States Navy with safe, militarily effective nuclear propulsion plants and to ensure the safe and reliable operation of those plants." All spent nuclear fuel from the Navy's nuclear-powered fleet is sent to NRF for examination, processing, dry storage, and eventual shipment to a permanent geologic repository. A reliable disposal path for remote-handled LLW generated during spent nuclear fuel handling and packaging operations is essential to NRF's continued receipt and processing of Navy spent nuclear fuel, to the Naval Nuclear Propulsion Program, and to national security. The mission need statement for the INL Remote-Handled LLW Disposal Project, created as a result of evaluating INL-generated LLW disposal options, is as follows:

> *The INL will develop replacement remote-handled low-level waste disposal capability ... to support cost-effective, efficient operations in support of INL's nuclear energy mission and the Naval Nuclear Propulsion Program. Such disposal capability is required to enhance ongoing Departmental and National mission-based research, defense, and energy programs.*

The proposed Remote-Handled LLW Disposal Project will provide concrete disposal vaults needed to dispose of remote-handled LLW. Specifically, the proposed Remote-Handled LLW Disposal Project will be designed to do the following:

- Provide a concrete vault disposal system that can accommodate transportation packages and waste containers that are currently being used for waste disposal of remote-handled LLW generated at NRF

- Provide a concrete vault disposal system that can accommodate transportation packages and waste containers that are anticipated to be used for disposal of activated metals generated at ATR and from new missions, including processing of waste currently in storage at RSWF

- Provide a concrete vault disposal system that can accommodate transportation packages and waste containers that are currently being used for disposal of remote-handled LLW ATR resins

- Accommodate waste container placement methods currently in use at RWMC and continue to use the existing remote-handled loading equipment and proven procedures for the NRF shipping cask

- Provide support equipment needed to unload waste containers that are anticipated to be used by ATR, RSWF, and other future mission activities

- Provide road access that can accommodate anticipated loads from transportation package transport vehicles

- Place waste containers into vaults while providing the appropriate level of shielding and worker protection

- Provide a vault/plug assembly to provide shielding and to act as a water barrier to prevent surface water intrusion into the concrete vaults

- Allow access to individual vaults without disturbing adjacent vaults

- Provide crane access areas to support placement of waste materials into the vaults, as needed, that will support the combined weight of a loaded crane during placement (crane, transportation package, transportation package-to-vault adapter components, shielding/sealing plug, and waste containers)

- Provide shielding sufficient to reduce radiation levels on top of the vaults to the levels specified in DOE Order 5400.5, "Radiation Protection of the Public and the Environment," when the plugs are in place.

1.3 Site Location

At this stage of project development, it is assumed that the facility will be a stand-alone facility that does not use the services of any existing INL facilities. Perimeter fencing will be constructed to provide protection from human and animal intrusions and to allow for proper access control. Specific interfaces between the proposed facility and existing INL facilities will be defined once a location is selected in accordance with the National Environmental Policy Act process.

An overall facility area for siting purposes will be based on the required site-specific components and is estimated to be between 4 to 6 acres (1.6 to 2.4 ha). The total number of vaults that are needed may change depending on the ability to support a vault depth capable of accepting three waste containers per vault.

A siting study (INL 2010d) has been conducted to support the National Environmental Policy Act process that considers possible locations within INL that are best suited for locating the proposed facility. The study considers five key elements: (1) regulations, (2) key assumptions, (3) conceptual design, (4) facility performance, and (5) previous INL siting studies. The study uses a five-step process to identify, screen, evaluate, score, and rank multiple sites located across INL.

Once the final location is decided, the safety-basis documents will be updated to reflect the specific siting location and any site-specific characteristics that could impact nuclear safety considerations for the proposed facility.

2. CONCEPTUAL DESIGN DESCRIPTION

2.1 Facility Structure and Layout

The proposed Remote-Handled LLW Disposal Project will be designed and constructed similar to the remote-handled LLW concrete disposal vaults currently in use in the RWMC Subsurface Disposal Area. This will accommodate, to the maximum extent possible, uninterrupted operations at the generating facilities and will capitalize on the operations experience and cost efficiencies of current remote-handled LLW disposal practices. The vaults will be constructed of precast concrete cylinders (i.e., pipe sections) stacked on end and placed in a honeycomb-type array (see Figure 1). A removable concrete plug will be set on top of the stacked precast concrete cylinders to serve as a radiation shield and water barrier.

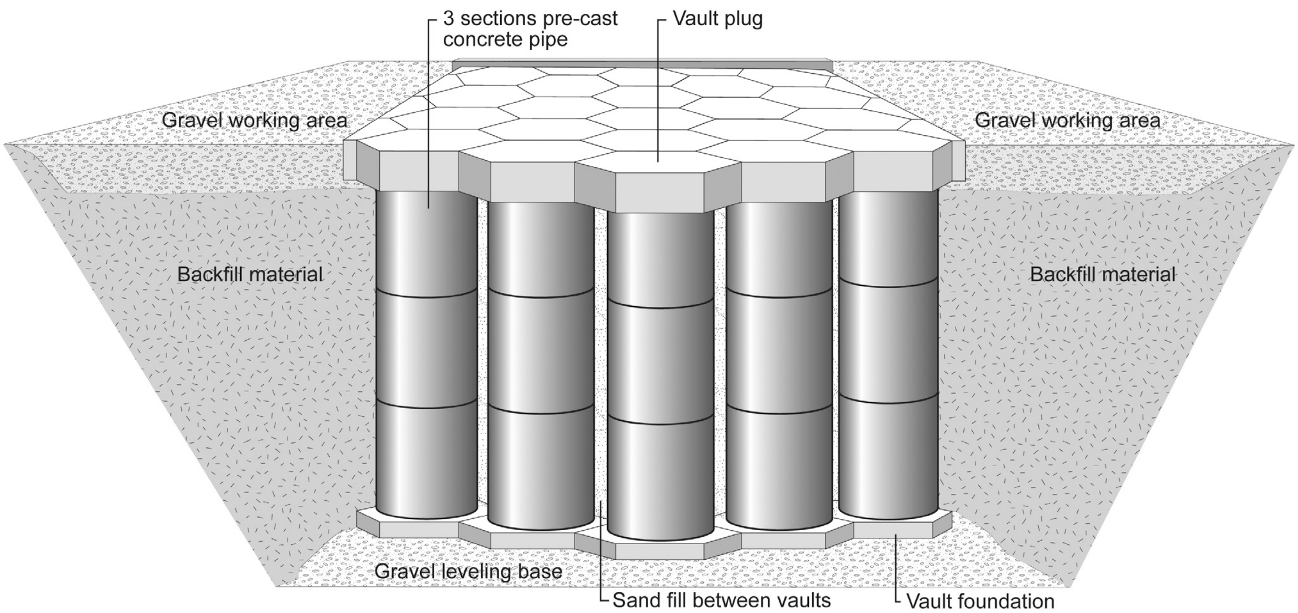

Figure 1. Concrete vault layout.

The proposed facility layout is based on the assumption that the facility would be a stand-alone facility and would provide its own administration buildings and infrastructure to support disposal operations. If a site is selected that is located in the vicinity of an existing facility, then new construction of some of the infrastructure components may not be needed (e.g., the administration building).

The facility would be laid out in a manner to allow trucks entering the disposal facility to have straight access to the unloading area next to the disposal vaults. The crane and other miscellaneous equipment required for completion of the transportation package-to-vault transfer operation will be staged before arrival of the waste containers. Figure 2 illustrates the facility configuration and includes a photo that shows the equipment currently staged for operation at RWMC. The new facility will use these same methods and will set up the necessary equipment in a similar configuration.

The total number of vaults that will be constructed will depend on the depth of surficial sediment at the specific site that is selected for the facility. The general layout in the conceptual design report shows the areal extent of the vaults, as determined using a vault depth that can accommodate disposal of two waste containers per vault. In this configuration, a minimum of 160 vaults will be needed for NRF waste, 60 vaults for ATR resins, and 23 vaults for activated metals from ATR processing of co-mingled, remote-handled LLW currently stored in RSWF and new INL programs. If the selected site has sufficient surficial sediment to accommodate three waste containers per vault, the total number of required vaults would be reduced by one-third.

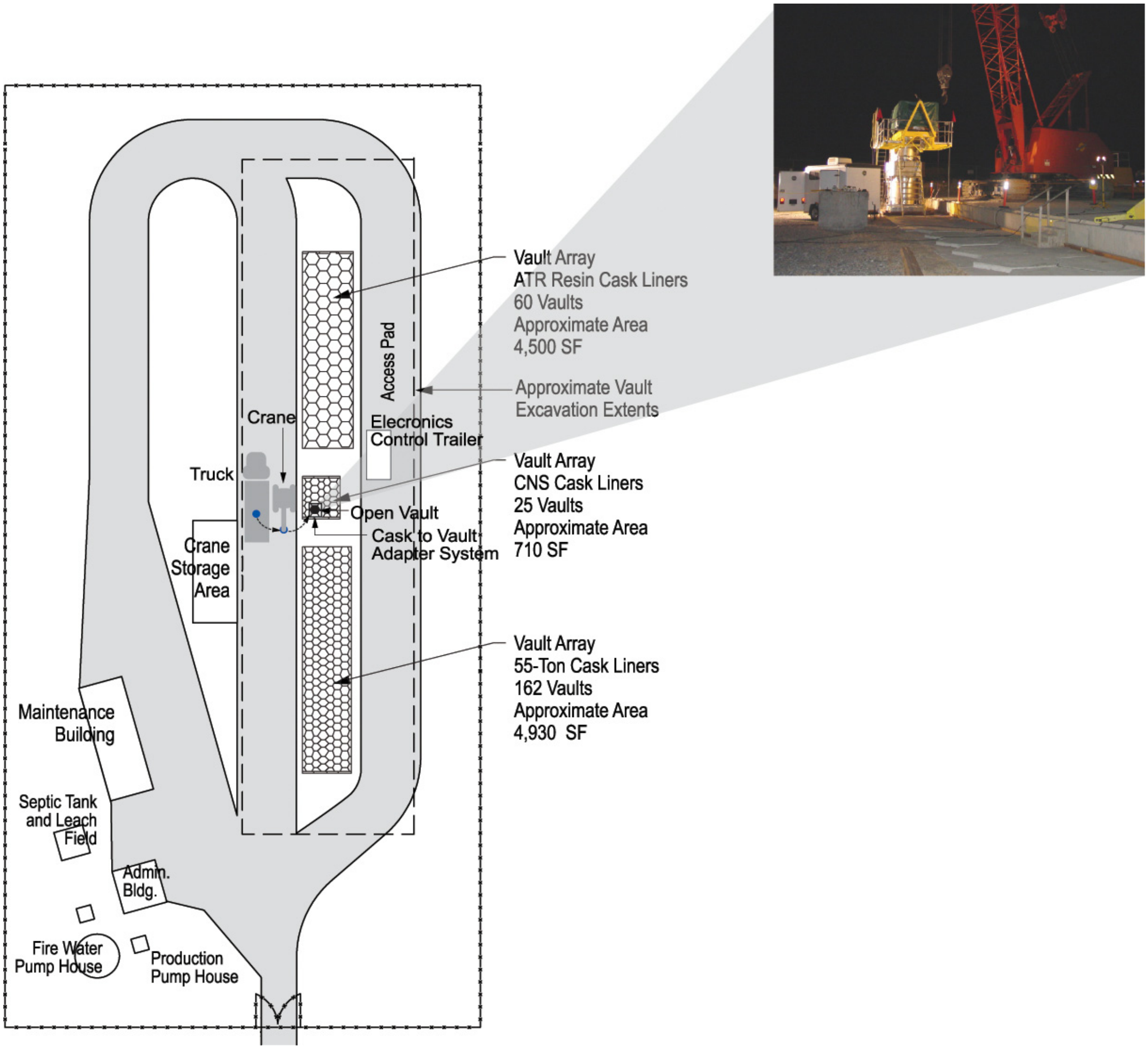

Figure 2. Proposed layout for the Remote-Handled Low-Level Waste Disposal Project.

The following are major components of the proposed facility:

- **Vaults**—The vaults will be aligned vertically to allow multiple remote-handled LLW containers to be stacked on top of the previous one inserted in a vertical orientation. Vaults used to dispose of NRF waste will be designed to interface with the existing cask-to-vault adapting structure (CVAS) and the 55-ton scrap cask. Remaining vaults will be designed to interface with the appropriate transportation package and associated transfer system.

- **Vault plugs**—A removable concrete plug will be placed on top of each of the stacked cylinder vaults. The plug will serve as a radiation shield for placed waste and will act as a water barrier to prevent surface water intrusion into the concrete vaults.

- **Crane**—The crane that is currently in use at RWMC will be disassembled, refurbished, and transported to the new disposal facility. This crane is a mobile two track crane with a lifting capacity of approximately 140 tons (127,000 kg). If it is determined that the existing crane will not be available, a new crane with similar lifting capacity will need to be procured for the facility.

- **Waste container**—Remote-handled LLW will be packaged into steel waste containers at the generating facilities. One liner at a time is shipped within a shielded transportation package from the generating facility to the disposal facility. Upon arrival at the appropriate vault array location, the waste container will be transferred from the transportation package into the concrete vault. These liners perform an important safety function as a contamination barrier. An example of a waste container is shown in Figure 3.

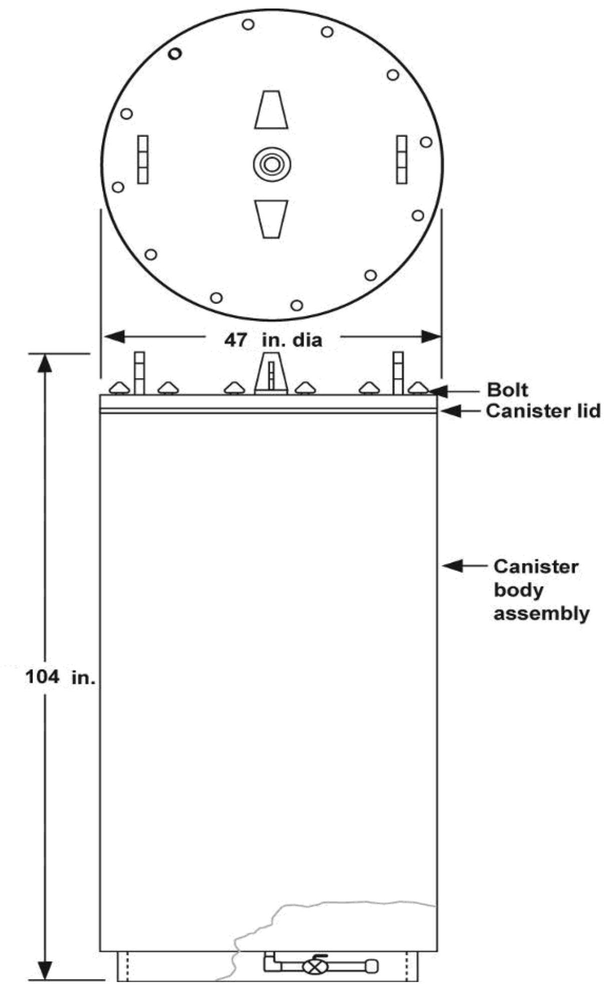

Figure 3. Waste container used inside the 55-ton scrap cask.

- **CVAS**—The CVAS currently located at RWMC will be transferred to the new disposal facility. All supporting equipment and components, such as the lifting rigging and control trailer, also will be made available for use.

- **Staging and storage area**—Staging and storage pads will be provided within the facility for operating equipment. These pads will be constructed using pit run gravel with a crushed gravel top surface. Areas will be provided for storage of the crane; the CVAS components, including the working platform; the bearing pad; the shield plugs; and the electrical control trailer.

- **Administrative and other supporting infrastructure**—Additional support and administrative structures and services are included in the conceptual design, which include the following:
  - Administration building
  - Electrical distribution
  - Maintenance enclosure
  - Temporary transportation package holding area
  - Equipment decontamination area
  - Access roads
  - Video monitoring
  - Firewater supply.

Additional details of these listed facility components may be found in the conceptual design report (INL 2010c).

2.2 Process Description

This section describes the overall process used for disposal of remote-handled LLW in concrete vaults at INL. Figure 4 shows the general process that is currently being used for NRF remote-handled LLW disposal in the vaults at RWMC. It is assumed that all future waste received from each of the INL generating facilities will be received and disposed of using this same, or similar, sequence of activities. This process is the basis for development of the technical and functional requirements, the conceptual design report, and hazard and accident analysis for the proposed disposal facility.

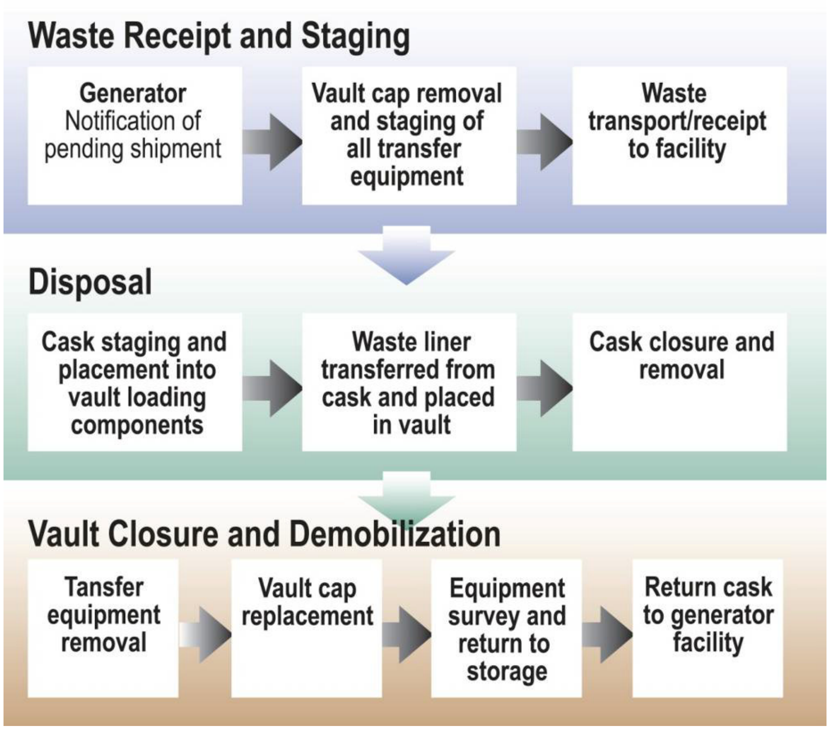

Figure 4. Facility process flow diagram.

Remote-handled LLW destined for disposal will be packaged into shielded transportation packages with waste containers. The waste containers will normally consist of cylindrical containers designed specifically for the transportation package systems used. It is assumed that remote-handled LLW will be transported from NRF to the proposed disposal facility using the same 55-ton scrap cask that is used at RWMC (see Figure 5). Operations involving this cask will be substantially the same as those used at RWMC. The operational system associated with the transportation packages and transfer systems used by other INL generators will be determined once specific waste container designs and transportation package systems are identified.

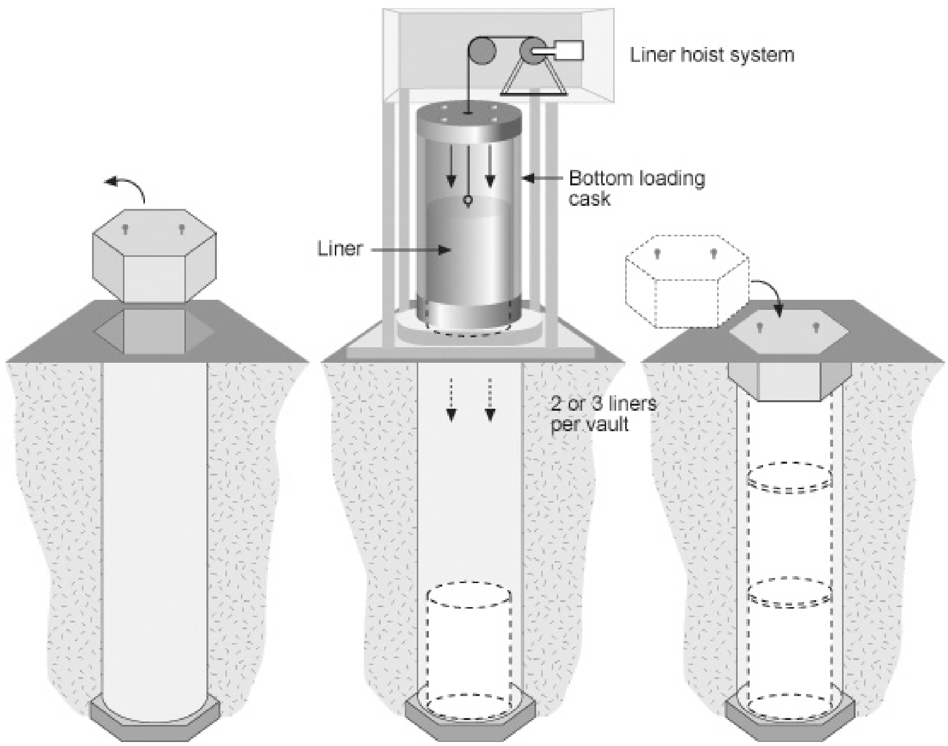

Figure 5. Cask waste container placement method at the Radioactive Waste Management Complex.

Only one waste container will be transported at any one time. No safety and health monitoring or surveillance, other than normal radiological surveys, is anticipated to be required as a part of normal operations. There may be additional surveillances required during transport and handling of specific waste containers with greater than 30,000 R/hour contact exposure rates. These requirements will be identified as part of the hazard and accident analysis process.

The current NRF waste container placement process consists of the following steps:

1. Once waste is transported to the site (using the NRF 55-ton scrap cask), a crane is used to remove the top plug on the vault and to position the CVAS on top of the open vault.

2. The 55-ton scrap cask (see Figure 6) is removed from the transporter and placed on the CVAS using the crane.

3. Using a remote-operated hoisting system, the waste package is unloaded from the bottom of the cask and lowered into the disposal vault.

4. The cask is then closed and the hoisting system with the associated equipment is removed from the top of the vault.

5. The vault is then closed.

Figure 6. A 55-ton scrap cask used for transporting waste to the disposal facility.

The specific operational systems and placement procedures that will be used in association with the other transportation package systems used for disposal of the remote-handled LLW at the proposed facility will be determined once the generating facilities identify their specific waste container configurations. It is assumed that the following general operational sequence would be used for placement of the waste containers into the associated disposal vaults:

1. Once waste is transported to the site, a crane will be used to remove the top plug on the vault and prepare the vault opening for waste container placement.

2. Using the crane, the waste container will be removed from the transportation package using the associated waste container handling equipment and appropriate shielding and positioned over the disposal vault.

3. The waste container will be lowered into the disposal vault.

4. The transfer equipment will be removed and the vault plug replaced.

3. PRELIMINARY HAZARD CATEGORIZATION

3.1 Hazardous Material Inventories

Remote-handled LLW is considered to be any waste container with a contact exposure rate (including neutron and beta radiation) greater than 200 mrem/hour. If internally or externally shielded, the greater than 200 mrem/hour threshold applies to the expected contact exposure rate without shielding.

Should shielded containers be designed for placement in the proposed Remote-Handled LLW Disposal Project, such containers will not have any Resource Conservation and Recovery Act-regulated metals used as shielding.

The waste streams that will be accepted for storage at the Remote-Handled LLW Disposal Project must meet the requirements for LLW as specified in DOE Order 435.1, "Radioactive Waste Management." These requirements specify that the material must contain less than 100 nCi/g transuranic (TRU) radionuclides. At this level, there would be less than 0.34 Ci in the largest (i.e., 3,400 kg) waste container. For a maximum of three waste containers per vault, this is equivalent to a maximum TRU inventory of 1.02 Ci.

In terms of radionuclides that contribute to direct radiation exposure consequences, the maximum inventory is based on identifying radioactive material inventories that do not exceed the 30,000 R/hr contact exposure rate specified in the technical and functional requirements and the conceptual design report as the shielding design basis for transport and loading activities involving individual waste containers. In addition, the technical and functional requirements specify 2000 Ci ^{60}Co as the design basis for the vault shield plugs.

3.2 Comparison of Inventories to Threshold Quantities

Based on the preliminary assessment of the anticipated remote-handled LLW streams and a comparison with DOE-STD-1027-92, "Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports," the Remote-Handled LLW Disposal Project would have an initial hazard categorization of a Hazard Category (HC)-2 nuclear facility. This preliminary categorization is documented in the safety design strategy (INL 2010e). The total Remote-Handled LLW Disposal Project radioactive material inventory anticipated to be present in the facility at a given time will exceed the HC-2 threshold quantity values for several radionuclides per DOE-STD-1027-92. However, DOE-STD-1027-92 supplemental guidance provides for facility categorization modification in the final hazard categorization process considering 1) alternative release fractions or 2) change in material subject to an accident due to facility features that preclude bringing material together or causing harmful interaction from a common severe phenomenon (facility segmentation).

The maximum vault inventory discussed in Section 3.1 is compared to threshold quantities for HC-3 and HC-2 as discussed in DOE-STD-1027-92. Fractions of the respective threshold quantities for HC-3 and HC-2 for ^{239}Pu are 2.0 and 0.02, respectively. Using ^{239}Pu as the representative TRU radionuclide is considered conservative because the inhalation dose conversion factor for ^{239}Pu is the highest for all TRU radionuclides that are likely to be present. For non-TRU radionuclides that are anticipated to be present in activated metal and resin waste streams, the inventory discussed in Section 3.1 is also less than the HC-2 threshold quantity for ^{60}Co of 190,000 Ci. These values exceed any anticipated Curie content for these radionuclides.

Based on these waste stream inventories, individual vaults can be categorized as HC-3. Individual vaults can be considered separate facility segments per guidance in DOE-STD-1027-92, based on the following:

- Each vault will consist of three pre-cast concrete cylinders stacked on end and vertically aligned. Each vault will be buried in an array with a sand backfill that will completely separate the individual vaults.

- Each vault will be placed on a concrete vault foundation and will have a separate removable concrete plug placed on top of the cylinder to serve as a radiation shield and a water barrier.

- Once these vaults are completely buried, there are no facility operations that can bring the contents of more than a single vault together. Once a vault is filled with up to three waste containers, the vault plug is put in place, thereby isolating the vault from the adjacent vaults comprising the array.

Because individual vaults at the Remote-Handled LLW Disposal Project individual facility segments and the maximum hazard categorization for a single vault is no greater than HC-3, the hazard category for the Remote-Handled LLW Disposal Project can be HC-3. This position will be further evaluated during development of the preliminary documented safety analysis (PDSA) and documented safety analysis per NS-18101, “INL Safety Analysis Process,” to determine if modification to the facility HC is appropriate based on alternative release fractions or additional facility segmentation consideration.

4. DESIGN BASIS ACCIDENTS

The list of potential hazards identified in the PHA (INL 2010b) is used as an outline for the development of a hazards assessment and facility safety-basis documents. It incorporates extensive experience and lessons learned from similar facility nuclear safety designs and operations. The current stage of the conceptual design process requires a preliminary identification of hazards and DBAs and the need for safety SSCs. Further detailed analyses will be completed in conjunction with development of the PDSA.

The waste containers associated with the various remote-handled LLW waste streams are anticipated to contain irradiated hardware, highly radioactive process materials, or nuclear reactor system resins. Though some variability in form is expected, most of the radioactive material will be in the form of irradiated solids or ion-exchange resin with contaminants attached to the resin particle surface.

4.1 Facility-Level Design Basis Accidents

A PHA (INL 2010b) was performed for the hazards that have the potential to result in an uncontrolled release of radioactive or hazardous material and affect the offsite public, collocated workers, facility workers, or the environment. In performing the PHA, the location, hazard, initiating conditions, likelihood, unmitigated consequences, and preventive and mitigative features are considered.

Consequence evaluation of the postulated accident scenarios associated with the proposed facility requires a qualitative evaluation of those hazards. This evaluation encompasses internal events, man-made external events, accident initiators at nearby facilities, and natural phenomenon hazards (NPHs). Sabotage and terrorism are not addressed in the analysis. Internal events occur as a result of facility operations and encompass all operational modes. Hazard identification involves determining the following for the facility:

- The material at risk (MAR) (i.e., the type and amount of radioactive and hazardous material that is potentially releasable), including the form and location of the material

- Potential energy sources and initiating events that could directly result in injury to workers or affect the inventory of radioactive and hazardous materials.

With respect to MAR, the maximum radionuclide content of any waste container for the purposes of evaluating inhalation dose consequences is 0.34 Ci TRU (see Section 3.1). Although there may be specific waste streams with significantly higher Curie quantities of other radionuclides, this level of TRU material results in the highest inhalation dose consequences from an airborne release due to the high dose conversion factors. Assuming that this material is released in powder form, the maximum inhalation dose consequences are less than 1 rem to the facility worker and are negligible.

With respect to direct radiation exposure consequences, a 30,000 R/hr contact exposure rate (see Section 3.1) results in consequences of ~10 rem to the facility worker for an exposure duration of two minutes in the absence of appropriate shielding and/or handling procedures during transfer operations and vault storage. For waste containers with contact exposure rates >30,000 R/hr, the consequences can be higher.

It should be noted that some of the waste streams may contain combustible materials such as plastics and other combustible radioactive waste. Waste containers with resins also may be subject to radiolytic water decomposition (i.e., hydrogen production) or corrosion-induced waste container failure during long-term storage. All of these specific materials are considered individually during development of the PHA, including likelihood and consequence evaluation.

Hazardous material inventories for construction and operation of the proposed Remote-Handled LLW Disposal Project are very low in comparison to other INL operations and commensurate with existing RWMC remote-handled LLW operations. No chemicals found in the Occupational Safety and Health Act substance-specific standards have been identified that would create a potential for exposure triggering medical surveillance during construction or operation. Additionally, no highly hazardous chemicals listed in 29 CFR 1910.119, "Process Safety Management of Highly Hazardous Chemicals," (Appendix A, List of Highly Hazardous Chemicals, or Toxics and Reactives) will be generated, used, or disposed of at this facility.

The likelihood category reflects a qualitative estimate of whether the hazardous event is anticipated, unlikely, extremely unlikely, or beyond extremely unlikely. The likelihood of a hazardous event is generally the frequency of the initiating event or cause. No credit is taken for controls (i.e., design or administrative) that prevent the event. For an internal event (i.e., events initiated by equipment failure or human error), this generally results in a likelihood category of anticipated (i.e., 10^{-2} to 10^{-1} per year) because the frequency can depend on facility design and operation. The likelihood category is based on available data, operating experience, and engineering judgment. If there is uncertainty in the likelihood category, the higher likelihood category is conservatively assumed.

The consequence category reflects a qualitative estimate of potential consequences to the offsite public, collocated workers, facility workers, and environment from the hazardous event. A consequence category of high, moderate, low, or negligible is assigned for each receptor and the environment based on the unmitigated quantity of radioactive or hazardous material potentially released and the energy source for dispersion. Unmitigated means that a material's quantity, form, location, dispersibility, and interaction with available energy sources are considered, but no credit is taken for safety features (e.g., ventilation system or fire suppression) that could mitigate a hazard. If there is uncertainty in the consequence category, the more severe consequence category is conservatively assumed.

Table 2 defines the INL radiological risk evaluation guidelines used for deriving the need for safety-class or safety-significant SSCs for facility and collocated workers and the offsite public. These evaluation guidelines are documented in NS-18104, "INL Guide to Safety Analysis Methodology," as supplemented in OS-QSD-050121, CCN 202983, "Nuclear Safety Rule Supplementation Information." Determining whether the consequences from a postulated event exceed the evaluation guidelines for the corresponding likelihood for that event determines if safety SSCs or safety analysis commitments are required as a control for that event. For each of the identified DBAs, the consequence and likelihood were identified and compared to the evaluation guidelines.

Table 2. Idaho National Laboratory risk evaluation guidelines.

Event Likelihood	Collocated and Facility Worker Consequences	Offsite Public Consequences
Anticipated (10^{-2} to 10^{-1}/year)	5.0 rem (L)	0.5 rem (L)
Unlikely (10^{-4} to 10^{-2}/year)	25 rem (M)	5 rem (M)
Extremely unlikely (10^{-6} to 10^{-4}/year)	100 rem (H)	25 rem (H)

The results of the PHA (INL 2010b) are found in Table 3, which includes hazards and initiators that should be considered as the design progresses and the safety-basis documentation is being prepared. This table lists identified hazards and causes and possible preventative and mitigative responses. The table is not intended to be all-inclusive and may be updated, as required. In addition, some hazards in the table may extend beyond the scope outlined in the technical and functional requirements and the conceptual design report (INL 2010c). This presentation of potential hazards will be used in a future analysis to determine whether further accident evaluation is warranted. At the time the safety-basis documentation is developed, some potential accidents may be eliminated from further consideration. They are included here because operational experience suggests that further consideration should be given. This list will be periodically reviewed and updated as additional hazards are identified in the facility design process.

4.2 Unmitigated Design Basis Accident Analyses

Based on the results of the PHA, several postulated events are selected as representative, bounding, or unique accidents. These accidents, defined as DBAs and derived from the PHA, include the following:

- Container drop accident (bounds all radioactive material release events)
- Vehicle fuel fire (bounds all fire and explosion events)
- Direct radiation exposure during waste container handling (representative for all direct radiation exposure events)
- Severe seismic event (representative for all NPH events)
- External events (consequences bounded by other events).

The unmitigated analyses for the DBAs are summarized in the following subsections. The accidents considered here are used to determine if safety SSCs would be considered at this stage of facility design and are preliminary in nature. Experience in other underground waste disposal facilities indicates that these events typically represent the highest risk in terms of likelihood and consequence. During subsequent safety document development, these accidents will be further evaluated, documented, and peer reviewed.

Table 3. Preliminary hazards identified for the Remote-Handled Low-Level Waste Disposal Project.

Scenario Description	Likelihood Category[a]	Unmitigated Consequence Category[b]	Safety Functions	Potential Preventive and Mitigative Features	
				Design[c]	Administrative[c]
Fire and Explosion Hazards					
Large transport vehicle fire, resulting in transportation package/waste container failure, ignition of combustible remote-handled LLW, and release of radioactive material	U	Offsite public: N Collocated workers: N Facility workers: N	Prevent/mitigate radioactive material release due to fire	Robust transportation packages/waste containers function as fire barrier (P) Facility fire suppression system (M)	Fire protection program (P) INL fire department response (M) Equipment maintenance and inspection (P)
Waste container drop during placement into vault, resulting in waste container failure, ignition of combustible remote-handled LLW, and release of radioactive material	U	Offsite public: N Collocated workers: N Facility workers: N	Prevent/mitigate radioactive material release due to fire	Robust waste containers function as fire barrier (P) Facility fire suppression system (M)	Fire protection program (P) INL fire department response (M) Equipment maintenance and inspection (P)
Hydrogen buildup inside waste container, resulting in waste container breach and release of radioactive material	U	Offsite public: N Collocated workers: N Facility workers: N	Prevent/mitigate radioactive material release due to hydrogen buildup in waste container	Robust waste containers function as fire barrier (P) Facility fire suppression system (M)	Fire protection program (P) INL fire department response (M) Equipment maintenance and inspection (P)
Radioactive Material Release Hazards					
Loss of confinement due to transportation package breach caused by human error or mechanical failure during unloading of truck, resulting in release of radioactive material and direct radiation exposure	A	Offsite public: N Collocated workers: N Facility workers: L	Prevent/mitigate radioactive material release and direct radiation exposure during transportation package handling operations	Solid waste form (M) ***Robust transportation package (M)***	Employee training (P) Equipment maintenance and inspection (P) Hoisting and rigging program (P) Immediate worker evacuation (M) Vehicle speed limits (P)
Loss of confinement due to waste container breach caused by human error or mechanical failure during placement of container into vault, resulting in release of radioactive material and direct radiation exposure	A	Offsite public: N Collocated workers: N Facility workers: L	Prevent/mitigate radioactive material release and direct radiation exposure during waste container handling operations	Solid waste form (M) Waste container integrity (M) ***Waste container transfer system shielding (P)***	Employee training (P) Equipment maintenance and inspection (P) Hoisting and rigging program (P) Immediate worker evacuation (M)

Table 3. Preliminary hazards identified for the Remote-Handled Low-Level Waste Disposal Project.

Scenario Description	Likelihood Category[a]	Unmitigated Consequence Category[b]	Safety Functions	Potential Preventive and Mitigative Features	
				Design[c]	Administrative[c]
Loss of confinement due to corrosion-induced waste container failure during storage caused by corrosive agents contained in resins, resulting in release of radioactive material	U	Offsite public: N Collocated workers: N Facility workers: N	Prevent/mitigate radioactive material release during storage	Vault storage design (P) Vault storage completely enclosed underground (M)	Radiation protection program (P)
Loss of confinement due to corrosion-induced waste container failure during storage caused by water intrusion into vault, resulting in release of radioactive material	U	Offsite public: N Collocated workers: N Facility workers: N	Prevent/mitigate radioactive material release during storage	Vault storage design (P) Vault storage completely enclosed underground (M) Vault storage designed to minimize potential water infiltration (M)	Radiation protection program (P)
Direct Radiation Exposure Hazards					
Direct radiation exposure due to loss of transportation package shielding caused by human error or mechanical failure during unloading of transport vehicle	A	Offsite public: N Collocated workers: N Facility workers: H	Prevent/mitigate direct radiation exposure during transportation package handling operations	***Robust transportation package (M)***	Employee training (P) Equipment maintenance and inspection (P) Hoisting and rigging program (P) Immediate worker evacuation (M) Vehicle speed limits (P)
Direct radiation exposure during waste container (less than 30,000 R/hour) transfer to vault	A	Offsite public: N Collocated workers: N Facility workers: H	Prevent/mitigate direct radiation exposure during waste container handling operations	***Waste container transfer system shielding (P)***	Radiation protection program (D, M) Employee training (P)
Direct radiation exposure during waste container (greater than 30,000 R/hour) transfer to vault	U	Offsite public: N Collocated workers: N Facility workers: H	Prevent/mitigate direct radiation exposure during waste container handling operations	***Waste container transfer system shielding (P)*** Temporary shielding (M)	Radiation protection program (D, M) Employee training (P) Specific procedures for transferring waste containers with high contact exposure rates (P)

Table 3. Preliminary hazards identified for the Remote-Handled Low-Level Waste Disposal Project.

Scenario Description	Likelihood Category[a]	Unmitigated Consequence Category[b]	Safety Functions	Potential Preventive and Mitigative Features	
				Design[c]	Administrative[c]
External Events					
Radioactive or hazardous materials released during vault storage due to external events (e.g., plane crash, vehicle crash, or adjacent building fire/explosion)	EU	Offsite public: N Collocated workers: N Facility workers: N	Prevent/mitigate radioactive material release during vault storage	Facility design (P) Vault storage completely enclosed underground (M)	Appropriate facility siting (M)
NPH Events					
Radioactive material released due to NPH (e.g., tornado, flood, range fire, lightning, or volcanoes)	U	Offsite public: N Collocated workers: N Facility workers: N	Prevent/mitigate radioactive material release during vault storage	Facility design (P) Vault storage completely enclosed underground (M)	Appropriate facility siting (M)
Direct radiation exposure due to loss of vault shield plug integrity caused by severe seismic event	EU	Offsite public: N Collocated workers: N Facility workers: H	Prevent/mitigate direct radiation exposure during vault storage	***Vault shield plug (P)***	Surveillance program (P) Immediate worker evacuation (M) Emergency response procedures (M)

A = anticipated (10^{-2} to 10^{-1}/year); U = unlikely (10^{-4} to 10^{-2}/year); EU = extremely unlikely (10^{-6} to 10^{-4}/year); BEU = beyond extremely unlikely (less than 10^{-6}/year)

N = negligible; L = low; M = moderate; H = high

Design and administrative features are identified as detection (D), prevention (P), or mitigation (M). Potential SSCs designed as safety-class or safety-significant, or specific administrative controls are highlighted in ***bold italics***. These indicators will be re-evaluated as the preliminary safety design report and PDSA are developed.

4.2.1 Container Drop Accident

Initially, this accident was considered in the PHA (INL 2010b) as separate events for transportation package drop and waste container drop. The consequences of either event are the same, and these events are considered together in this evaluation.

This DBA event involves dropping a waste container or transportation package that results in the release of radioactive material. The transportation package is lifted from the transport vehicle using the facility crane. The waste containers are transferred to the vault using the facility crane. The drop of a waste container or transportation package results from failure of the crane or lifting and handling equipment, and is considered an anticipated event for the proposed facility. In this postulated event, the waste container or transportation package is dropped to the ground breaching multiple layers of confinement and releasing radioactive material.

Dose to the worker in this case is from both uptake of radioactive material and from direct radiation exposure to the unshielded and high radiation levels associated with this material.

The MAR for release of radioactive material in this scenario is limited to the contents of a single waste container. An unmitigated analysis is performed for which the hazardous material's quantity, form, location, dispersibility, and interaction with available energy are considered, but no credit is taken for safety features that could mitigate the consequences of a hazard. The bounding MAR, defined in Section 3.1, is 0.34 Ci ^{239}Pu in a single waste container (based on 100 nCi/g TRU). If other radionuclides are considered, then the maximum MAR for ^{60}Co a single waste container is2,000 Ci. Values used to calculate a source term for uptake in this accident are as follows:

Damage ratio (DR) = 0.1

Airborne release fraction (ARF) = 1.0 E-3

Respirable fraction (RF) = 1.0

Leak path factor (LPF) = 1.0.

The DR represents the fraction of MAR that could be affected by the postulated accident and is a function of the accident initiator and the operational scenario being evaluated. DR is determined based on engineering judgment, best available information, and prior analysis. The DR for the container drop scenario is evaluated at 0.1. It assumes that although the drop is significant enough to fail the transportation package and the waste container, only 10% of the available source term is damaged enough to be released.

The ARF and RF were taken from DOE-HDBK-3010-2004, "Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities," and are applicable to contaminated, combustible solids that experience airborne release of surface contamination by shock and vibration.

A LPF of 1.0 is a standard assumption for DOE facilities and activities and is appropriate for an outdoor event.

The source term (ST) is found by multiplying the above factors as follows:

$$ST = MAR \times DR \times ARF \times RF \times LPF$$

Dose consequences are calculated as follows:

Dose (rem) = ST × breathing rate × DCF × χ/Q

Breathing rate (3.33E-4 m^3/s) is the assumed breathing rate described in DOE Order 440.1B, "Worker Protection Program for DOE." Dose conversion factors are published in ICRP-68, "Dose Coefficients for Intakes of Radionuclides by Workers," for workers and ICRP-72, "Age-dependent Doses from Intakes of Radionuclides," for the public.

The airborne dispersion value, χ/Q, for facility workers is based on dispersion into a 10-m diameter hemisphere with an exposure time of 10 sec (i.e., the time that it takes to evacuate the 10-m hemisphere), or 3.82E-02 s/m^3.

The airborne dispersion values for downwind distances were obtained from RSAC-7 modeling for instantaneous releases using 95 percentile meteorological conditions specific to INL as described in INL/EXT-09-15275, "Radiological Safety Analysis Computer (RSAC) Program Version 7.0 Users' Manual" (INL 2009). These values are 4.08E-03 s/m^3 at 100 m and 5.32E-5 s/m^3 at 4,000 m, respectively. It should be noted that the site-specific airborne dispersion value at 100 m is higher than the value of 3.4E-03 s/m^3 specified in DOE-STD-1189 and is therefore conservative. It should also be noted that the dispersion value for the off-site public is evaluated at 4,000 m, which is the distance to the location of the nearest public receptor based on the preferred site alternatives described in the siting evaluation (INL 2010d).

Using appropriate values for each factor gives inhalation dose consequences shown in Table 4.

Table 4. Inhalation dose consequences for container drop accident.

Radionuclide	Source Term (Ci)	Facility Worker		Collocated Worker		Offsite Public	
		Inhalation DCF (rem/Ci)	Dose (rem)	Inhalation DCF (rem/Ci)	Dose (rem)	Inhalation DCF (rem/Ci)	Dose (rem)
^{239}Pu	3.40E-05	1.18E+08	5.12E-02	1.18E+08	5.47E-03	4.44E+08	2.07E-04
^{60}Co	2.00E-01	6.29E+04	1.60E-01	6.29E+04	1.71E-02	1.15E+05	4.60E-04

DCF = dose conversion factor

These results are considered conservative for this accident and are most applicable to the activated metal waste streams that most likely do not contain combustible materials. These results are also considered conservative with respect to debris and resin waste streams because, although ARF/RF values may be higher, the inventory of ^{60}Co in a single waste container would be considerably lower. The likely fission products present in the resin waste streams have significantly lower DCF values and would also result in negligible inhalation dose consequences.

Dose from direct radiation exposure may be calculated using the maximum anticipated contact exposure rate expected at the Remote-Handled LLW Disposal Project of 30,000 R/hour at near-contact, then calculated to the positions of the facility worker and collocated worker.

Assuming a point source, the dose is reduced at the rate of the inverse to the square of the distance as shown in the inverse square law:

$I_1/I_2 = (r_2/r_1)^2$, where I is radiation intensity and r is distance from the source.

The radiation intensity, I_1, of the dropped transportation package/waste container is assumed to be 30,000 R/hour at 0.1 m as a point source. Assuming an evacuation time, t = 2 min, dose to the facility worker and collocated worker is calculated as follows:

Dose (rem) = $I_2 \times t$ (min) / (60 min / hr).

Using appropriate values for each factor gives the direction radiation dose consequences shown in Table 5.

Dose rates to the public from direct radiation at an assumed 5,000 m would be negligible based on the negligible dose at 100 m for the collocated worker.

Table 5. Direct radiation exposure consequences for container drop accident.

Distance (m)	I_2 (R/hour)	Facility Worker Dose (rem)	Collocated Worker Dose (rem)
1	300	10	NA
10	3	0.1	NA
100	0.03	NA	0.001

Dose consequence evaluation guidelines for the public and workers are 0.5 rem and 5 rem, respectively, for anticipated events. As can be seen in the evaluation of dose to public and workers, the dose consequences may exceed evaluation guidelines for this event when considering both inhalation dose from uptake of radioactive material and direct radiation exposure to facility workers. Therefore, safety-significant SSCs and/or specific administrative controls may be required as shown in Table 3.

4.2.2 Vehicle Fuel Fire

The lifting and handling of transportation packages and waste containers requires use of trucks, tractor/trailer combinations, and a crane. These vehicles introduce the potential for a vehicle fire that is postulated to occur during transport or during transportation package/waste container unloading activities. Such a fire is postulated to initiate from fuel from the transport vehicle or crane and entirely engulfs the transportation package/waste container being transported or unloaded, resulting in volatilization of a fraction of the waste material being handled. This accident analysis considers only material that is affected by the thermal stresses from the fire as container boundaries are breached. Material already in storage is not involved in this accident. The likelihood of this accident is judged as unlikely based on the limited number of miles that transport vehicles would travel on the Remote-Handled LLW Disposal Project site, low speeds, and robust transportation package/waste container design that would prevent a fire from spreading to engulf the entire contents.

Dose to the facility worker, collocated worker, and public in this case is assumed to be only from intake of radiological material made airborne in the fire.

The MAR in this scenario is limited to the contents of a single waste container. The unmitigated analysis performed for this event takes no credit for safety features that could mitigate the consequences. For the unmitigated analysis, the MARs are the same as those discussed in Section 4.2.1.

Values used to calculate the source term for uptake in this accident are as follows:

DR = 0.1

ARF = 5.0E-4

RF = 1.0

LPF = 1.0.

The DR of 0.1 is based on engineering judgment of the amount of material impacted from an engulfing fire where the source material contains combustibles and originates within multiple layers of protection.

The ARF and RF were taken from DOE-HDBK-3010-2004 and are applicable to contaminated, combustible solids exposed to thermal stress. Materials adhering to the surface are ejected by the expansion and contraction of the material during heating and oxidation.

LPF of 1.0 is a standard assumption for DOE facilities and activities and is appropriate for an outdoor event such as what is being evaluated.

The source term and inhalation dose consequences are evaluated as discussed in Section 4.2.1 and results are shown in Table 6. Although a fire may be a longer duration event than a container spill, airborne dispersion is evaluated as discussed in Section 4.2.1 for an instantaneous release. This is conservative because the airborne dispersion parameters for longer-duration events or events such as fires with significant plume rise are lower than those for instantaneous releases (INL 2009).

Table 6. Inhalation dose consequences for vehicle fuel fire.

Radionuclide	Source Term (Ci)	Facility Worker		Collocated Worker		Offsite Public	
		Inhalation DCF (rem/Ci)	Dose (rem)	Inhalation DCF (rem/Ci)	Dose (rem)	Inhalation DCF (rem/Ci)	Dose (rem)
^{239}Pu	1.70E-05	1.18E+08	2.56E-02	1.18E+08	2.73E-03	4.44E+08	1.04E-04
^{60}Co	1.00E-01	6.29E+04	7.99E-02	6.29E+04	8.55E-03	1.15E+05	2.03E-04

DCF = dose conversion factor

As noted above, these results are considered conservative for this accident and are most applicable to the activated metal waste streams that most likely do not contain combustible materials. These results are also considered conservative with respect to debris and resin waste streams because, although ARF/RF values may be higher, the inventory of ^{60}Co in a single waste container would be considerably lower. The likely fission products present in the resin waste streams have significantly lower DCF values and would also result in negligible inhalation dose consequences.

Dose consequence evaluation guidelines for the public and facility workers are 5 rem and 25 rem, respectively, for unlikely events. This analysis shows that evaluation guidelines are not exceeded.

4.2.3 Direct Radiation Exposure During Waste Container Handling

Waste containers are routinely transferred from transportation packages to storage vaults using a variety of container handling equipment and appropriate shielding during the transfer operations. The contact exposure rates for nominal waste containers can be as high as 30,000 R/hour, potentially resulting in consequences to facility workers that exceed evaluation guidelines. Therefore, shielding is required to ensure that facility workers are not exposed to high exposure levels. For waste containers with greater than 30,000 R/hour contact exposure rates, additional controls, such as maintaining minimum distances from the containers or addition of temporary shielding during the transfer operations, may be required to ensure that evaluation guidelines for facility workers are not exceeded.

4.2.4 Severe Seismic Event

This event bounds the primary NPHs typically considered credible at the INL site. Other NPH events were considered as required by DOE Order 420.1B and DOE Guide 420.1-2, "Guide for the Mitigation of Natural Phenomena Hazards for DOE Nuclear Facility and Non-Nuclear Facilities," and were determined to be beyond credible, or the effects would fall within the effects of these primary hazards. Of these potential NPH events, high winds, fires, lightning strikes, and floods are assumed to have little or no impact on the underground concrete vaults, leaving seismic events as posing the greatest potential to impose structural loads sufficient to cause waste container failure. A seismic event with sufficient energy to cause failure of the concrete vault and waste container is extremely unlikely. As delineated in the safety design strategy (INL 2010e), based on an initial review of the applicable facility hazards and in accordance with ANSI/ANS-2.26-2004, "Categorization of Nuclear Facility Structures, Systems, and Components for Seismic Design," the Remote-Handled LLW Disposal Project seismic design category (SDC) will be SDC-1. This determination is based on the assumption that a failure of a vault will not cause radiological material to be brought to the surface and that it will remain in place without causing significant radiological exposure to workers, the public, or the environment. Even if such an event were to occur and the waste container and vault were both to fail, it is probable that the failure would result in at least a partial filling of backfill sand and soil into the vault that would prevent a significant release of radiological material.

A severe seismic event is postulated to cause loss of shield plug integrity. Consequences to this postulated event are direct radiation exposure to a worker in the immediate vicinity of a failed shield plug. The shield plugs to be used are robust and placed in a solid array over the entire waste disposal area. Failure of a shield plug requires that the plug rupture and be physically removed vertically from its installation because, while the plug is in place, it is laterally constrained from horizontal movement by the presence of adjacent plugs and the facility boundary. A strong seismic event is an assumed condition that could result in a failure of the 5-ft-thick concrete vault plugs. A seismic event with enough ground motion to fracture or rubbilize the vault plug is considered an extremely unlikely event. Other events that could potentially fracture a shield plug could not credibly cause the shield plug material to be completely removed and fail to provide shielding.

Dose to the facility worker in this case is assumed to be primarily from direct radiation exposure. Contact exposure rates with the waste container can be as high as 30,000 R/hour. Therefore, it is possible that a facility worker could be exposed to high consequences in the immediate vicinity of a failed shield plug. At the distance of the collocated worker at 100 m and the public at 5,000 m, dose rates would be negligible. Because the consequences can exceed evaluation guidelines, the shield plug is required to ensure that facility workers are not exposed to high consequences.

4.2.5 External Events

Events under this category of accidents include plane crash, vehicle crash, and adjacent building fire/explosion. Plane crashes on INL are judged to be beyond extremely unlikely due to diversion of air traffic and based on air transportation safety information. Vehicle crashes and adjacent building fire/explosions do not have a significant impact on the waste containers buried at the remote-handled LLW disposal facility. There is no anticipated release of radioactive material from this category of accidents should they occur.

4.3 Preliminary Selection and Classification of Safety Structures, Systems, and Components

Safety-class SSCs are hazard controls for which credit is taken, either preventive or mitigative, to meet the evaluation guidelines for the offsite public. Based on the results in this CSDR, evaluation guidelines for the public are not challenged for unmitigated releases. Therefore, no safety-class SSCs are identified for this facility.

Safety-significant SSCs are hazard controls for which credit is taken to prevent or mitigate postulated anticipated or unlikely accidents that could result in consequences to collocated or facility workers exceeding evaluation guidelines. Based on the results in this CSDR, it is concluded that the potential exists for an accident that could result in consequences exceeding these guidelines to the facility worker. The 5-ft-thick concrete shield plugs are identified as a component that would protect the facility worker from these consequences after the waste containers are placed in the vaults. In addition, the CVAS and any shielding required for top-unloading transportation packages are identified as components that would protect the facility worker from these consequences during placement of the waste containers in the vaults. The shield plugs, CVAS, and shielding required for top-unloading transportation packages may, therefore, be designated as safety-significant SSCs for design and facility planning purposes. As the facility design matures, further analyses will be performed evaluating direct radiation exposure to the facility worker from specific material being transferred and stored.

The primary mechanical system of the facility is related to operation of the hoisting system associated with the working platform that is used to lower the waste containers into the vaults. The system currently at RWMC is owned by the Office of Naval Reactors and is planned to be transferred to the new disposal facility for use in waste placement operations. This system has not been designated as a safety SSC for handling the NRF activated metal waste. Because the waste stream is the same and handling operations will be similar in the proposed new facility, it is not expected that the safety designation would change. Preliminary dose calculations made for a single waste container also supports this position. Development of other waste container transfer systems will need to ensure that all applicable mechanical systems are designed using the appropriate protocols. Any ancillary equipment specifically required to interface with the waste container for transport and unloading, other than the typical hoisting and rigging components, will be provided by the generating facility.

The pre-cast concrete storage vaults are considered defense-in-depth design features and perform the safety function of shielding and confinement; however, they are not derived as safety significant. The vaults are located below ground surface, isolating contents from facility workers, and, upon failure, would not impose any risk of fatality or serious injury to workers. There are no failure scenarios for the vaults that result in a loss of function in an emergency that may be needed to preserve the health and safety of workers. Furthermore, in the improbable event of vault or shield plug failure, there would be no significant offsite consequences.

5. SECURITY HAZARDS AND DESIGN IMPLICATIONS

The waste stream destined for the proposed Remote-Handled LLW Disposal Project is classified as remote-handled LLW. The waste contains little fissionable material and poses little or no risk of criticality or diversion. Based on the characteristics of the remote-handled LLW, the facility (as proposed) will require, at most, a property protection area (perimeter fence) as security controls.

The facility will be equipped with a security system that includes remote visual capabilities and wireless alarms that can be monitored at an offsite location. This system will be comprised of a camera network that will monitor the access gate to the facility and other locations as warranted. The network signals will provide remote video surveillance and indications of when the facility gate is open and closed at a monitoring location determined by the INL security organization. This system should be equipped with 8 hours of backup power.

Insider threats and sabotage risks are similar in nature to the current operational portfolio in place at RWMC. Therefore, the security impact of the Remote-Handled LLW Disposal Project should have minimal additional impact to the existing INL security program and should be consistent with current practices and operational risks at RWMC. Based on an initial review of the proposed operations in relation to ongoing disposal operations at RWMC, analyses are expected to focus on radiological sabotage events associated with transport and disposal of the remote-handled LLW streams.

NRF-generated remote-handled waste that is currently disposed of at RWMC and will be disposed at the proposed facility is generally classified. The waste containers are bolted shut and are considered self-protected due to the high contact radiation levels. Continued management of NRF classified waste in this manner is planned and has been approved by the Naval Reactors Laboratory Field Office. Other waste streams identified for disposal at the proposed facility do not contain classified material.

A formal review will be conducted, based on the final design and location of the facility, to validate the foregoing conclusions and to determine if additional security concerns exist based on material types and quantities. This analysis will be conducted before the start of operations to ensure adequate security measures are in place and operational. Security requirements that may be imposed are not anticipated to impact the facility safety basis, and any such additional security measures should not introduce additional hazards.

Consistent with DOE Order 420.1B, facility design will accommodate all requirements for safeguards and security, access control, and emergency egress. Where conflict occurs between such requirements, life safety requirements have precedence. As designed, this facility complies with the letter and intent of the order and presents no risk to employees with respect to NFPA 101, “Life Safety Code.”

6. NUCLEAR SAFETY DESIGN CRITERIA

6.1 Approach for Compliance with Design Criteria

DOE Order 420.1B provides attributes to which design criteria may be compared to ensure that DOE nuclear facilities are constructed in a manner that will protect the public, workers, and the environment from nuclear hazards. Applicable criteria from DOE Order 420.1B are provided as follows:

- **Integration of design with safety analyses**—Safety analyses will be performed during the appropriate phases of the critical decision process in accordance with DOE-STD-1189-2008. Those analyses will be used to identify hazards associated with the facility and the safety SSCs and the safety functions of those SSCs. Those analyses will be performed starting with preliminary analyses performed in the conceptual design phase and progressively provide more detail as the design matures. At this point in the facility design, only the vault shield plugs and shielding associated with waste container transfer operations have been designated as safety SSCs derived from accident analysis for hazards within the facility.

- **Nuclear facility design**—Design features for the Remote-Handled LLW Disposal Project will include multiple layers of confinement for the material being stored. Defense-in-depth will include the concrete storage vaults and steel waste containers. The nature of the facility as an underground facility provides defense-in-depth protection. Control processes will be identified for transferring material from the transportation packages to the storage vault locations. Lifting equipment will be managed in a way to maintain safe lifting operations. Worker safety will be enhanced through the use of procedures, safety management programs, qualification and training programs, as-low-as-reasonably-achievable (ALARA) programs, and equipment measurement and testing requirements. The use of shielded transportation packages under proper administrative controls provides defense-in-depth protection to facility operations.

- **Fire protection program**—The waste streams identified for disposal in the proposed Remote-Handled LLW Disposal Project can contain combustible materials. An evaluation will be performed and documented to assess the need for fire protection programs and systems as part of development of the PDSA.

- **Nuclear criticality safety considerations**—Further evaluation will be made on the need for criticality safety requirements (i.e., specific packaging configurations for high quantities of fissile materials) pertaining to the proposed Remote-Handled LLW Disposal Project during development of the PDSA.

- **Natural phenomena mitigation design—**Facility components will be designed, constructed, and operated in a manner that will withstand NPH and ensure the safety function of each system or component will remain viable. For most NPH events, facility consequences are very low due to the vaults being placed in the ground where releases are very unlikely. A seismic study will be performed during development of the PDSA, which will evaluate the effect of seismic events on the concrete vaults in particular.

6.2 Exceptions to Design Criteria

No exceptions to safety design criteria in DOE Order 420.1B have been identified for this proposed facility. In all cases, design criteria should be commensurate with the importance of the safety function performed and the needed reliability of that component or system.

7. OTHER CONSIDERATIONS

7.1 Planned Studies or Analyses

As the project design matures, generation of other safety documents and analyses will be required. These supporting documents, other than operational procedures, will include, as appropriate, a fire hazard analysis, preliminary safety design report, PDSA, documented safety analysis (DOE approval required) to supplement the INL's standardized documented safety analysis, hoisting and rigging plan, engineering design files, ALARA reviews, radiation work permits, operational job safety analyses, and industrial hygiene exposure assessments prepared in accordance with associated INL procedures:

- **Seismic evaluation—**Damage from seismic events results from differential movement in a structure. Because of the decoupled nature of the vault system, in that each of the storage vaults is independent of the other vaults, the facility is characterized by numerous small structures rather than a single large structure. The relatively compact size of each vault, the design and materials of construction, and placement method (installed entirely in soil and never in bedrock) make it improbable that seismic motion of adjacent soils could generate forces sufficient to damage a storage vault. However, a structural evaluation of the proposed vault system should be performed to address the capability of the vaults to withstand the loads and forces as defined in applicable guidance from the International Building Code. The evaluation should include the effects of differential displacement, mechanical loading (i.e., horizontal and vertical vibration of the vault and waste containers stored within), and seismic excitation of the soil where the vaults are set. The results of the evaluation should indicate whether the seismic-induced differential displacement is a concern for the vaults and will be used in making the final hazard categorization. A seismic study will be performed during development of the PDSA, which will evaluate the effect of seismic events generally and on the concrete vaults in particular.

- **Criticality evaluation**—The need for a criticality study will be evaluated in a later phase of facility design. In the conceptual stage of the facility, preliminary evaluations indicate that the waste streams for the facility do not contain significant quantities of fissionable material to make nuclear criticality a credible accident. The need for a criticality evaluation will be performed during development of the PDSA.

- **Consequence evaluations**—The DBAs noted in this document were qualitatively evaluated in order to determine magnitude of postulated accidents involving release of radioactive material or direct exposure to high radiation levels. These results are considered preliminary and should be followed up with more complete and rigorous analyses. The evaluations should consider probable MAR from specific waste streams and container profiles and also should consider waste acceptance criteria that will be developed for the facility. Likewise, direct radiation exposure analyses are preliminary in this report. As the design is developed and new waste container configuration completed, more detailed analyses can be modeled. Further dose consequence evaluations will be performed during development of the PDSA.

- **Vulnerability assessment**—Security hazards will be better understood upon completion of an assessment of the risks associated with disposal of this material. Based on an initial review of the proposed operations in relation to ongoing disposal operations at RWMC, analyses are expected to focus on radiological sabotage events associated with transport and disposal of the remote-handled LLW streams. This assessment is tied closely to facility siting; therefore, it will be prepared following identification of a proposed siting location.

7.2 Safety-in-Design Risks and Opportunities

The project risk management plan (PLN-2541, "Risk Management Plan for the Remote-Handled Low-Level Waste Disposal Project") defines the scope, responsibilities, and methodology for identifying and evaluating impacts of and managing risks that could affect successful completion of the project. The objective of the risk management plan is to enable project success by identifying project risks, including programmatic, technical, cost, and schedule risks, and appropriate response actions to effectively manage the risks through project completion. The safety-in-design considerations for the risk and opportunity analysis based on the risk management plan and the results of the CSDR are shown in Table 7. The results presented in Table 7 will be evaluated for addition to the risk management plan risk register.

The highest ranked risk from PLN-2541 is the decision whether or not the facility will be underlayed with an impervious liner. Significant cost and schedule impact could occur as a result of increased design requirements, more complex construction (e.g., liner and monitoring systems), and regulatory coordination. Given the constraints of the planned schedule, any delay could result in the inability to maintain continuous disposal capability for remote-handled LLW, adversely impacting both the project and the waste generators' abilities to fulfill their respective missions. The use of an impervious underlayment could impact the risk of an analyzed accident of corrosion-induced waste container failure. Trapped environmental water could increase the likelihood of that event. A decision to include an impervious liner also could result in fundamental changes to facility design, which could impact completed and planned nuclear safety analyses and documentation.

Other risks related to the safe design of the facility involve validation of assumptions used in calculations and analyses associated with this CSDR. Preliminary hazard categorization and performance of preliminary accident identification and consequence analyses were made using assumptions and information obtained from the project technical and functional requirements, conceptual design report, and other facility documents reflecting this stage of conceptual design. The assumptions used in these evaluations were understood to be conservative. If those assumptions change or are found to be non-conservative, results of the analyses could change. The risk associated with a change in assumptions is the possibility that previously unidentified safety SSCs could be identified if the analyses changed significantly.

In summary, the program risks identified in PLN-2541 are primarily associated with successful completion of the facility rather than of a technical nature for storing waste in belowground vaults and the design features that are needed. Overall, the programmatic risk is that the facility would not be built, or would not be operational by the target date. The consequence of this risk would be significant impact on the DOE Office of Naval Reactors and Office of Nuclear Energy programs, which rely on a clear disposal path for remote-handled LLW. Other risks are in performance of evaluations on which design criteria will be established. New findings could result in revised dose consequence evaluations, potentially identifying new or higher-classed SSCs.

An opportunity that will require evaluation during the design phase of the project will be final hazard categorization of the facility based on facility segmentation. This will be based on an evaluation of alternate release fractions and on the results of the seismic analysis evaluation that also will be performed. The outcome of these evaluations may result in an opportunity to downgrade the facility hazard category.

7.3 Lessons Learned From Previous Experience Involving Major Systems

The proposed Remote-Handled LLW Disposal Project will be designed and constructed similar to the remote-handled LLW concrete disposal vaults currently in use in the RWMC Subsurface Disposal Area. This will accommodate, to the maximum extent possible, uninterrupted operations at the generating facilities and will capitalize on the operations experience and cost-efficiencies of current remote-handled LLW disposal practices.

Table 7. Safety-in-design considerations for risk and opportunity analysis.

Functional Area	Risk	Mitigation Strategy/Opportunity
Design		
Undefined, incomplete, unclear process or safety function or requirements		
• Potential impact to confinement ventilation strategy	None; no confinement ventilation required	NA
• Potential impact to functional classification of SSCs	Low; robust transportation packages required as safety-significant SSC; design for some packages currently undefined	Transportation packages selected to meet safety functional requirements.
Facility Design Features		
• Security requirements and impact on safety analysis	None, stored waste does not pose a security risk	NA
• Safety-related control system design, interface with safety analysis, and implementation	None; no control systems are required	NA
• Vault design to minimize water intrusion	Low, corrosion-induced waste container failure may result in radioactive material release to aquifer	Vault liner and/or monitoring system
• Waste container design to minimize corrosion failures	Low, corrosion-induced waste container failure may result in radioactive material release inside vault	Performance assessment.
Assumptions on key utility interfaces (capacity, equipment compatibility, and reliability)	Low; fire suppressions system, including water supply identified as defense-in-depth	Restrict facility operations if fire suppression system water supply is unavailable.
Design-basis threat requirements	None, similar to existing facility	NA

Table 7. Safety-in-design considerations for risk and opportunity analysis.

Functional Area	Risk	Mitigation Strategy/Opportunity
Deferred Capability Decisions (where hazards could be introduced or increased with added capability in the future)		
• Potential for added capacity (MAR and SSC functional classification impact)	Low, potential for increased total facility MAR if storage capacity is increased	Ensure that additional stored waste meets waste acceptance criteria.
• Potential for addition of significant mass to structure affecting seismic design	None; storage vaults not affected by mass of stored waste containers	NA
• Potential for impacting confinement ventilation system	None; no confinement ventilation required	NA
Safety-class SSC selection confidence	None, there is no potential for identification of safety-class SSCs.	NA
Safety-significant SSC selection confidence		
• Management judgments related to selection of borderline SSC classifications should be identified	Low, all SSC classifications derived from hazard and accident analysis	Final SSC classification during PDSA and documented safety analysis development.
• Assumptions critical to inhalation dose consequence results with potential for change (e.g., ARF)	Low, uncharacterized waste streams may have waste form and radionuclide content that can impact MAR and ARF.	Waste acceptance criteria ensure that waste streams do not have the potential to result in airborne releases that exceed evaluation guidelines.
• Assumptions critical to direct radiation exposure consequence results with potential for change	Low, uncharacterized waste streams may have contact exposure rates that exceed shielding design criteria.	Waste acceptance criteria ensure that waste streams do not have the potential to exceed evaluation guidelines due to direct radiation exposure.
Assumptions regarding production objectives	None, Remote-Handled Low-Level Waste Disposal Project is not a production facility	NA
Errors and omissions in design	None, facility design and operation based on existing technology	NA

Table 7. Safety-in-design considerations for risk and opportunity analysis.

Functional Area	Risk	Mitigation Strategy/Opportunity
Seismic design margin	Low, potential for seismic motion of soils to generate forces sufficient to damage a storage vault	Perform seismic analysis and incorporate results into design and safety basis documentation.
Criticality design criteria	Low, preliminary evaluation indicates that waste streams do not contain enough fissionable materials to make nuclear criticality a credible accident	Perform criticality evaluation and incorporate results into design and safety basis documentation.
Fire protection	Low, minimal risk based on transportation packages/waste containers acting as fire barriers	Perform PHA and incorporate results into design and safety basis documentation.
Field quality control during construction leading to design changes that impact seismic basis, separation requirements, etc.	Low, potential impact on facility segmentation justification for HC-3 designation	Implementation of quality control program during construction.
Technology		
New technology application or new application of existing technology	None, facility design and operation based on existing technology	NA
Unknown or undecided technology	Low, transportation packages and waste container handling system not yet selected.	Transportation packages or waste containers selected to meet safety functional requirements.
Scale-up of bench-scale technology or process or technology application maturity	None, no new technology involved in handling or storage operations	NA

8. REFERENCES

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