PNL-10210 Rev. 0
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2020

Characterization Plan for Hanford Spent Nuclear Fuel

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December 1994

Prepared for
the U.S. Department of Energy
under Contract DE-AC06-76RLO 1830

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Preface

The U.S. Department of Energy's (DOE's) Hanford Site in eastern Washington State contains about 80% of DOE's inventory of unprocessed spent nuclear fuel (SNF). This SNF, particularly that large portion residing in the K-East and K-West water basins at the 100 Area of the Site, has been identified with environmental vulnerabilities because of the proximity of the basins to the Columbia River. The mission of the Hanford SNF Project is to develop disposition strategies for the SNF at the Hanford Site. The project objectives include 1) temporary isolation of the SNF within the 105-K basins, 2) transportation of the SNF to safe and environmentally sound extended interim storage in a facility away from the Columbia River, and 3) preparation of the SNF for final disposition. Decisions regarding the accomplishment of these objectives will be supported by clear documentation of project planning, supportive testing and data analysis, data quality/quality assurance requirements, regulatory compliance requirements, and disposition pathway selection criteria.

The relationship of this document to other SNF project requirement documents and project plans is shown in the document hierarchy figure (Figure P.1).

The SNF FY95 Multiyear Program Plan is a Westinghouse Hanford Company (WHC) document that defines DOE-approved WHC SNF project activities and budgets.

The Hanford SNF Characterization Program Management Plan is a joint WHC and Pacific Northwest Laboratory (PNL) document that defines the scope and objectives of the joint WHC and PNL effort. This document also defines the relationship of the Hanford SNF characterization activities to both the overall Hanford SNF mission and the National Spent Fuel Program Characterization Plan.

This document, prepared by PNL, defines the issues to be resolved to meet DOE objectives of isolation, secure interim storage, and final disposition of the 105-K basins SNF. It describes the information needs required to support DOE's development of the SNF disposition pathway, and defines the associated SNF sampling and testing required to satisfy these information needs.

The SNF Project Characterization Data Quality Objectives Strategy is a WHC document that describes the application of the data quality objectives process to the overall Hanford SNF project characterization activities. It explains the role of the process, the steps in the process, and the review/approval authority level required for the various data quality objective activities.

The Sampling Strategy document is a joint WHC and PNL document that describes the approach to be used in selecting 105-K basins SNF samples for characterization testing at PNL. It relates this approach to WHC and DOE strategies for developing the pathway for 105-K basins SNF disposition.

The SNF Project Disposition Decision Data Quality Objectives are WHC documents that describe the data quality objectives required to support DOE disposition pathway decisions. These documents include those aspects of the data quality objective process related to problem definition and associated information requirements.

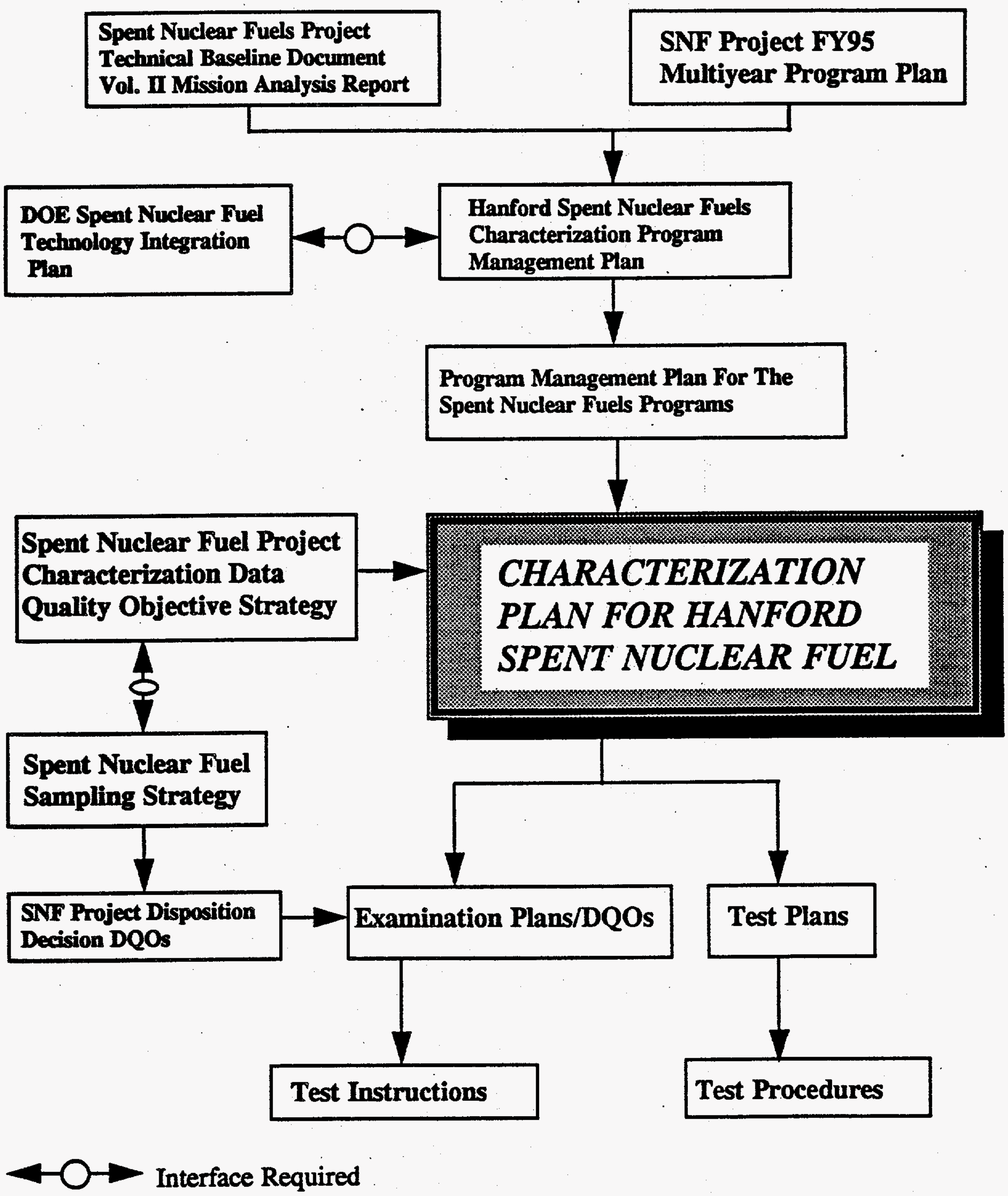

Figure P.1. Spent Nuclear Fuel Characterization Program Document Hierarchy

Each Examination Plan/Data Quality Objective is a PNL document that describes the issues to be addressed, data and information needs to be obtained, and the SNF samples to be tested in a given shipment of SNF from the 105-K basins to the PNL test facilities. The Examination Plans include aspects of the data quality objective process related to the laboratory tests and analyses to be performed on the SNF samples contained in a shipment, justification for the sample selection and testing, definition of the tests required, and data accuracy requirements. Each Examination Plan will reference the applicable SNF Project Disposition Decision Data Quality Objectives to relate the sample characterization testing to the disposition decision requirements.

Test Plans are PNL documents that describe the individual laboratory tests involved in providing data for an information need. The Test Plans describe the test types and procedures, equipment requirements, accuracy capabilities, and data reduction requirements. Each information need identified in this document has an associated Test Plan.

Procedures are step-by-step guidance for the operation of laboratory and hot cell apparatuses and test equipment at PNL.

Test Instructions are internal directives from PNL project management to technical staff that describe in detail the samples to be tested and procedures to be used in meeting the requirements of the Examination Plans/Data Quality Objectives.

Responsibility for the development of a disposition strategy, or pathway, for the Hanford SNF has been delegated by DOE to WHC. Responsibility for the SNF characterization activities associated with the disposition has been delegated by DOE to PNL. The successful completion of the Hanford SNF Project requires close interaction between WHC and PNL in the planning, documentation, and conduct of the characterization activities in support of the SNF disposition. Thus, although individual documents may be the assigned responsibility of PNL or WHC, considerable participation of both organizations is common in their generation.

Summary

Reprocessing of spent nuclear fuel (SNF) at the Hanford Site Plutonium-Uranium Extraction Plant (PUREX) was terminated in 1972. Since that time a significant quantity of N Reactor and Single-Pass Reactor SNF has been stored in the 100 Area K-East (KE) and K-West (KW) reactor basins. Approximately 80% of all U.S. Department of Energy (DOE)-owned SNF resides at Hanford, the largest portion of which is in the water-filled KE and KW reactor basins. The basins were not designed for long-term storage of the SNF and it has become a priority to move the SNF to a more suitable location. As part of the project plan, SNF inventories will be chemically and physically characterized to provide information that will be used to resolve safety and technical issues for development of an environmentally benign and efficient extended interim storage and final disposition strategy for this defense production-reactor SNF.

Pacific Northwest Laboratory (PNL) is responsible for setting the sample-selection criteria and subsequent characterization of the Hanford SNF in support of SNF isolation, interim storage, and final disposition effort. The pathway for final disposition of the 105-K basins SNF has not been firmly established, and several options are being considered by DOE. These disposition pathway options include

- packaging SNF in 105-K basins; transfer of the packaged SNF to a new wet storage facility, such as Fuel Material Examination Facility at the 200-East Area; conditioning the SNF for an extended dry-interim storage; and conditioning, if necessary, for final disposal

- packaging SNF in basins; conditioning packaged SNF for dry storage; conditioning for final disposition (Independent Technical Assessment Team option)

- encapsulating KE SNF and storing it in 105-K basins; conditioning the encapsulated SNF for interim wet or dry storage; conditioning, if necessary, for final disposition (current Westinghouse Hanford Company option).

The Characterization Program described here will provide sufficient SNF characterization data and analyses to evaluate the acceptability and support licensing of the disposition strategy eventually adopted by DOE. However, the data and analyses would also serve in the evaluation of alternative pathways, should programmatic decisions dictate a change in the chosen pathway.

Sufficient information will be gathered to support demonstration of safety margins and regulatory compliance for isolation and interim storage activities. The tests and analyses will also help establish conformance to waste acceptance criteria for ultimate disposition of SNF in a geologic repository. The testing described here will resolve key technical issues associated with the SNF isolation, interim storage, and final disposal, which are generic to any pathway to final disposition. The testing will do this in part by establishing the initial condition of the SNF as input to pathway development. These issues and the SNF characterization information required to resolve them are described in Sections 2.1 and 2.2.

Characterization tests are described for the SNF stored in the 105-K basins and PUREX. Detailed information concerning the inventories of this SNF is given in Appendix A. It is expected that the information obtained for the 105-K basins SNF will be a sufficient basis for decisions concerning the small amount of SNF at PUREX. The characteristics of the expected sludge in the canisters, resulting from fuel corrosion since the circa 1982 SNF encapsulation campaign, and the basin sludge being evaluated by WHC are included in the program. The existing historical database of the SNF will be reviewed to establish the baseline for the current condition of the fuel. The physical and chemical characteristics of the SNF will be determined by both in-basin and laboratory activities such as 1) visual examinations and other nondestructive examination tests of the fuel in the basins, 2) metallographic examinations in hot cells, and 3) tests on the dissolution, oxidation/corrosion, and hydriding behavior of the spent fuel in air, moist environment, and water. Additional information about the products of corrosion or degradation of the SNF now in storage will be analyzed with mass spectrometry, scanning electron microscopy, x-ray diffractometry, and transmission electron microscopy. Justification for the various tests and analyses is elaborated in Section 3.0, and relationship of the tests to the information needed for issue resolution is shown in Table 3.1.

Acronyms

AEA	*Atomic Energy Act of 1954*
ALARA	As Low As Reasonably Achievable
CAA	*Clean Air Act*
CERCLA	*Comprehensive Environmental Response, Compensation and Liability Act of 1980*
CWA	*Clean Water Act*
DOE	U.S. Department of Energy
DSC	Differential Scanning Calorimeter
EM	Environmental Restoration and Waste Management
EPA	U.S. Environmental Protection Agency
FMEF	Fuel Material Examination Facility
ITA	Independent Technical Assessment (Team)
KE	105-K East
KW	105-K West
MCO	multicanister overpack
MS	Mass Spectrometer
MTU	Metric ton of Uranium
MWD	Megawatt-Day
NDE	Non-destructive examinations
NEPA	*National Environmental Policy Act*
NESHAP	National Emission Standard for Hazardous Air Pollutants
NRC	U.S. Nuclear Regulatory Commission
NWPA	*Nuclear Waste Policy Act of 1988*
OCRWM	Office of Civilian Radioactive Waste Management
PNL	Pacific Northwest Laboratory
PUREX	Plutonium-Uranium Extraction Plant
RCRA	*Resource Conservation and Recovery Act of 1976*
SARA	*Superfund Amendments and Reauthorization Act of 1986*
SEM	Scanning Electron Microscope
SNF	Spent Nuclear Fuel
SPR	Single-Pass Reactor
TEM	Transmission Electron Microscope
TGA	Thermogravimetric Analyzer
WAC	Waste acceptance criteria
WHC	Westinghouse Hanford Company
XRD	X-ray Diffractometer

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1.0 Introduction

1.1 Background

At the direction of the Secretary of Energy in 1992, the Assistant Secretary for Environmental Restoration and Waste Management (EM) initiated an integrated, U.S. Department of Energy (DOE) Complex-wide spent nuclear fuel (SNF) management and disposal program. The integrated program encompasses all the existing DOE-owned SNF inventory, 80% of which resides at the DOE Hanford Site in southeastern Washington State. The SNF is stored primarily in the 105-K East (KE) and 105-K West (KW) basins; a relatively small amount of SNF is stored in the Plutonium-Uranium Extraction Plant (PUREX). Westinghouse Hanford Company (WHC) is the Site operating contractor. The mission of this program is to safely, reliably, and efficiently manage and dispose of DOE-owned SNF.

A number of vulnerabilities (focused upon the proximity of the radionuclide sources to the Columbia River) have been identified [1] with the storage conditions at the two basins holding the N Reactor fuel; these vulnerabilities require near-term remediation actions. The Hanford SNF Project has been established to develop these required remediation actions, in particular to meet the project objectives of 1) temporary isolation of the SNF in the basins, 2) transportation to safe and environmentally sound extended interim storage in a facility away from the Columbia River, and 3) preparation for final disposition. This characterization plan, prepared by Pacific Northwest Laboratory (PNL),[(a)] describes the testing required to support WHC and DOE in the development of the Hanford SNF disposition pathway.

The SNF in the KE and KW basins is primarily spent N Reactor fuel with a much smaller inventory of Single-Pass Reactor (SPR) fuel. The N Reactor fuel is composed of uranium alloy clad in Zircaloy-2, while the SPR fuel is composed of uranium alloy clad in aluminum alloy 8001.

The KE basin contains about 1,150 metric tons of uranium (MTU) of SNF in 50,683 assemblies in open canisters exposed to the basin water. Detailed information concerning the fuel fabrication methodology, as-fabricated characteristics, and fuel inventories of the SNF in the 105-K basins is given in Appendix A. It has been estimated that the cladding on at least 7% of the fuel was breached during reactor discharge activities [2]. Visual examinations of the canisters during storage show extensive corrosion of the canisters and SNF and sludge accumulation in KE basin and canisters. The combination of the breached cladding and corrosion of the exposed fuel since its storage has led to radionuclide contamination of the basin water. Additionally, the KE basin has leaked periodically over its life. These effects have resulted in radiation exposure to workers and radionuclide contamination to the ground.

An estimated 65 m^3 of sludge have accumulated on the KE basin floor [2]. This sludge is a combination of intrusion sand and corrosion products of the fuel, canisters, and basin racks. Sludge has also formed in the open KE canisters that have solid bottoms and in the sealed KW canisters with

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(a) The Pacific Northwest Laboratory is operated for the U.S. Department of Energy by Battelle Memorial Institute, under Contract DE-AC06-76RLO 1830.

corroding fuel. DOE is currently evaluating whether the KE basin sludge is to be treated as SNF; the characterization effort will assume that only the canister sludge (KE and KW) will be treated as SNF. DOE has not chosen a preferred methodology for the packaging or treatment of the canisters or KE basin sludge for interim storage and disposal.

The KW basin SNF currently resides in sealed, vented canisters similar in size and design to the KE canisters. About 3,821 sealed canisters with about 952 MTU of N Reactor fuel are in the KW storage basin. These canisters show no visible signs of significant corrosion or other degradation. Although the activity of cesium and tritium in the water (0.14 μCi/L for Cs and 0.064 μCi/L for ^{3}H in January 1994) is increasing gradually, KW basin water is relatively free of sludge accumulation. The SNF inside the KW canisters has not been examined since the encapsulation campaign ended in 1984.

This characterization plan identifies the 105-K basins disposition pathway issues and information needs to resolve those issues. The plan also identifies tests that could be performed to gather the information needs. Other documents will be generated by the characterization program to address how data will be collected and analyzed to meet the SNF project objectives. The relationship between these documents is shown in Figure 1.1.

1.2 Characterization Program Support to Disposition Pathway

A number of disposition pathways are being considered by DOE, including pathways envisioned by WHC and the Independent Technical Assessment (ITA) team [3] for disposition of the 105-K basins SNF. An option to ship the SNF to Europe (United Kingdom and France) for reprocessing is also being considered.

The WHC pathway involves

- encapsulation of the KE SNF, possibly including some canister sludge, by the basic methodology used in the circa 1982 KW basin fuel encapsulation, with no further conditioning of the KW SNF

- transfer of the encapsulated KE and KW SNF to a dry extended interim storage facility, where the SNF will undergo minimum processing or conditioning to meet repository waste acceptance criteria (WAC)

- transfer of the conditioned SNF to a geologic repository.

The ITA pathway involves

- packaging the 105-K basins SNF and sludge into a multicanister overpack (MCO) designed by the ITA [3]

- conditioning the MCO packages for dry interim storage

- conditioning, if necessary, for final disposal.

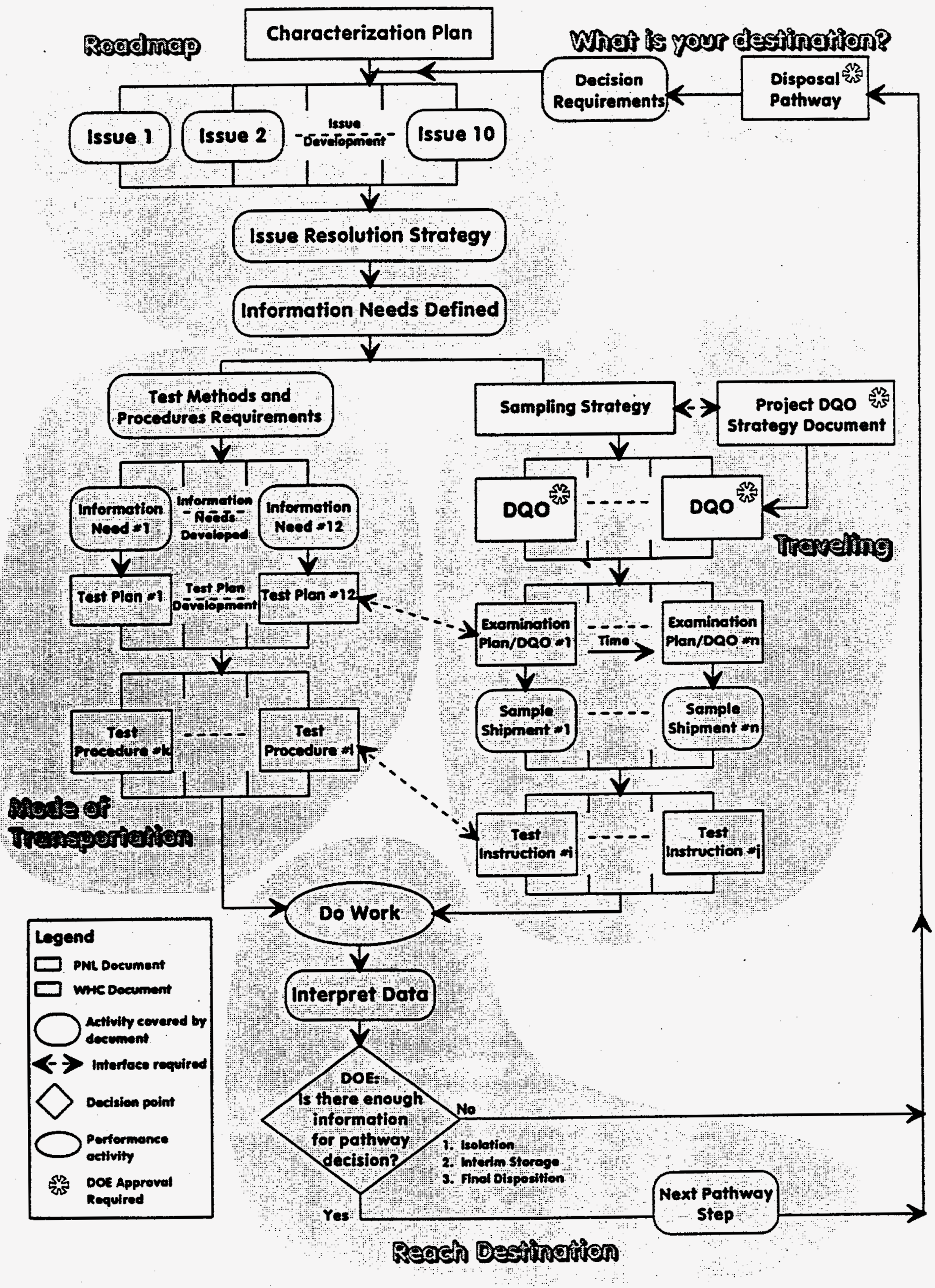

Figure 1.1. Relationship Between SNF Characterization Documents

The final pathway to disposition developed by DOE will evolve from programmatic and technical considerations. The SNF characterization program will provide sufficient information to evaluate any potential pathways, such as fuel stabilization, before extended interim storage or conditioning prior to repository disposal. The SNF characterization program will provide this information by providing a technical basis for the resolution of existing SNF storage issues, conditioning technologies for interim storage and final disposal, and WAC for final disposal.

The evaluation of options for the treatment or conditioning of the SNF in the interim storage facility and subsequent disposal in a geologic repository are complicated by the lack of WAC for DOE SNF at the proposed repository at Yucca Mountain, Nevada. The WAC for DOE SNF will be controlled by DOE's Office of Civilian Radioactive Waste Management (OCRWM), which has responsibility for the licensing and construction of the geologic repository. Repository waste package and engineered barrier system designs, and their associated performance assessment activities related to DOE SNF, are not being performed or planned. Although it is not a goal of the near-term characterization effort to provide sufficient information to evaluate long-term (i.e., geologic repository) disposal, it is expected that much of the data acquired will be necessary to meet repository WAC established by OCRWM.

2.0 Characterization Objectives

The SNF Characterization program will initially focus the testing to support 105-K basins SNF disposition pathway decision-making by DOE. The programmatic issues are discussed in Section 2.1. The characterization program will provide supportive information for the resolution of those issues. Toward this end, the initial objective of the SNF characterization program is to provide sufficient data and analyses to make decisions about the following:

- **K Basin SNF Isolation**: the isolation by packaging of the SNF assemblies and sludge in the KE Basin and the need, if any, to repackage the KW SNF for removal to an interim storage facility
- **Interim Storage**: the level of SNF/canister stabilization/conditioning required, if any, for approximately 40-year extended interim storage
- **Final Disposal**: the level of SNF conditioning required, if any, for meeting a geologic repository WAC.

Because a preferred treatment or conditioning methodology for basin and canister sludges (to the extent that sludges cannot be packaged) has not been identified, characterization of sludge samples will focus on providing enough physical and chemical information to serve as an "initial condition" for process selection and treatment decisions. Additionally, the characterization of basin sludges will provide information helpful in KE and KW basins remediation. However, because KE SNF repackaging activities could significantly change the nature of the current basin sludge, samples would need to be taken after the repackaging campaign.

2.1 Issues to be Resolved to Meet Project Objectives

To meet the SNF project objectives for isolation, interim storage, and final disposal of the 105-K basins SNF, sufficient information concerning the SNF must be obtained to resolve the following issues:

- SNF Isolation Issues

  1. Can most of the KE fuel be packaged and isolated?
  2. Can the sludge and/or severely damaged fuel be packaged?
  3. If not, can another packaging process be developed?
  4. Is most of the KW fuel adequately packaged for transfer to an interim storage facility?
  5. Will any KW canisters need repackaging?

- Extended Interim Storage Issues

  6. Is the packaged SNF adequate for extended interim storage?
  7. If not, can the packaged KW SNF be provided with an additional barrier for extended interim storage?
  8. If not, can the SNF be conditioned for extended interim storage?

- SNF Ultimate Disposition Issues

    9. Can the SNF or its conditioned form meet repository WAC?
    10. If not, is an alternate disposal pathway required?

These issues have been derived from already known characteristics of the SNF and potentially applicable regulatory drivers, such as U.S. Nuclear Regulatory Commission (NRC), U.S. Environmental Protection Agency, and state codes and regulations. Details of the perceived regulatory drivers are listed in Appendix B. Figure 2.1 shows how resolution of these issues relates to meeting project objectives.

2.2 Information Needs Required to Resolve Issues

The technical information required to resolve the issues below will be provided by the SNF characterization program. These information needs are data and/or analyses of data required to resolve the safety, technical, and programmatic issues relating to the program objectives. The information needs are:

- chemical and isotopic composition of SNF
- radionuclide release characteristics
- chemical-phase stability/degradation
- corrosion characteristics and rate
- dry/oxidation/dissolution kinetics
- combustibility/pyrophoricity
- nuclear criticality characteristics
- chemical toxicity
- size/weight/density characteristics
- SNF physical properties
- physical condition/integrity
- thermal properties.

The relationship between information needs and issue resolution shown in Table 2.1 was derived by considering the necessity of the information need for adequate resolution of each issue. Section 3.0 of this document describes the various tests and analyses required to satisfy these information needs. The priority of the information needs is that considered for resolving the current preferred-pathway issues.

2.3 Disposition Pathway

The pathway currently envisioned by WHC for disposition of the 105-K basins SNF involves encapsulation of the KE basin SNF by a method similar to that used for the KW basin SNF circa 1982, removal of the encapsulated SNF to a dry interim storage facility, and ultimate disposal in a geologic repository. There is no current preferred pathway for the disposition of KE basin or canister sludge; preliminary encapsulation evaluations by WHC will determine whether the sludge can be encapsulated with the SNF assemblies. WHC is also investigating a pathway that involves packaging the 105-K basins SNF in an overpack, transferring the packaged SNF to the Fuel Material Examination Facility (FMEF) for temporary storage followed by conditioning for an interim storage. These pathways are considered as potentially the most expeditious and effective ways to remediate the 105-K basins vulnerabilities [1]; however, the choice is contingent on information obtained in the Characterization Program or other programmatic redirection.

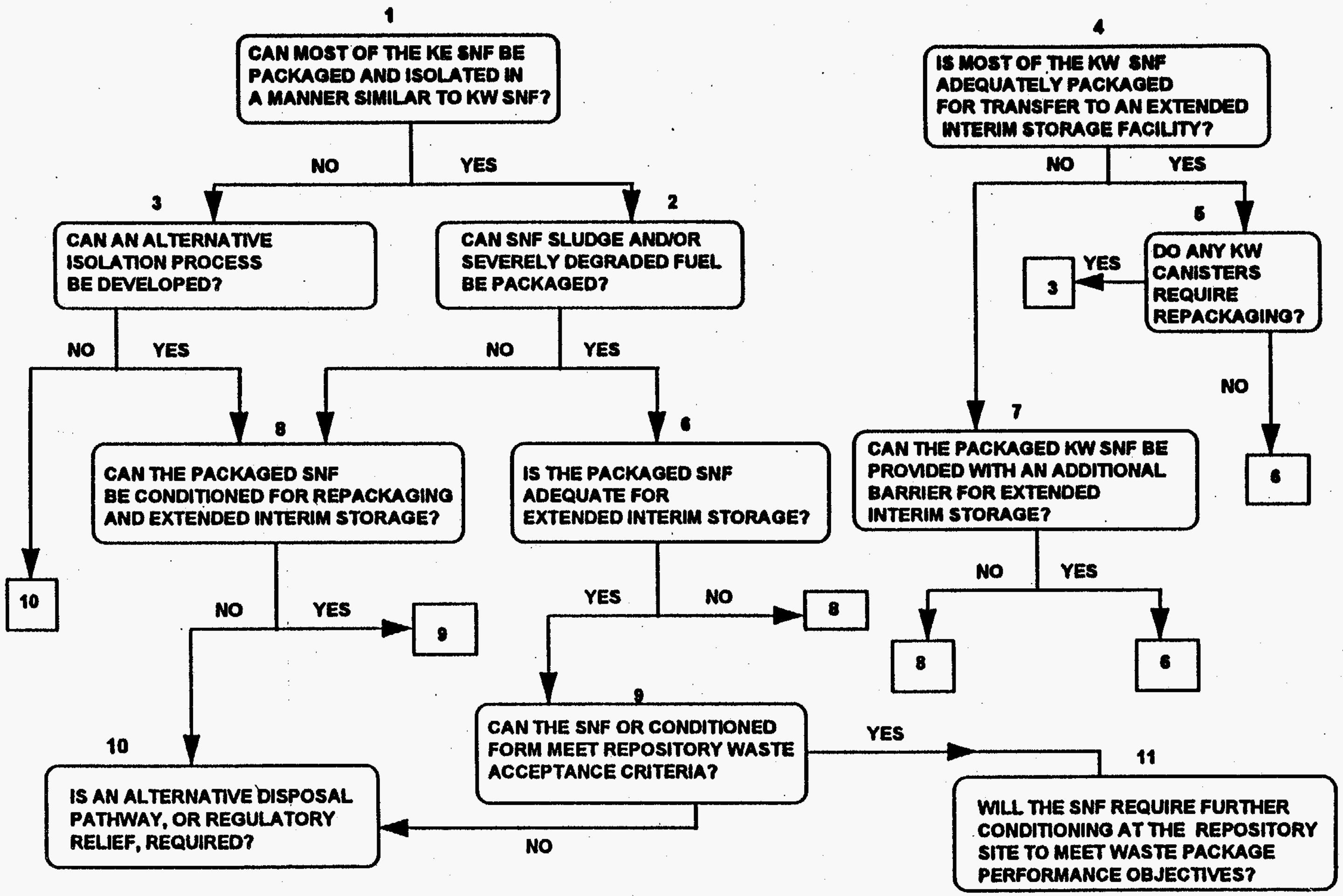

Figure 2.1. Issues Hierarchy

Table 2.1. K-Basin SNF Technical Information Needs for Issue Resolution

Issues	Information Needs: Chemical & Isotopic Composition		Radionuclide Release		Chemical Phase Stability		Corrosion		Combustibility		Criticality		Pyrophoricity		Toxicity		Size/ Weight		Physical Properties		SNF Physical Condition		Thermal Properties	
	H	L	H	L	H	L	H	L	H	L	H	L	H	L	H	L	H	L	H	L	H	L	H	L	H	L
1. Can most KE SNF be isolated?			X				X				X			X			X				X					
2. Can sludge and/or damaged SNF be packaged?			X			X	X				X			X				X		X	X					
3. Can an alternative package be developed?				X	X			X			X			X			X		X		X					
4. Is most KW SNF adequately packaged?				X			X					X									X					
5. Do any KW canisters require repackaging?			X			X	X				X			X			X		X		X					
6. Is packaged SNF adequate for extended interim storage?	X		X		X		X					X		X		X					X					
7. Can SNF be provided with an additional barrier?		X	X				X			X		X		X				X				X		X		
8. Can the SNF be conditioned for extended interim storage?	X		X		X				X			X	X													
9. Can the SNF or conditioned SNF meet repository WAC?	X		X					X	X					X	X				X			X	X			
10. Is an alternative disposal pathway required?	X		X					X		X			X		X					X		X	X			

H = High priority.
L = Low priority.

The ITA-proposed pathway for disposition of the 105-K basins SNF and sludge involves transition from wet storage to dry storage using MCOs followed by transfer to a geologic repository [3].

The Characterization Program is designed to fulfill information needs that could apply to any preferred pathway. Thus the characterization studies will provide information that will support

- evaluating the suitability of the packaging of the 105-K basins SNF for transfer to the temporary storage and/or transfer to the extended dry interim storage
- determining if SNF conditioning is required for extended dry interim storage and the behavior of conditioned SNF in extended dry interim storage
- evaluating WAC for repository disposal or alternative disposition.

3.0 Characterization

3.1 SNF Characterization Tests

The characterization tests will provide data pertinent to

- near-term isolation of the SNF by packaging in the 105-K basins (Isolation Objective)
- transport to, potential conditioning for, and waste package performance during extended interim storage (Interim Storage Objective)
- waste form conditioning for final disposition (Final Disposition Objective).

Although the current focus involves dry interim storage, it is expected that much of the characterization data would also be pertinent to wet interim storage. Testing the SNF will address the information needs most pertinent to the disposition pathway chosen by DOE. Potential disposition pathways could have significant commonality of information needs. Once the disposition path is chosen, the data quality objective (DQO) process will be applied to the information need. The testing and analysis activities required to provide the information needs are discussed below.

Each information need summarized in this section may involve several kinds of tests or analyses to obtain the data necessary for issue resolution. Table 3.1 summarizes the information needs for the three major objectives, the test methods to provide the data, and the functions the data needs fulfill.

3.1.1 Information Need: Chemical and Isotopic Composition

Functions

- *Isolation objective*: N/A
- *Interim storage objective*: To enable demonstration of conformance of the encapsulated (and potentially conditioned) SNF during transport to, and residence in, the interim storage facility to the as-low-as-reasonably-achievable (ALARA) requirements of 40 CFR 191 (Subpart B for containment and assurance requirements, and Appendix A for release limits); the radiation protection requirements of 10 CFR 20 (Subpart C for occupational dose limits, Subpart I for storage and control, and Appendix B for annual intake limits); the packaging requirements of DOE Order 5480.3; and the monitored retrievable storage requirements of 10 CFR 60 and the *Nuclear Waste Policy Act of 1988* (NWPA).

Table 3.1a. SNF Information Needs and Tests (Preferred Pathway)

Information Need	Test(s)	Purpose of Data (Preferred Pathway)		
		Isolation Objective	Interim Storage Objective	Final Disposal Objective
Chemical and Isotopic Composition	Wet chemical analysis and radiochemistry	NA	To determine radioisotopes release to air and water; comparison with ORIGEN2 code results; for package design and process selection	To determine radioisotopes release to air and water; comparison with ORIGEN2 code results.
Radionuclide Release Characteristics	Scintillation counting and radiochemistry on leachants to determine radionuclides that could be released to the SNF environment.	For environmental release and ALARA determination.	For environmental release and ALARA determination.	For environmental release and ALARA determination.
Chemical Phase Stability	Chemical reactions of the SNF with water, air, and moist atmospheres using a TGA system; drying tests.	NA	Developing SNF conditioning process for disposal.	To address WAC requirements
Drying/Oxidation/ Dissolution	Oxidation and drying studies with a TGA system; nitric acid dissolution tests	NA	To provide the conditions for drying SNF; oxidation and dissolution kinetics for stabilization processes.	NA
Corrosion Characteristics and Rate	Corrosion/oxidationexperiments with TGA; metallographic exams; product identification by SEM/XRD	For predicting fuel degradation; fuel swelling and hydride formation in the wet storage mode.	For fuel degradation and products prediction in wet storage.	For fuel degradation and product prediction in moist environment.
Pyrophoricity/ Combustibility	Fuel burning and ignition temperature measurements; hydride product determination by metallographic exams and MS coupled vacuum effusion test.	For fuel handling issues associated with encapsulation; hydrogen generation rate by water reaction.	Fuel handling and transportation safety issues associated with fuel reactivity with air and water.	Fuel handling and transportation safety issues associated with conditioning processes.
Criticality	Radiochemistry and gamma scan to measure fissile material content	For safety criticality calculations during fuel movement and packaging.	For safety calculations of criticality configurations of the package and storage facility.	NA
Chemical Toxicity	Chemical analysis to determine toxic elements and compounds in the SNF	NA	For safety requirements for handling, transportation and storage.	For safety requirements for handling, transportation, and storage.
Size/Weight/Density	Immersion density for fuel swelling; fuel dimensions measurements.	For resolving retrieval and packaging issues; SNF accountability.	For resolving retrieval and packaging issues; SNF accountability.	NA
Physical Properties	Charpy impact test; dewatering and filtering tests; metallographic exams.	Evaluate encapsulation	Evaluate interim storage design	NA
Physical Condition	Visual and optical exams of the SNF;.	For resolving retrieval, handling and transportation issues.	For resolving retrieval, handling and transportation issues.	For resolving retrieval, handling, and transportation issues.
Thermal Properties	Thermal diffusivity measurements using thermocouples.	NA	To calculate the temperature of the package in storage; safe operating parameter limits.	NA

NA = Not applicable.

Table 3.1b. SNF Information Needs and Tests (Pathway Options)

Information Need	Test(s)	Purpose of Data (Pathway Options)				
		Passivation	Conditioning (oxidation/consolidation)	Reprocessing/Separation	Pyroprocessing	Others
Chem/Isotopic	Wet chemistry and radiochemistry	NA	NA	Cell radioisotopes burden.	Cell radioisotope burden	TBD
Radionuclide Release Charact.	Scintillation counting and radiochemistry on leachants	Radionuclide release to passivating media.	Radionuclide release to conditioning media	NA	NA	TBD
Chemical Phase Stability	Reactions in water, air, moist atmospheres using TGA; drying tests.	Effects of prepassivation exposure on SNF surfaces	Developing SNF conditioning process for disposal.	NA	NA	TBD
Drying/Oxidation/Dissolution	Oxidation/drying studies with a TGA system; nitric acid dissolution tests	Evaluate oxidation passivation	Oxidation kinetics for conditioning processes.	Evaluation of dissolution kinetics in nitric acid .	NA	TBD
Corrosion Characteristics and Rate	Corrosion/oxidationtests by TGA; metallographic exams; product identification by SEM/XRD	Prepassivation corrosion aspects and hydride formation in wet storage.	Preconditioned corrosion aspects after wet storage.	NA	Amount of corrosion product for cell performance	TBD
Pyrophoricity/ Combustibility	Fuel burning and ignition temperature tests; hydride by metallography and MS coupled vacuum effusion.	Fuel handling associated with passivation; hydrogen generation rate.	Fuel handling and safety related to conditioning steps.	Fuel handling and safety issues related to acid dissolution steps.	Fuel handling and safety issues related to cell refining steps.	TBD
Criticality	Radiochemistry and gamma scan to measure fissile material content	Criticality calcs for fuel handling and passivation.	Criticality calculations for conditioning facilities.	NA	NA	TBD
Chemical Toxicity	Chemical analysis to determine toxic elements and compounds	NA	Safety requirements for conditioning and storage.	Safety requirements for reprocessing and storage.	Safety requirements for processing and storage	TBD
Size, Weight, Density	Immersion density, weight, dimensional measurements.	Resolve SNF handling issues during passivation.	Resolve preconditioning handling issues.	NA	NA	NA
Physical Properties	Charpy impact, dewatering, filtering tests, metallographic exams.	Evaluate encapsulation	Evaluate interim storage design	NA	NA	NA
Physical Condition	Visual and optical exams of the SNF	Retrieval, handling, and transportation issues.	Retrieval, handling, and transportation issues.	Retrieval, handling, transportation issues.	Retrieval, handling, and transportation issues.	NA
Thermal Properties	Thermal diffusivity measurements using thermocouples.	NA	NA	NA	NA	NA

NA = Not applicable.
TBD = To be determined.

Testing will provide radionuclide data to support the ORIGEN2 calculation of SNF radionuclide inventories and the calculation of heat loads in the conditioned or unconditioned SNF during transport and interim storage.

- *Final disposition objective*: To provide radionuclide inventory estimates in support of the DOE demonstration of compliance to the overall geologic repository requirements, the engineered barrier system radionuclide release limits, waste package heating and temperature restrictions of 10 CFR 60, and the reactive materials requirements of 10 CFR 60.135.

Test Methods

Elemental and isotopic inventory of the SNF will be measured by radiochemistry techniques coupled with inductively coupled plasma mass spectrometer system. Redistribution of the elements in the SNF because of the degradation process will be determined by taking specimens from different sections of the fuel.

3.1.2 Information Need: Radionuclide Release Characteristics

Functions

- *Isolation objective*: To enable demonstration of conformance of the 105-K basins packaging activities to the ALARA requirements of 40 CFR 191, the radiation protection requirements of 10 CFR 20 (Subpart C for occupational dose limits, Subpart I for storage and control, and Appendix B for annual intake limits), and the packaging and transportation safety requirements of DOE Order 5480.3. Testing will provide data to support the ORIGEN2 calculations of radionuclide inventories, and the determination of radionuclides that could be released to the basin water during wet encapsulation of the KE SNF (and possible re-encapsulation of the KW SNF) in the 105-K basins.

- *Interim storage objective*: To enable demonstration of the conformance of the SNF package during transport to and residence in the interim storage facility to the ALARA requirements of 40 CFR 191 (Subpart B for containment and assurance requirements, and Appendix A for release limits), the radiation protection requirements of 10 CFR 20 (Subpart C for occupational dose limits, Subpart I for storage and control, and Appendix B for annual intake limits), the packaging requirements of DOE Order 5480.3, the transportation requirements of 10 CFR 71 (Subpart I for general transportation requirements), and the monitored retrievable storage requirements of 10 CFR 72 (Subparts E for radioactive materials criteria and L for spent fuel storage casks). Testing will provide radionuclide release data to support the calculation of package release rates for the unconditioned SNF under both wet and dry conditions and for SNF that has been passivated or conditioned in a manner to be determined for extended interim storage. (Near-term characterization of the SNF does not involve passivating/conditioning of the SNF.)

- *Final Disposition Objective*: To enable evaluation of whether the SNF package in interim storage can meet WAC, to be determined by OCRWM, for disposal in a geologic repository or if SNF will require further treatment or conditioning to meet the WAC. WAC are expected to be based on whether the SNF waste form can be incorporated into a waste package and engineered barrier system conforming to the requirements of 10 CFR 60 (Sections 102, 111, and 113 for waste isolation and retrievability). Testing will provide wet and dry radionuclide release rate data for the SNF in its current condition.

Test Methods

SNF samples will be dissolved to obtain upper values for radionuclide release and leached to obtain realistic values for radionuclide release in water environments. A thermogravimetric analyzer (TGA) will be used to obtain realistic release values in air environments. Radiochemistry methods will be used to measure the concentrations of the major gaseous and soluble radionuclides (H^3, C^{14}, I, Kr, and oxides of Cs) in dissolved SNF samples and leachant samples to determine quantities that could be released to air and water. The radiochemistry data will be analyzed to estimate the radiation levels in the water and air. Worker radiation doses will be monitored by personal dosimetry.

3.1.3 Information Need: Chemical and Phase Stability/Degradation

Function

- *Isolation objective*: N/A

- *Interim storage objective*: To enable WHC to demonstrate the conformance of the SNF package during residence in the interim storage facility to the transportation requirements of 10 CFR 71 (Subpart I for general transportation requirements) and the monitored retrievable storage chemical stability requirements of 10 CFR 72 (Subparts E for radioactive materials criteria and Subpart L for spent fuel storage casks). Testing will provide data about the initial SNF chemical-phase condition and the stability of the material under ambient and inerted wet and dry conditions.

- *Final disposition objective*: To enable evaluation of the conformance of the SNF package in interim storage to WAC, to be determined by OCRWM, for disposal in a geologic repository, and/or the need for further treatment or conditioning to meet the WAC. WAC are expected to be based on whether the SNF waste form can be incorporated into a waste package and engineered barrier system conforming to the requirements of 10 CFR 60 (Sections 102, 111, and 113 for waste isolation and retrievability). Testing will provide data on the chemical stability of the SNF in its current condition as input for OCRWM evaluations of the reactivity of the fuel in the repository waste-package environment.

Test Methods

The corrosion kinetics of the SNF and its degraded products will be measured by metallographic and thermogravimetric techniques. The study will measure oxide thickness and weight changes of samples exposed to various atmospheric moisture conditions. The hydrogen generation rate by the water oxidation of SNF material will be measured to understand uranium hydride formation caused by fuel degradation.

3.1.4 Information Need: Drying/Oxidation/Dissolution Kinetics

Function

- *Isolation objective*: N/A

- *Interim storage objective*: To aid in the evaluation of SNF conditioning options for interim storage, determine the capability of these options to meet the monitored retrievable storage requirements of 10 CFR 72 (Subpart E for radioactive materials criteria and Subpart L for spent fuel storage casks), and other uses to be determined by DOE. Testing will provide data concerning the drying, oxidation, and dissolution kinetics of various SNF samples in their current condition. The data will enable evaluation of 1) drying as a possible initial step in passivation and/or conditioning of SNF for interim storage, 2) oxidation as a step in forming chemically stable oxides as either final waste forms for interim storage or precursors to oxide-based waste forms, and 3) dissolution of the SNF in nitric-acid-based solvents as a step in the formation of chemically stable species for interim storage.

- *Final disposition objective*: N/A

Test Methods

Sludge and fuel samples will be dried in a furnace to provide the drying conditions for the SNF. The weight change caused by loss of water and the volatile products will be monitored by a TGA/mass spectrometer (MS) system. The dried product will be further analyzed by scanning electron microscope (SEM) and x-ray diffractometer (XRD) for particle sizes, surface morphology and chemical phase. Acid dissolution of pieces of SNF and small samples (about one gram) will be conducted in a hot cell and a differential scanning calorimeter (DSC) system, respectively, to measure the dissolution kinetics and heat of reaction to support the conditioning processes.

3.1.5 Information Need: Corrosion Characteristics and Rate

Function

- *Isolation objective*: To provide sufficient data about the current state of corrosion/degradation of the KE and KW basins SNF to evaluate the adequacy of the packaging methodology used to encapsulate the KW SNF assemblies, and to demonstrate the conformance of encapsulation activities to the packaging and transportation requirements of DOE Order 5480.3. Testing will provide data concerning the type and quantity of corrosion products that have formed in the canistered SNF during storage in the KE and KW basins and the effect of the corrosion on the mechanical integrity of the SNF assemblies.

- *Interim storage objective*: To provide sufficient information concerning the corrosion of the SNF and the canisters during residence in the 105-K basins to choose between wet and dry interim storage options for unconditioned SNF, evaluate potential SNF conditioning options for interim storage (such as oxidation or passivation), and other possible uses related to corrosion during extended interim storage to be determined by DOE. The testing will provide data on the type and amount of corrosion product that formed in KE (open) and KW (sealed) environments. The data will also help provide bounding estimates of corrosion rates for unconditioned SNF under wet and dry interim storage options.

- *Final disposal objective*: To provide information that can be used to compare the expected condition of the SNF (in the event it is not extensively conditioned in the interim storage facility) and canisters, with OCRWM-generated WAC for repository emplacement.

Test Methods

The corroded SNF and canister will be metallographically examined to determine the corrosion products (oxides, hydrides and the metal). Thicknesses of each product will be used to estimate the corrosion rate. The detailed corrosion kinetics of the SNF will be determined by TGA/MS measurement of weight changes and hydrogen generation. Dimensional and immersion density measurements will be used to estimate the swelling characteristics of the fuel.

3.1.6 Information Need: Combustibility/Pyrophoricity

Function

- *Isolation objective*: To provide enough information concerning the pyrophoricity of the 105-K basins SNF to demonstrate that the in-basin packaging activities conform to the requirements of DOE Order 5480.3. The testing will provide data concerning the reactivity/pyrophoricity of the SNF during handling and examination in hot cell air, inert gas, and oxygen-saturated water conditions.

- *Interim storage objective*: To provide sufficient information concerning the pyrophoricity and combustibility of the SNF to make decisions concerning the need for conditioning before interim storage and demonstrate that the SNF waste form in interim storage conforms to the monitored retrievable storage requirements of 10 CFR 72 (Subpart E for radioactive materials criteria and Subpart L for spent fuel storage casks) and other applications concerning the safe handling of the SNF to be determined by DOE. The testing will measure the amount of potentially pyrophoric/combustible metal (U, Zr) hydrides, that may have formed in the SNF during long-term corrosion in the 105-K basins, and the energetics of their potential oxidation and decomposition reactions in dry and wet environments.

- *Final disposition objective*: To provide information concerning the potentially pyrophoric behavior of unconditioned SNF to evaluate its potential for conformance to OCRWM-generated WAC. Testing will quantify the metal hydride that formed in the SNF samples during 105-K basin storage, thereby enabling estimates of hydride formation during further wet storage of unconditioned SNF.

Test Methods

The reaction energetics of the SNF will be measured with a DSC apparatus. The heat generated will help in estimating the hydride, oxide, and metallic components of the sample. Ignition temperature of SNF will be determined by heating samples in different atmospheres. Measuring the surface area of any particulate product of SNF corrosion by the BET method will provide a specific surface area, a parameter needed to estimate the pyrophoric behavior of the SNF.

3.1.7 Information Need: Nuclear Criticality

Function

- *Isolation objective*: To enable demonstration that packaging activities will not allow nuclear criticality in the SNF and thereby conform to the ALARA requirements of 40 CFR 191, the radiation protection requirements of 10 CFR 20 (Subpart C for occupational dose limits, Subpart I for storage and control, and Appendix B for annual intake limits), and the packaging and transportation safety requirements of DOE Order 5480.3. Testing will confirm the quantity of fissile material in the SNF and the geometry basis for criticality calculations.

- *Interim storage objective*: To demonstrate that the interim storage SNF configuration meets the criticality requirements of 10 CFR 71 and DOE Order 5480.3. Testing will confirm the quantity of fissile material in the SNF and, in the case of unconditioned SNF, confirm the SNF geometry basis for criticality calculations.

- *Final disposition objective*: N/A

Test Methods

Fissile material content of the SNF will be determined by radiochemistry and mass spectrometry as input parameters to criticality configuration calculations.

3.1.8 Information Need: Chemical Toxicity

Function

- *Isolation objective*: N/A

- *Interim storage objective*: To demonstrate that the interim storage configuration meets the chemical toxicity restrictions imposed by 10 CFR 72 (Subpart A) and DOE Order 5400.3 (Section 4.a - Hazardous Waste). Testing will measure the level of hazardous constituents in the SNF samples.

- *Final disposal objective*: To aid in analyzing conformance of the waste to restrictions placed on chemically toxic materials by the repository WAC or the requirements of 10 CFR 60 (Section 135 - repository waste package component criteria). Testing will measure the level of hazardous constituents in the SNF samples.

Test Methods

The toxicity limits of the elements in the SNF will be measured by radiochemistry methods coupled with MS techniques.

3.1.9 Information Need: Size/Weight/Density Characteristics

Function

- *Isolation objective*: To aid in conducting and evaluating the packaging operations for the partially degraded KE SNF assemblies and in making decisions concerning their use to package the canister and basin sludge. Testing will provide information on the existing canister and assembly dimensions and weights and the density of the sludge.

- *Interim storage objective*: To aid in evaluating interim storage package design in the event that extensive conditioning of the SNF is not required and in demonstrating compliance of the SNF to the requirements of DOE Order 5480.3 and 10 CFR 72 (Subpart L - for spent fuel storage casks). Testing will provide information on the canister and assembly dimensions and the density of the sludge.

- *Final disposition objective*: N/A

Test Methods

SNF element weight and dimensions will be measured to estimate swelling of the fuel. The wet and dry weight of the sludge will be measured.

3.1.10 Information Need: SNF Physical Properties

Function

- *Isolation objective*: To aid in evaluating the adequacy of the packaging of the 105-K basins SNF, in conducting the packaging operations for the (partially degraded) KE basin SNF assemblies, and in demonstrating the conformance of the SNF isolation activities with the requirements of DOE Order 5480.3. Testing will provide information on the mechanical strength and integrity of KE and KW SNF and the capability of the equipment to handle canister sludge.

- *Interim storage objective*: To evaluate interim storage package design in the event that extensive conditioning of the SNF is not required and in demonstrating compliance of the SNF to the requirements of DOE Order 5480.3 and 10 CFR 72 (Subpart L - for spent fuel storage casks). Testing will provide information on the SNF assembly strength and integrity and the density of the consolidated sludge.

- *Final disposition objective*: N/A

Test Methods

The mechanical integrity of the SNF elements will be determined by crush tests. The settling time of sludge in a graduated cylinder will be measured. SEM and optical microscopy will be used to measure particle sizes. The sludge settling time is needed to schedule SNF retrieval activities in KE because visibility in the basin water is poor. The particle sizes of the sludge will be used to determine the filtration needs during sludge movement.

3.1.11 Information Need: SNF Physical Condition/Integrity

Function

- *Isolation objective*: To aid in evaluating the adequacy of the previous encapsulation of the KW SNF, in conducting the packaging operations for the (partially degraded) KE basin SNF assemblies, in making decisions concerning whether these operations can package the canister and basin sludge, and in demonstrating the conformance of the 105-K basin SNF isolation activities with the requirements of DOE Order 5480.3. Testing will provide information on degradation, swelling, and degree of cladding failure in KE and KW SNF and estimates of the amount of canister sludge.

- *Interim storage objective*: To aid DOE in evaluating the performance of the SNF in the interim storage environment in the event that the SNF is not extensively conditioned before interim storage and in demonstrating the conformance of the SNF in interim storage to the requirements of 10 CFR 72. Testing will provide information on the extent of cladding failure in the encapsulated SNF and the amount of sludge in the canisters.

- *Final disposition objective*: N/A

Test Methods

SNF relocation in canisters will be detected by visual examination of the KE canisters during sample acquisition. A non-destructive examination (NDE) using borescope and ultrasound methods will be performed in KW basin to ascertain gross fuel degradation in the sealed canisters. A detailed visual inspection of the SNF elements will be done in the hot cell to detect intact or broken elements, cracks and visible pinholes.

3.1.12 Information Need: SNF Thermal Properties

Function

- *Isolation objective*: N/A

- *Interim storage objective*: To aid in evaluating the thermal performance of the SNF package in interim storage and in demonstrating the conformance of the interim storage package to the design requirements of 10 CFR 72 (Subpart F) and DOE Order 5480.3.

- *Final disposition objective*: N/A

Test Methods

The thermal diffusivity or heat capacity of SNF samples will be measured in a controlled furnace; the decay-heat generation will be measured calorimetrically. The burning curves (i.e., temperature versus time curve of SNF material when subjected to a temperature ramp in a furnace) of fuel pieces will be determined by monitoring the temperature with an attached thermocouple. The data will be used to estimate the ignition temperature of the SNF material.

3.2 Initial SNF Sampling for Characterization

The following factors may have contributed to the current condition of the SNF and will therefore be considered in selecting the fuels for initial characterization:

1. Breached cladding: The basins contain elements whose cladding was breached during reactor discharge and/or subsequent handling. The cladding failures range from cracks to broken fuel elements, creating a variety of effective surface areas that can affect degradation rates.

2. Storage mode: The two major storage modes are open (KE basin) and encapsulated (KW basin) canisters. The open conditions are closed or open bottom (i.e., screen bottom) in the KE basin. The uranium corrosion rate is strongly influenced by the storage environment. In the sealed canisters in KW basin, portions of many fuel elements extend into the gas space of the canisters, which allows a possibility of gas-phase oxidation of the SNF. In addition, the enclosed atmosphere could deplete oxygen (by reaction with exposed fuel) in the canister water, changing the water corrosion chemistry of the fuel.

3. Irradiation history: The irradiation histories of the SNF created isotopic compositions generally between 3% ^{240}Pu and 16% ^{240}Pu of the total plutonium concentration. However, most of the fuel is either 6% ^{240}Pu (weapons grade material with burnup of 907 and 1,098 megawatt-day (MWD)/MTU for Mark IV and Mark IA fuels, respectively) or 12% ^{240}Pu (fuel grade material with burnup of 2,268 and 2,720 MWD/MTU for Mark IV and Mark IA fuels, respectively). Irradiation history affects the fuel degradation mechanism mainly through fuel swelling and embrittlement, which can generate microcracks as potential reaction sites for uranium oxidation/corrosion.

4. Fuel type and enrichment: The basins contain three main types of spent nuclear fuel: Mark IV, Mark IA and SPR (which constitutes a minor fraction of SNF inventory). The fuel types vary in size and in degree of ^{235}U enrichment. Most of the Mark IV elements extend above the water level in the sealed canisters and may have undergone vapor-phase corrosion. Most of the N Reactor spent fuel stored in the basins is enriched with 0.947% ^{235}U, except for the outer tubes of the Mark IA which have 1.25% ^{235}U, and a small inventory of Mark IV tubes with 0.71% ^{235}U. The SPR fuel was primarily in two ^{235}U concentrations; natural (0.71%) and 0.947%. The enrichment affects fuel burnup and thus embrittlement and corrosion characteristics.

5. Galvanic effects during storage: Two materials were used for the SNF canisters and racks, aluminum and stainless steel. The galvanic coupling of zirconium, and in some cases uranium, with the storage containers may affect the degradation of the fuel. SPR fuel elements with aluminum cladding could display different corrosion characteristics compared to the Zircaloy-clad N Reactor fuel. Aluminum, for example, is reported to be anodic [4] to uranium at the initial stages of corrosion, resulting in coupled uranium with a higher corrosion/hydriding rate than that of uncoupled uranium.

6. Basin water quality: Over the years the water quality of the basins has varied. Adding chlorine as a fungicide was stopped when evidence of pitting corrosion was seen on the aluminum storage

canisters. The current practice is to demineralize the basin water. Because the SNF storage spans a range of time, different fuel groups have been exposed to water of different qualities.

7. Storage time: The length of time the fuel has been in wet storage directly affects the current condition of the fuel. Fuel with long storage history has undergone considerable handling, which could have mechanically damaged the fuel cladding, increasing the area of metallic element exposed to corrosion.

8. Fuel assembly configuration: The N Reactor fuel assembly is made of two concentric tubes held together by locking spacers. During discharge and handling, the outer elements experienced much more mechanical damage than the inner tubes. Hence it is likely that the outer elements experienced more fuel degradation.

The selection of representative characterization samples for the SNF inventory must take into account these factors and the total population of fuel assemblies in the basins. The large number of fuel assemblies (103,000) in the KE and KW basins and their different storage modes and histories make a statistical sampling process impractical. Further hindrances to sampling are high cost and regulatory requirements involved in working with radioactive material (e.g., *National Environmental Policy Act*, EPA, ALARA). Thus, for the characterization activity to provide data that is representative of the fuel inventory in the two basins and at the same time to support the WHC decision-making process, a phased sampling approach has been adopted. The initial samples will be chosen to obtain the general condition and physical properties of the SNF. Sampling sufficient to support a licensing position for extended interim storage would be more extensive than described here and would be determined after the disposition pathway decision-making phase.

These initial samples for the characterization program will provide 1) minimum data to support the encapsulation of the KE basin fuel and any repackaging of the fuel stored in the KW basin, 2) bounding or worst-case samples of SNF and canister sludge for extended-interim storage treatment or conditioning evaluations by WHC, and 3) a basis for statistical sampling and analysis for SNF final disposition. These considerations, information in the literature on metallic uranium oxidation/corrosion [5-22], and the results of visual examinations in the basins will contribute to selection of the following types of fuel elements for the initial characterization:

1. *Fuel-grade and weapons-grade materials from both basins.* The SNF inventory contains more fuel-grade than weapons-grade material, and the fuel-grade material has been in storage longer than the weapons-grade material. The fuel-grade material with higher burnup is expected to degrade faster than the weapons-grade material because of defects and fission-gas accumulation in the fuel matrix. Comparisons of weapons-grade and fuel-grade fuel will help in deciding whether this factor will be significant in the next phase of fuel sampling.

2. *A breached Mark IV-type fuel with length codes S or E from KW basin.* Sampling an element that extends above the water level in the closed canisters and might have undergone vapor-phase corrosion will help determine whether significant degradation of the SNF has occurred in the closed canisters and whether any stabilized hydride is formed during the gas-phase corrosion.

3. *A KE basin outer fuel element with breached cladding.* Because there are probably more outer fuel elements with breached cladding, an outer element better represents fuel with damaged cladding. The outer elements, having a greater quantity of uranium than the inner elements, could provide a bounding example of fuel degradation.

4. *A visual intact KE basin fuel element.* Detailed examination of a sample of KE SNF that appeared intact and uncorroded during the visual examinations will show whether such SNF is in fact substantially undegraded. An intact element would be easier to transport to the hot cell for examination. Intact, uncorroded fuel could also be used to study irradiated uranium corrosion kinetics. Samples of this fuel element will undergo radiochemical analysis and mechanical integrity tests to evaluate handling methodologies for encapsulation and isolation of KE SNF.

The results of the basin visual examinations and the characterization studies on these samples will lead to selection of some or all of the following types of SNF for characterization in the subsequent fuel-sampling phases:

5. *A visual intact KW basin fuel element.* An outer element with cladding intact at the time of encapsulation may be selected from the database for the circa 1982 encapsulation. The literature indicates that the oxygen-depleted environment assumed to exist in the closed canisters could accelerate corrosion of the SNF assemblies. These elements could provide a bounding example of the fuel degradation in the closed canister environment of the KW basin.

6. *Breached fuel elements with the shortest and longest storage time.* The corrosion rate of the spent fuel in the basin might depend on the storage history. The variation of the storage history in the two basins can be analyzed by comparing fuels with long and short storage times.

7. *SPR fuel.* The SPR fuel with the longest storage history has aluminum cladding, which can influence both the corrosion rate and the products of corrosion. The Zircaloy cladding of the N Reactor fuel, unlike aluminum, is a potential sink for hydrogen generated during uranium corrosion. This sink might affect the amount of uranium hydride formed in the two kinds of fuel and provide a basis for comparison with uranium metal fuel hydriding data from the literature.

8. *Fuel from a "chip can."* Chip cans are broken pieces of fuel with large areas exposed to water. The extent of corrosion will show the effect of increased surface area on the corrosion rate and products. The examination results could indicate the time required for metallic uranium fuel pieces to corrode completely to an oxide form and the potential suitability of the oxide form for interim storage.

Sludge from all the selected canisters, when found in quantity sufficient to be analyzed (approximately 1 gram sample), will also be examined.

These samples will provide a means for determining the worst-case conditions of the SNF, establish a sampling strategy in support of characterization for extended-interim storage, and establish a basis for statistical sampling and analysis for final disposition of the fuel. Additional sampling might be required to resolve all the issues relevant to the encapsulation and the recanning activities, conditioning and process selection for long-term interim storage, and final disposition.

3.3 SNF Isolation Characterization

The first phase of characterization will evaluate the bounding initial conditions for the isolation of KE basin SNF by determining reaction products (hydrides, metals and oxides) and environmental effects of the current storage. This phase will address the issues relating to the 105-K basins SNF isolation activities by 1) review of the existing database on SNF, 2) visual inspection of the SNF in the storage basins, and 3) laboratory tests to determine the physical and chemical characteristics of the fuel, canisters, and sludge. Emphasis will be put on the potential for pyrophoricity of the degraded fuel for safety evaluations; the physical properties (such as mechanical strength of fuel or canister) for retrieval and handling activities; and corrosion characteristics to detect swelling and predict fuel degradation. Radioisotopic composition will be measured to determine what radionuclides could be released to the environment (basin water) and workers, and concentrations of fissile material (to estimate criticality configurations). These characteristics are important both in configuring the testing program and in sampling the SNF.

3.3.1 Review of Historical Database

Several 105-K basin activities will be performed in conjunction with WHC personnel to ascertain the historical factors that contributed to the existing conditions. Information gathered from these activities will be used in conjunction with the testing data to predict the degradation rate of the fuel. The following information will be reviewed:

- the existing records (e.g., reports, data sheets, video tapes, procedures) relevant to the fuel discharge activities, subsequent handling, and storage treatment and history
- the fuel attributes from the fuel fabrication processes and burnup history
- detailed description of the storage basins including their temperature history, basin water treatment, and operations.

3.3.2 In-Basin Examinations

PNL and WHC will collaborate in examining the fuel in the KE basin with an underwater video camera to ascertain the extent of degradation and the fraction of damaged fuel elements. Additionally, DOE and WHC plan a pilot-scale run as a training exercise for the KE basin fuel encapsulation activity. The intent of the pilot-scale run is to determine the safety and technical problems associated with the retrieval of the KE canisters and fuel for repackaging. Visible corrosion damage to the fuel and the canisters will be documented. A map of the fuel arrangement on the basin storage floor will be developed to the extent possible from the records before implementing the video taping. The map will present the following information:

- canister type: Mk 0, Mk I - Al or SS, and Mk II
- fuel type: Mark IV or Mark IA
- storage history

- irradiation history - weapons grade and fuel grade material
- N Reactor fuel and SPR fuel
- chip cans containing broken fuel pieces.

The in-basin visual examination will facilitate the selection of canisters for other basin actions.

In the KW basin, a series of examinations of the SNF will be performed to determine the degradation status of the fuel that will help identify canister to select for characterization. The examinations will address

- evolution of gas bubbles from the sealed canisters
- visible changes of the canisters and fuel in the KW basin; any indication of excessive corrosion or relocation of the fuel in the sealed canister
- gas and liquid sampling for laboratory analyses (see section 3.3.3)
- ultrasound measurement of liquid levels in the gas trap to identify leaking canisters.

3.3.3 KW Canister Gas and Liquid Sampling and Tests

Canisters from KW basin will be selected for gas and liquid sampling tests and for hot cell examinations. The video tapes of the SNF elements that were loaded into these canisters will be reviewed before sampling the gas and liquid for analysis. The gas and liquid samples will then be analyzed by MS and scintillation counting, and the information will help direct subsequent testing. The analytical data will include the concentrations of hydrogen, nitrogen, krypton-85, other gaseous radionuclides, and tritium, cesium, and strontium in solution. These data will indicate the extent of fuel degradation in the sealed canisters.

The pH of the water will be measured for water-chemistry changes caused by corrosion and radiolytic decomposition of water. Finally, the gas and liquid sampling will show whether the Grafoil seal failed during storage.

3.3.4 SNF Sample Integrity

Care will be taken to minimize physical and chemical changes in the SNF samples during transportation to the hot cell. Steps will be taken in particular to avoid decomposition of unstable uranium hydrides so that the samples examined in the hot cells truly represent the condition in the K basins. Each selected element will be shipped in a single-element canister filled with the same basin water. Samples from the fuel element will be cut under argon gas at low temperatures. Unused samples will be stored in ultra-pure helium-sealed capsules.

3.3.5 Hot Cell Examinations

The initial characterization of the SNF in the hot cell will involve careful optical examination of fuel elements (outer and inner). Outstanding features of any form of fuel degradation will be photographed for analysis. A preliminary visual examination of the fuel element will seek corrosion, location of the corrosion, cracks, missing fuel pieces, cladding failure (e.g., splitting, bulging), and other mechanical damage to the fuel (e.g., pinholes, embrittlement). The dimensions and immersion densities of the fuel and fuel pieces will be measured to determine the extent of swelling. The fuel element will then be sectioned for further detailed examinations.

SNF corrosion/degradation examinations: Sections of the fuel will be examined to estimate the type and rate of corrosion and the amounts, compositions, and properties of corrosion products (e.g., hydrides and oxides). Metallographic examinations of the reaction zones, loose corrosion products, and the grain boundaries of the metallic matrix will use optical microscopy, SEM, and other electron microprobe techniques. The information gathered in these examinations can be used in conjunction with data from other tests to estimate the quantities of SNF corrosion products. If needed, the identification of the phases formed will be determined using XRD and TEM. Care will be taken to maintain sample integrity. The examinations will be conducted with stringent control of the sample environment to minimize reactions that could compromise sample integrity.

Corrosion kinetics: Tests on the kinetics of the fuel degradation reactions will help identify the failure mechanisms and the type of corrosion products. These tests will involve thermogravimetric measurement of corrosion rate (Section 3.1) in controlled atmospheres. The hydrogen generation rate resulting from the reaction will be monitored by a quadruple MS. The data from these measurements will help in estimating the quantity of hydride formed during corrosion (Section 3.1.5) as well as the rate of the corrosion reaction. Concurrently, the heat of the reactions will be determined with a DSC. To supplement the data for determining the hydride formation during corrosion, hydrogen evolution during vacuum annealing of samples will be measured directly. At temperatures above 300°C, uranium hydride decomposes by the reaction:

$$UH_3 = U + 3/2H_2 \tag{1}$$

The hydride determination test is expected to measure uranium hydride in the sample, but the associated water reaction with uranium also generates hydrogen. However, careful drying techniques, such as long-term storage in a vacuum should provide a reasonable estimate of the hydride fraction for use in conjunction with other tests. Additional chemical analysis using the reaction

$$12AgNO_3 + 2UH_3 + 4H_2O \longleftrightarrow 12Ag + 2UO_2(NO_3)_2 + 8HNO_3 + 3H_2 \tag{2}$$

may also be conducted to accurately measure the hydride content of the fuel. The hydrogen gas produced by reaction (2) can be monitored by a MS.

Radionuclide tests: Fuel samples will be analyzed radiochemically to determine the isotopic inventory of the fuel and its corrosion products. This information is required for radionuclide release data. Isotopic inventories calculated using ORIGEN2 code are available for comparison with data from the laboratory tests.

3.3.6 Basin Sludge Sampling and Tests

The two basins contain different sludges. In KE, the floor is covered with sludge from corrosion of SNF in 1,405 screen-bottom Mk 0 canisters, corrosion of the racks and canisters, and intrusion sand. The sealed-bottom Mk I and II canisters in KE contain sludge from SNF corrosion. Last-sealed canisters in KW with corroding SNF also contain sludge. Each of these canister sludges will be sampled for characterization.

Sludge in the canisters

The sludge in each canister from which a fuel element is selected for hot cell examination will be sampled to determine chemical composition, chemical phases, particle size (to determine filtration needs during sludge movement), density (wet and dry), and radionuclide content.

Sludge in KE basin floor

The decision to encapsulate the sludge for near-term storage requires information about particle size and density and chemical phase composition of the sludge. The initial sludge characterization tests will focus on identifying the following:

- radionuclide inventory and chemical state
- variability of sludge composition across the basin floor
- particle size (for filtration requirements) and density
- fraction of fuel converted to sludge.

The following tests are planned to characterize the sludge in the KE basin:

- sludge settling test: Measurement of time for a water-mixed sludge to settle to the bottom of a graduated container.
- density and particle size measurement: The sludge density will be measured by weighing the settled sludge into graduated centrifuge containers after removal of the water by evaporation. The drying test would use a TGA system to measure water content of the sludge. Both optical and electron microscopy will be used to determine the size distribution of the sludge particles.
- compositional analysis: The chemical and phase composition of the sludge will be measured by XRD and TEM.
- radioisotope concentrations measurement: Isotopic content of the sludge will be determined by radiochemical techniques.

Samples of the sludge will be taken from various areas of the basin, including the locations where fuels of different characteristics are stored. For example, fuel-grade and weapons-grade material may present different radionuclide and isotopic sludge compositions. If the sludge in the sealed-bottom canisters is to be dumped into the basin during the scheduled encapsulation, sampling of the basin sludge should occur before and after that activity.

3.4 Interim Storage Characterization

Dry interim storage is expected to last at least 40 years and this requires data adequate for use in establishing the safety envelopes, predicting the long-term performance of the materials in the interim storage environment, and evaluating the need for SNF processing. Thus, the characterization activities and stabilization processes must ensure that the in situ chemical, physical and nuclear properties of the fuel and its interactions with the storage environment and container do not compromise the performance of the storage facility or change the fuel condition to a state that precludes available disposal options. The stabilization process may also serve as an intermediate (and possibly final) step for conditioning before disposal in a repository. Several options are under consideration:

- move all SNF in canisters moved to the interim storage facility
- remove SNF from canisters and repackage
- stabilize SNF by converting it to an oxide (i.e., by air oxidation, including fluid bed oxidation, enhanced fuel exposure oxidation, dissolution, calcination; water oxidation, including steam and hot water; or molten salt oxidation)
- use fuel separation processing (high-level waste constituents to borosilicate glass; all other fuel constituents to a stable waste form)
- apply multiple barriers to the fuel
- dewater/dehydrate sludge and repackage.

Ultimately, the characterization data must support design and operation of the interim storage facility. The regulatory requirements are such that, whatever the process option selected, the characterization activities must provide information relating to fissile material content for criticality estimates; oxidation/reduction reactions for stabilization processes; corrosion reactions in the interim storage package; hydriding for chemical phase stability and pyrophoricity properties; gas generation for addressing flammability issues resulting from hydrogen gas and gaseous radionuclides for airborne release; thermal properties to determine the temperature of the package; mechanical integrity for handling activities; radiolytic inventory for radioactive releases; radiation damage to determine effect on the mechanical strength of the canister; fire and explosion hazards.

The data from the preliminary examinations of the selected fuel samples in Section 3.2 might identify the variables to be considered during this stage of SNF sampling. The information gathered will serve a data quality objective process used to optimize the next sampling strategy. The data quality objective process will determine the fuel samples for the interim storage characterization activities.

3.4.1 Laboratory Tests with Unirradiated N Reactor Fuel Material

Preliminary tests on unirradiated samples of the N Reactor fuel materials will determine the rate of water corrosion of metallic uranium at the temperature ranges typical of the basin water and the ignition tests of the N Reactor material in different gaseous atmospheres. The experimental data from these tests can be compared with data from the SNF material to determine whether unirradiated N Reactor fuel could be used to do some of the processing tests. Use of unirradiated fuel material in some experiments to evaluate SNF conditioning options would reduce the cost and time necessary to carry out the tests.

3.4.2 Hot Cell Examinations

The hot cell characterization activities will begin with careful optical examination of selected fuel elements. Samples will then be cut for other detailed studies, such as corrosion and oxidation kinetics, radiochemistry, fuel dissolution and ignition tests. All examinations will be conducted with stringent control of the sample environment (i.e., provision of inert atmosphere) to minimize any occurrence of additional reactions during sample transfer activities. Corrosion of the SNF will be studied at slightly higher temperature than that of basin water. The resulting product will be subjected to detailed metallographic examinations.

SNF corrosion studies: Additional data will be gathered to elucidate fuel corrosion rate, corrosion type, and the amounts, compositions, and reaction properties of the different types of corrosion products (hydrides and oxides) by the methods described in Section 3.3.4. Lessons learned will be used to make any changes needed to improve data quality.

Oxidation kinetics: The corrosion kinetics of the SNF will be studied to support decisions on selection of the conditioning process and the stability of the interim storage package. The tests will include the measurement of corrosion rate in controlled atmosphere; TGA and DSC will be used to measure the heat of reaction by the method described in Section 3.3.4.

Process Development Tests

SNF ignition test: The ignition temperature in various atmospheres (i.e., air, moist air, or moist inert) of samples of the fuel will be measured in the hot cell. Sample size will be in the range of grams to tens of grams. Experiments with larger pieces of fuel will minimize the inhomogeneities that might affect tests of small samples. Such studies provide information relevant to the conditions necessary for pyrophoric behavior of the SNF.

Dry storage environmental changes tests: The SNF will be dried to determine the conditions necessary for dewatering the fuel for a dry storage facility. The drying will remove all the water of hydration from the oxides formed during corrosion. Drying the fuel at temperatures above 300°C will also decompose any hydride product in the fuel. Tests will use a vacuum or inert atmosphere coupled with an MS.

Dissolution tests: The rate of nitric acid dissolution of sections of the fuel element will be measured in the hot cell to determine the heat generation rate and the dissolution behavior of the different fuel forms during the fuel-degradation processes.

Radionuclide Inventory Tests

Radiochemical techniques will be used to determine the isotopic inventory of the SNF and its degradation products. These techniques are well established at the PNL analytical chemistry laboratory. The data will be compared to ORIGEN2 code output.

3.4.3 Basin Sludge Sampling and Tests

Sludge in KW canisters

The sludge in every canister from which a fuel element is selected for hot cell examination will be sampled for testing to determine the chemical and phase composition, particle size, density (wet and dry), and radionuclide concentrations.

Sludge in KE canisters

The envisioned encapsulation activity will put the basin sludge in sealed canisters. The physical and chemical properties of the canistered basin-sludge have to be characterized for the extended-interim storage. The particle size and density for compacting, radionuclide concentration for determining radionuclides that could be released, and the chemical and phase composition will be measured to address the sludge handling issues and container design. Treatment options for the interim storage facility will require knowledge about moisture and waters of hydration in the sludge. The following tests are planned to characterize the basin sludge in the sealed canisters:

- *Density and particle size measurement:* The sludge density will be measured by weighing the settled sludge into graduated centrifuge containers after removal of the water by evaporation. The drying test could be done with a TGA system to measure water content. Optical and electron microscopy will be used to determine the size distribution of the sludge particles.

- *Dewatering/dehydration tests:* To support compacting processes for a dry storage facility, the water of hydration and the moisture in the sludge will be removed. A thermogravimetric system will be used to determine the temperature and environment needed to effectively dry the sludge.

- *Compositional analysis:* The compositional makeup of the sludge will be measured by wet chemistry techniques, XRD, and TEM.

- *Radioisotope concentrations measurement:* Isotopic content of the sludge will be determined by radiochemistry techniques.

Samples of the sludge will be taken from a number of canisters to check the variability of its physical and chemical properties.

3.4.4 Monitoring Programs

After establishing the initial conditions of the SNF in the interim storage facility, PNL will monitor the stabilization and degradation behavior of the SNF. This will involve periodic examinations of representative samples of the SNF in storage and will require development of examination equipment and instrumentation to obtain the data.

3.5 Final Disposition Characterization

The WAC for geologic disposal of the SNF have not been developed nor will they be developed by the characterization program. However, if the SNF is to be disposed of in a geologic repository, the waste form must meet WAC established by the OCRWM. The WAC will be driven by the performance assessment process for the waste package. Data generated during the SNF characterization program will be used for waste form processing and certification evaluations. The following processing options are being considered:

- direct disposal in the current form (most unlikely)
- encapsulation by applying multiple barriers (e.g., ceramic, glass, metal) to SNF substantially in its current form and/or after conversion to an oxide form
- encapsulation by incorporation into a waste matrix (e.g., borosilicate glass, ceramic).

Tests will be designed to ascertain the waste form conformance to WAC for those options selected for development by DOE.

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Appendix A

Hanford Spent Nuclear Fuel

Appendix A

Hanford Spent Nuclear Fuel

Different types of spent nuclear fuels [2] are stored in various locations at the U.S. Department of Energy's (DOE) Hanford Site (shown in Figure A.1). Much of the spent fuel, approximately 2100 metric tons of uranium (MTU), is stored in two water pools, the 105-K East (KE) and 105-K West (KW) basins. As a result of safety issues relating to leakage of radionuclides and fuel degradation, the Site custodians, Westinghouse Hanford Company (WHC) and DOE, ordered that part of the fuel inventory be characterized in support of the spent nuclear fuel (SNF) disposition effort. This appendix describes the stored fuel, its origins, inventory, and storage condition.

A.1 Reactors

The SNF under the characterization program were discharged from two types of defense production reactors at the Hanford Site, the dual-purpose N Reactor and the Single-Pass Reactors (SPR).

A.1.1 N Reactor

The pressurized water-cooled 105-N Reactor, moderated by graphite, began operation in 1963. It was initially designed to produce plutonium for the defense production industry but was later modified to also generate electricity for the Washington Public Power Supply System.

The 1800-ton core reactor is 10 m high by 10 m wide by 12 m long. The fuel and the cooling water were contained in 1003 horizontal Zircaloy-2 process tubes, with a total fuel load of 366 MTU. Operating reactivity, neutron flux shaping, and emergency shutdown were provided by 34 horizontal boron-containing control rods perpendicular to the process tubes. The N Reactor produced 4000 megawatts of heat, to generate 860 megawatts of electricity. The reactor ceased operations in 1987, and the final core was discharged in April 1989.

A.1.2 Single-Pass Reactor

Eight single-pass, water-cooled, graphite-moderated reactors produced plutonium, beginning in 1944 with the B Reactor and ending in 1971 with the KE Reactor shutdown [23-28]. The reactors and their operating periods are listed in Table A.1. The older reactors B, C, D, DR, F, and H were of similar design. Their graphite block cores were 11 m high by 11 m wide by 8.5 m long. The fuel and the cooling water were contained in a total of 2004 horizontal aluminum tubes. Reactors B, D, DR, and F had 9, and C and H had 15 horizontal water-cooled, boron-containing control rods running perpendicular to the process tubes. These rods provided operating reactivity, neutron-flux shaping, and emergency shutdown controls. Additional emergency shutdown was provided by vertical channels penetrating the core, which could be filled with either boron-containing safety rods or boron-containing steel balls.

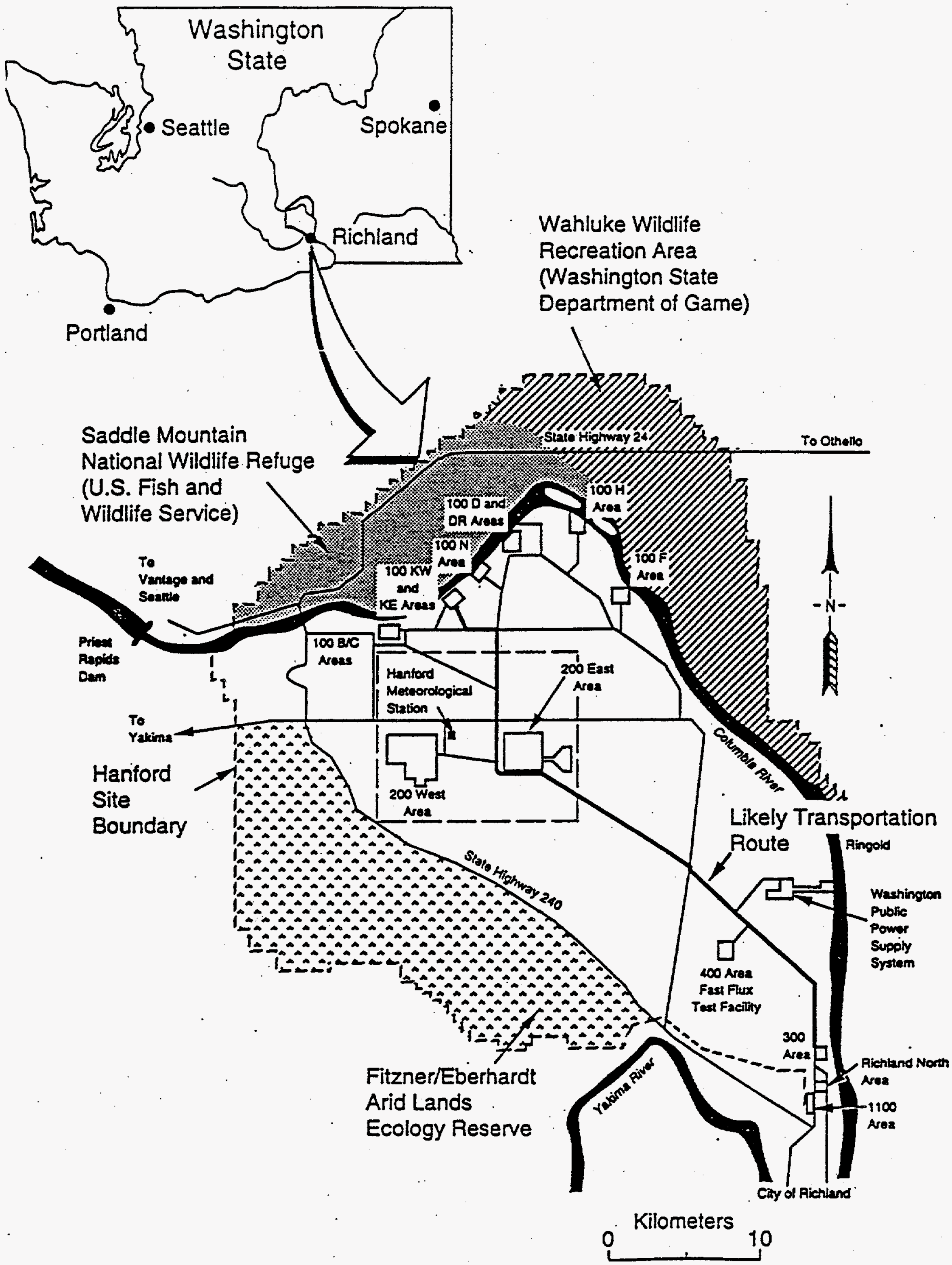


Figure A.1. Hanford Site Map

Table A.1. Operating Periods of the Hanford Single-Pass Reactors

Reactor	Hanford Area	Operating Period	
		Start Up	Shut Down
B	100-B/C	1944	1968
C	100-B/C	1952	1969
D	100-D/DR	1944	1967
DR	100-D/DR	1950	1964
F	100-F	1945	1965
H	100-H	1949	1965
KW	100-KE/KW	1955	1970
KE	100-KE/KW	1955	1971

The design of the KE and the KW reactors differed from the older reactors as follows. Their graphite cores were 12.5 m high by 12.5 m wide by 10.7 m long; each had 3220 horizontal aluminum process tubes, 2400 of which were later replaced with Zircaloy tubes, and 20 horizontal control rods.

A.2 Fuel Description

A.2.1 N Reactor Fuel

The N Reactor fuel elements consisted of two concentric tubes made of uranium metal co-extruded into Zircaloy-2 cladding. The two basic types of fuel elements (Mark IV and Mark IA) are differentiated by their uranium enrichment. Mark IV fuel elements had an initial enrichment of 0.947% ^{235}U in both tubes and an average uranium weight of 22.7 kg. Mark IA fuel elements had an initial enrichment of 1.25% ^{235}U in the outer tube and 0.947% ^{235}U in the inner tube. Different lengths of the Mark IA fuel element gave variation in the quantity of uranium and an average uranium weight of about 16.3 kg. A small amount of Mark IV fuel had 0.71% ^{235}U content. A Mark IV fuel element is shown in Figure A.2. Mark IA fuel elements are similar in design. The physical characteristics of Marks IV and IA fuel elements are given in Table A.2. Table A.3 shows the chemical compositions of the N Reactor fuel.

A.2.2 Single-Pass Reactor Fuel

The SPR fuel consisted of a machined uranium core that was sealed within and metallurgically bonded to an aluminum can. SPR fuel elements had three basic designs: solid cylinders, hollow-core cylinders, and tubes. All three types are illustrated in Figure A.3. The solid fuel was used from 1944 until the late 1950s. Increasing reactor power levels and fuel exposure caused significant fuel failures beginning in 1951. The hollow-core fuel was used concurrently with the solid fuel from late 1954 until

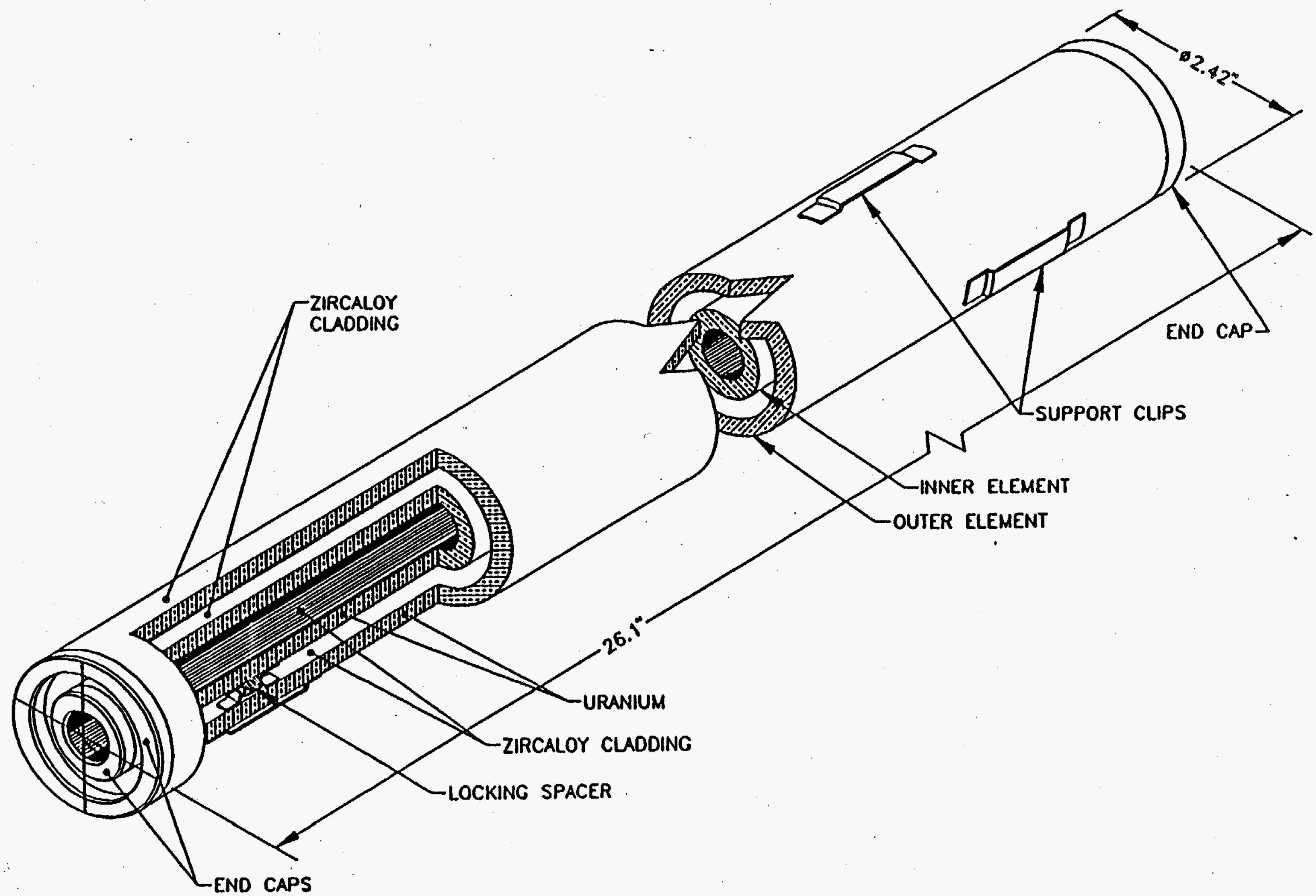


Figure A.2. 105-N Reactor Mark IV Fuel Element Assembly

Table A.2. 105-N Reactor Fuel Element Description

	Mark IV				Mark 1A		
Pre-irradiation enrichment	0.947% of ^{235}U				1.25 or 0.947% ^{235}U		
Type-Length Code[(a)]	E	S	A	C	M	T	F
Outer Length (cm)	66.3	62.5	58.9	44.2	53.1	49.8	37.8
Element Diameter (cm)							
1. Outer of outer	6.15				6.10		
2. Inner of outer	4.31				4.50		
3. Outer of inner	3.25				3.18		
4. Inner of inner	1.29				1.12		
Cladding Weight (Kg)							
Outer element	1.09	1.04	0.99	0.79	0.88	0.83	0.66
Inner element	0.55	0.52	0.50	0.40	0.53	0.51	0.40
Weight of Uranium in Outer Element (Kg)							
1. (0.947% ^{235}U)	15.95	15.00	14.14	10.47			
2. (1.25% ^{235}U)					11.06	10.38	7.84
Weight of U in inner	7.48	7.02	6.62	4.94	5.48	5.12	3.90
% of total elements	63				37		
(a) Letter code differentiates the different lengths of the Mark IV or Mark 1A fuel							

1957 in an attempt to reduce fuel failures, but the design was unsuccessful. Late in 1955 the tube design, called internally and externally cooled (I & E) fuel, was introduced to reduce fuel failures.

The SPR fuel was made primarily in two ^{235}U concentrations, natural (0.71%) and 0.94%; although other enriched fuels (1.25%, 1.7% and 2.1% ^{235}U), were also made. The lengths of the fuel elements ranged from approximately 12 cm to 23 cm; specific lengths were generally associated with specific ranges of enrichment. The diameters of most of the fuel elements ranged from 3.45 cm to 3.63 cm for compatibility with slight differences in the process tubes of different reactors. The fuel for the KE and KW Reactors was generally the largest and included some 5.1-cm diameter elements. The fuel for the C Reactor was larger than that used in the other five reactors. These variations in length, diameter, and secondary features, such as tru-line ends, led to the need for many similar but distinct fuel elements. Typical impurities of the materials in the SPR fuel elements are shown in Table A.4.

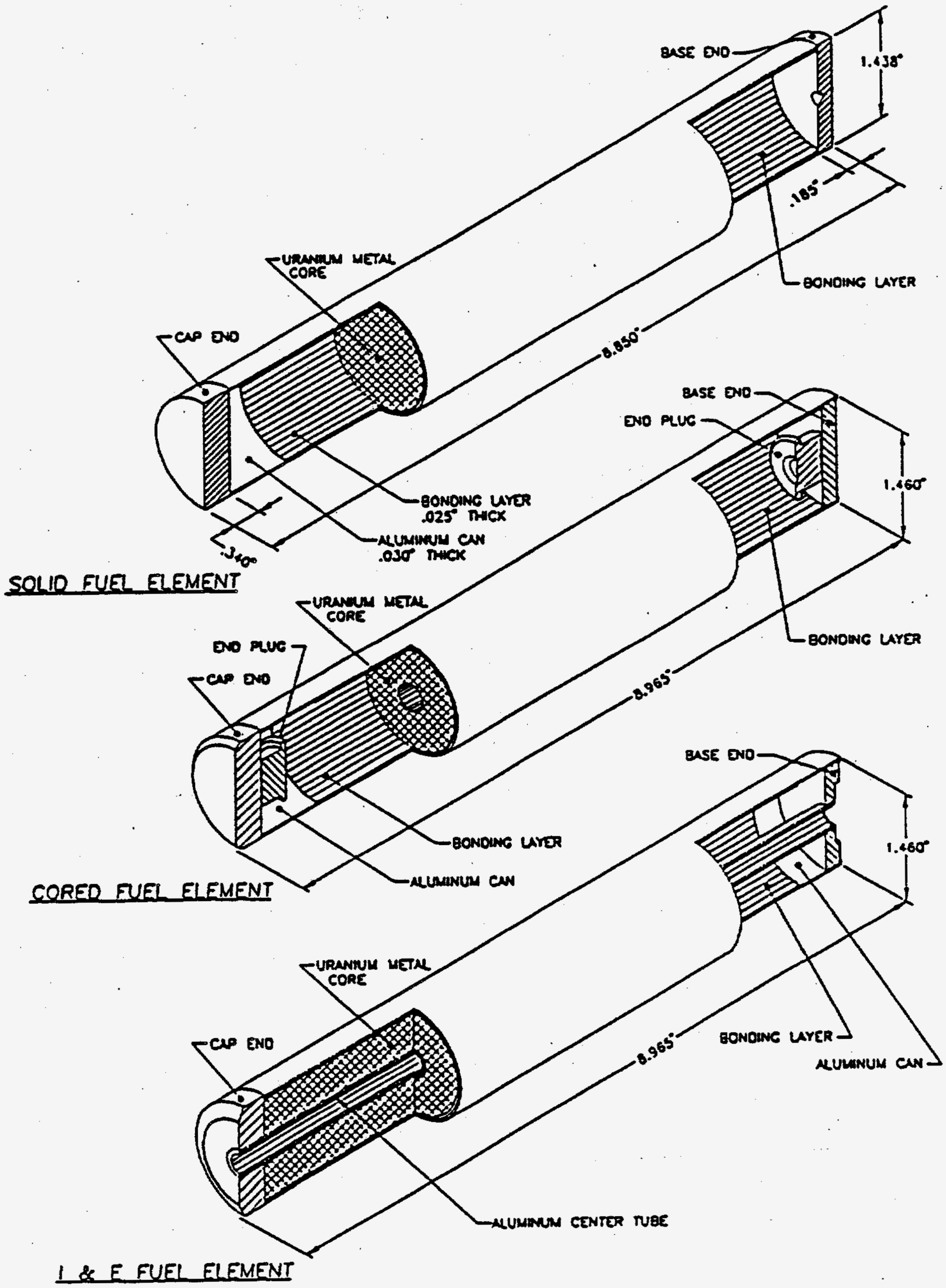


Figure A.3. Single-Pass Reactor Fuel Elements

Table A.3. Chemical Composition of 105-N Reactor Fuel Elements[(a)]

Element	Uranium Alloy 601	Zircaloy-2	Braze Filler
Aluminum	700 - 900	75	145
Beryllium	10	-	4.75 - 5.25 wt%
Boron	0.25	0.5	0.5
Cadmium	0.25	0.5	0.5
Carbon	365 - 735	275	500
Chromium	65	0.05 - 0.15 wt%	0.05 - 0.15 wt%
Cobalt	-	10	20
Copper	75	50	60
Hafnium	-	200	200
Hydrogen	2.0	25	50
Iron	300-400	0.07 - 0.20 wt%	0.06 - 0.21 wt%
Lead	-	100	130
Magnesium	25	20	60
Manganese	25	50	60
Molybdenum	-	50	50
Nickel	100	0.03 - 0.08 wt%	0.03 - 0.08 wt%
Nitrogen	75	80	200
Oxygen	-	-	2300
Silicon	124	100	250
Sodium	-	20	20
Tin	-	1.20 - 1.70 wt%	1.14 - 1.70 wt%
Titanium	-	50	50
Tungsten	-	50	100
Uranium	Balance	3.5	4
Vanadium	-	50	50
Zirconium	65	Balance	Balance

(a) Concentrations given in parts per million maximum or ppm range.

Table A.4. Impurity Limits in the Metallic SPR Fuel Elements[a]

Impurity	Uranium Alloy 501	Aluminum Alloy X-8001	Al-Si Braze
Boron		10	
Cadmium		30	
Carbon	150-750		
Chromium	65		
Cobalt		10	
Copper		1500	1000
Iron	150	4500-7000	5000
Lithium		80	
Magnesium	25		
Manganese	25		
Nickel	100	9000-13000	
Nitrogen	100		
Silicon	75	1700	
Other, each		500	500
Others, total		1500	2000

(a) The units are in parts per million.

The physical characteristics of some of the SPR fuel elements are listed in Table A.5. To permit easy differentiation, each fuel element is stamped with a 6-character code. Partial codes are listed in column 1 of Table A.5. The first letter indicates the reactor in which the fuel was used (K for KE and KW, C for C reactor, O for the other five reactors); the number indicates the number of times the design was modified; and the third character indicates the general uranium content and other features (N for natural uranium, E for enrichment, W for natural enrichment with a 5.08 cm cooling water mixing attachment, and D for depleted uranium).

A.3 Fuel Fabrication

A.3.1 N-Reactor Fuel

The N Reactor fuel was fabricated by a co-extrusion process. In this process, each uranium tube was encased in an inner and outer sleeve of Zircaloy-2 and a copper outer sheath. This assembly was

Table A.5. Physical Characteristics of Single-Pass Reactor Fuel

Code	% ^{235}U	Fuel Element			Uranium Core			
		Length (cm)	Diameter (cm)		Length (cm)	Diameter (cm)		Weight (Kg)
			Outer	Inner		Outer	Inner	
C2N	0.714	22.77	3.72	0.95	22.28	3.48	1.22	3.34
K4N	0.714	22.77	3.71	0.98	22.28	3.48	1.26	3.30
K5N	0.714	22.77	3.86	1.07	21.15	3.63	1.35	3.56
O3N	0.714	22.77	3.67	0.79	22.28	3.44	1.07	3.37
C6N	0.714	22.77	5.04	0.85	21.15	4.78	1.15	6.75
C3E	0.947	16.87	3.71	0.95	15.37	3.48	1.24	2.40
K5E	0.947	16.61	3.83	1.10	15.24	3.60	1.38	2.49
O3E	0.947	16.87	3.67	0.79	15.37	3.44	1.07	2.43
K4W	0.714	16.87	3.71	1.02	15.37	3.48	1.30	2.36
K5	1.250	14.07	3.82	1.11	12.70	3.61	1.38	2.08

evacuated and sealed to prevent oxidation during preheating and extrusion. As the tube assembly was extruded under high pressure and elevated temperature, a solid-state diffusion bond formed between the uranium core and the Zircaloy-2 cladding.

The extruded fuel was cut to the desired lengths and a recess was machined into the uranium at each end of the fuel section. The copper sheath was then acid-stripped, and an acid etching process removed residual uranium from the cladding. The tube ends were closed by placing a braze ring, made of Zircaloy-2 and 5% beryllium, and a Zircaloy-2 end cap in the recess at each end of the fuel and induction-heating the assembly to brazing temperature (approximately 1050°C) in a vacuum. The junction of the end cup, braze, and cladding was fusion-welded to alloy the braze material with the cladding and thereby improve corrosion resistance and provide a hermetic seal. The fuel was then heat-treated and cleaned with an abrasive grit blast and acid baths.

Six spacers were welded to the outside of the inner tube to ensure proper alignment and locking with the outer tube when the two tubes were assembled together. Eight Zircaloy-2 support clips were welded to the outside of the outer tube to ensure proper alignment of the fuel when inserted into the N Reactor core. Low-carbon steel "shoes" were crimped onto the clips to reduce the effect of their rubbing against the reactor tubes during fueling and discharge of the fuel elements. The shoes varied in thickness from 10 to 18 mils and contained 0.42 to 0.74 grams of iron. Finishing operations on each tube included nondestructive testing, heating in an autoclave, inspecting, and inserting an inner tube into an outer tube to produce the finished fuel element.

A.3.2 Single-Pass Reactor Fuel

The lead-dip fabrication process was used after 1954. In this process, a machined uranium core was sealed in and metallurgically bonded to an aluminum can. The core was first subjected to a degreasing and pickling operation to clean its surface and heated to bonding temperature by immersion in a duplex bath of aluminum-silicon alloy floating on lead. The core was preheated to brazing temperature by the lead and wet with the brazing alloy by brief agitation in the Al-Si layer. The core was then transferred to a heated Al-Si canning bath, where it was inserted into a preheated aluminum can and topped with an aluminum cap. After quenching, the fuel element was welded to create a continuous bond between the uranium core, the Al-Si braze ($\sim$20-mil thick) and the aluminum can. The ends of the fuel element were machined to the proper length and contour,and the exposed braze area was closed by a tungsten inert-gas weld bead. Finally, the completed fuel element was heated in an autoclave and inspected visually and by other non-destructive methods.

A.4 Fuel Storage and Inventory

The N Reactor fuel is stored in the KE and KW basins in the 100-K Area of the Hanford Site. The basins are constructed of reinforced concrete walls and floors. Their dimensions are 38 m long, 20 m wide, and 6 m deep. Figures A.4 and A.5 are schematics of the basins.

Water levels are maintained in each basin at a minimum of 3 m above the irradiated fuel to cool the fuel and provide radiological shielding for personnel working in the facility. The water in each basin is recirculated through a closed water-cooling system with mechanical chillers. Filters and ion exchange systems maintain basin-water clarity and remove radionuclides. The irradiated N Reactor fuel inventory is listed in Table A.6.

The KW Basin contains approximately 952 MTU of N Reactor fuel in 3821 closed (1773 Mk Is and 2048 Mk IIs) canisters, shown in Figure A.6, and the basin has not experienced water leakage to the ground [2]. Of the Mk I canisters, 777 are made of aluminum and 996 are made of stainless steel. The Mk II canisters are made of 304L stainless steel. The lids and seals of the Mk I and Mk II canisters are designed differently. The Mk I canister lid and seal were designed so that when installed, the Grafoil seal is compressed between the outside of the lid and the inner wall of the canister. Sticking of the Grafoil seal to the canister and fragmentation into the canister when the lid was removed led to modifications in the Mk II lid and seal design that retain the GRAFOIL seal.

The KE basin was constructed in 1951 to provide interim storage for the SPR fuel discharged from the KE reactor. This basin was reactivated in 1970 to serve as temporary storage for the N Reactor fuel. The KE basin is made of unlined concrete filled with water of 1.3 million gallons. An asphaltic membrane beneath the basin was intended to direct all leaked water to a collection sump. The basin contains approximately 1150 MTU of N Reactor fuel stored in 3668 open-top canisters (Mk 0, Mk I, and Mk II), and its water is contaminated with radionuclides. There are about 1405 Mk 0 aluminum canisters with screen bottoms, 1406 Mk I stainless-steel canisters and 857 Mk II stainless-steel canisters in the KE basin. The arrangement of the canisters is shown in Figure A.6.

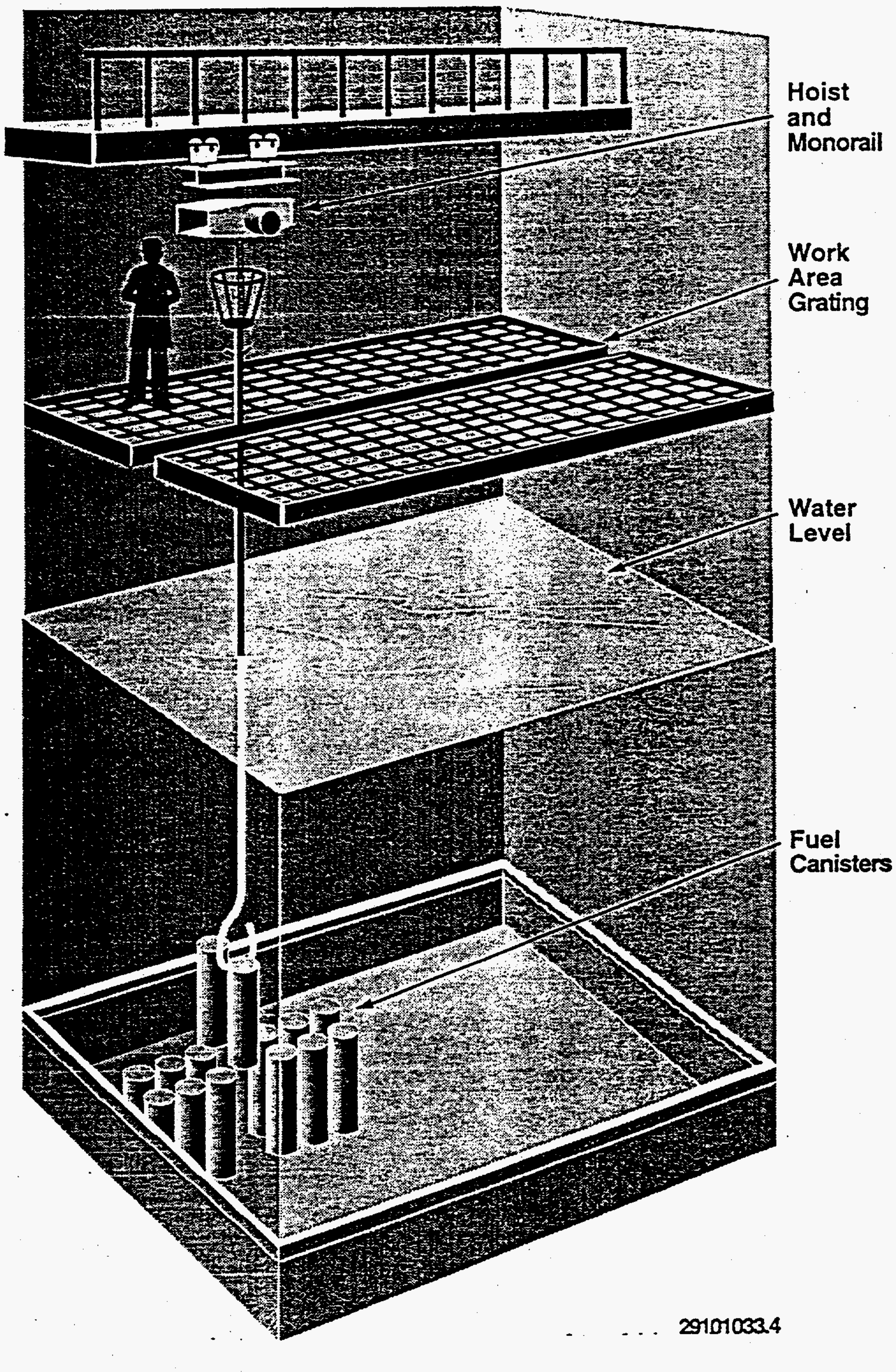


Figure A.4. Schematic of K-Basins

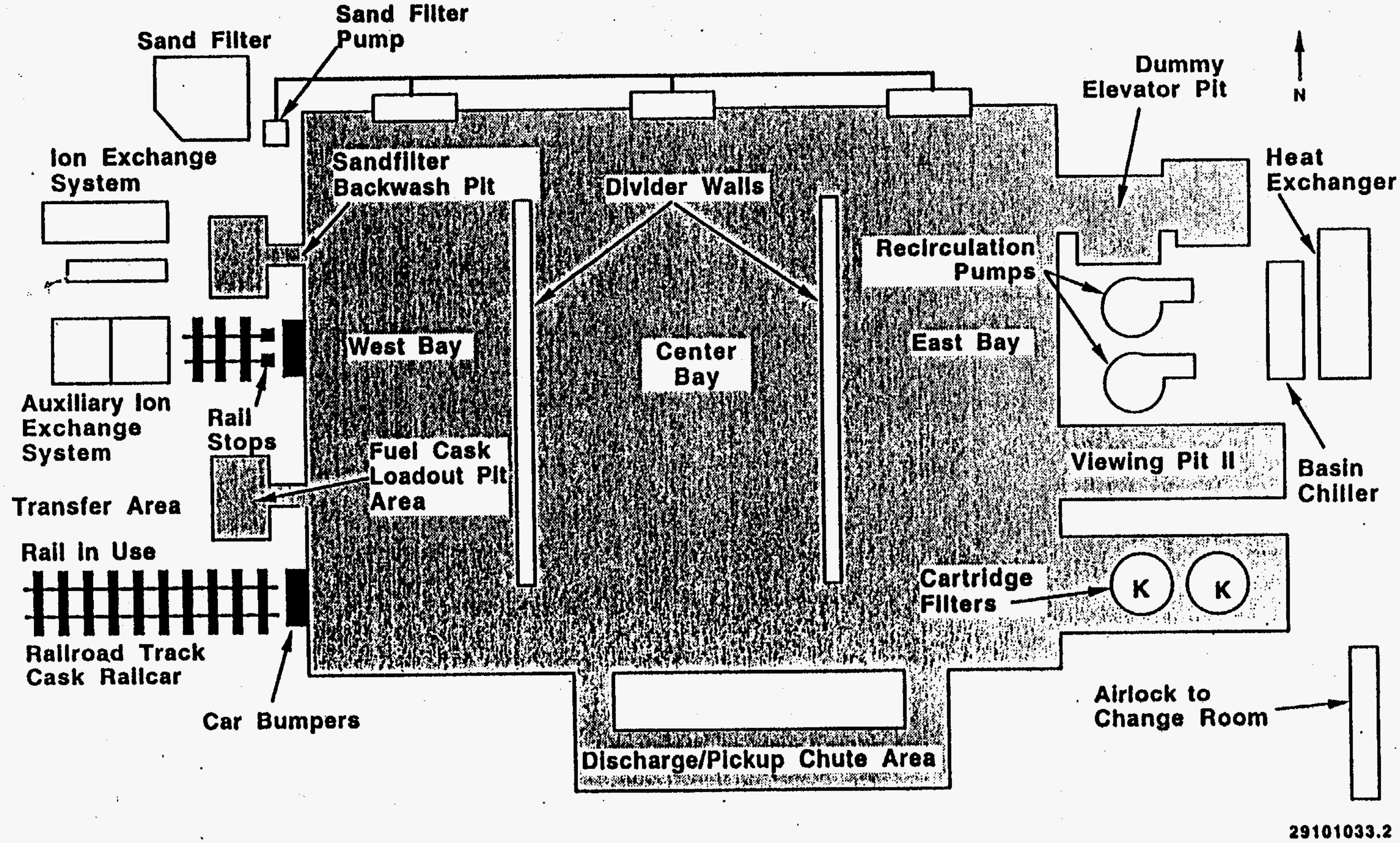


Figure A.5. The K-Basin Layout

Table A.6. 105-N Reactor Fuel Inventory

Fuel Grade	Fuel Type						Total (MTU)
	Mark IV			Mark IA			
	Weight (MTU)	Burnup (MWd/MTU)	Decay Time (yrs)	Weight (MTU)	Burnup (MWd/MTU)	Decay Time (yrs)	
Weapon	291.9	907	13-23	39.3	1089	13-23	331.2
Fuel	1176.3	2268	7-8	588.0	2720	7-8	1764.3
Unknown	0.3			0			0.3
Total (MTU)	1468.5			627.3			2095.8

In addition, each basin contains a small amount of SPR fuel; KW contains 0.1 MTU and KE contains 0.4 MTU. The specific types and number of the SPR fuel elements are listed in Table A.7.

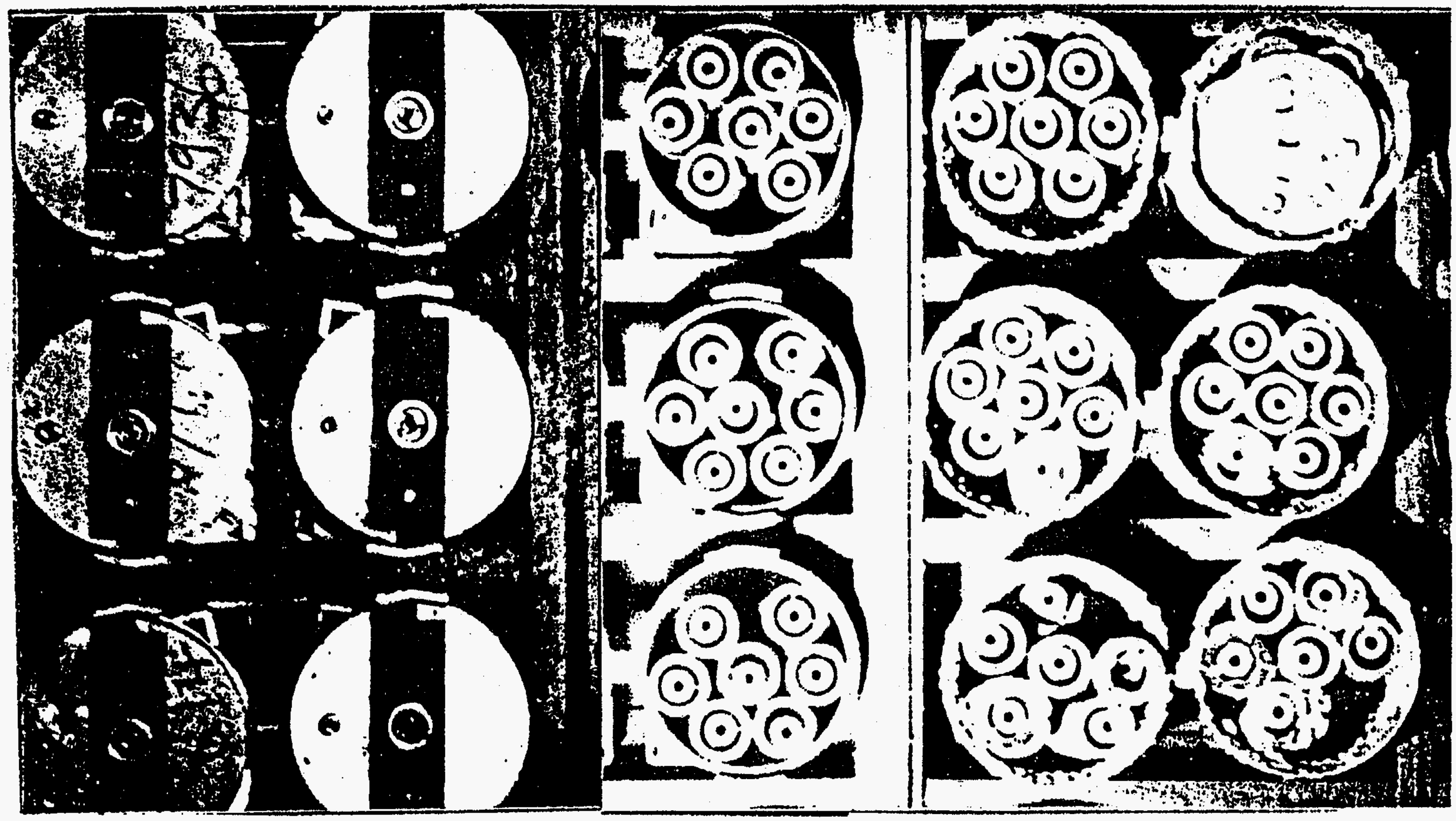
Three different types of fuel canisters (Mark 0, Mark I, and Mark II) hold the irradiated fuel elements. Each has two cylindrical barrels that hold seven elements vertically. Mark 0 canisters have screen bottoms and open tops, and are fabricated of aluminum. The canister designs are shown in Figure A.7. Closed canisters are of two designs, Mk I and Mk II. They have solid bottoms and sealed tops, although some canisters do not have the tops installed. Each sealed canister barrel is vented through an external gas trap to prevent buildup of pressure in the canister. The Mk I canisters were made of either aluminum or stainless steel. All Mk II canisters were made of stainless steel.

The canisters contain either Mark IA or Mark IV fuel. Some canisters contain fuel elements of different lengths or with missing inner or outer tubes. Some canisters contain a mixture of odd pieces [29].

All three types of canisters were filled under water. The fuel, water, and gas volumes in the canisters of the KW basin vary with the type and condition of the fuel. When the Mk I and Mk II KW canisters were originally filled with fuel and closed, the top 6.35 cm of water in each barrel was displaced with inert nitrogen gas. As a result they contain 2.8 to 3.9 gallons of water and 1.34 to 2.10 liters of nitrogen gas [29].

The Plutonium-Uranium Extraction Plant (PUREX) dissolver cells A, B, and C, also contain some intact Mark IV fuel elements and fuel element pieces that cannot be retrieved once they spill to the floor during dissolver charging operations, which occurred over the past 23 years. An estimate of the amount and location of fuel on the floor of the PUREX dissolver cells (given in Table A.8) was based on reports and records of the fuel spills [30-37].

Most of the SPR fuel, about 2.87 MTU contained in 779 fuel elements, is stored at PUREX [38]. Most of this fuel is residual material from the KE and the KW Reactors, but it also includes fuel from the clean-out of the C and the D storage basins. The PUREX storage basin is at the east end of the



105-KW Basin **105-KE Basin**

Figure A.6. N Reactor Fuel Storage at KE and KW Basins

Table A.7. SPR Fuel Elements Stored in KE and KW Basins

Fuel Type	Basin	Quantity
C2N	KE	2
K4N	KE	3
K5N	KE	42
O3N	KE	15
C3E	KE	42
K5E	KE	29
O3E	KE	4
K4W	KE	1
Uncertain[a]	KW	47

(a) Records of these fuel elements do not indicate the specific fuel type.

202A Building in the 200 East Area of Hanford. Its dimensions are 9.5 m long, 6.1 m wide, 5.2 m deep. It is open to the PUREX canyon and contains 3.7 m of water to limit the radiation level at the surface. The fuel is stored in four buckets, which are 43-cm square by 52-cm tall with a grid of 1.27 cm diameter drain hole in the bottom. Figure A.8 is a schematic of the SPR fuel in the PUREX storage basin.

A.5 Fuel Condition

The prevailing physical and chemical characteristics of the SNF result from breaching of the fuel cladding during discharge and subsequent handling of the fuel and the corrosion of the fuel in the storage-basins water. The metallic uranium fuel and the Zircaloy cladding, being reactive with water and air, started corroding under the current storage conditions. The corrosion rate can be influenced by a number of factors, such as the storage mode (open or encapsulated), the fuel type (Mark IV and Mark IA), the outer or inner fuel element, the temperature of the storage medium, the presence of alloys and impurities in the fuel, and the fuel irradiation history.

A.5.1 Damaged/Breached Fuel During Discharge

The fuel inventory in the storage basins contains elements whose cladding was breached during reactor discharge and subsequent handling. The cladding failures range from cracks to severed fuel elements and were generally consequences of the ~0.7-m free-fall during discharge. Visual inspection above a pool of water suggested that as many as 7% of the fuel elements suffered significant damage during the discharge operation.

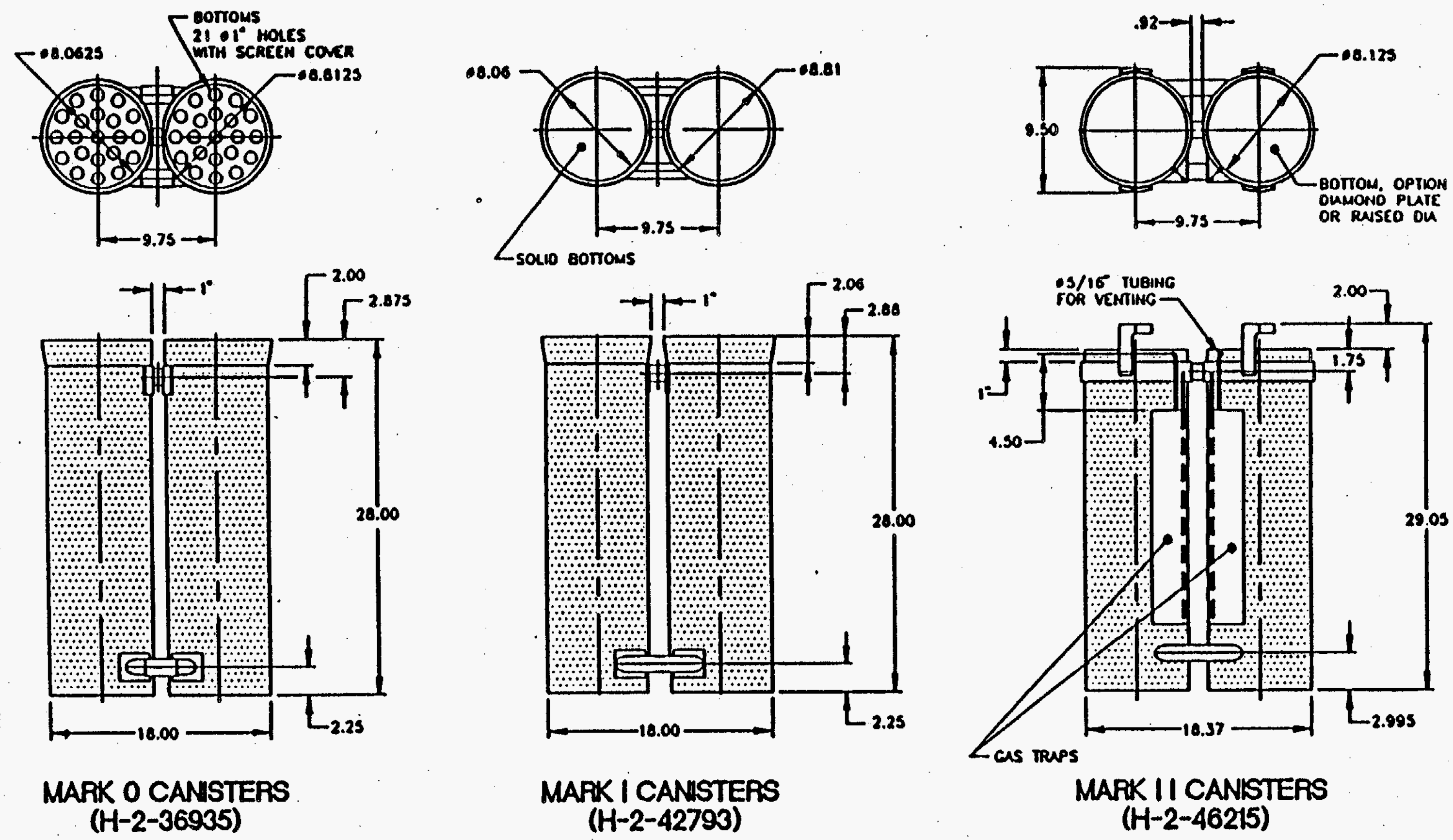


Figure A.7. N Reactor Fuel Storage Canisters

Table A.8. Estimated N-Reactor Fuel Elements on PUREX Canyon Floor

PUREX Cell	Uranium (Kg)	Number of Fuel Elements	Mark IV Fuel Length Type
A	26	3.5 inner	E
B	230	22.5 inner 11.5 outer	C
C	4 - 8	1.0 inner	E

A.5.2 Corrosion Damage in the Basins

The reactivity of uranium with water and air are strongly effected by surface area, temperature, alloys and impurities, as well as other factors. Under certain conditions, metallic uranium and zircaloy are pyrophoric.

In the storage basins, uranium fuel with breached cladding corrodes when it comes into contact with the basin water or the water in the sealed canisters. Corrosion products include the oxides of uranium, uranium hydride, and hydrogen gas. Uranium corrosion generally proceeds by the following reactions.

The water-metallic uranium reaction:

$$U + 2H_2O \rightarrow UO_2 + 2H_2 \tag{A.1}$$

A fraction of the hydrogen generated by Equation (A.1) further reacts with the uranium to give uranium hydride:

$$U + 3/2H_2 \rightarrow UH_3 \tag{A.2}$$

The generated hydride formed in Equation (A.2) is reported to be generally located ahead of the oxide layer. The uranium hydride in the proposed mechanism reacts with water by Equation (A.3) to yield unprotective uranium dioxide, hence the linear corrosion rate of uranium in water. The uptake of the hydrogen by either the metal or the oxide film could enhance the oxidation reaction by two possible mechanisms:

(1) the incorporation of hydroxyl ions in the oxide lattice can increase the ionic diffusion

(2) formation of uranium hydride, although a transitory product, would deleteriously affect the integrity of the protective oxide layer.

$$UH_3 + 2H_2O \rightarrow UO_2 + 7/2H_2 \tag{A.3}$$

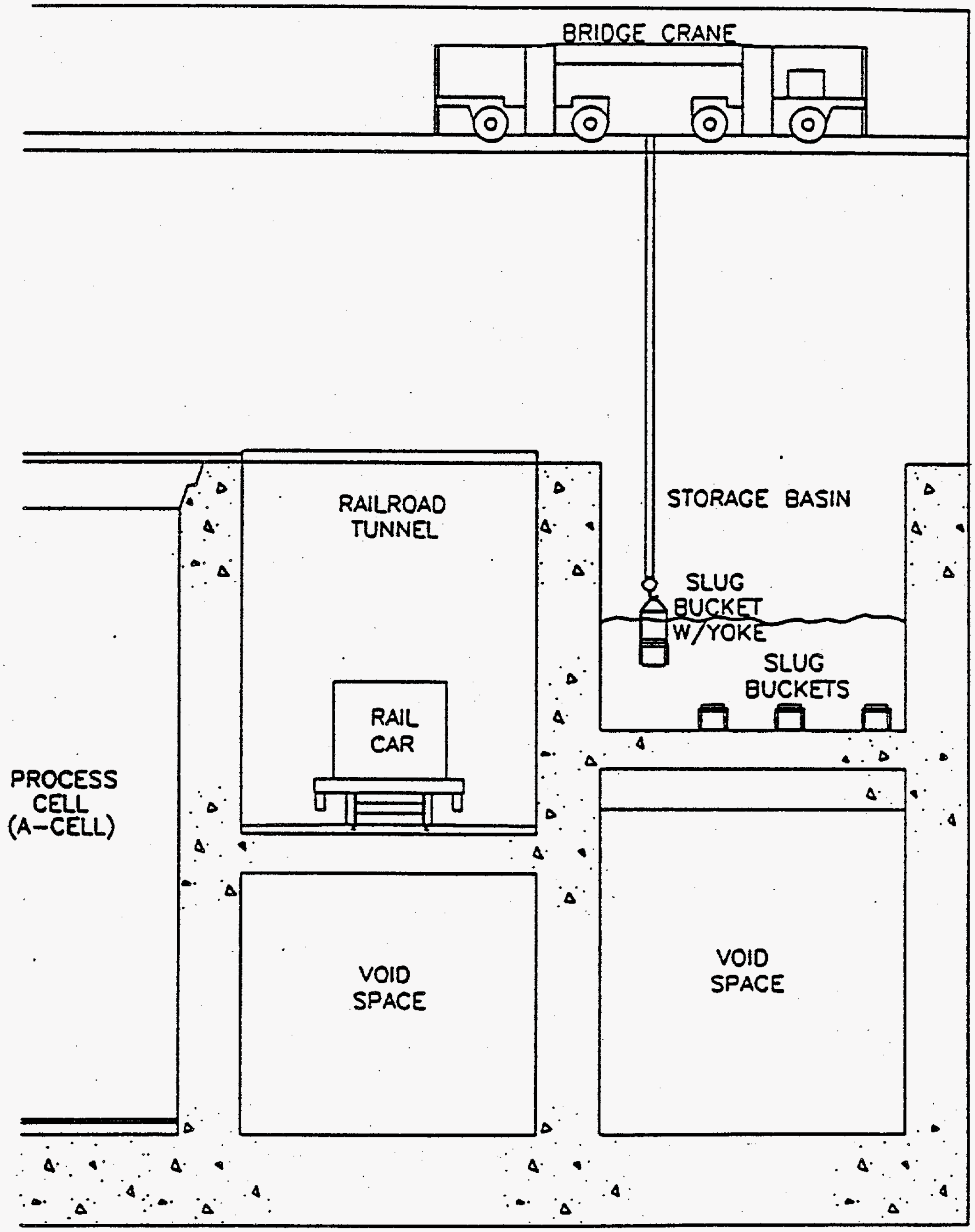


Figure A.8. Single-Pass Reactor Fuel Storage in PUREX Storage Basin

In the absence of oxygen in the water, the hydride can be partly oxidized to form a protected uranium oxide layer [12]. If oxygen is present in the water, the nature of the corrosion product has been reported by Orman and Robertson [17] to change from uranium dioxide [Equation (A.1)] to a yellow hydrated trioxide [Equation (A.4)]. The hydride also can undergo secondary reaction with the dissolved oxygen to yield uranium dioxide [Equation (A.5)].

$$UO_2 + H_2O + 1/2O_2 \rightarrow UO_3.H_2O \tag{A.4}$$

$$UH_3 + 7/4O_2 \rightarrow UO_2 + 3/2H_2O \tag{A.5}$$

Zircaloy cladding also reacts with water, producing oxide and hydride of zirconium.

$$2Zr + 2H_2O \rightarrow ZrO_2 + ZrH_2 + H_2 \tag{A.6}$$

Any of the corrosion reactions listed above could occur preferentially along the grain boundaries or microcracks in the fuel by diffusion of surface-dissociated radicals (e.g., OH and H). The consequence of such a reaction is a friable matrix that can disintegrate to increase the uranium surface area. Swanson observed such an effect [39] when an extensively water-corroded, irradiated N Reactor fuel element section disintegrated into pieces during partial decladding and dissolution operations.

The oxidation products with lower densities (oxides of uranium and uranium hydride) cause swelling of the fuel. Consequently, the corrosion reactions can cause further damage to the original breached cladding, exposing more uranium to the water and thereby increasing the corrosion reaction.

The above corrosion mechanism of a breached fuel element predicts the formation of uranium hydride, oxides of uranium and hydrogen gas from reaction of uranium with water. Further, the cladding-water reaction produces hydrides and oxides of zirconium and hydrogen gas. It is therefore reasonable to assume that these products are present in the basins. However, the depletion reactions (A.3) and (A.5) of uranium hydride and the limited solubility of hydrogen in water suggest that only small quantities may be present. The KW sealed canisters might have some hydrogen gas trapped in the head gas space.

A.5.3 Storage Mode

The rate of corrosion of the uranium (and consequent formation of UH_3) depends on the storage environment. The encapsulation of the SNF into sealed canisters in the KW storage basin may have enhanced the corrosion environment relative to that of the open canisters of KE fuel and thereby accelerated fuel degradation. British and French experiences indicate that an anoxic environment resembling that assumed to exist in the sealed KW canisters could accelerate corrosion and hydriding of the uranium metal in the fuel elements. However, that experience involved bare or magnesium-clad uranium alloy. The electrochemical environment for the corrosion of zircaloy-clad uranium metal could differ significantly from that for bare or magnesium-clad uranium. Previous investigations of N Reactor fuel corrosion [40] revealed no significant accumulation of uranium hydride. The KW encapsulation method involved sealing the fuel elements in a canister filled with water diluted with potassium nitrite and topped with about 6.35-cm space of nitrogen gas. The canisters are vented through traps, which allow gaseous products to escape but prevents exchange of the canister liquid with the basin water.

A large fraction 95% of the Mark IV fuel elements with length codes E and S are long enough to extend above the liquid phase in the sealed canisters. Thus, a high probability of vapor phase oxidation existed in the barrels from the very beginning of sealed storage. The hydrogen generated by the uranium-water reaction will eventually sweep out the nitrogen in the gas phase. The corrosion will also decrease the water volume and increase the gas space. The ratio of water volume to fuel volume in the barrels loaded with seven intact fuel elements of the same type ranges from 1.04 to 2.74. That ratio corresponds to a water to uranium mole ratio of ~0.7 to ~1.9. Thus, from Equation (A.1), ~35 to ~95% of the uranium can be corroded by the water. In the limit all the water in the canister could be consumed by uranium-water reaction leaving metallic uranium, oxides of uranium, uranium hydride and hydrogen gas in the barrel.

Published reports suggest that breached fuel stored in the sealed canisters at the KW basin might be undergoing greater degradation than the fuel stored in the open canisters in the KE basin. Metallic uranium corrosion rates are reported [41] to be more severe in oxygen-depleted water than in aerated water. The fraction of uranium hydride remaining in the corrosion product in oxygen deficient water has been reported to increase [41]. A recent report from Lawrence Livermore National Laboratory [42] concludes that uranium hydride is formed when an iron clad metallic uranium is exposed to moisture.

Thus, in the KE basin aerated water, the corrosion rate may be reduced by the dissolved oxygen and hydride formation in the KE fuel might be minimal. Any trapped molecular hydrogen in the sealed KW canisters would not be expected to increase the uranium hydride fraction or enhance corrosion of the fuel [22].

Appendix B

Regulatory Requirements for Spent Nuclear Fuel Characterization

Appendix B

Regulatory Requirements for Spent Nuclear Fuel Characterization

B.1 Introduction

This review provides brief summaries of the major regulations applicable to the characterization for disposal of Single-Pass Reactor (SPR) and N Reactor fuels in the 105-K basins. The objectives of the characterization project, as discussed in the Characterization Plan, are as follows:

1. **Isolation**: Isolating the fuel in the KE basin from the basin water by repackaging the fuel-containing open canisters, and to prevent further contamination of the KE basin water by the fuel and canister corrosion products.) The isolation phase will require initial visual examination of the canisters and fuel to determine their condition, and to facilitate prediction of problems in handling them. Air emission and personnel exposure will be the major regulated aspects of this step. Thereafter, regulated activities include handling radioactive and contaminated fuel and canisters to move them from the storage racks to the encapsulating bench, to empty them, and to place the fuel (and as much of the sludge as possible) into new canisters; transporting fuel, pieces of fuel, and sludge to the examination and analytical facilities (which may entail transportation on public roads, which in turn requires adherence to such regulations as those governing choice of cask type, cask licensing, detailed declaration of contents, radiation and contamination limits, and labeling); taking samples, performing the tests, and disposing of the waste material (which may be mixed waste) products of the tests.

2. **Interim Storage**: At present, interim storage is assumed to be in a dry-storage facility. For storage in the 100-K Area, only on-site transportation would be needed, but for storage at the Fuel Material Examination Facility (FMEF) or a new site far removed from the basins, the repackaged fuel would have to be transported on public roads, requiring additional regulation as noted above.

   Additional regulations would apply to stabilizing and drying, for such aspects as air emissions, and waste generation, and disposal. Not strictly part of the characterization, but an intrinsic part of the cost of any chosen option, are the regulations on the storage facility (monitored retrievable storage [MRS], independent spent fuel storage installation [ISFSI], or other), such as the specifications to which it must be built. Interim storage will also require monitoring, which includes determining criteria for maintaining safe storage, maintaining fuel stability, and monitoring environmental changes, all of which must meet certain standards.

3. **Fuel Waste Form and Disposal**: For ultimate disposition, a final waste form must be chosen and the spent nuclear fuel (SNF) will have to be processed into that form. The required processes and the resulting waste form suitable for placement in the final repository, as well as the repository itself, all must meet standards and conform with regulations governing long-term public safety.

This review is limited to the regulations that directly impact handling, transporting, analyzing and processing the spent fuel elements, and designing transportation, storage, and disposal containers for the fuel elements and any resulting waste forms. The intent is to ensure that the characterization activities are performed in conformity with all applicable regulations and with least cost and least total radiation and toxic material exposure to the public and the environment in both the short and the long term.

Administrative topics, such as personnel training, record-keeping, licensing (except as pertains to contents of certifiable packages), and reporting, that are applicable at any facility in which radioactive materials are contained or handled, are not discussed here.

Protection of the public and the environment is the basis for a body of public law that has been enacted to address a host of synthetic hazardous materials, emissions, and wastes, both nonnuclear and nuclear. Taken together, these laws form the basis for regulatory domains of the U.S. Nuclear Regulatory Commission (NRC), the U.S. Environmental Protection Agency (EPA), the U.S. Department of Energy (DOE), and other agencies that govern directly or indirectly all the aspects of nuclear materials handling. The major governing laws are listed chronologically:

- *The Atomic Energy Act of 1954* (AEA) specifies the framework for the safety and licensing of nuclear facilities and activities involved in the management of source, special nuclear and byproduct materials. The *Energy Reorganization Act of 1974* identifies the functions of the NRC.

- *The Clean Air Act of 1963* (CAA and amendments) provides regulatory standards for all toxic or hazardous air pollutants under the National Emission Standards for Hazardous Air Pollutants (NESHAPs), for which 40 CFR 61 is the EPA interpretation and DOE Order 5400.5 is the DOE guideline.

- *The National Environmental Policy Act of 1969* (NEPA, Public Law 91-190 and amendments) states that Federal plans, functions, programs, and resources must be used to achieve six general goals, including the assurance of "safe, healthful, productive, and aesthetically and culturally pleasurable surroundings" for all Americans.

- *The Resource Conservation and Recovery Act of 1976* (RCRA, Public Law 94-580 and amendments) regulates waste that meets two criteria: 1) it must be solid, and 2) it must exhibit certain hazardous characteristics (interpreted in 40 CFR 261). RCRA establishes a cradle-to-grave regulatory program for current hazardous waste activities.

- *The Clean Water Act* (*Federal Water Pollution Control Act Amendments of 1977*, FWPCA, and amendments) is concerned with surface water and, most applicable to waste-disposal container and disposal site acceptability, drinking water (and its sources).

- *The Nuclear Waste Policy Act of 1982* (NWPA, Public Law 97-425 and amendments; successor to the AEA) provides environmental protection standards for management and disposal of SNF, high-level waste (HLW), and transuranic waste (TRU) and specifies the requirements for characterization and licensing of a federal HLW repository.

- *The Comprehensive Environmental Response, Compensation, and Liability Act of 1980* (CERCLA, Public Law 96-510) and the *Superfund Amendments and Reauthorization Act of 1986* (SARA) and amendments establish a comprehensive response program for past hazardous substance activities.

- *The Federal Facilities Compliance Act of 1992* mandates the development of plans for treatment capabilities for mixed wastes for each DOE site where mixed wastes are generated or stored.

The primary interpretation of the governing legislation is provided in the Code of Federal Regulations (CFR) by the NRC and the EPA and the U.S. Department of Transportation (DOT). DOE orders are supplementary to the CFR, and are generally written for specific application to DOE tasks.

Note, however, that SNF is in a materials class of its own as far as most of the regulations are concerned. It is not waste inasmuch as the determination has not been made that it has no further usefulness and it is not yet isolated in a disposal system; it is not HLW unless it has been dismantled and reprocessed. Although it contains TRU and fission products, it is not classified with either. The SNF of concern in this Characterization Plan is the property of the DOE; hence, strictly speaking, in its current storage mode the NRC regulations governing commercial fuel do not apply. Although SNF contains radioactivity which can be hazardous, it is not hazardous waste or hazardous material under RCRA or CERCLA; only the hazardous materials portions of a mixture of spent fuel material and hazardous materials are regulated under RCRA and/or CERCLA, although this remains an open subject for interpretation. Nevertheless, guidelines are needed for the activities leading to the interim storage and ultimate disposition of SNF. For that reason, the requirements of the NRC (particularly with respect to MRS), the EPA, and the DOE regulations on TRU, HLW, and low-level waste (LLW) should be met to a common-sense extent. The final form of SNF could then be licensable to the same standards as commercial fuel when a final repository is ready.

The regulations providing the most applicable guidance for SNF are given in Table B.1. Table B.2 in Section B.2 lists the sections of the regulations that are pertinent to each phase of the characterization activities. Pertinent parts of the regulations are summarized in Section B.3. Where the pertinent sections are too extensive to include here, the location of the sections of interest will be cited, so that the reader may consult the regulations directly. Fundamental terms and their definitions, as given in the various regulations, are given in Section B.4.

Major lists of hazardous waste or materials are found in 40 CFR 61 (air pollutants), 40 CFR 261 Subpart B (wastes from manufacturing and other sources) and 49 CFR 172 (materials that must be marked for shipping). Hazardous waste regulations 264, 265, and 268 were written for nonradioactive hazardous materials or wastes. However, they apply to any listed hazardous materials when the hazardous materials are mixed with radioactive waste; in that case, the regulations on radioactive mixed waste apply also.

This review is not exhaustive, in that special-case exemptions and modifying clauses may have escaped notice or have been judged to be not pertinent. Furthermore, the summaries and excerpts that follow should be used as general guidelines only; many qualifying statements have been omitted in the

Table B.1. List of Regulations Reviewed

10 CFR 20	Standards for Protection against Radiation (NRC)
10 CFR 60	Disposal of High-Level Radioactive Wastes in Geologic Repositories (NRC)
10 CFR 61	Licensing Requirements for Land Disposal of Radioactive Waste (NRC)
10 CFR 71	Packaging and Transportation of Radioactive Material (NRC)
10 CFR 72	Licensing Requirements for the Independent Storage of SNF and High-Level Radioactive Waste (NRC)
40 CFR 61	National Emission Standards for Hazardous Air Pollutants
40 CFR 191	Environmental Radiation Protection Standards for Management and Disposal of SNF, HLW, and TRU (EPA)
40 CFR 261	Identification and Listing of Hazardous Waste (EPA)
40 CFR 264	Standards for Owners and Operators of Hazardous Waste Treatment, Storage, and Disposal Facilities (EPA)
40 CFR 265	Interim Status Standards for Owners and Operators of Hazardous Waste Treatment, Storage, and Disposal Facilities (EPA)
40 CFR 268	Land Disposal Restrictions (EPA)
49 CFR 172	Hazardous Materials Table, Special Provisions, Hazardous Materials Communications, Emergency Response Information, and Training Requirements (DOT)
49 CFR 173	Shippers - General Requirements for Shipments and Packaging (DOT)
DOE Order 5400.3	Hazardous and Radioactive Mixed Waste Program
DOE Order 5480.3	Safety Requirements for the Packaging and Transportation of Hazardous Materials, Hazardous Substances, and Hazardous Wastes
DOE Order 5633.3A	Control and Accountability of Nuclear Materials
DOE Order 5820.2A	Radioactive Waste Management

interests of brevity. Detailed evaluation of applicability and interpretation of the fine points of these required regulations can be made more effectively after decisions have been reached on the specific characterization activities.

B.2 Pertinent Regulations

In Table B.2, the number and title of sections of relevant regulations are given and grouped by similar topics where possible; e.g., the several documents treating packaging and transportation are presented in parallel. The pertinent excerpt from the text or, where there is too much detail to include, an indication that there is detail that should be read, is then found in Section B.3, where the regulations are arranged in the same order as they appear in Table B.1.

Table B.2. Listing of Regulations Applicable to SNF Characterization

Radiation Protection Standards	
10 CFR 20	Standards for Protection against Radiation
40 CFR 191	Environmental Radiation Protection Standards for Management and Storage of SNF and Wastes
DOE 5400.3	Hazardous and Radioactive Mixed Waste Program
DOE 5633.3	Control and Accountability of Nuclear Materials
DOE 5820.2A	Radioactive Waste Management
10 CFR 20.1003,1004, and 1005 (Subpart A)	Definitions, Units
40 CFR 191.02,.03,12 (Subpart A, B)	Definitions, Standards (mrem exposure to body)
DOE 5400.3 4. DOE 5400.3 5.	Definitions, Hazardous Waste, Radioactive Waste, Radioactive Mixed Waste, Byproduct Material, Radioactive Material
DOE 5633.3	Definitions, #33 Nuclear materials, 45 Source material, 46 Special Nuclear Material
DOE 5820.2A/A	Definitions, Hazardous Wastes, HLW, LLW, Mixed Waste, SNF, TRU
10 CFR 20.1201-1208 (Subpart C)	Occupational Dose Limits
10 CFR 20.1301-1302 (Subpart D)	Radiation Dose Limits for Individual Members of the Public
40 CFR 191.15 (Subpart B)	Individual Protection Requirements
10 CFR 20.1501-1502 (Subpart F)	Surveys and Monitoring
40 CFR 191.14 (Subpart B)	Assurance Requirements

Table B.2. (contd)

Radiation Protection Standards (contd)	
DOE 5633.3	Chap I - Basic Requirements; Chap II - Materials Accountability; Chap III - Materials Control
DOE 5820.2A Chap I (HLW) 3.b.(3)	Monitoring, Surveillance, and Leak Detection
DOE 5820.2A Chap II (TRU) 3.a.(4)	Waives Analysis if Hazardous to Personnel
10 CFR 20.1801-1802 (Subpart I)	Storage and Control of Licensed Material
40 CFR 191.13, 14 (Subpart B)	Containment Requirements, Assurance Requirements
DOE 5820.2A Chap I (HLW) 3. b. (2)	Storage and Transfer Operations
DOE 5820.2A Chap II (TRU) 3. b. (1)-(4)	TRU Generation and Treatment (and Minimization)
10 CFR 20.2001-2007 (Subpart K)	Waste Disposal
10 CFR 20.2005(1)	Disposal as if Not Radioactive (Limits for Scintillation Counting Liquid for ^{3}H & ^{14}C)
40 CFR 191.11-.17 (Subpart B)	Environmental Standards for Disposal
10 CFR 20 Appendix B (new)	Annual Limits on Intake (ALIs) & Derived Air Concentrations.....
40 CFR 191 Appendix A	Release Limits in Support of Subpart B
40 CFR 191 Appendix B	Guidance for Implementation of Subpart B
Requirements for Transportation	
10 CFR 71	Packaging and Transportation of Radioactive Material
49 CFR 173	Shippers - General Requirements for Shipments and Packagings
DOE 5480.3	Safety Requirements for Packaging and Transportation of Hazardous Materials, Substances, and Waste
10 CFR 71.4 (Subpart A)	Definitions
49 CFR 173.2 (Subpart A)	Hazardous Materials Classes & Index to Hazard Class Definitions
49 CFR 173.403 (Subpart I)	Definitions
DOE 5480.3 5.	Definitions
10 CFR 71.7-10 (Subpart B)	Exemptions
49 CFR 173.3-4 (Subpart A)	Packaging and Exceptions; Exceptions for Small Quantities
49 CFR 173.12 (Subpart A)	Exceptions for Shipment of Waste Materials
DOE 5480.3 7.f	Exemptions (49 CFR 107.103)
10 CFR 71.12-24 (Subpart C)	General Licenses

Table B.2. (contd)

Requirements for Transportation (contd)	
49 CFR 173.448-459 (Subpart I)	General Transportation Requirements; Classification of Fissile Materials; Transportation of Class III Fissile Materials
DOE 5480.3 7.d.	DOE Certificates of Compliance/in Excess of Type A
10 CFR 71.31-39 (Subpart D)	Application for Package Approval
10 CFR 71.41-65 (Subpart E)	Package Approval Standards; General and Specific Package Standards and Radioactivity Limits
49 CFR 173.24-40 (Subpart B)	Preparation of Hazardous Materials for Transportation
49 CFR 173.411-419 (Subpart I)	Radioactive Materials - Design Requirements; Authorized Type A&B Packages; Packaging Fissile, Pyrophoric, Oxidizing Materials..
49 CFR 173.441-443 (Subpart I)	Radiation Level Limitations, Thermal Limitations, Contamination Control
DOE 5480.3 7.(c)	Package Standards for Radioactive Materials in Amounts Greater than Type A quantities
DOE 5480.3 8.(a)	General Standards for All Packaging
10 CFR 71.63 (Subpart C)	Special Requirements for Plutonium Shipments
DOE 5480.3 7.b.	Special Packaging for Plutonium Pu-bearing Wastes
10 CFR 71.71-77 (Subpart F)	Package and Special Form Tests - Normal Conditions of Transport and Hypothetical Conditions
DOE 5480.3	11. Normal Conditions and 12. Hypothetical Accident Conditions
10 CFR 71.83-87 (Subpart G)	Assumptions as to unknown properties; Preliminary Determinations; Routine Determinations
49 CFR 173.433(5)(6) (Subpart G)	Assumptions if Radionuclides Known/Unknown
49 CFR 173.461-463 (Subpart I)	Compliance, Testing, Requirements for NRC and DOT approval,...
DOE 5480.3 10.(b)(d)	Assumptions as to Unknown Properties; Routine Determinations
10 CFR 71.97 (Subpart G)	Advance Notice of Shipment of Nuclear Waste
49 CFR 173.476-478 (Subpart I)	Approval of Special Form and for Export Shipments; Notification to Authorities
DOE 5480.3 10.(g)	Notification Procedures for Shipment and Nonreceipt of Radioactive Materials
10 CFR 71.101-107 (Subpart H)	Quality Assurance and Package Design Control
49 CFR 173,474-475 (Subpart I)	Quality Control for Packaging and Prior to Shipment
DOE 5480.3 9.(a)(b)	Quality Assurance Procedures for the Fabrication, Assembly, and Testing of Offsite Shipping Containers

Table B.2. (contd)

List of Hazardous Materials	
40 CFR 61	National Emission Standards for Hazardous Air Pollutants
40 CFR 261	Identification and Listing of Hazardous Waste
49 CFR 172	Hazardous Materials Table
DOE 5400.3	Hazardous and Radioactive Mixed Waste Program
40 CFR 261.3 (Subpart A)	Definition of Hazardous Waste
DOE 5400.3 (4a) and (4c)	Definitions; Hazardous Waste and RCRA/AEA Inconsistencies; Radioactive Waste; Radioactive Mixed Waste
40 CFR 261.10,11 (Subpart B)	Criteria for Identifying and Listing Hazardous Waste
40 CFR 61.01 (Subpart A)	List of Pollutants....
40 CFR 261.30-.35 (Subpart D)	Lists of Hazardous Wastes, from Specific and Non-Specific Sources
49 CFR 172.101,102 (Subpart B)	Table of Hazardous Materials and Special Provisions
49 172.403(a)&(g) (Subpart E)	Need for Radionuclide Content List on Label
Requirements for Independent Storage in MRS, ISFSI, Geologic, and Land Surface Storage/Disposal Sites	
10 CFR 72	Licensing Requirements/Independent Storage of SNF and High Level Radioactive Waste
10 CFR 72.2 (Subpart A)	Scope, ISFSI or MRS
10 CFR 72.3 (Subpart A)	Definitions (HL Radioactive W, ISFSI, MRS, Byproduct Material, Special Nuclear Material, Source Material, SNF
10 CFR 72.6 (Subpart A)	License required; Types of Licenses
10 CFR 72.16-34 (Subpart B)	License Application, Form and Contents
10 CFR 72.40-62 (Subpart C)	Issuance and Conditions of License
10 CFR 72.70-86 (Subpart D)	Records, Reports, Inspections, and Enforcement
10 CFR 72.90-108 (Subpart E)	Siting Evaluation Factors
10 CFR 72.104 (Subpart E)	Criteria/Radioactive Materials in Effluents and Direct Radiation from ISFSI/MRS
10 CFR 72.120-130 (Subpart F)	General Design Criteria for Storage Installation
10 CFR 72.140-220 (Subparts G-K)	Quality Assurance, Administrative Requirements
10 CFR 72.230-240 (Subpart L)	Approval of Spent Fuel Storage Casks
40 CFR 61	National Emission Standards for Hazardous Air Pollutants
40 CFR 61.01 (Subpart A)	List of Pollutants and Applicability

Table B.2. (contd)

Requirements for Independent Storage in MRS, ISFSI, Geologic, and Land Surface Storage/Disposal Sites (contd)	
40 CFR 61.92,.93 (Subpart H)	Standard Emission Limit; Emission Monitoring and Test Procedures
40 CFR 61.100,101 (Subpart I)	Emissions Other than Radon from DOE Facilities; Applicability, Definitions, Emission Determination; Exemptions
40 CFR 61.102 (Subpart I)	Standard Emission Limit, (same as 61.92 but includes iodine)
40 CFR 61.103,107 (Subpart I)	Dose Compliance Using EPA or DOE Models or Measurements (61.107 and Appendix E)
40 CFR 61.192 (Subpart Q)	Emission from Storage and Disposal Facilities (see also 61.90)
40 CFR 61 Appendix D	Methods for Estimating Radionuclide Emissions (see also 40 CFR 61.107)
40 CFR 264	Standards for Owners and Operators of Hazardous Waste Treatment, Storage, and Disposal Facilities
40 CFR 265	Interim Status Standards for Owners and Operators of Hazardous Waste Treatment, Storage, and Disposal Facilities
40 CFR 264.13	(same as 265.13) Requires full and updated analyses
40 CFR 265.13	General Waste Analysis
40 CFR 264.17	(same as 265.17) Adds the need to document the analyses
40 CFR 265.17	General Requirements for Ignitable, Reactive, or Incompatible Wastes
40 CFR 264.170-178	See 264.175 details on containment; 264.178 on Closure
40 CFR 265.170-177 (Subpart I)	Use and Management of Containers
40 CFR 264.220-231	See 264.220 detail on liners; 264.221 information on mono-fill deleted; no trial tests -.225; 264.226 monitoring different adds 264.229 and .230 special requirements for ignitable and reactive waste and incompatible wastes.
40 CFR 265.220-229 (Subpart K)	Surface Impoundments
40 CFR 264.1030	Air Emission Standards for Process Vents
40 CFR 265.1030	Air Emission Standards for Process Vents
10 CFR 60	Disposal of High Level Radioactive Wastes in Geologic Repositories
10 CFR 60.43	License Specification
10 CFR 60.102 (e)	Isolation of Waste (the Containment Period)
10 CFR 60.111 (b)	Retrievability of Waste
10 CFR 60.113 (a)(1)	Engineered Barrier System

Table B.2. (contd)

Requirements for Independent Storage in MRS, ISFSI, Geologic, and Land Surface Storage/Disposal Sites (contd)	
10 CFR 60.131 (b)(7)	General Design Criteria for Geologic Repository Operations (Criticality Control)
10 CFR 60.135	Criteria for the Waste Package and its Components (HLW Design, Specific Criteria)
10 CFR 61	Licensing Requirements for Land Disposal of Radioactive Waste
10 CFR 61.41	Protection of the General Public from Releases of Radioactivity (Limits)
10 CFR 61.52	Land Disposal Facility Operation and Disposal Site Closure (Class C Depth)
10 CFR 61.55	Waste Classification
10 CFR 61.56	Waste Characteristics
40 CFR 268	Land Disposal Restrictions
40 CFR 268.1(a)(b) & 268.30-35	Restrictions/Materials Disposed in Landfill
40 CFR 268.2 (c)	Land Disposal Regulations Cover Vaults and Bunkers
40 CFR 268.40-45	Standards for Treatment Expressed as Concentration, etc.

B.3 Pertinent Excerpts from the Regulations

The activities required for the characterization of the SNF in the 105-K basins before either short- or long-term storage will consist of

- removing representative samples of the SNF and sludge from the basins
- transporting the samples to one or more of several laboratories in the 300 Area and possibly the 200 Area for specific tests
- subjecting judiciously chosen samples to various nondestructive and destructive tests in the laboratories
- returning the samples to Westinghouse Hanford Company (WHC) for storage or disposition.

The sample characterization results that will be used to evaluate pathways to ultimate disposition of the 105-K basins SNF include

- isolation of the 105-K East SNF from the accessible environment
- extended interim storage for up to 40 years in a facility away from the Columbia River

- developing or adapting and implementing one or more stabilization processes in preparation for ultimate disposal

- storing the stabilized products for up to 40 years (these final products may be SNF or may be processed products, including TRU, liquid and solid process effluents, and various nuclides of beneficial use)

- eventually providing ultimate disposition of the material in a geologic repository, with or without additional processing

- carrying out all these activities with maximum protection for the workers, the public and the environment.

Selected excerpts covering these requirements follow. An attempt to group the regulations by topic was made in Table B.2. For simplicity in reading and finding the sources of specific regulations, summaries of the pertinent excerpts will be listed in the same numerical order as in Table B.1.

B.3.1 Transportation

10 CFR 71 - Packaging and Transportation of Radioactive Material - states that no licensed material (originally, commercial nuclear material) may be moved without authorization from the NRC, for which the requirements are set forth. Part 71 is the key authority on requirements for licensing shipments, with definitions of categories that determine package types and license requirements. These requirements include details of categories of fissile material and levels of radioactivity of the contents, with extensive instructions on determining the permissible levels for each category of package (requiring adequate knowledge of the composition of the contents); acceptable radiation levels outside the package; and packaging requirements to prevent leakage under a wide variety of scenarios, for which prescribed tests must be passed. The tables of radioactivity limits for most radionuclides for shipment under the "Type A" category are given in Appendix A of Part 71.

10 CFR 71.3: A license to ship radioactive materials is required, with few exceptions (71.3). The details and the exemptions are covered in this regulation and in 49 CFR 173.

10 CFR 71.4 - Definitions: See Section B.4 for definitions from all regulations, where they can be viewed together and compared. The following are definitions basic to an understanding of the regulations on packaging and transporting.

Type A Quantity - A quantity of radioactive material, the aggregate radioactivity of which does not exceed A_1 for special form radioactive materials or A_2 for normal form radioactive material, where A_1 and A_2 are given in Appendix A of 10 CFR 71 or may be determined by procedures described in that Appendix.

Type B Quantity - A quantity of radioactivity greater than a Type A quantity.

Special form radioactive material - Radioactive material which satisfies certain conditions. Note: Intact spent fuel probably satisfies those conditions.

10 CFR 71.7 - Specific Exemptions: On application of any interested person or on its own initiative, the NRC may grant an exemption from the requirements of the regulations in this part that it determines is authorized by law and will not endanger life or property or the common defense and security.

10 CFR 71.10 - Exemption for Low Level Materials:

(a) A licensee is exempt from all requirements of this part with respect to shipment or carriage of a package containing radioactive material having a specific activity not greater than 0.002 microcurie/gram.

(b) A licensee is exempt from all requirements of this part, other than Sections 71.5 and 71.88, with respect to shipment or carriage of the following packages:

(1) a package containing no more than a Type A quantity of radioactive material if the package contains no fissile material or if the fissile material exemption standards of Section 71.53 are satisfied

(2) a package transported between locations within the United States that contains only americium or plutonium in special form with an aggregate radioactivity not to exceed 20 curies, if the package contains no fissile material or if the fissile material exemption standards of Section 71.53 are satisfied.

10 CFR 71.12-24 - General Licenses: Conditions for general licenses and for exemptions are given. Tables are given that show permissible masses allowable under each category of fissile material. The reader is referred to the document itself for details applicable to a specific case.

10 CFR 71.31-39 - Application for Package Approval: Of particular importance is the following requirement. The application must include a description of the proposed package in sufficient detail to identify the package accurately and provide a sufficient basis for evaluation of the package. The description must include, with respect to the contents of the package:

(1) identification and maximum radioactivity of radioactive constituents
(2) identification and maximum quantities of fissile constituents
(3) chemical and physical form
(4) extent of reflection, the amount and identity of nonfissile materials used as neutron absorbers or moderators, and the atomic ratio of moderator to fissile constituents
(5) maximum normal operating pressure
(6) maximum weight
(7) maximum amount of decay heat
(8) identification and volumes of any coolants.

10 CFR 71.41-65 (Subpart E) - Package Approval Standards: Specific category standards for package compliance, including the amount of allowable external radiation are given.

10 CFR 71.47 - External Radiation Standards for all Packages: A package must be designed and prepared for shipment so that the radiation level does not exceed 200 millirem per hour at any point on the external surface of the package and the transport index does not exceed 10.

10 CFR 71.65 - Special Requirements for Plutonium Shipments: Plutonium in excess of 20 curies per package must be shipped as a solid. Solid plutonium in the following forms is exempt from the requirement of this paragraph:

(1) reactor fuel elements
(2) metal or metal alloy
(3) other plutonium bearing solids that the NRC determines should be exempt from the requirements of this section.

Extensive additional details are provided for plutonium shipments.

10 CFR 71.71-77 - Package and Special Form Tests

10 CFR 71.71 - Normal Conditions of Transport

10 CFR 71.73 - Hypothetical Accident Conditions

10 CFR 71.75 - Qualification of Special Form Radioactive Material

10 CFR 71.77 - Tests for Special Form Radioactive Material

10 CFR 71.83 - Assumptions as to Unknown Properties: When the isotopic abundance, mass, concentration, degree of irradiation, degree of moderation, or other pertinent property of fissile material in any package is not known, the licensee shall package the fissile material as if the unknown properties have credible values that will cause the maximum nuclear reactivity.

10 CFR 71.85 - Preliminary Determinations: Where the maximum normal operating pressure will exceed 34.3 kilopascal (5 psi) gauge, the licensee shall test the containment system at an internal pressure at least 50% higher than the maximum normal operating pressure to verify the capability of that system to maintain its structural integrity at that pressure.

10 CFR 71.87 - Routine Determinations

10 CFR 71.97 - Advance Notification of Shipment of Nuclear Waste: Written notification of the intent to ship licensed material to, through, or across the boundary of the state must be sent to the governor of the state (or designee), postmarked at least 7 days before the 7-day period in which the shipment is expected to take place. Notable among the types of information to be provided is the description of the material, as required by DOT document 49 CFR 172.202 and .203 (Subpart C, Description of Hazardous Materials on Shipping Papers). See 71.97(b) for radioactivity limits covered by this rule. This item lists all the information needed prior to the shipment of nuclear waste and the conditions that are exempted from some of the requirements.

Table B.3. Removable External Radioactive Contamination Wipe Limits

Contaminant	Maximum Permissible (mCi/cm^2)	Limits (dpm/cm^2)
Beta-gamma emitting radionuclides	10 - 5	22
All radionuclides with half-lives less than ten days	10 - 5	22
Natural uranium and thorium	10 - 5	22
Uranium 235 and 238; thorium 232, 230 and 228	10 - 5	22
All other alpha-emitting radionuclides	10 - 6	2.2

10 CFR 71.101-106 - Quality Assurance in Records, Procurement, etc.

10 CFR 71.107 - Package Design Control

49 CFR 173 - Shippers - General Requirements for Shipments and Packaging

49 CFR 173 Subpart A 173.2 - Hazardous Materials Classes and Index to Hazard Class Definitions: The hazard class of a hazardous material is indicated either by its class (or division) number, its class name, or by the letters ORM-D. The class numbers, division numbers, class or division names, and sections of this subchapter that contain definitions for classifying hazardous materials, including forbidden materials, are given. Radioactive material is Class 7 (40 CFR 173.403).

49 CFR 173 Subpart A 173.2a - Classification of a Material Having More Than One Hazard: In general, a material having more than one hazard shall be classed according to the highest applicable hazard. Exceptions include certain highly hazardous materials (such as explosives, peroxides, highly infectious materials, and all but very small quantities of radioactive materials), which must be regulated according to both hazards, depending on the specific combination and type of hazards. The reader is referred to the document for the complete list of classes. Radioactive materials, in all but very small quantities, are Class 7.

49 CFR 173 Subpart A 173.3 - Packaging and Exceptions: The packaging of hazardous material for transportation by air, rail, highway, or water must be as specified in this section. Requirements for salvage drums are given in detail.

49 CFR 173 Subpart A 173.4 - Exceptions for Small Quantities: Details are given for conditions under which quantities of 1 ounce or less of hazardous material are not subject to the other requirements of this chapter.

49 CFR 173.21 - Forbidden Materials and Packages: Details on materials that may not be shipped.

49 CFR 173.24 - General Requirements for Packagings and Packages: Details on limits on releases, types of packaging, conformance to specifications of the DOT and the United Nations.

49 CFR 173 Subpart I 173.403 - Definitions: Definitions are compiled in Section B.4.

49 CFR 173 Subpart I 173.411 - General Design Requirements and .412 - Additional Design Requirements for Type A Packages: Extensive details of requirements on all aspects of the design of transportation containers for radioactive material.

49 CFR 173 Subpart I 173.413 - Design Requirements for Type B Packages: See 10 CFR 71.

49 CFR 173 Subpart I 173.415 - Authorized Type A Packages and .416 - Authorized Type B Packages: Package capabilities to withstand heat, etc.; content limits; marking; etc.

49 CFR 173 Subpart I 173.417 - Authorized Packaging - Fissile Material: Requirements, including tables on:

- fissile material content and transport index for specification 6J or 17H packages
- allowable content of uranium hexafluoride (UF6) heels in a Specification 7A cylinder
- authorized contents in kg and conditions for specification 6L packages
- authorized contents for specification 6M packages

49 CFR 173 Subpart I 173.418 - Authorized Packaging-Pyrophoric Radioactive Materials

49 CFR 173 Subpart I 173.419 - Authorized Packaging-Oxidizing Radioactive Materials

49 CFR 173 Subpart I 173.421 - Limited Quantities of Radioactive Materials: Radioactive materials whose activity per package does not exceed specified limits are excepted from the specification packaging, shipping paper and certification, marking, and labeling requirements of this subchapter if they meet a set of conditions of packaging, radioactivity level, nonfixed contamination, etc., as listed in this subpart, for example:

(1) the radiation level at any point on the external surface of the package does not exceed 0.5 millirem per hour

(2) The nonfixed (removable) radioactive surface contamination on the external surface of the package does not exceed the limits specified in 173.443.

49 CFR 173 Subpart H 173.423 - Table of Activity Limits-Excepted Quantities and Articles: The limits applicable to instruments, articles, and limited quantities subject to exceptions under 173.421 and 173.422 are shown in Table B.4.

49 CFR 173 Subpart H 173.433 - Requirements for Determination of A1 and A2 Values for Radionuclides: The procedures for determining the value of A1 and A2 for known radionuclides and various parameters are given. In addition, for unknown radionuclides:

(1) When the identity of each radionuclide is known but the individual activities of the radionuclides are not known, the most restrictive value of A1 or A2 applicable to any one of the radionuclides present is the applicable value.

Table B.4. Activity Limits for Limited Quantities, Instruments, and Articles

Nature of Contents	Instrument and Articles		Material Package Limits
	Instrument and article limits[(a)]	Package limits	
Solids:			
Special form	10-2A1	A1	10-3A1
Other forms	10-2A2	A2	10-3A2
Liquids:			
Tritiated water:			
<0.1 Ci/L			1000 Ci
0.1 - 1.0 Ci/L			1000 Ci
>1.0 Ci/L	10-3A2		1 Ci
Other liquids		10-1A2	10-4A2
Gases:			
Tritium[(b)]	20 Ci	200 Ci	20 Ci
Special form	10-3A1	10-2A1	10-3A1
Other forms	10-3A2	10-2A2	10-3A2

(a) For mixture of radionuclides see 173.433(b).
(b) These values also apply to tritium in activated luminous paint and tritium adsorbed on solid carriers.

(2) When the identity of the radionuclides is not known, the value of A1 is 2 curies and the value of A2 is 0.002 curies. However, if alpha emitters are known to be absent, the value of A2 is 0.4 curies.

49 CFR 173 Subpart H 173.441 - Radiation Level Limitations: Each package of radioactive material offered for shipment shall be designed and prepared so that the radiation level does not exceed:

(1) 200 millirem per hour (2 millisievert per hour) on the external surface of the package unless the following conditions are met, in which case the limit is 1000 millirem per hour (10 millisievert per hour)

(2) 10 millirem per hour (0.1 millisievert per hour) at any point 2 m (6.6 ft) from the outer lateral surfaces of the vehicle (excluding the top and underside of the vehicle)

(3) 2 millirem per hour (0.02 millisievert per hour) in any normally occupied space, except that this provision does not apply to private carriers if exposed personnel under their control wear radiation dosimetry devices and operate under provisions of a state or federally regulated radiation protection program.

49 CFR 173 Subpart H 173.442 - Thermal Limitations: Each package of radioactive material shall be designed, constructed, and loaded so that:

(1) The heat generated within the package because of the radioactive contents will not, at any time during transportation, affect the integrity of the package under conditions normally incident to transportation

(2) The temperature of the accessible external surfaces of the loaded package will not, assuming still air in the shade at an ambient temperature of 38°C (100°F), exceed either:

- 50°C (122°F) for other than an exclusive use shipment
- 82°C (180°F) for an exclusive use shipment.

49 CFR 173 Subpart H 173.443 - Contamination Control:

(1) The level of nonfixed (removable) radioactive contamination on the external surfaces of each package offered for shipment shall be kept as low as practicable. Details are given as to the procedure to use and a table is given for removable external radioactive contamination-wipe limits.

Readers may find various useful details in the following sections:

49 CFR 173 Subpart H 173.461 - Demonstration of Compliance with Tests

49 CFR 173 Subpart H 173.462 - Preparation of Specimens for Testing

49 CFR 173 Subpart H 173.463 - Packaging and Shielding-Testing for Integrity

49 CFR 173 Subpart H 173.465 - Type A Packaging Tests

49 CFR 173 Subpart H 173.466 - Additional Tests for Type A Packagings Designed for Liquids and Gases

49 CFR 173 Subpart H 173.467 - Tests for Demonstrating the Ability of Type B and Fissile Radioactive Materials Packagings to Withstand Accident Conditions in Transportation

49 CFR 173 Subpart H 173.469 - Tests for Special Form Radioactive Materials

49 CFR 173 Subpart H 173.474 - Quality Control for Construction of Packaging

49 CFR 173 Subpart H 173.475 - Quality Control Requirements Prior to Each Shipment of Radioactive Materials

49 CFR 173 Subpart H 173.476 - Approval of Special Form Radioactive Materials

49 CFR 173 Subpart H 173.478 - Notification to Competent Authorities for Export Shipments

For details of the procedures described in Subparts 173.461 through 173.476, consult the documents.

DOE 5480.3 - Safety Requirements for Packaging and Transportation of Hazardous Materials, Hazardous Substances, and Hazardous Waste - The requirements in DOE 5480.3 will be implemented by the organization managing transportation and the acquisition of casks and other containers. The only information presented in this excerpt is that which may impact the samples being shipped in conjunction with the characterization activities. Therefore, the details are included here.

DOE 5480.3 7. - Requirements:

7b. Special packaging requirements apply to solid plutonium and plutonium-bearing wastes (in addition to other packaging requirements in this order) for quantities greater than A2 quantities for normal form or greater than A1 quantities for special form.

(3) Plutonium packaging requirements for any surface mode of transportation.

(a) Plutonium in excess of 20 curies per package must be shipped as a solid and meet special packaging requirements (refer to (b) of this item for details and special tests).

(4) Solid plutonium in excess of 20 curies per package in the following forms is not subject to the requirements of paragraph 7b(3):

(a) reactor fuel elements
(b) metal or metal alloy
(c) special Form materials
(d) other forms of plutonium-bearing materials, e.g., wastes or contaminated equipment, as approved by the office of Operational Safety.

7c. Package standards for radioactive materials in amounts greater than Type A quantities. These quantities depend on the nuclide and are given in the table in the appendix to this regulation. (Refer to 10 CFR 71.37; 10 CFR 71.121; 10 CFR 71.137 for QA requirements.)

7d. Applications for a DOT exemption shall be prepared in accordance with 49 CFR 107.103, and shall be forwarded through the Safety Engineering and Analysis Division to the DOT.

7f. Exemption. Packages that do not meet the standards in the DOT Hazardous Materials Regulations and that do not qualify for shipment under the National Security Exemption may be shipped only under the provisions of an exemption issued by the DOT, or on public vehicles or aircraft if approved.

DOE 5480.3 8. - Packaging:

a. General standards for all packaging.

(1) Reference 10 CFR 71.
(2) For determination of transport indexes for packaging, see paragraph 5 of this order.
(3) Excluded from the standards, testing requirements, packaging certification, and documentation described in this order are low specific activity shipments consigned as exclusive use. The requirements for this type of shipment are contained in 49 CFR 173.425.
(4) Type A packaging requirements are contained in 49 CFR 173.411 and 412.

b. For structural standards for type B packaging, see 10 CFR 71.

(2) External Pressure. Packaging shall be adequate to assure that the containment vessel will suffer no loss of contents if subjected to an external pressure of 25 pounds per square inch gauge.

c. The criticality standards for fissile material packages require that if water leaks in or the contents become liquid, the contents will not reach criticality under the worst reasonable postulated conditions.

d. The evaluation of a single package requires that tests be made on a sample or scale model to simulate normal and reasonably postulated accident transport conditions.

e. Standards for normal conditions of transport for a single package. The major items of concern to the Characterization Project are that:

(c) there will be no mixture of gases or vapors in the package that could, through any combination of pressure or an explosion, significantly reduce the effectiveness of the package

(d) radioactive contamination of the liquid or gaseous primary coolant will not exceed 10-7 curies of activity or Group radionuclides per milliliter, 5 x 10-6 curies of activity of Group II radionuclides per milliliter, and 3 x 10-4 curies of activity of Group III and Group IV radionuclide per milliliter.

(2) A package used for the shipment of fissile material shall be designed and constructed, and its contents so limited, that under normal conditions of transport specified in paragraph 11, considered individually:

(a) the package will be subcritical.

(b) the geometric form of the package contents would not be substantially altered.

(c) there will be no leakage of water into the containment vessel. This requirements need not be met if, in the evaluation of undamaged packages under paragraph 8h, 8i, or 8j below, it has been assumed that moderation is present to such an extent as to cause maximum reactivity consistent which the chemical and physical form of the material.

(d) there will be no substantial reduction in the effectiveness of the packaging, including:

i) reduction by more than 5 percent in the total effective volume of the packaging of which nuclear safety is assessed.

ii) reducing by more than 5 percent in the effective spacing on which nuclear safety is assessed between the center of the containment vessel and the outer surface of the packaging.

iii) occurrence of any aperture in the outer surface of the packaging large enough to permit the entry of a 4-inch cube.

(3) A package used for the shipment of more than type A quantity of radioactive material shall be so designed and constructed, and its contents so limited, that under normal conditions of transport specified in paragraph 11, considered individually, the containment vessel would not be vented directly to the atmosphere.

f. Standards for hypothetical accident conditions for a single package.

g. Criticality standards for packaging fissile materials. The fissile characteristics of each package and array of packages shall be evaluated for criticality and the assignment of the proper fissile Class I, II, or III (10 CFR 71).

DOE 5480.3 9. - Quality Assurance Procedures for the Fabrication, Assembly, and Testing of Offsite Shipping Containers.

DOE 5480.3 10. - Operating procedures

b. Assumptions as to Unknown Properties. When the isotopic abundance, mass, concentration, degree of irradiation, degree of moderation, or other pertinent property of fissile material in any package is not known, the shipper shall package the fissile material as if the unknown properties have such credible values as will cause the maximum nuclear reactivity. Any special instructions needed to safely open the package are to be made available to the consignee.

c. and d. Outline of required package inspection before use.

g. Notification Procedures for Shipment and Nonreceipt of Radioactive Materials. To reduce to a minimum the number of shipments that must ultimately be considered lost, the following procedures shall be implemented:

(3) packaging shall be marked conspicuously and durably with its model number. Prior to applying the model number, an inspection shall be made to determine that the packaging has been fabricated in accordance with the approved design.

DOE 5480.3 11. - Normal Conditions of Transport: Each of the following normal conditions of transport is to be applied separately to determine its effect on a package. The reader is referred to the 10 CFR 72 document for extensive details of normal and accident condition package tests.

DOE 5480.3 12. - Hypothetical Accident Conditions of Transport: See document for details of required tests.

DOE 5480.3 13. - A1 and A2 Values for Radionuclides: These values are found in 49 CFR 173.435.

DOE 5480.3 14. - Tests for Special Form Material: See document if details are needed.

B.3.2 Treatment and Storage

40 CFR 264 - Standards for Owners and Operators of Hazardous Waste Treatment Storage and Disposal Facilities - This regulation, 40 CFR 264, differs from 40 CFR 265 primarily in that the latter covers the owner/operator while waiting for a RCRA permit or until certification of final closure or, if the facility is subject to post-closure requirements, until post-closure responsibilities are fulfilled (265.1). It applies to all owners/operators of facilities that treat, store, or dispose of hazardous wastes referred to in 40 CFR 268, Land Disposal Restrictions, after receiving the appropriate permits. Most of the sections of 40 CFR 264 are similar to, and some are identical to 40 CFR 265.

40 CFR 265 - Interim Status Standards for Owners and Operators Hazardous Waste Treatment, Storage, and Disposal Facilities

40 CFR 265 (Subpart A) - General: This regulation establishes national standards for handling an interim storage of hazardous (nonradioactive) waste. The fine points of applicability of 40 CFR 265 are detailed in this first subpart.

40 CFR 264.340 and 265.340 - Applicability: Check the list of hazardous waste in 40 CFR 261 to ensure compliance with any specific conditions. The following are fundamental requirements of this regulation:

40 CFR 264.13 and 265.13 - General Waste Analysis: "Before an owner treats, stores, or disposes of any hazardous wastes, or nonhazardous wastes if applicable under 40 CFR 264.113(d), he must obtain a detailed chemical and physical analysis of a representative sample, which must contain all the information required to treat, store, or dispose of the wastes in accordance with this part and part 268 (*Land Disposal Restrictions*) of this chapter." The information may be obtained from existing published or documented data on waste generated by the same or similar processes.

40 CFR 264.17 and 265.17 - General Requirements for Ignitable, Reactive, or Incompatible Wastes: Ignitable or reactive material is implied to be that which ignites easily from open flame, hot surfaces, frictional heat, sparks, spontaneous ignition from chemical reactions. Graphite is not ignitable, even though it can be made to burn. Ignitable or reactive material must not be placed in a landfill or burned unless special precautions are taken to avoid conditions that would cause it to ignite, (precautions given in 40 CFR 268). For requirements governing these characteristics in waste piles see 265.312 and 264.256; in surface impoundments see 265.229 and 264.229. Also, see 10 CFR 60.135 with respect to combustible material, which appears to imply that burning that may have to be sustained by an outside heat source, may still be included as hazardous under the ignitability category.

40 CFR 264.94 - Concentration Limits: The concentration of hazardous constituents in the ground water must be below the background level given in Table 1 of this section, if the level in the table is below the background level. Values are given for the following elements: arsenic, barium, cadmium, chromium, lead, mercury, selenium, silver, endrin, lindane, methoxychlor, toxaphane, 2,4-d (2,4-dichlorophenoxyacetic, and 2,4,5-tp silvex (2,4,5-trichlorophenoxypropionic acid).

40 CFR 264.97 and 265.91 - General Ground-Water Monitoring Requirements: A ground-water monitoring system must be put into operation, including a sufficient number of wells, situated at appropriate locations and distances apart, to provide representative samples of the ground-water content and allow for early detection of contamination. Appropriate records, statistical methods, and analysis must be applied to control contaminants and protect human health and the environment, 264.98 Detection monitoring program (details of ground-water detection requirements), and 40 CFR 264.99 (compliance monitoring program and requirements).

40 CFR 264.170-172 and 265.170-.174 (Subpart I) - Use and Management of Containers: Containers must be in good condition, must be inspected weekly, and must not interact with the contents or with the residue in a previously used and unclean container (compatibility requirements). Containers holding ignitable or reactive waste must be located at least 15 m (50 ft) from the facility's property line.

40 CFR 264.177 & 265.177 - Special Requirements for Incompatible Waste:

(a) Incompatible wastes (and/or materials) must not be placed in the same container unless 40 CFR 264.17 or 265.17 are complied with.

(b) Hazardous waste must not be placed in an unwashed container that previous held an incompatible waste or material.

(c) A storage container holding a hazardous waste that is incompatible with any waste or other materials stored nearby in other containers, piles, open tanks, or surface impoundments must be separated from the other materials or protected from them by means of a dike, berm, wall, or other device.

40 CFR 264 and 265 (Subpart K) - Surface Impoundments: This subpart appears to apply to MRS installations and drywells. (See 265.220-229 and 264.220-231). There are some differences between 265 and 264. 40 CFR 265.225 requires additional analyses if the surface impoundment is used for processing waste. Each new or replacement impoundment must be equipped with at least two liners with leachate collection system between the liners. Leakage limit is specified and a plan for response if the leakage rate is exceeded must be in place before waste can be accepted. Special requirements are stipulated for wastes that are ignitable, reactive or incompatible.

40 CFR 264 and 265 (Subpart AA) - Air Emission Standards for Process Vents: This section applies to process vents associated with distillation, fractionation, thin-film evaporation, solvent extraction, air or steam stripping operations that manage hazardous wastes with organic concentrations of at least 10-ppmw, if these operations are conducted in, among other things, hazardous waste recycling units that are located on hazardous waste management facilities. Detailed requirements for emission limits for a variety of emission control devices (open- and closed-vent) are listed. Auxiliary software (which is not included here) is used for heating value and maximum allowed velocity of the gases. Requirements are given for monitoring the input to the process.

10 CFR 72 - Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High Level Radioactive Waste - Applies to both wet and dry modes of storage in ISFSI and MRS facilities.

This NRC regulation establishes requirements, procedures, and criteria for the issuance of licenses to DOE to receive, transfer, package, and possess power reactor spent fuel, high-level radioactive waste, and other radioactive materials associated with spent fuel and high-level radioactive waste storage in an MRS or an ISFSI installation.

10 CFR 72 Subpart A 72.2 - Scope:

(c) The requirements of this regulation are applicable, as appropriate, to both wet and dry modes of storage of 1) spent fuel in an ISFSI and 2) spent fuel and solid high-level radioactive waste in an MRS.

(d) Licenses covering the storage of spent fuel in an existing spent fuel storage installation shall be issued in accordance with the requirements of this part as stated in 72.40, as applicable.

10 CFR 72 Subpart A 72.3 - Definitions: See Glossary.

10 CFR 72 Subpart A 72.6 - License Required; Types of Licenses

(a) Licenses for the receipt, handling, storage, and transfer of spent fuel or high-level radioactive waste are of two types: general and specific.

10 CFR 72 Subpart B 72.24 - Contents of Application: Technical Information: Each application for a license under this part must include a Safety Analysis Report (SAR) describing the proposed ISFSI or MRS for the receipt, handling, packaging, and storage of spent fuel or high-level radioactive waste, including how the ISFSI or MRS will be operated. The minimum information to be included in this report must consist of the following:

(1) a description and safety assessment of the site

(2) a description and discussion of the ISFSI or MRS structures with special attention to design and operating characteristics, unusual or novel design features, and principal safety consideration

(3) an analysis and evaluation of the design and performance of structures, systems, and components important to safety

(4) the means for controlling and limiting occupational radiation exposures

(5) the features of ISFSI or MRS design and operating modes to reduce to the extent practicable radioactive waste volumes generated at the installation.

(6) an analysis of the potential dose equivalent or committed dose equivalent to an individual outside the controlled area from accidents or natural phenomena events that result in the release of radioactive material to the environment or direct radiation from the ISFSI or MRS.

10 CFR 72 Subpart C 72.40 - Issuance of License: This part provides, in great detail, the requirements for obtaining a license.

10 CFR 72 Subpart C 72.42 - Duration of License; Renewal: Each license issued under this part must be for a fixed period of time to be specified in the license. The license term for an ISFSI must not exceed 20 years from the date of issuance. The license term for an MRS must not exceed 40 years from the date of issuance. Licenses for either type of installation may be renewed by the NRC.

10 CFR 72 Subpart C 72.44 - License Conditions: Each license issued under this part shall include license conditions. The license conditions may be derived from the analyses and evaluations included in the SAR and amendments thereto submitted pursuant to 72.24. License conditions pertain to design, construction, operation and any additional conditions the NRC finds appropriate. The license must include technical specifications, which must in turn include: 1) functional and operating limits and monitoring instruments and limiting control settings, 2) limiting conditions - lowest operating performance levels of equipment required for safe operation, 3) surveillance requirements, 4) design features, and 5) administrative controls - full management procedures.

10 CFR 72 Subpart C 72.46,48-62: These parts provide requirements for public hearings; changes, tests and experiments; application for transfer, termination, modification, revocation and suspension, and amendment of license; and backfitting.

10 CFR 72 Subpart D 72.70-86 - Records, Reports, Inspections, and Enforcement: See full details in document.

10 CFR 72 Subpart E 72.90-108 - Siting Evaluation Factors: This part covers environmental impact considerations. Of note for radioactive materials:

10 CFR 72.96 - Siting Limitations:

(1) An ISFSI owned and operated by DOE must not be located at any site within which there is a HLW repository.

(2) An MRS must not be located in any state in which there is located any site approved for site characterization for a HLW repository

(3) If an MRS is located within 50 miles of the first HLW repository, the quantity of spent fuel or HLW that may be stored in both the repository and the MRS must be limited to 70,000 metric tons of heavy metal or the quantity of HLW resulting from the reprocessing of that quantity of spent fuel until the second repository is in operation.

10 CFR 72 Subpart E 72.104 - Criteria for Radioactive Materials in Effluents and Direct Radiation from an ISFSI or MRS:

(1) During normal operations and anticipated occurrences, the annual dose equivalent to any real individual who is located beyond the controlled area must not exceed 25 mrem to the whole body, 75 mrem to the thyroid and 25 mrem to any other organ.

10 CFR 72 Subpart F 72.120-130 - General Design Criteria: These design criteria establish the design, fabrication, construction, testing, maintenance, and performance requirements for structures, systems, and components important to safety as defined in 72.3. The general design criteria identified in this subpart establish minimum requirements for the design criteria for an ISFSI or MRS.

The reader is referred to the following sections for details:

10 CFR 72 Subpart F 72.124 - Criteria for Nuclear Criticality Safety

10 CFR 72 Subpart F 72.126 - Criteria for Radiological Protection

10 CFR 72 Subpart F 72.128 - Criteria for Spent Fuel, High-Level Radioactive Waste, and Other Radioactive Waste Storage and Handling

10 CFR 72 Subpart L - Approval of Spent Fuel Storage Casks

40 CFR 191 - Environmental Radiation Protection Standards for Management and Storage of SNF, HLW, and TRU Wastes - This regulation does not apply to the Yucca Mountain site,[(a)] but it can be used as guidance while the regulations for Yucca Mountain are being written.

40 CFR 191.02,03,12 - Definitions and Standards: See Section B.4.

40 CFR 191.11 - Applicability:

(a) This part applies to:

   (1) Radioactive materials released into the accessible environment as a result of the disposal of SNF or high-level or transuranic radioactive wastes

   (2) Radiation doses received by members of the public as a result of such disposal

   (3) Radioactive contamination of certain sources of ground water in the vicinity of disposal systems for such fuel or wastes.

40 CFR 191.13 - Containment Requirements: Releases to the environment for 10,000 years after disposal shall have a likelihood of less than one chance in 10 of exceeding the listed quantities for the listed radionuclides or one chance in 1,000 of exceeding 10 times the quantities listed in Table B.5.

40 CFR 191.14 - Assurance Requirements:

(1) Active institutional controls over disposal sites should be maintained for as long a period of time as is practicable after disposal; however, performance assessments that assess isolation of the wastes from the accessible environment shall not consider any contributions from active institutional controls for more than 100 years after disposal.

(2) Disposal systems shall be monitored after disposal to detect substantial and detrimental deviations from expected performance until there are no significant concerns.

(a) The Energy Policy Act of 1992 (Public Law 1102-486) directs the EPA to prescribe the maximum annual effective dose equivalent to individual members of the public from radioactivity released from the Yucca Mountain site.

Table B.5. Release Limits for Containment Requirements*

Radionuclide	Release Limit per 1,000 MTHM, Curies
Americium-241 or -243	100
Carbon-14	100
Cesium-125 or -137	1,000
Iodine-129	100
Neptunium-237	100
Plutonium-238,-239,-240,-or -242	100
Radium-226	100
Strontium-90	1,000
Technetium-99	10,000
Thorium-230, or -232	10
Tin-126	1,000
Uranium-233,-234,-235,-236, or -238	100
Any other alpha-emitting radionuclide with a half-life >20 years	100
Any other radionuclide with a half-life >20 years that does not emit alpha particles	1,000
*Cumulative releases to the accessible environment for 10,000 years after disposal from 40 CFR 191, Subpart B, Appendix A.	

(3) Both natural and engineered barriers shall be used to isolate the wastes from the environment.

40 CFR 191.15 - Individual Protection Requirements: "Disposal systems for SMF or high-level or transuranic radioactive wastes shall be designed to provide a reasonable expectation that, for 1,000 years after disposal, undisturbed performance of the disposal system shall not cause the annual dose equivalent from the disposal system to any member of the public in the accessible environment to exceed 25 mrem to the whole body or 75 mrem to any critical organ. All potential pathways (associated with undisturbed performance) from the disposal system to people shall be considered, including the assumption that individuals consume 2 liters per day of drinking water from any significant source of ground water outside of the controlled area."

40 CFR 191.16 - Ground Water Protection Requirements: Ground water protection requirements, with 1,000-year pCi/L limits, have been remanded and are being rewritten.

40 CFR 191.17 - Alternative Provisions for Disposal: Alternative provisions must be submitted for public comment which must be fully considered in developing the final version of the alternative.

40 CFR 191 Appendix B - Guidance for Implementation of Subpart B: This appendix provides details on the procedures to use in determining acceptable limits for emission releases.

B.3.3 Disposal

10 CFR 60 - Disposal of High Level Radioactive Wastes in Geologic Repositories - This regulation prescribes rules governing the licensing of the DOE to receive and possess source, special nuclear and byproduct material at a geologic repository operations area as indicated in the Nuclear Waste Policy Act of 1982. SNF, as defined in 10 CFR 72.3, includes special nuclear material, byproduct material, source material and other radioactive materials associated with fuel assemblies and is specifically defined as High Level Radioactive Waste in 10 CFR 60.2. Hence, indirectly, 10 CFR 60 applies to SNF.

10 CFR 60.15-18 - Site Characterization: Full details are given.

10 CFR 60.21-23 - Content of Application: Pages of details are given.

10 CFR 60.43 - License Specification: The ability to obtain a license requires certain characterization, as there are restrictions and conditions on the physical and chemical form and radioisotope content of the licensed material. The specifics of characterization required are not given in this article.

10 CFR 60.46 - Particular Activities Requiring License Amendment and 10 CFR 60.111 (b) - Retrievability of Waste: Disposal under 10 CFR 60.46 (a)(1) and 60.111 (b) requires that the material be retrievable, inasmuch as a license amendment is required for any action that would make the emplaced HLW irretrievable or difficult to retrieve during the first 50 years after emplacement.

10 CFR 60.102 (e) - Isolation of Waste

10 CFR 60.113 (a)(1) - Engineered Barrier System: The licensed materials must be contained within the disposal package for at least the first 300 years.

10 CFR 60.131 - General Design Criteria for the Geologic Repository Operations Area (b)(7) Criticality Control: All systems involving the materials to be placed in the repository must be designed to avoid criticality. This implies knowledge of the characteristics of the material to be deposited.

10 CFR 60.135 - Criteria for the Waste Package and its Components: Specific criteria for HLW package design

(1) The waste package shall not contain explosive or pyrophoric materials or chemically reactive materials in an amount that could compromise the ability of the underground facility to contribute to waste isolation.

(2) The waste package shall not contain free liquid in an amount that could compromise the ability of the waste packages to contain the HLW.

(3) All radioactive wastes shall be in solid form. Particulate wastes shall be consolidated into an encapsulating matrix. Combustibles radioactive wastes shall be reduced to noncombustible form unless it can be demonstrated that a fire would not compromise the waste package integrity.

10 CFR 60.140 - General Requirements: There must be a performance confirmation program, started during site characterization and continued until permanent closure. It will involve in situ monitoring and laboratory tests to ensure that changes in the environment caused by the repository are within license specifications. The monitoring likewise must not adversely affect the performance requirements.

10 CFR 60.143 - Monitoring and Testing Waste Packages: A program for monitoring waste packages under conditions of the repository, consistent with safe operation of the repository and including laboratory tests focussing on the internal condition of the packages, shall continue until the time of permanent closure.

10 CFR 61 - Licensing Requirements for Land Disposal of Radioactive Waste - This regulation establishes the details for licensing for land disposal of radioactive wastes containing byproduct, source and special nuclear material received from other persons.

10 CFR 61.3 - Licensed Required: No person may receive, possess, and dispose of radioactive waste containing source, special nuclear, or byproduct material at a land disposal facility unless authorized by the NRC pursuant to this part.

10 CFR 61.7(a) - The Disposal Facility: Near-surface means no deeper than 30 meters, i.e., a trench. Here, waste means low-level radioactive wastes containing source, special nuclear, or byproduct material that are acceptable for disposal in a land disposal facility. As in the Low-Level Waste Policy Act, low-level radioactive waste means radioactive waste not classified as high-level radioactive waste, TRU, SNF, or byproduct material.

10 CFR 61.7(b) - Waste Classification and Near-Surface Disposal: The classification of LLW is based on the content of long-lived radionuclides (and their shorter-lived precursors), the content of shorter-lived radionuclides, and the stability of the waste (i.e., its tendency to decompose like ordinary trash). (61.7 and 61.55). The classes of waste (A, B, and C) can be determined from the tables and the procedures given in 61.55. Additional information pertaining to this form of classification is found in 61.7(b)(5). The required depth of burial of Class C is given in 61.52(a)(2).

10 CFR 61.10 - Content of Application: An application to receive from others, possess and dispose of waste containing or contaminated with source, byproduct or special nuclear material by land disposal must consist of general, specific technical, institutional, and financial information as set forth in Sections 61.11 through 61.16.

10 CFR 61.12 - Specific Technical Information: Needed to show that performance objectives are met must include a description of the kind, amount, classification and specifications of the radioactive material proposed to be received, possessed and disposed of at the land disposal facility.

10 CFR 61.16(b)(12) - Safety Information Concerning Criticality: An applicant for a license to receive and possess special nuclear material in quantities must demonstrate how the requirements to prevent criticality will be met.

10 CFR 61.23 - Standards for Issuance of a License

10 CFR 61.24 - Condition of License

10 CFR 61.41 - Protection of the General Population from Releases of Radioactivity: Concentrations of radioactive material which may be released to the general environment in ground water, surface, water, air, soil, plants, or animals must not result in an annual dose exceeding an equivalent of 25 millirems to the whole body, 75 millirems to the thyroid, and 25 millirems to any other organ of any member of the public. Reasonable effort should be made to maintain releases of radioactivity in effluents to the general environment as low as reasonably achievable.

10 CFR 61.52 - Land Disposal Facility Operation and Disposal Site Closure: This section provides details of the depth and segregation required for waste packages according to general waste classifications A through C. These classifications are defined in detail in 10 CFR 61.55.

10 CFR 61.56: Minimum requirements for waste characteristics are listed here and include:

- Structural stability (to withstand the weight of overburden and compaction equipment)
- Elimination of void spaces within the waste and between the waste and its package
- Minimization of capability for detonation or explosive decomposition at normal temperatures and pressures
- Minimization of capability for generating toxic gases (except for radioactive gases, which may be packed at a pressure not to exceed 1.5 atmospheres at 20°C, with total activity not to exceed 100 Ci per container)
- Content of no more than 1% by volume of free liquid in a container designed to ensure stability or 5% of the volume of waste processed to a stable form.

40 CFR 268 - Land Disposal Restrictions

40 CFR 268.1 - Purpose, Scope, and Applicability: All who generate or transport hazardous waste or who operate hazardous waste treatment, storage or disposal facilities must observe restrictions on hazardous wastes that can be deposited in landfills (268.1).

40 CFR 268.2 - Definitions Applicable in this Part: It is implied that land disposal includes concrete vaults or bunkers.

40 CFR 268.3 - Dilution Prohibited as a Substitute for Treatment: Dilution is not acceptable as a substitute for treatment.

40 CFR 268.30-35 (Subpart C) - Prohibitions on Land Disposal: Specific definitions of hazardous nonradioactive waste types are given with correlated specific restrictions for land disposal and standards for treatment. These restrictions might apply to wastes from certain separation, leaching and other disposal processing steps.

40 CFR 268.40-45 (Subpart D) - Treatment Standards: Treatment standards are expressed in a variety of ways.

40 CFR 268 - Appendices: Appendices give Toxicity Characteristic Leaching Procedure (I), A list of Regulated Organic Compounds (III), and Recommended Technologies to Achieve Deactivation of Characteristics (VI), to name the most pertinent-looking of them.

B.3.4 General Management of Hazardous Materials

10 CFR 20 - Standards for Protection against Radiation - This section sets forth the standards to be used in NRC licensed facilities for protecting the public.

10 CFR 20.1003-1005 - Definitions: See Section B.4., where 10 CFR 20 definitions are given and compared with those in other regulations.

10 CFR 20.1201 - Occupational Dose Limits for Adults: The essential points are copied below; for many additional details, the reader is referred to the regulatory document.

(a) The licensee shall control the occupational dose to individual adults, except for planned special exposures under Section 20.1206, to the following dose limits:

(1) An annual limit, which is the more limiting of:

(i) The total effective dose equivalent being equal to 5 rems (0.05 Sv)

(ii) The sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue other than the lens of the eye being equal to 50 rems (0.5 Sv).

(2) The annual limits to the lens of the eye, to the skin, and to the extremities, which are:

(i) An eye dose equivalent of 15 rems (0.15 Sv)
(ii) A shallow-dose equivalent of 50 rems (0.50 Sv) to the skin or to any extremity.

10 CFR 20.1202 - Compliance with Requirements for Summation of External and Internal Doses:

20.1203 - Determination of External Dose from Airborne Radioactive Material

20.1204 - Determination of Internal Exposure

20.1206 - Planned Special Exposures

20.1207 - Occupational Dose Limits for Minors

20.1208 - Dose to an Embryo/Fetus

10 CFR 20.1301 - Dose Limits for Individual Members of the Public:

(a) Each licensee shall conduct operations so that

(1) the total effective dose equivalent to individual members of the public from the licensed operation does not exceed 0.1 rem (1 mSv) in a year, exclusive of the dose contribution from the licensee's disposal of radioactive material into sanitary sewerage in accordance with Section 20.2003

(2) the dose in any unrestricted area from external sources does not exceed 0.002 rem (0.02 mSv) in any one hour.

(b) If the licensee permits members of the public to have access to controlled areas, the limits for members of the public continue to apply to those individuals.

(c) A licensee or license applicant may apply for prior NRC authorization to operate up to an annual dose limit for an individual member of the public of 0.5 rem (5 mSv). For the information to be included in the application and additional details, the reader is referred to the document.

10 CFR 20.1501 - General

(a) Each licensee shall make or cause to be made, surveys that

(1) may be necessary for the licensee to comply with the regulations in this part
(2) are reasonable under the circumstances to evaluate

i) the extent of radiation levels
ii) concentration or quantities of radioactive material
iii) the potential radiological hazards that could be present.

10 CFR 20.1801 - Security of Stored Material

10 CFR 20.1802 - Control of Material not in Storage

10 CFR 20.2001(b) - General Requirements: A licensee must be specifically licensed to receive waste containing licensed material for:

(1) treatment prior to disposal
(2) treatment or disposal by incineration
(3) decay in storage
(4) disposal in a land disposal facility licensed under 10 CFR 61
(5) disposal at a geologic repository under 10 CFR 60.

10 CFR 20.2002 - Method for Obtaining Approval for Proposed Disposal Procedures...for approval...to Dispose of Licensed Material Generated in the Licensee's Activities: Each application shall include a description of the waste containing licensed material to be disposed of, including the physical and chemical properties important to risk evaluation, and the proposed manner and conditions of waste disposal.

10 CFR 20.2005 - Disposal of Specific Wastes: Material used for liquid scintillation counting may be disposed of as if it were not radioactive if it contains 0.05 μCi or less of ^{3}H or ^{14}C per gram of medium.

10 CFR 20 Appendix B: The introduction is included here to give an indication of the content of this appendix, which is extensive and complex and introduces the terms ALI and DAC, which may be useful.

"For each radionuclide Table 1 indicates the chemical form which is to be used for selecting the appropriate ALI (annual limit on intake) or DAC (derived air concentration-hours) value. The ALIs and DACs for inhalation are given for an aerosol with an activity median aerodynamic diameter (AMAD) of 1 mm and for three classes (D,W,Y) of radioactive material, which refer to their retention (approximately days, weeks or years) in the pulmonary region of the lung. This classification applies to a range of clearance half-times of less than 10 days for D, for W from 10 to 100 days, and for Y greater than 100 days. The class (D, W, or Y) given in the column headed "Class" applies only to the inhalation ALIs and DACs given in Table 1, columns 2 and 3. Table 2 provides concentration limits for airborne and liquid effluents released to the general environment. Table 3 provides concentration limits for discharges to sanitary sewer systems." (These tables are to be found in the reference, not in this Appendix.)

DOE 5633.3 - Control and Accountability of Nuclear Materials - For detailed accountability instructions, primarily of an administrative nature, the reader is referred to the DOE 5633.3 as a whole. In general, nuclear materials designated as wastes are included in the accountability requirements unless specifically exempted. See Chapter I, Section 1q.

DOE 5633.3, Chapter I: Provides minimum requirements for the control and accountability of nuclear materials. The level of control and accountability shall be consistent with the economic and strategic value of these materials. The minimum reportable quantities are given in Table B.6.

DOE 5633.3, Chapter I - Section Graded Safeguards: Graded safeguards is the concept of providing the greatest relative amount of control and effort to the types and quantities of SNM that can be most effectively used in a nuclear explosive device. The grade considerations of specific configurations are provided in the order in Figures 1 and 2.

DOE 5633.3, Chapter II - Materials Accountability, Section 4 Measurements and Control: Provides details on the inventory measurement methods, statistics and controls.

DOE 5820.2A - Radioactive Waste Management

DOE 5820.2, Chapter I - High-Level Waste: HLW will be considered to be radioactive mixed waste and subject to AEA and RCRA. Most requirements listed are concerned with tank wastes and the Waste Isolation Pilot Plant.

Table B.6. Minimum Reportable Quantities of Nuclear Materials

Material Type	SNM	Source	Other Nuclear Materials	Reportable Quantity
Depleted Uranium		x		Kilogram
Enriched Uranium[1]	x			Gram
Normal Uranium		x		Kilogram
Uranium-233	x			Gram
Plutonium-242[2]	x			Gram
Plutonium-239 - 241	x			Gram
Plutonium-238[3]	x			Tenth of a Gram
Americium-241			x	Gram
Americium-243			x	Gram
Berkelium			x	Microgram
Californium-252			x	Microgram
Curium			x	Gram
Deuterium			x	Tenth of a Gram
Lithium-6			x	Kilogram
Neptunium-237			x	Gram
Thorium		x		Kilogram
Tritium[4]			x	Hundredth of a Gram

1. Uranium in cascades is treated as enriched uranium.
2. Report as Pu-242 if the contained Pu-242 is 20% or greater of total Pu by weight; otherwise report as Pu-239 to Pu-241.
3. Report as Pu-238 if the contained Pu-238 is 10% or greater of the total by weight Pu; otherwise report as Pu-239 to Pu-241.
4 Tritium contained in water (H_2O or D_2O) used as a moderator in a nuclear reactor is not an accountable material.

DOE 5820.2, Chapter II - Management of Transuranic Waste: Also predicated on the use of Waste Isolation Pilot Plant, with much detail concerning siting and site management, but the management requirements for TRU are generic. In this chapter, as in other places throughout the NRC and DOE regulations, the statement is made that radioactive waste, in this case TRU wastes that are also mixed wastes are subject to the requirements of the AEA and the RCRA.

One of the sections of most pertinence is the following (note in particular that Section 4.c):

DOE 5820.2, Chapter II - Management of Transuranic Waste - 3.a. - Waste Classification;

(1) Any material that is known to be, or suspected of being contaminated with transuranium radionuclides shall be evaluated as soon as possible in the generating process, and determined to be either recoverable material, TRU, LLW, mixed waste, or nonradioactive trash in order to avoid commingling the various material streams.

(2) The lower concentration limit for TRU (100 nCi/g of waste) shall apply to the contents of any single waste package at the time of assay. The mass of the waste container including shielding shall not be used in calculating the specific activity of the waste.

(3) Radioactive wastes with quantities of transuranic radionuclides in concentrations of 100 nCi/g of waste or less shall be considered to be LLW, and shall be managed according to the requirements of Chapter III of this order.

(4) Mixed transuranic waste:

(a) Mixed transuranic waste meeting the requirements of the Waste Isolation Pilot Plant-Waste Acceptance Criteria shall be sent to the Waste Isolation Pilot Plant.

(b) The data package prepared by the generators for the Waste Isolation Pilot Plant shall include information on the kinds and quantities of hazardous components contained in a waste package in accordance with applicable RCRA regulations.

(c) The determination whether the TRU exhibits any hazardous characteristics or contains listed hazardous components may be based on knowledge of the waste generating process when the performance of a chemical analysis would significantly increase the radiation hazard to personnel.

DOE 5820.2, Chapter II b. - Transuranic Waste Generation and Treatment:

(1) Technical and administrative controls shall be directed to reducing the gross volume of waste generated and/or the amount of radioactivity requiring disposal. TRU reduction efforts shall be based on the implementation of techniques such as process modification, process optimization, materials substitution, decontamination, assay of suspect waste, and new technology development. Volume reduction techniques, such as incineration, compaction, extraction, and shredding, shall be implemented wherever cost effective and practical. Treatment facilities shall be permitted by the appropriate regulatory authority.

(2) TRU shall be assayed or otherwise evaluated to determine the kinds and quantities of transuranic radionuclides present before storage. Additionally, hazardous waste components shall be estimated or analyzed, whichever is appropriate.

(3) Mixed transuranic waste shall be treated, where feasible and practical, to destroy the hazardous waste component.

DOE 5820.2, Chapter II e. - Temporary Storage at Generating Sites: A list of activities is prescribed to ensure the safe storage of TRU consistent with the requirements of applicable RCRA regulations, the most pertinent of which is:

(1) TRU shall be segregated or otherwise clearly identified to avoid the commingling of TRU streams with HLW or LLW

(2) Certified TRU shall not be commingled with noncertified TRU and shall be stored in a manner unlikely to alter its certification status.

DOE 5820.2, Chapter III - Low Level Waste:

DOE 5820.2 2. - Policy:

(a) DOE LLW operations shall be managed to protect the health and safety of the public, preserve the environment of the waste management facilities, and ensure that no legacy requiring remedial action remains after operations have been terminated.

(b) DOE LLW shall be managed on a systematic basis using the most appropriate combination of waste generation reduction, segregation, treatment, and disposal practices so that the radioactive components are contained and the overall system cost effectiveness is maximized.

(c) DOE LLW shall be disposed of on the site at which it is generated, if practical, or if onsite disposal capability is not available, at another DOE disposal facility.

(d) DOE LLW that contains nonradioactive hazardous waste components (mixed waste) shall conform to the requirements of this order, applicable EH Orders, and shall also be regulated by the appropriate regional authorities under the RCRA.

DOE 5820.2 3. - Requirements d. - Waste Characterization:

(1) LLW shall be characterized with sufficient accuracy to permit proper segregation, treatment, storage, and disposal. This characterization shall ensure that, upon generation and after processing, the actual physical and chemical characteristics and major radionuclide content are recorded and known during all stages of the waste management process.

(2) Waste characterization data shall be recorded on a waste manifest, as required by paragraph 3m, and shall include:

  (a) the physical and chemical characteristics of the waste
  (b) volume of the waste (total of waste and any solidification or absorbent media)
  (c) weight of the waste (total of waste and any solidification or absorbent media)
  (d) major radionuclides and their concentrations
  (e) packaging date, package weight, and external volume.

(3) The concentration of a radionuclide may be determined by direct methods or by indirect methods such as use of scaling factors which relate the inferred concentration of one radionuclide to another that is measured, or radionuclide material accountability, if there is reasonable assurance that the indirect methods can be correlated with actual measurements.

DOE 5820.2, Chapter III, 3.i. - Disposal:

(4) Disposition of waste designated as greater-than-class C, as defined in 10 CFR 61.55, must be handled as special cases. Disposal systems for such waste must be justified by a specific performance assessment through the NEPA process and with the concurrence of DP-12 for all DP-1 disposal facilities and of NE-20 for those disposal facilities under the cognizance of NE-1.

(5) The following are additional disposal requirements intended either to improve stability of the disposal site or to facilitate handling and provide protection of the health and safety of personnel at the disposal site:

(a) waste must not be packaged for disposal in cardboard or fiberboard boxes, unless such boxes meet DOE requirements and contain stabilized waste with a minimum of void space. For all types of containers, void spaces within the waste and between the waste and its packaging shall be reduced as much as practical.

(b) liquid wastes, or wastes containing free liquid, must be converted into a form that contains as little freestanding and noncorrosive liquid as is reasonably achievable, but, in no case, shall the liquid exceed 1 percent of the volume of the waste when the waste is in a disposal container, or 0.5 percent of the volume of the waste processed to a stable form.

(c) waste must not be readily capable of detonation or of explosive decomposition action at normal pressures and temperatures, or of explosive reaction with water.

(d) waste must not contain, or be capable of generating, quantities of toxic gases, vapors, or fumes harmful to persons transporting, handling, or disposing of the waste. This does not apply to radioactive gaseous waste packaged as identified in paragraph 3i(5)(e).

(e) waste in a gaseous form must be packaged at a pressure that does not exceed 1.5 atmospheres at 20°C.

(f) waste must not be pyrophoric. Pyrophoric materials contained in waste shall be treated, prepared, and packaged to be nonflammable.

(6) Waste containing amounts of radionuclides below regulatory concern, as defined by federal regulations, may be disposed of without regard to radioactivity content.

B.3.5 Lists of Hazardous Materials

40 CFR 61 - National Emission Standards for Hazardous Air Pollutants - This document in one of the defining regulations for the CAA. It contains extensive lists of pollutants, to which the reader is referred (40 CFR 61.01). Specific regulations apply for beryllium, mercury, benzene, asbestos, arsenic from glass manufacturing plants, among others.

These emission standards could be important for N Reactor fuel characterization, where releases are encountered in the KE basin and airborne radionuclide releases could result during handling and testing, because of the possibility of oxides and sludge becoming dry.

40 CFR 61 (Subpart H) - National Emission Standards for Emissions of Radionuclides Other Than Radon from Department of Energy Facilities: (Subpart H does not apply to the disposal facilities subject to 40 CFR 191 Waste Isolation Pilot Plant or 40 CFR 192.)

40 CFR 61.92 - Standard (Emission): The standard for emission to the ambient air from any facility operated by DOE must not cause any member of the public to receive an effective dose equivalent exceeding 10 mrem/year.

40 CFR 61.93 - Emission Monitoring and Test Procedures: Monitoring requirements are outlined and limits set for emissions from stacks and vents.

40 CFR 61 (Subpart I) - National Emission Standards for Radionuclide Emission from Facilities Licensed by the Nuclear Regulatory Commission and Federal Facilities not Covered by Subpart H: The applicability of this subpart is the same as for Subpart H.

40 CFR 61.102 - Standard: Same as in Subpart H but includes iodine.

40 CFR 61.103 - Determining Compliance: Dose limit compliance may be calculated using EPA or DOE approved computer modelling codes or measurement procedures described in 61.107 and Appendix E of CFR 61.

40 CFR 61.107 - Emission Determination: References are given to various methods of monitoring flow velocity, volumetric rates, etc.

40 CFR 61 (Subpart Q) - National Emission Standards for Radon Emissions from Department of Energy Facilities

40 CFR 61.192 - Standard: The limit for emission of radon into the air from all DOE storage and disposal facilities is 20 pCi/m^2-s of radon-222 as an average for the entire source (61.190).

40 CFR 261 - Identification and Listing of Hazardous Waste

40 CFR 261.3 (Subpart A) - Definition of Hazardous Waste

40 CFR 261.4 (Subpart A) - Exclusions: The definition of hazardous waste and the types of materials that are included are set forth at length in 40 CFR 261.3, Subpart A; the exclusions are listed in 40 CFR 261.4. The complexities are such that the reader is directed to determine applicability of the regulations on a case-by-case basis. The RCRA toxic limits are listed in Table B.7.

40 CFR 261.10 (Subpart B) - Criteria for Identifying the Characteristics of Hazardous Waste:

(a) The Administrator shall identify and define a characteristic of hazardous waste in Subpart C only upon determining that:

  (1) A solid waste that exhibits the characteristic may:

    (i) cause, or significantly contribute to, an increase in mortality or an increase in serious irreversible, or incapacitating reversible, illness

Table B.7. RCRA Toxic Elements from 40 CFR 261.3

RCRA Toxic Element	Maximum for any single composite sample, in mg/L
Antimony	0.063
Arsenic	0.055
Barium	6.3
Beryllium	0.0063
Cadmium	0.032
Chromium (total)	0.33
Lead	0.095
Mercury	0.009
Nickel	0.63
Selenium	0.16
Silver	0.30
Thallium	0.013
Vanadium	1.26

(ii) pose a substantial present or potential hazard to human health or the environment when it is improperly treated, stored, transported, disposed of or otherwise managed

(2) The characteristic can be:

(i) measured by an available standardized test method which is reasonably within the capability of generators of solid waste or private sector laboratories that are available to serve generators of solid waste

(ii) reasonably detected by generators of solid waste through their knowledge of their waste.

40 CFR 261.11 - Criteria for Listing Hazardous Waste:

(a) The Administrator shall list a solid waste as a hazardous waste only upon determining that the solid waste meets one of the following criteria:

(1) it exhibits any of the characteristics of hazardous waste identified in Subpart C.

(2) it has been found to be fatal to humans in low doses or, in the absence of data on human toxicity, it has been shown in studies to have an oral LD 50 toxicity (rat) of less than 50 milligrams per kilogram, an inhalation LC 50 toxicity (rat) of less than 2 milligrams per

liter, or a dermal LD 50 toxicity (rabbit) of less than 200 milligrams per kilogram or is otherwise capable of causing or significantly contributing to an increase in serious irreversible, or incapacitating reversible, illness. (Waste listed in accordance with these criteria will be designated Acute Hazardous Waste.)

(3) it contains any of the toxic constituents listed in Appendix VIII and, after considering the following factors, the Administrator concludes that the waste is capable of posing a substantial present or potential hazard to human health or the environment when improperly treated, stored, transported or disposed of, or otherwise managed:

  (i) the nature of the toxicity presented by the constituent

  (ii) the concentration of the constituent in the waste

  (iii) the potential of the constituent or any toxic degradation product of the constituent to migrate from the waste into the environment under the types of improper management considered in paragraph (a)(3)(vii) of this section

  (iv) the persistence of the constituent or any toxic degradation product of the constituent

  (v) the potential for the constituent or any toxic degradation product of the constituent to degrade into non-harmful constituents and the rate of degradation

  (vi) the degree to which the constituent or any degradation product of the constituent bioaccumulates in ecosystems

  (vii) the plausible types of improper management to which the waste could be subjected

  (viii) the quantities of the waste generated at individual generation sites or on a regional or national basis

  (ix) the nature and severity of the human health and environmental damage that has occurred as a result of the improper management of wastes containing the constituent

  (x) action taken by other governmental agencies or regulatory programs based on the health or environmental hazard posed by the waste or waste constituent

  (xi) such other factors as may be appropriate.

Substances will be listed on Appendix VIII only if they have been shown in scientific studies to have toxic, carcinogenic, mutagenic or teratogenic effects on humans or other life forms. (Wastes listed in accordance with these criteria will be designated toxic wastes.)

40 CFR 261.21-24 (Subpart C) - Characteristics of Hazardous Waste: There are four characteristics by which hazardous waste is identified: ignitability, corrosivity, reactivity, and toxicity. Details are given in Sections 21 through 24. The maximum concentration of contaminants for the toxicity characteristic is given in Table B.8.

Table B.8. Maximum Concentration of Contaminants for the Toxicity Characteristics[a]

EPA HW No.[b]	Contaminant	CAS No.[c]	Regulatory Level (mg/L)
D004	Arsenic	7440-38-2	5.0
D005	Barium	7440-39-3	100.0
D018	Benzene	71-43-2	0.5
D006	Cadmium	7440-43-9	1.0
D019	Carbon Tetrachloride	56-23-5	0.5
D020	Chlordane	57-74-9	0.03
D021	Chlorobenzene	108-90-7	100.0
D022	Chloroform	67-66-3	6.0
D007	Chromium	7440-47-3	5.0
D023	o-Cresol	95-48-7	200.0[d]
D024	m-Cresol	108-39-4	200.0[d]
D025	p-Cresol	106-44-5	200.0[d]
D026	Cresol		200.0[d]
D016	2,4-D	94-75-7	10.0
D027	1,4-Dichlorobenzene	106-46-7	7.5
D028	1,2-Dichloroethane	107-06-2	0.5
D029	1.1-Dichloroethylene	75-35-4	0.7
D030	2,4-Dinitrotoluene	121-14-2	0.13[e]
D012	Endrin	72-20-8	0.02
D031	Heptachlor (and its epoxide)	76-44-8	0.008
D032	Hexachlorobenzene	118-74-1	0.13[e]
D033	Hexachlorobutadiene	87-68-3	0.5
D034	Hexachloroethane	67-72-1	3.0
D008	Lead	7439-92-1	5.0
D013	Lindane	58-89-9	0.4
D009	Mercury	7439-97-6	0.2
D014	Methoxychlor	72-43-5	10.0
D035	Methyl ethyl ketone	78-93-3	200.0
D036	Nitrobenzene	98-95-3	2.0
D037	Pentachlorophenol	87-86-5	100.0
D038	Pyridine	110-86-1	5.0[e]
D010	Selenium	7782-49-2	1.0
D011	Silver	7440-22-4	5.0
D039	Tetrachloroethylene	127-18-4	0.7
D015	Toxaphene	8001-35-2	0.5
D040	Trichloroethylene	79-01-6	0.5
D041	2,4,5-Trichlorophenol	95-95-4	400.0
D042	2,4,6-Trichlorophenol	88-06-2	2.0
D017	2,4,5-TP (Silvex)	93-72-1	1.0
D043	Vinyl chloride	75-01-4	0.2

(a) From 40 CFR 261.24.
(b) Hazardous waste number.
(c) Chemical abstracts service number.
(e) Quantitation limit is greater than the calculated regulatory level. The quantitation limit therefore becomes the regulatory level.
(d) If o-, m-, and p-Cresol concentrations cannot be differentiated, the total cresol (D026) concentration is used. The regulatory level of total cresol is 200 mg/l.

40 CFR 261.31-35 (Subpart D) - Lists of Hazardous Wastes, from Specific and Non-Specific Sources: The reader should inspect the lists for the specific chemicals of concern.

49 CFR 172 - Hazardous Materials Table

49 CFR 172.101 (Subpart B) - Table of Hazardous Materials: Table 1 - Hazardous Substances other then Radionuclides (and reportable quantities; Table 2 - Radionuclides (and reportable quantities). These are extensive tables to which the reader is referred.

DOE 5400.3 - Hazardous and Radioactive Mixed Waste Program

DOE 5400.3 4.b. - Definitions: See Section B.4. Inconsistency between RCRA and the AEA occurs if the requirements of both laws are incompatible. RCRA applies to hazardous or radioactive mixed waste to the extent it is not inconsistent with the requirements of the AEA.

DOE 5400.3 5.b. - Background: Any radioactive material, as used in subsection (a) of the AEA (42 U.S.C. 2011 et seq.), refers only to the actual radionuclides dispersed or suspended in the waste substance. The nonradioactive hazardous component of the waste substance will be subject to regulation under the RCRA.

DOE interprets these definitions to mean that whenever any hazardous waste identified or listed in Title 40 CFR Part 261 is inadvertently mixed with any source material, special nuclear material, or byproduct material, the hazardous waste component is subject to regulation under Subtitle C of RCRA. The May 1, 1987 Federal Register notice did not affect materials that are defined as byproduct material under Section 11e(2) of the AEA.

DOE 5400.3-6 - Policy: The radioactive component of radioactive mixed waste is subject to the requirements of DOE 5820.2A.

B.4 Glossary

A1 [49 CFR 173.403 (a)] - The maximum activity of special form radioactive material permitted in a Type A package.

A2 [49 CFR 173.403 (b)] - The maximum activity of radioactive material, other than special form or low specific activity radioactive material, permitted in a Type A package. These values are either listed in 173.435 or may be derived in accordance with the procedure prescribed in 173.433. Also 49 CFR 173.403 (cc) through (hh) - definitions of Type A, B, B(M) and B(U) packaging.

Byproduct material - 1) Any radioactive material (except special nuclear material) yielded in, or made radioactive by, exposure to the radiation incident to the process of producing or utilizing special nuclear material, and 2) the tailings or wastes produced by the extraction or concentration of uranium or thorium from ore processed primarily for its source material content, including discrete surface wastes resulting from uranium solution extraction processes. Underground ore odies depleted by these solution extraction operations do not constitute byproduct material within this definition.

Byproduct material [10 CFR 72.3] - Any radioactive material (except special nuclear material) yielded in or made radioactive by exposure to the radiation incident to the process of producing or utilizing special nuclear material.

Byproduct material [DOE 5400.3 5.a.] - 1) Any radioactive material (except special nuclear material) yielded in or made radioactive by exposure to the radiation incident to the process of producing or utilizing special nuclear material, and 2) the tailings or wastes produced by the extraction or concentration of uranium or thorium from any material processed primarily for its source material content.

Class (or lung class or inhalation class) - A classification scheme for inhaled material according to its rate of clearance from the pulmonary region of the lung. Materials are classified as D, W, or Y, which applies to a range of clearance half-times: for Class D (Days) of less than 10 days, for Class W (Weeks) from 10 to 100 days, and for Class Y (Years) of greater than 100 days.

Collective dose - The sum of the individual doses received in a given period of time by a specified population from exposure to a specified source of radiation.

Disposal [10 CFR 60.2] - The isolation of radioactive wastes from the accessible environment.

Fissile classification [DOE 5480.3 d.] - Classification of a package or shipment of fissile materials according to the controls needed to provide nuclear criticality safety during transportation as follows (DOE 5480.3):

(1) **Fissile Class I** - Packages that may be transported in unlimited numbers and in any arrangement and that require no nuclear criticality safety controls during transportation. For purposes of nuclear criticality safety control, a transport index is not assigned to Fissile Class I packages. However, the external radiation levels may require a transport index number.

(2) **Fissile Class II** - Packages that may be transported arrangement but in numbers that do not exceed a transport index of 50. For purposes of nuclear criticality safety control, individual packages may have a transport index if not less than 0.1 and not more than 10. However, the external radiation levels may require a higher transport index number but not to exceed 10. Such shipments require no nuclear criticality safety control by the shipper during transportation.

(3) **Fissile Class III** - Shipments of packages that do not meet the requirements of Fissile Class I and II and that are controlled in transportation by special arrangements between the shipper and the carrier to provide nuclear criticality safety.

Fissile materials [DOE 5480.3 e.] - Uranium-233, uranium-235, plutonium-23X, plutonium-239, plutonium-241, neptunium-237, and curium-244.

Hazardous waste [DOE 5400.3 4.a.] - Waste defined as hazardous in 40 CFR Part 261. The radionuclides of source material, special nuclear material, and byproduct material as defined by the AEA, as amended, are specifically excluded from the term hazardous waste. The hazardous components of waste mixed with the radionuclides of source, special nuclear, or byproduct material are not excluded from the term hazardous waste.

Hazardous wastes [DOE 5820.2 17] - Those wastes that are designated hazardous by EPA regulations (40 CFR 261).

High-level radioactive waste (HLW) [10 CFR 72.3] - 1) The highly radioactive material resulting from the reprocessing of SNF, including liquid waste produced directly in reprocessing and any solid material derived from such liquid waste that contains fission products in sufficient concentrations, and 2) other highly radioactive material that the NRC, consistent with existing law, determines by rule requires permanent isolation.

High-level waste [DOE 5820.2 18] - The highly radioactive waste material that results from the reprocessing of SNF, including liquid waste produced directly in reprocessing and any solid waste derived from the liquid, that contains a combination of TRU and fission products in concentrations requiring permanent isolation.

Independent spent fuel storage installation (ISFSI) [10 CFR 72.3] - A complex designed and constructed for the interim storage of SNF and other radioactive materials associated with spent fuel storage. An ISFSI that is located on the site of another facility may share common utilities and services with such a facility and be physically connected with such other facility and still be considered independent, provided, that such sharing of utilities and services or physical connections does not 1) increase the probability or consequences of an accident or malfunction of components, structures, or systems that are important to safety, or 2) reduce the margin of safety as defined in the basis for any technical specification of either facility.

Low-level waste [DOE 5820.2 20] - Waste that contains radioactivity and is not classified as HLW, TRU, SNF, or byproduct material as defined by this order. Test specimens of fissionable material irradiated for research and development only, and not for the production of power or plutonium, may be classified as LLW, provided the concentration of transuranic is less than 100 nCi/g.

Low specific activity material (LSA) [49 CFR 173.403 (n)] - Any of the following:

(1) uranium or thorium ores and physical or chemical concentrates of those ores.

(2) unirradiated natural or depleted uranium or unirradiated natural thorium.

(3) tritium oxide in aqueous solutions provided the concentration does not exceed 5.0 mCi/ml.

(4) material in which the radioactivity is essentially uniformly distributed and in which the estimated average concentration of contents does not exceed:

   (i) 0.0001 millicurie per gram of radionuclides for which the A2 quantity is not more than 0.05 curie

   (ii) 0.005 millicurie per gram of radionuclides for which the A2 quantity is more than 0.05 curie, but not more than 1 curie

   (iii) 0.3 millicurie per gram of radionuclides for which the A2 quantity is more than 1 curie.

(5) objects of nonradioactive material externally contaminated with radioactive material, provided that the radioactive material is not readily dispersible and the surface contamination, when averaged over an area of 1 square meter, does not exceed 0.0001 millicurie (220,000 disintegrations per minute) per square centimeter of radionuclides for which the A2 quantity is not more than 0.05 curie, or 0.001 millicurie (2,200,000 disintegrations per minute) per square centimeter for other radionuclides.

Mixed waste [DOE 5820.2 22] - Waste containing both radioactive and hazardous components as defined by the AEA and the RCRA.

Monitored Retrievable Storage Installation (MRS) [10 CFR 72.3] - A complex designed, constructed, and operated by DOE for the receipt, transfer, handling, packaging, possession, safeguarding, and storage of SNF aged for at least one year and solidified high-level radioactive waste resulting from civilian nuclear activities, pending shipment to a HLW repository or other disposal.

Radioactive material [49 CFR 173.403 (y)] - Any material having a specific activity greater than 0.002 microcuries per gram (uCi/g). (See Specific activity.)

Radioactive mixed waste [DOE 5400.3 4.d.] - Waste containing both radioactive and hazardous components regulated by the AEA and RCRA. The term radioactive component refers only to the actual radionuclides dispersed or suspended in the waste substance.

Radioactive waste or Waste [10 CFR 60.2] - HLW and other radioactive materials other than HLW that are received for emplacement in a geologic repository.

Radioactive waste [DOE 5400.3 4.c. and DOE 5820.2 29] - Solid, liquid, or gaseous material that contains radionuclides regulated under the AEA, as amended, and of negligible economic value considering costs of recovery.

Source material -

(1) Uranium or thorium or any combination of uranium and thorium in any physical or chemical form

(2) Ores that contain, by weight, one-twentieth of 1 percent (0.05 percent), or more, of uranium, thorium, or any combination of uranium and thorium. Source material does not include special nuclear material.

Source material [10 CFR 72.3] -

(1) Uranium or thorium, or any combination thereof, in any physical or chemical form

(2) Ores that contain by weight one-twentieth of one percent (0.05%) or more of uranium, thorium, or any combination thereof.

Source material does not include special nuclear material.

Special form radioactive material [49 CFR 173.403 (z)] - Radioactive material that satisfies the following conditions:

(1) it is either a single solid piece or is contained in a sealed capsule that can be opened only by destroying the capsule

(2) the piece or capsule has at least one dimension not less than 5 millimeters (0.197 inch)

(3) it satisfies the test requirements of 173.469. Special form encapsulations designed in accordance with the requirements of 173.389(g) in effect on June 30, 1983, and constructed prior to July 1, 1985 may continue to be used. Special form encapsulations either designed or constructed after June 30, 1985 must meet the requirements of this paragraph.

Special form radioactive material [DOE 5480.3 n.] - To qualify as special form the radioactive material must either be in massive solid form or encapsulated. Special tests which are required of special form material are explained in 49 CFR 173.403.

Special nuclear material [10 CFR 72.3] - 1) Plutonium, uranium-233, uranium enriched in the isotope 233 or in the isotope 235, and any other material which the NRC, pursuant to the provisions of Section 51 of the AEA, determines to be special nuclear material, but does not include source material, or 2) any material artificially enriched by any of the foregoing but does not include source material.

Spent nuclear fuel or Spent fuel [10 CFR 72.3] - Fuel that has been withdrawn from a nuclear reactor following irradiation, has undergone at least one year's decay since being used as a source of energy in a power reactor, and has not been chemically separated into its constituent elements by reprocessing. Spent fuel includes the special nuclear material, byproduct material, source material, and other radioactive materials associated with fuel assemblies.

State hazardous waste [DOE 5400.3 4.e] - Waste defined as hazardous by a state. Pursuant to RCRA Section 6001, DOE is subject to and must comply with state requirements respective to solid and hazardous waste management.

Transport index [49 CFR 173.403 (bb)] - The dimensionless number (rounded up to the first decimal place) placed on the label of a package to designate the degree of control to be exercised by the carrier during transportation. The transport index is determined as follows:

(1) the number expressing the maximum radiation level in millirem per hour at 1 m (3.3 ft) from the external surface of the package

(2) for Fissile Class II packages or packages in a Fissile Class III shipment, the number expressing the maximum radiation level at 1 m (3.3 ft) from the external surface of the package, or the number obtained by dividing 50 by the allowable number of packages which may be transported together, whichever is larger.

Transport index [DOE 5480.3 o.] - The number placed on a package to designate the degree of control to be exercised by the carrier during transportation. The transport index to be assigned to a package of radioactive material shall be determined by either paragraph (1) or (2) below, whichever is larger. The number expressing the transport index shall be rounded up to the next higher tenth (e.g., 1.01 becomes 1.0 and 1.06 becomes 1.1).

(1) The highest radiation dose rate in millirem per hour at 1 m from any accessible external surface of the package.

(2) The transport index of each Fissile Class II package is calculated by dividing the number 50 by the number of such Fissile Class packages that may be transported together as determined under the limitations of 10 CFR 71.

Transuranium radionuclide [DOE 5820.2 38] - Any radionuclide having an atomic number greater than 92.

Transuranic waste [DOE 5820.2 39] - Without regard to source or form, waste that is contaminated with alpha-emitting transuranium radionuclides with half-lives greater than 20 years and concentrations greater than 100 nCi/g at the time of assay. Heads of Field Elements can determine that other alpha contaminated wastes, peculiar to a specific site, must be managed as TRU.

Waste classification [10 CFR 61.55 (a)(2)] - Classes A through C are waste definitions based on nuclide activity, nuclide half-life, and general waste stability. If they exceed the radioactivity levels of C-14, Ni-59, Nb-94, Tc-99, I-129, TRU, Pu-241, and Cm-242, or the radioactivity levels of H-3, Co-60, Ni-63, Sr-90, or Cs-137, in general they are not suitable for disposal in near-surface facilities. The reader is referred to the document, which contains a great deal of detail.

Weighting factor (wT) - For an organ or tissue (T) is the proportion of the risk of stochastic effects resulting from irradiation of that organ or tissue to the total risk of stochastic effects when the whole body is irradiated uniformly. For calculating the effective dose equivalent, the values of wT are:

Table B.9. Organ Dose Weighting Factors

Organ or tissue	wT
Gonads	0.25
Breast	0.15
Red bone marrow	0.12
Lung	0.12
Thyroid	0.03
Bone surfaces	0.03
Remainder	n1 0.30

Table B.9. (contd)

Organ or tissue	wT
Whole body	n2 1.00
n1 - 0.30 results from 0.06 for each of 5 remainder organs (excluding the skin and the lens of the eye) that receive the highest doses. n2 - For the purpose of weighting the external whole body dose (for adding it to the internal dose), a single weighting factor, wT=1.0, has been specified. The use of other weighting factors for external exposure will be approved on a case-by-case basis until such time as specific guidance is issued.	

Working level (WL) - Any combination of short-lived radon daughters (for radon-222: polonium-218, lead-214, bismuth-214, and polonium-214; and for radon-220: polonium-216, lead-212, bismuth-212, and polonium-212) in 1 liter of air that will result in the ultimate emission of 1.3105 MeV of potential alpha particle energy.

Working level month (WLM) - An exposure to 1 working level for 170 hours (2,000 working hours per year/12 months per year = approximately 170 hours per month).

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