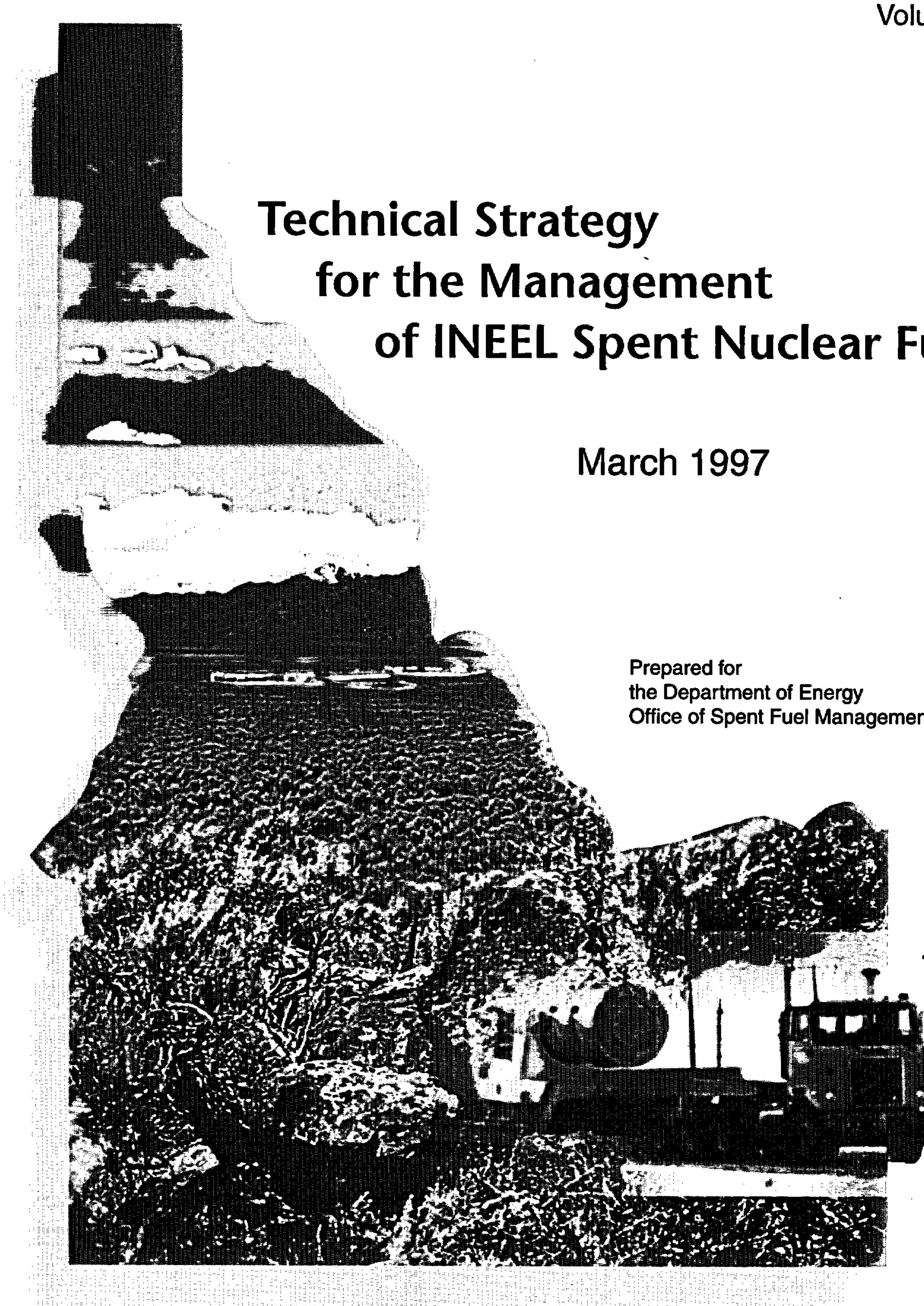
Volume I

Technical Strategy for the Management of INEEL Spent Nuclear Fuel



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Prepared for
the Department of Energy
Office of Spent Fuel Management

A Report of the INEEL Spent Nuclear Fuel Task Team

Foreword

The Idaho National Engineering and Environmental Laboratory (INEEL) has been a center of nuclear technology in the United States for over half a century, supporting the national defense in the development of Naval nuclear submarine and surface vessel propulsion systems, as well as in civilian and military nuclear applications. INEEL is operated by various contractor organizations under the direction of the U.S. Department of Energy (DOE).

A substantial part of the work done at INEEL involves use or testing of nuclear fuel from various sources. Fuel that has been withdrawn from a nuclear reactor following irradiation is called spent nuclear fuel (SNF). In years past SNF generated at INEEL or received from off-site sources would be processed. However, fuel processing capability at INEL was shut down in 1992 and some SNF remains in storage today.

In June 1996, DOE assembled a group of specialists in SNF matters - the INEEL Spent Nuclear Fuel Task Team - to evaluate the situation at INEEL and develop a technical strategy for INEEL SNF, including stabilization (as required), near term storage, packaging, transport and ultimate disposal. The Team's work is intended to supplement existing baseline and ten-year plans by providing conceptual strategies, identifying necessary further study, and other actions.

This is a report of the evaluations, findings and recommendations of the Task Team.

The INEEL Spent Nuclear Fuel Task Team

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Robert Pahl (Argonne National Laboratory) – Co-Chairman

Jack DeVine (Polestar Applied Technology) – Co-Chairman

Hugh Benton (Framatome Cogema Fuels) – OCRWM Waste Package Development

Denny Fillmore (Lockheed Martin Idaho Technologies Co.) – Fuels Behavior and Peformance

Leroy Lewis (Lockheed Martin Idaho Technologies Co.) – Fuels Behavior and Peformance

Phyllis Lovett (TRW) – Repository Design and Requirements

Richard Murphy (Westinghouse Savannah River Co) – Fuels Behavior and Peformance

Roger McCormack (Duke Engineering) – Fuels Behavior and Peformance

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Section 1

The Evaluation, in Overview

1.1 Background

About 565 cubic meters of spent nuclear fuel (SNF), more than one-third by volume of the total SNF inventory managed by DOE, is in storage at the INEEL. More significant than its quantity is the diversity and complexity of the SNF at INEEL – it comprises more than 250 individual fuel types, including various fuel materials and configurations, with fissile material enrichments ranging from depleted to 97%. Much is intact, but some SNF is in a degraded condition[1] ranging from minor cladding breaches to completely declad, crushed or melted fuel elements. The SNF is stored in various facilities, both wet and dry, at locations around the site.

For some time, work has been proceeding at INEEL to achieve and maintain safe SNF storage. Current plans focus on onsite needs, and include three central elements:

1. Resolution of near-term vulnerabilities, on a priority basis. Some of the storage facilities are outdated and require upgrading or phase-out. Dry storage is preferred to wet storage (most of the SNF is currently stored wet) to minimize further degradation of the fuel, and because it is much lower in operating costs.
2. Consolidation of storage locations. The INEEL site is very large (890 sq. miles). To simplify and strengthen the management of SNF and to permit release of portions of the site for other use, the SNF will be consolidated onsite at a few, well designed storage facilities.
3. Achieving dry storage in transportable packages. An integral part of the consolidation effort is to ensure the transportability of the SNF, so that it can be shipped out of Idaho when a permanent repository, or a suitable national interim storage facility, becomes available.

The existing INEEL SNF Management Plan, prepared by the INEEL staff in 1995, incorporates the above elements. Building on this existing planning base, DOE seeks to examine the broader issues attendant to ultimate disposal of the SNF and to put in place a technical strategy for packaging, storing, transporting, and ultimate disposal of the INEEL SNF. The INEEL SNF Task Team was engaged to develop such a strategy.

1.2 The Task Team Evaluation

The Task Team assembled by DOE included specialists, from public and private sectors, with broad experience and capability in the technical matters at issue. A synopsis of the Team members' professional credentials is provided as an appendix to this report.

The Team was established in June 1996 and worked through the balance of the year. The Team's charter was to evaluate the situation, examine alternatives, and develop for DOE consideration a path forward for the

[1] Some of the SNF is stored disassembled, or otherwise mechanically or chemically degraded, as a direct consequence of long-term storage or of the examinations and experimental work performed at INEEL. Others, such as the TMI-2 fuel, were received in a degraded condition.

INEEL SNF.[2] They compiled and evaluated SNF technical information, met with INEEL and other personnel responsible for SNF management, conducted scoping studies and quantitative analyses in key areas (such as repository criticality potential and repository performance of representative fuel types), and reached consensus on the findings and recommendations reported herein.

In the course of their work, the Team adopted several ground rules and assumptions. Among the more important:

- Road ready packaging

  DOE's objective at INEEL has been to achieve a road ready[3] storage condition. For the purpose of this evaluation, SNF is considered to be road ready once it has been conditioned (if required) and packaged, in a configuration suitable for over-the-road transportation and likely to meet repository handling and disposal requirements, with only reasonable and conventional shipping preparations. For example, transfer of the packaged SNF to a shipping cask at the time of shipment would be consistent with the "road ready" concept; major treatment or reconfiguration of the SNF at the time of shipment would not meet the road ready objective.

- Reliance on proven technology

  As a matter of technical philosophy, the Team has pursued engineering approaches that are rooted in demonstrated, commercially available technology and equipment, or modest extrapolations from proven technology.

  Similarly, the Team considered standardized solutions (e.g., for SNF packaging) to the degree practical. It is clear that a "one size fits all" universal packaging approach, while hypothetically possible, would be neither practical nor cost-effective.

- Compliance with regulatory requirements

  In all cases, the Team has attempted to construct a path forward that can meet both existing and anticipated regulatory requirements such as DOT, NRC, IAEA. DOE is committed to the phase-in of NRC licensing of new facilities.

  The Team notes that there is some uncertainty here. In the final analysis, compliance will be a matter of technical detail that is not available at this conceptual stage. Further, some requirements are as yet undefined (e.g., translation of NRC requirements to DOE facilities different from those presently licensed by NRC), or are subject to change in the years ahead. The Team has attempted to apply reasonable judgments in these cases.

- Repository availability

  For the purposes of this study, it is assumed that the repository will be operational in 2010 as currently projected, and that its configuration, and related technical requirements applicable to the SNF to be disposed there, will be as currently planned. Acceptance of DOE-owned SNF is anticipated to begin in 2015.

- Potential funding limitations

  Substantial cuts in funding have taken place in recent years. The level of funding available in future years may dictate both schedule and priorities. In this evaluation, the Team has considered cost-effectiveness to be a primary factor in selecting path forward concepts.

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[2] The Naval SNF at the INEEL was not addressed because the Department of the Navy is managing its own SNF. The aluminum-based SNF currently located at INEEL will be sent to the Savannah River Site for preparation for disposal in accordance with the Programmatic EIS.

[3] Road ready is defined in reference A.

The Team's evaluation, findings and recomendations are all of a conceptual nature. Given the complexities and uncertainties (technical and otherwise) inherent in this matter, it is not possible for an evaluation such as this one to develop definitive, final, precise conclusions. The results presented here reflect the Team members' best judgement, validated to the degree possible by analysis and peer review. These results should not be viewed as rigid conclusions but as a starting point for more definitive planning.

1.3 Findings and Recommendations

Through the course of its investigations, the Team reached consensus on many points. These are presented in this report as "findings" (information considered by the Team to be important to decision-makers) and "recommendations" (near-term DOE actions to implement the proposed path forward). These are presented throughout this report and compiled in Section 5. They are restated here, in synopsis form.

Findings:

- Presently planned actions at INEEL to resolve near-term SNF storage vulnerabilities are appropriate.
- In many cases, it may be acceptable (and consistent with the "road ready" concept) to utilize the existing SNF canisters for onsite staging.
- Most (over 90%) of the INEEL SNF will not require treatment as a prerequisite for repository disposal; properly designed and conservatively analyzed packaging can provide the necessary confidence that the disposed material will be critically safe over the long term.
- Sodium-bonded fuels (approximately 3% by volume of the INEEL inventory) are not considered suitable for repository disposal and therefore must be treated.
- Characterization requirements for the DOE SNF are not yet completely defined. SNF characterization should be limited to that information necessary to permit reasonable prediction of performance in storage, transportation and repository disposal. Adequate information for performance based characterization already is available for the bulk of the INEEL SNF, and existing facilities at INEEL seem adequate to perform additional characterization which may be needed.
- Repository criticality safety for high- and medium-enriched fuels (HEU[4] and MEU[5]) can be achieved through proper package design. Design features for criticality control can include limitations in allowable neutronic reactivity arangements and/or incorporation of neutron poison or moderator exclusion materials. Simple, standardized and relatively small cylindrical canisters (nominal diameters of 10, 17, and 24 inches have been evaluated) appear to provide adequate criticality safety and packaging flexibility for the INEEL HEU and MEU.
- Co-disposal of packaged HEU or MEU with high-level waste is a simple and conservative way to achieve long-term repository criticality safety.
- The most cost-effective approach for repository disposal of several small quantities of SNF is likely to be combining several SNF types in a single package. Performance assessments for such packages should be based on conservative bounding assumptions, with characterization requirements limited accordingly.
- Repository disposal of INEEL SNF would be only a small contributer to the overall projected peak annual dose to persons in the accessible environment.

Recommendations:

[4] HEU fuel has Uranium 235 greater than 20% of the total uranium or significant quantities of Plutonium.
[5] MEU fuel has uranium 235 between 5% and 20% of the total uranium.

The Team recommends that DOE:

- Begin the development of standard canister designs suitable for disposal of HEU and MEU fuel.
- Continue the working interfaces between EM and RW.
- Continue to refine characterization requirements for SNF, based strictly on the need to determine SNF performance.
- Engage NRC in the refinement of this path forward, particularly those aspects that will ultimately require regulator concurrence. These include: SNF characterization and analysis strategy, criticality analysis methods, and the development of safe packaging concepts.
- Continue more extensive evaluations, including respository criticality evaluations and more detailed performance assessments for the INEEL SNF, using refined inputs.
- Conduct repository evaluations for aggregate packages of selected small quantities INEEL SNF.
- Continue the technical work needed to qualify the Electro-metallurgical process, or an alternative process, for treatment of the sodium-bonded fuel.

Section 2

SNF Management at INEEL – Defining the Problem

Developing a viable strategy for dealing with this SNF requires a thorough understanding of the fuel, the storage and handling facilities available onsite, and the primary technical and institutional issues and constraints that must be considered. This section summarizes those considerations.

2.1 Fuels

The DOE SNF is a small part (3% by mass, 6% by volume) of the total SNF that is to be disposed in a geologic repository. This relatively small amount presents a significant challenge because it includes many fuel types of varying fissile content and structural characteristics. Much of the DOE SNF differs significantly from commercial fuel. These differences need to be considered in determining repository performance.

The INEEL manages approximately 38% (by volume) of the DOE fuel, (Figure 2.1-1). Additionally, INEEL is scheduled to receive future SNF shipments from seven DOE facilities, 20 U. S. universities, nine non-DOE research facilities, and 19 foreign countries. The projected total INEEL SNF inventory will encompass about 60% (by volume) when all shipments have been received.

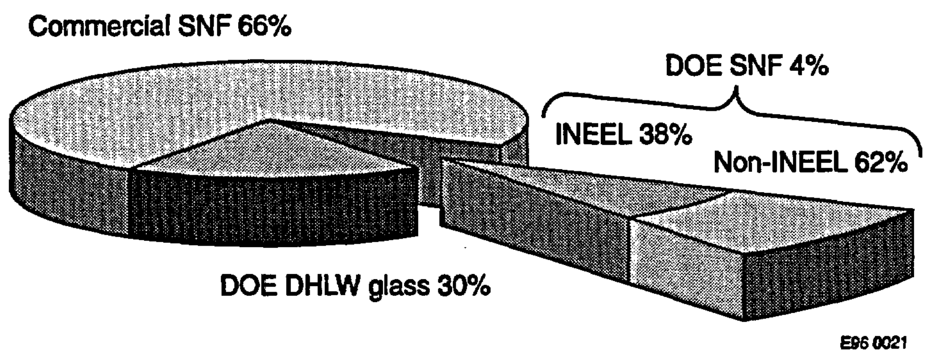


Figure 2.1-1 Volume of DOE SNF compared to commercial SNF and HLW glass and comparative volume at the INEEL, for 1997 inventory.

The overall inventory at the INEEL includes about 250 specific fuel types of which ten constitute more than 90% of the total by both mass and volume. Many of the other fuel types comprise only very small quantities of fuel. Ninety types of INEEL SNF consist of less than six fuel handling units each.

For purposes of analysis and discussion the Task Team classified the INEEL SNF into seven broad categories (16 individual groups) based on those characteristics most important to interim storage and/or long-term repository performance. These characteristics are: fissile material content, which affects nuclear criticality concerns; chemical composition, chemical reactivity, and solubility, which affects repository analysis and performance; physical condition and cladding integrity (i.e., intact, declad, or severely breached cladding) which affect handling and packaging requirements. Table 2.1-1 is a summary of the important characteristics of each INEEL SNF group. The relative quantities of fuels in each group is shown graphically in Figure 2.1-2. Appendix A lists the specific fuels comprising each group.

Category (Volume Percent)		Group		Quantity	Representative Fuel
I	Intact oxide fuel (7.2%)	1	LEU	31 fuel types 30 cubic meters 76.8 MTHM	Commercial
		2	MEU	8 fuel types 1.4 cubic meters 4 MTHM	PBF & EBWR
		3	HEU	21 fuel types 9.3 cubic meters 8.74 MTHM	Shippingport PWR
II	Disrupted oxide fuel (28%)	4	LEU	34 fuel types 145.5 cubic meters 87.5 MTHM	TMI-2
		5	HEU	44 fuel types 23.6 cubic meters 6.2 MTHM	TORY
III	Uranium zirconium hydride fuel (1.4%)	6	MEU	6.6 cubic meters 1.8 MTHM	TRIGA MEU
		7	HEU	1.3 cubic meters 0.2 MTHM	TRIGA HEU
IV	Uranium metal and uranium alloy fuel (0.5%)	8	LEU	14 fuel types 0.8 cubic meters 2 MTHM	HWCTR
		9	HEU	6 fuel types 2 cubic meters 3.9 MTHM	Fermi driver
V	Uranium carbide fuel (42%)	10	High integrity particles	196 cubic meters 23.4 MTHM	FSVR
		11	Lower integrity particles	7 fuel types 35 cubic meters 3 MTHM	Peachbottom graphite
		12	Metal clad	2 fuel types 5 cubic meters 0.06 MTHM	SRE
VI	Intact uranium and thorium oxide fuel (11%)	13	U-233 thorium	51.5 cubic meters 39 MTHM	Shippingport LWBR
VII	Other (10%)	14	Metalic sodium	33 fuel types 14.6 cubic meters 60 MTHM	Fermi blanket, EBR-II
		15	Al clad	14 fuel types 37.5 cubic meters 3.4 MTHM	ATR
		16	Other	5 fuel types 4.3 cubic meters 0.2 MTHM	MSRE

Table 2.1-1 INEEL SNF groups and categories characteristic.

	Decription	Fissile Loading	Physical Condition
	Uranium oxide, some mixed with other oxides. Clad with zirconium or stainless steel	Low (<5%) U-235 enrichment	Intact cladding Some assemblies dismantled
		Medium (5-20%) U-235	
		High (>20%) U-235 and/or Pu	
		Low (<5%) U-235 enrichment	Disrupted cladding or melted fuel and cladding
		High (>20%) U-235 and/or Pu enrichment	Ground up unclad ceramic fuel, disrupted cladding, metallurgical mounts
	Uranium zirconium hydride, clad with stainless steel, zirconium, or aluminum	Medium (5-20%) U-235 enrichment	Mostly Intact Cladding
		High (>20%) U-235 enrichment	
	Uranium metal or uranium alloys of zirconium or molybdenum with various claddings	Low (<5%) U-235 enrichment	
		High (>20%) U-235 enrichment	
	Uranium carbide, thorium carbide in separate or mixed condition. Formed into particles and placed in a graphite matrix	High (>20%) U-235 & U-233 enrichment	Intact particles in intact assemblies
			Mostly intact
	Uranium and thorium ceramic oxide	High (>20%) U-233 enrichment	Intact cladding and assemblies
	Uranium metal containing metallic sodium	Varied	Mostly intact cladding
	Aluminum based	High (>20%) U-235 enrichment	
	Other	High	Various

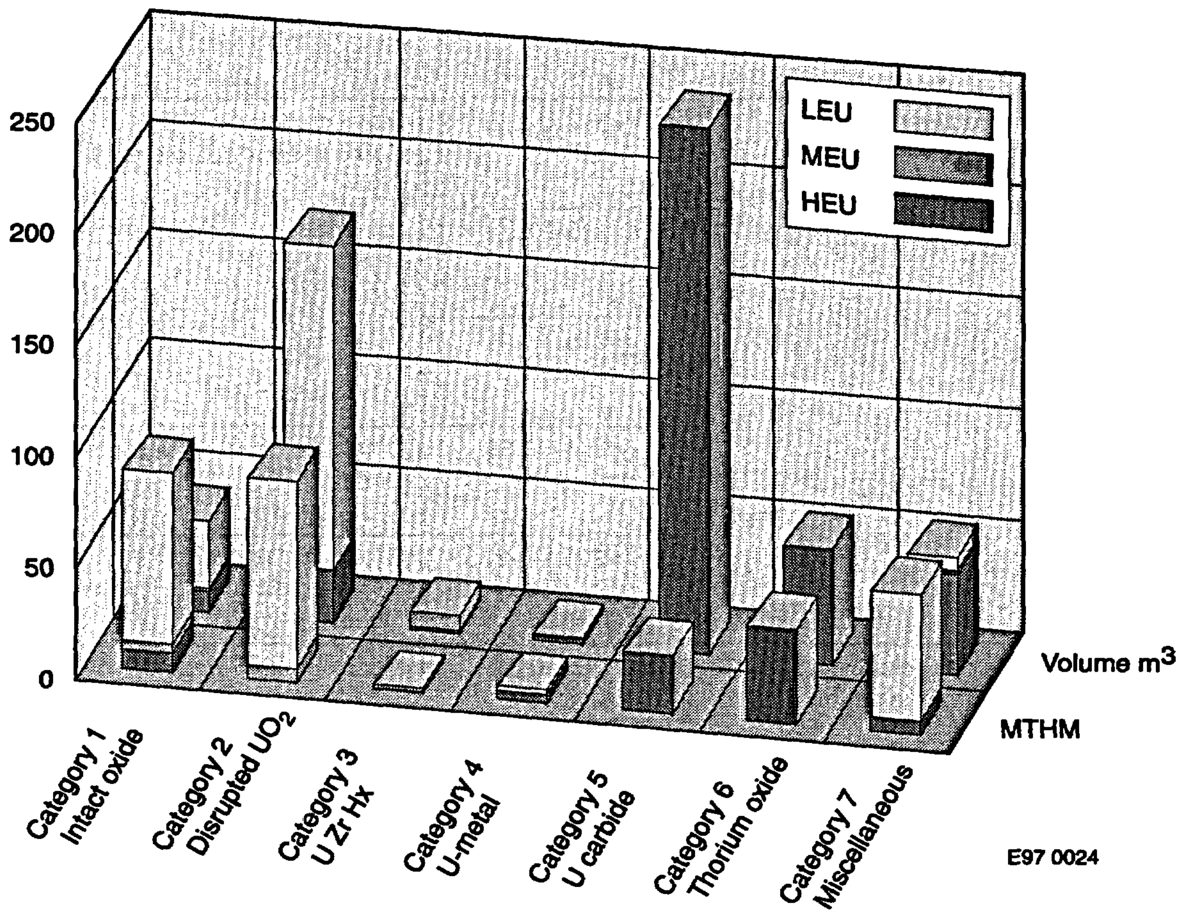


***Figure 2.1-2** INEEL SNF inventory by category.*

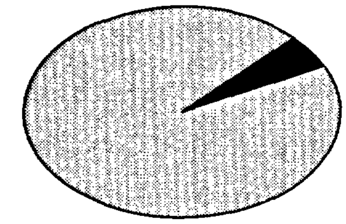
Some of the fuel types contain materials that may present challenges for direct disposal in a geologic repository, (metallic uranium, metal carbides, and fluoride salt). Some of the fuels are known not to be intact, composed of degraded or powdered material, or contain sufficient moisture to produce hydrogen gas by radiolysis.

The fuels in storage at the INEEL have been characterized to varying degrees. For most of the newer fuels, design and operating information is readily available. However, little information is available on many fuel types, particularly the older fuels.

The fuel fact sheets on the following pages describe representative fuels in the larger groups. These fact sheets are intended to provide perspective regarding the spectrum of SNF material in storage at INEEL, particularly for the more significant fuel types. Only groups with a significant quantity of fuel are represented by a fact sheet. Those groups with an insignificant amount of fuel are not represented.

Category I, Group 1 Fuel
INTACT URANIUM OXIDE (LEU)

The INEEL inventory originated from operating commercial utility master plants, and is identical to that used in most of the existing commercial utilities reactors. It is made by hot pressing uranium oxide into pellets. The pellets are loaded into Zircaloy or stainless steel tubes and the tubes are made into assemblies, typically a 14- or 15-tube square array. There are 31 types of commercial fuel in Group 1.



Group 1 volume % of INEEL total

Repository Disposal Considerations

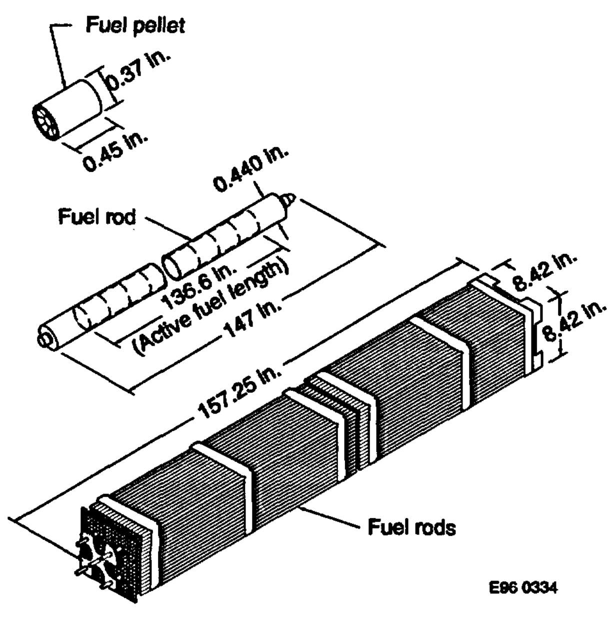
- The consolidated fuel requires additional packaging analysis.
- This fuel should be handled consistent with other commercial SNF.

Group 1 representative fuel: Westinghouse 15 x 15 PWR

Fuel Description

The commercial fuel was brought to the INEEL for examination or testing programs. Some of the fuel has been reconfigured for the Dry Rod Consolidation Test (DRCT) program. The reconfiguration involved consolidating the fuel by removing the rods and placing them into canisters that have twice as many rods as an assembly. The examination or testing program involved taking some of the assemblies and the rods apart for post-irradiation examination.

Physical Characteristics



PWR Assembly

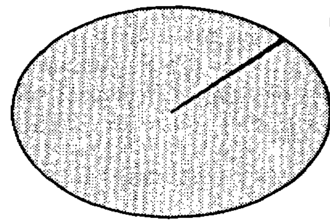
- Approximate dimensions –
  Assembly: 8" X 8" X 160" long
- Materials –
  Uranium oxide pellets clad with zircaloy or stainless steel
- Uranium loading –
  456 kg per assembly BOL
  912 kg per DRCT assembly BOL
- Uranium enrichment –
  4% BOL
- Condition-
  Rods are intact, some assemblies have been reconfigured
- Storage configuration –
  87 assemblies stored dry in canisters
  431 assemblies stored bare in water
- Location –
  TAN dry storage pad
  TAN pool
  Other DOE and Non-DOE sites
- Storage condition –
  Good

Quantity Summary

	PWR	Group 1 Totals
Number of assemblies -	163	519
Volume, m^3 -	27	30
Mass, MTHM (EOL) -	74	76.8

Category I Group 2 Fuel
INTACT URANIUM OXIDE (MEU)

This fuel is uranium oxide that is medium enriched, in which the enrichment is between 5% and 20%. The fuel is similar to commercial fuel which is made by pressing the uranium oxide into pellets. The pellets are loaded into zircaloy or stainless steel tubes. The fuel is intact. There are 8 types of fuel in Group 2.



Group 2 volume % of INEEL total

Repository Disposal Considerations

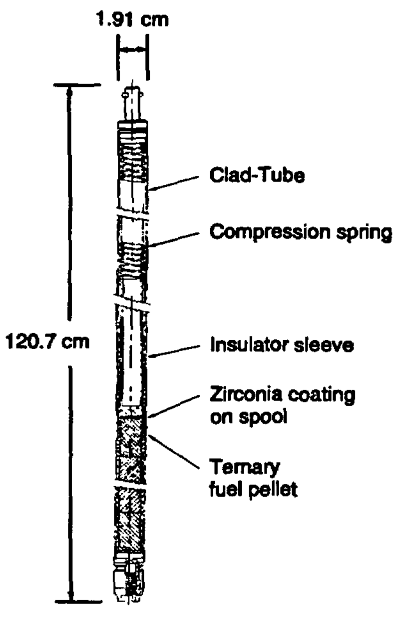
- Criticality potential of MEU in the repository

Group 2 representative fuel: Power Burst Facility (PBF)

Fuel Description

The PBF fuel element consists of an 18.5 percent enriched pelletized ternary oxide ceramic fuel (UO_2-ZrO_2-CaO) clad in 304L stainless steel. The pellets are surrounded by a helium gas annulus and an insulator sleeve of ZrO_2-CaO before being inserted into the 304L stainless steel tubes. The fuel matrix is extremely stable in all environments.

Physical Characteristics



PBF Fuel Rod

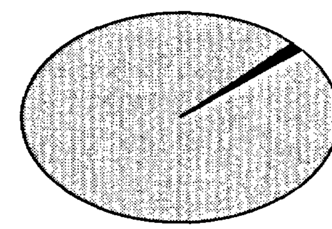
- Approximate dimensions –
  Rod: 0.8" OD X 48" long
  Fuel Pellet: 0.6" dia.
- Materials –
  Fuel: UO_2-ZrO_2-CaO
  Sleeve: ZrO_2-CaO
  Cladding: 304L stainless steel
- Uranium Loading –
  0.24 kg per assembly BOL
- Uranium enrichment –
  18.5% BOL
- Condition –
  Good
- Storage configuration –
  Intact rods, stored wet
- Location –
  2,427 irradiated elements in PBF reactor and canal.
- Storage condition –
  Good

Quantity Summary

	PBF	Group 2 Totals
Number of assemblies -	2,427	2,542
Volume, m^3 -	0.84	1.4
Mass, MTHM (EOL) -	0.56	4.02

Category I, Group 3 Fuel
INTACT URANIUM OXIDE (HEU)

This group contains intact SNF assemblies that contain highly enriched uranium oxide or mixed uranium and plutonium oxides. Some of the oxides are sintered and some are more stable ceramics. All rods or plates are intact; however, some of the assemblies have been disassembled. There are 22 fuel types in Group 3.



Group 3 volume % of INEEL total

Repository Disposal Considerations

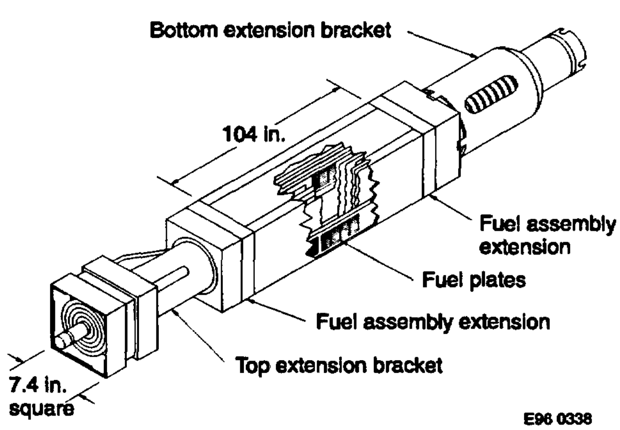
- Criticality potential of HEU in the repository

Group 3 representative fuel: Shippingport Pressurized Water Reactor (PWR)

Fuel Description

The Shippingport PWR Core 2, Seed 2 fuel was made by a process of cold pressing and sintering a free-flowing powder of the fuel oxide mixture into rectangular wafers. Calcium oxide was added as a stabilizing compound to the fuel meat. The wafers were inserted into the individual compartments of the receptacle Zircaloy-4 plate and pressure bonded to Zircaloy-4 plates. Seed 2 has 1,000 wafers per plate, fifteen plates per subassembly, and 4 subassemblies per assembly.

Physical Characteristics



PWR Assembly

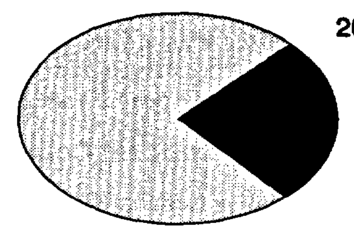
- Approximate dimensions-
  Wafer: 1.5 in. x 0.25 in. x 0.1-in. thick
  Plate: 2.0 in. x 0.4 in. by 72-in. long
  Assembly: 7.4 in. x 7.4 in. x 104 in.
- Materials-
  Fuel: uranium, zirconium, oxide ceramic
  Cladding: Zircaloy-4
- Uranium loading-
  21 kg per subassembly BOL
- Uranium enrichment-
  93% BOL
- Condition-
  Intact assembly
- Storage configuration-
  Intact assemblies
- Location-
  Stored wet in ICPP-666
- Storage condition-
  Good

Quantity Summary

	Shippingport PWR	Group 3 Totals
Number of assemblies -	39	729
Volume, m^3 -	3.64	9.35
Mass, MTHM (EOL) -	1.15	8.74

Category II, Group 4 Fuel
DISRUPTED URANIUM OXIDE (LEU)

This group is composed of low-enriched uranium oxide fuels that have been disrupted from their original configuration. The cladding has been severely disrupted through operational activities, testing, accidents, or destructive examination. There are 34 fuel types in Group 4.



Group 4 volume % of INEEL total

Repository Disposal Considerations

- Material is composed of debris and some particles.
- Material needs to be dried.
- Leaching of fission products and actinides in repository environment is not well understood.
- Potential for leaching of cadmium from the disrupted control rods.
- No individual canister content characterization data

Group 4 representative fuel: Three Mile Island unit 2 (TMI-2)

Fuel Description

The fuel was a typical commercial pressurized water nuclear reactor fuel until it melted in a reactor accident. It now consists of material with sizes ranging from fines to nearly intact assemblies, some of which have been melted and cooled. The fuel debris was placed into three types of stainless steel canisters: filter canisters that contain the fines, knockout canisters that contain gravel consistency material, and fuel canisters that contain large pieces of melted or unaffected assemblies. The material has been extensively characterized as part of the TMI-2 reactor accident analysis.

Physical Characteristics

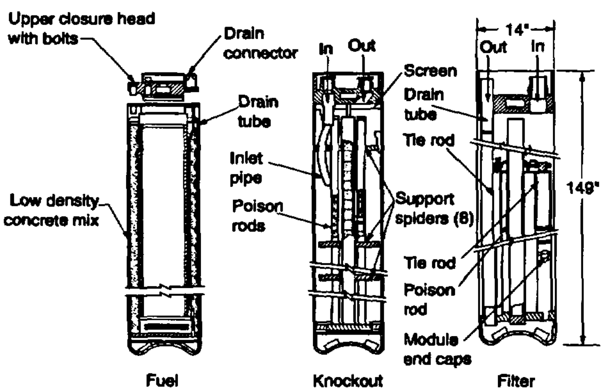


Diagram of the the TMI-2 canister types

J96 0200

- Approximate dimensions –
  Canister: 14-in. dia. X 149-in. long
- Materials –
  Canister is Stainless Steel
  Contents are rubble from melted products of uranium oxide, Zircaloy, and control rods.
  Canisters contain concrete
  Canisters are full of water
- Uranium loading –
  No per canister data, 81,500 kg total
- Uranium enrichment – 4% BOL
- Condition – Rubble in canisters
- Storage configuration
  Water filled steel canisters
- Location – TAN pool
- Storage condition –
  Stable - canisters vented to remove hydrogen

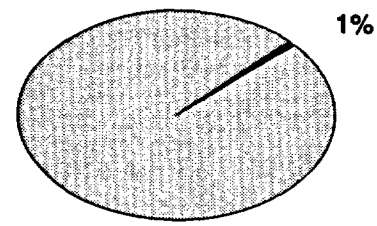
Quantity Summary

	TMI-2	Group 4 Totals
Number of canisters -	344	462
Volume, m^3 -	129	145.5
Mass, MTHM (EOL) -	81.6	87.5

FUEL FACTS

Category III, Group 6 Fuel
URANIUM ZIRCONIUM HYDRIDE (MEU)

This group is composed of medium enrichment (5-20%) uranium/zirconium hydride SNF that is clad in aluminum or stainless steel. The sources are primarily universities, foreign countries, and some non-DOE facilities. Only MEU TRIGA fuel is in Group 6.



Group 6 volume % of INEEL total

Repository Disposal Issues

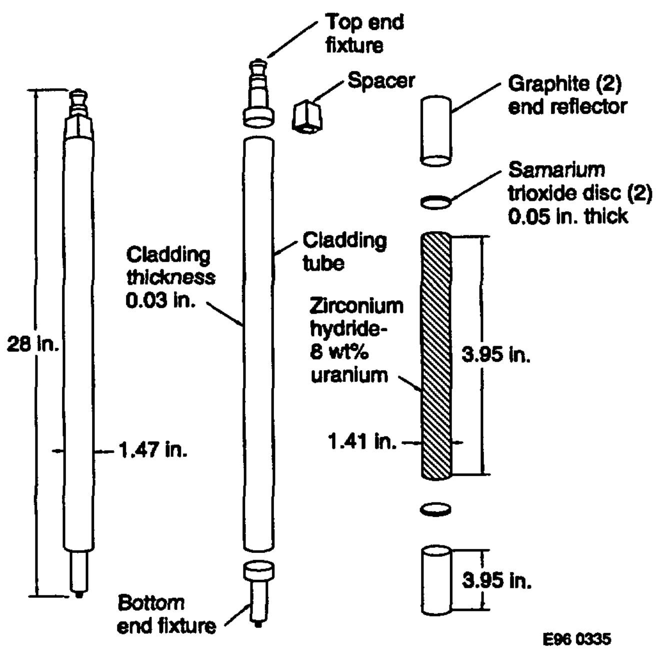
- Criticality potential of MEU in the repository.
- Leaching of fuel meat in repository conditions is not known.
- Potential diffusion of hydrogen from fuel meat in repository.

Group 6 representative fuel: MEU TRIGA

Fuel Description

The TRIGA fuel is made from uranium/zirconium hydride that was formed into either solid or hollow rods that are 1.4 inches in diameter and 14 or 15 inches long. Graphite plugs and samarium discs were placed on the end of the fuel. After cladding and addition of end pieces the rods are 28 inches long. The rods are not placed into assemblies, but each is handled separately.

Physical Characteristics



TRIGA Rod

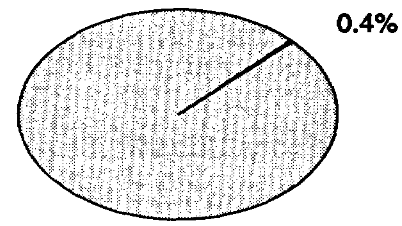
- Approximate dimensions –
  Rod: 1.5-in. dia. x 28-in. long
- Materials –
  Uranium/zirconium hydride clad with SST, Zr, or Al
- Uranium loading –
  0.19 kg uranium per rod BOL
- Uranium enrichment – 19.9% for MEU BOL
- Condition – Intact rods
- Storage configuration-
  Most is stored bare in wet pools
  Some is stored bare in dry storage
  One type is canned
- Location –
  ICPP-603 wet pool: 854 rods
  ANL-W TREAT: 40 rods
  ANL-W HFEF: 2 rods
  6,748 rods at Universities and test reactors in the US and in foreign countries
- Storage condition –
  Some aluminum cladding is corroding, possible physical damage to the fuel rods.

Quantity Summary

	TRIGA at INEEL	Group 6 Totals (all TRIGA)
Number of units -	896	7,644
Volume, m^3 -	0.72	6.63
Mass, MTHM (EOL) -	0.16	1.76

Category IV, Group 9 Fuel
URANIUM METAL OR ALLOY (HEU)

This group is composed of highly enriched uranium metal or uranium alloy fuel with intact cladding. The uranium metal has not been exposed to water. There are six fuel types in Group 9.



Group 9 volume % of INEEL total

Repository Disposal Considerations

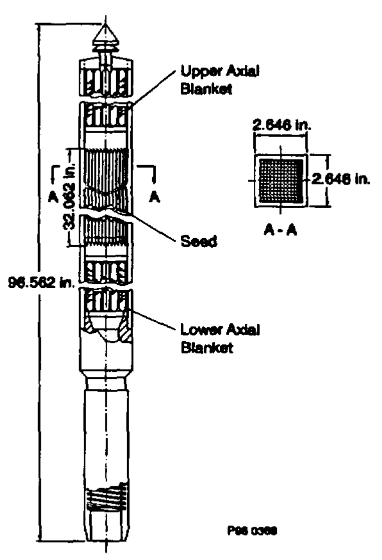
- Criticality potential of HEU in the repository.
- Potential chemical reactivity of uranium metal.

Group 9 representative fuel: FERMI Core 1 & 2

Fuel Description

The FERMI fuel was made of a uranium/molybdenum alloy that is formed into pins or rods that were clad with zirconium by co-extrusion. The pins were placed into assemblies. The fuel assembly consisted of three distinct sections, the upper LEU axial blanket, the HEU driver and the lower LEU axial blanket. When the fuel assemblies were removed from the reactor the assemblies were sectioned to remove the axial blanket sections from the fuel section. After approximately 14 years in storage the fuel sections were disassembled and the pins repackaged into aluminum cans. The Fermi fuel that was segmented for post irradiation examination, damaged by melting, or declad is not included in Group 9, but is included in Group 5.

Physical Characteristics



Fermi Fuel Subassembly

- Approximate dimensions –
  Pins: 0.16-in. dia. x 33-in. long
  Assembly 2.6 in. x 2.6 in. x 36-in. long
- Materials –
  Uranium -10% molybdenum alloy clad in zirconium
- Uranium loading – 0.134 kg per rod BOL
- Uranium enrichment – 25.7% BOL
- Condition –
  most is intact
  disrupted SNF is not in this group
- Storage configuration –
  Fuel assemblies have been disassembled and the pins are stored 140 per aluminum can, 3.2-in. diameter x 42.4-in. long.
- Location – Stored wet in CPP-666
- Storage condition – Cans are corroding
  Fuel pin cladding is intact

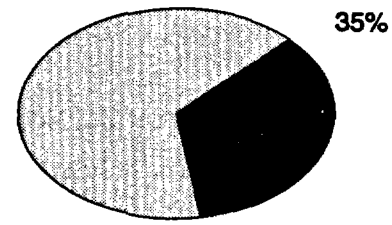
Quantity Summary

	Fermi	Group 9 Totals
Number of units -	205 cans, 140 pins ea.	257
Volume, m^3 -	1.2	2.02
Mass, MTHM (EOL) -	3.78	3.9

FUEL FACTS

Category V, Group 10 Fuel
URANIUM AND THORIUM CARBIDE (HEU)

All of the fuel in group 10 is Fort St. Vrain Reactor (FSVR) fuel. DOE has taken ownership of the fuel presently stored in Colorado as well as the fuel at the INEEL.



Group 10 volume % of INEEL total

Repository Disposal Considerations

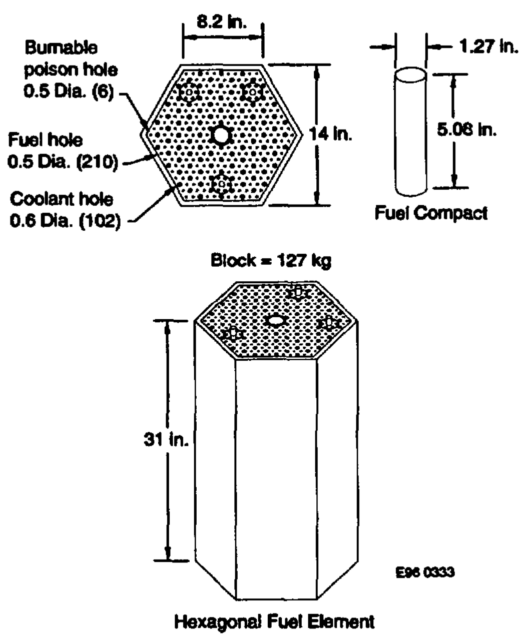
- Criticality potential of HEU in the repository.
- Reactivity of metallic carbides with water.
- Potential combustibility of graphite.
- Uranium 233 and thorium effects on repository performance.

Group 10 representative fuel: Fort St. Vrain Reactor (FSVR) Fuel

Fuel Description

The FSVR fuel is a graphite based fuel that was used only in the FSVR. An assembly is composed of a hexagonal shaped graphite block drilled with 102 coolant holes and 210 fuel holes. The fuel is made of highly enriched uranium carbide and thorium carbide spheres coated with layers of pyrolitic carbon followed by a coating of silicon carbide, which is very durable, and an outer pyrolitic coating. The fuel spheres are sintered with carbon and formed into rods, called compacts, and then stacked into fuel holes. A fully loaded graphite block holds 3,132 fuel compacts.

Physical Characteristics



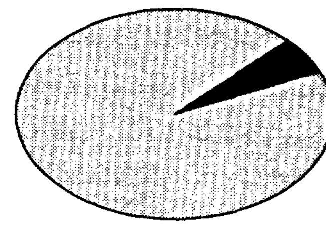
- Approximate dimensions –
  Shape: hexagonal block
  Fuel spheres: 200 and 450 microns
- Materials –
  Graphite block
  Uranium carbide and thorium carbide particles
  Silicon carbide coated
- Uranium loading –
  0.44 kg per assembly BOL
- Thorium loading – 8.5 kg per assembly BOL
- Uranium enrichment –
  93% uranium 235 BOL, 54% EOL
  0% uranium 233 BOL, 29% EOL
- Condition – Intact assemblies
- Storage configuration –
  Idaho: 4 intact elements per can
  Colorado: 6 intact elements per can
- Location –
  ICPP-603 IFSF: 744, stored dry
  Colorado: 1,464, stored dry
- Storage condition – Good

Quantity Summary

	FSVR at INEL	Group 10 Totals (all FSVR fuel)
Number of elements -	744	2,208
Volume, m^3 -	66	196.47
Mass, MTHM (EOL) -	8.6	23.4

Category V, Group 11 Fuel
URANIUM AND THORIUM CARBIDE (HEU)

This group is composed of fuel made of mixed uranium carbide and thorium carbide coated particles that are dispersed in a graphite matrix material. There are seven fuel types in Group 11.



Group 11 volume % of INEEL total

Repository Disposal Considerations

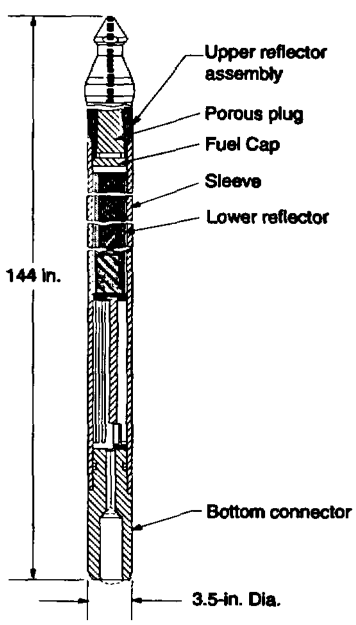
- Criticality potential of HEU in the repository.
- Chemical reactivity of metal carbides with water.
- Potential combustibility of graphite.
- Uranium 233 and thorium effects on repository performance.

Group 11 representative fuel: Peachbottom Core 1 and 2

Fuel Description

The Peachbottom Cores 1 & 2 are a graphite based fuel that is made of mixed uranium carbide and thorium carbide that is made into particles ranging from 295 to 630 microns in diameter and coated with pyrolytic carbon. The particles are formed into annular compacts 2.98 inches high with a center hole diameter of 1.75 inches and an outside diameter of 2.7 inches. The compacts are stacked on a 30 inch long graphite spine. Three units make up the 90 inch long fuel section. An annular low-permeability graphite sleeve is slipped over the fuel compacts.

Physical Characteristics



Peach bottom assembly J96 0201

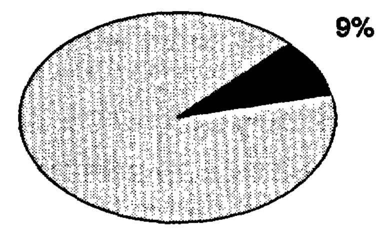
- Approximate dimensions –
  Particle: 295 - 630 micron dia.
  Compact: Flat annular cylinders, 1.7-in. ID, 2.7 in. OD x 3 in. long.
  Assembly: 3.5-in. dia. X 144-in. long
- Materials –
  Mixed U and Th carbide particles,
  Core 1 isotropic pyrolytic coated
  Core 2 coated with anisotropic pyrolytic carbon, both in graphite matrix.
- Uranium loading – 0.312 kg per element BOL
- Thorium loading – 1.79 kg per element BOL
- Uranium enrichment – 93% BOL
- Condition –
  Intact assembly
  Core 1 has up to 70% breached particles
- Storage configuration –
  Core 1 (814 units): canned
  Core 2 (786 units): stored bare
- Location – ICPP dry storage
- Storage condition –
  1st generation facility has moisture in wells

Quantity Summary

	Peachbottom at INEEL	Group 11 Totals
Number of units -	1,600	1,875
Volume, m^3 -	34.09	35.34
Mass, MTHM (EOL) -	2.95	3.03

Category VI, Group 13 Fuel
Uranium Oxide and Thorium Oxide Meat (HEU)

This group is made up entirely of LWBR fuel. The fuel is made of uranium oxide and thorium oxide in a ceramic matrix and is clad in Zircaloy-4 tubing. The cladding is intact.



Group 13 volume % of INEEL total

Repository Disposal Considerations

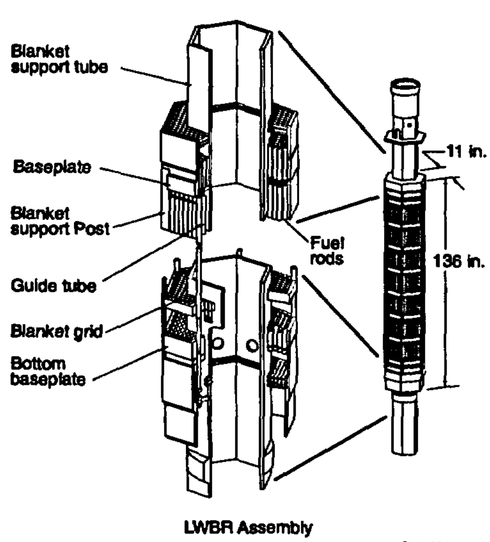
- Criticality potential of HEU in the repository.
- Uranium 233 and thorium effects on repository performance.

Group 13 representative fuel: Shippingport Light Water Breeder Reactor (LWBR)

Fuel Description

The Shippingport LWBR was used to demonstrate the production of fissile uranium 233 from thorium in a water-cooled operating reactor. The fuel was made of uranium oxide, enriched up to 98% in uranium 233 mixed with thorium oxide and made into cylindrically shaped ceramic pellets. The fuel pellets were loaded into 0.3" diameter Zircaloy-4 tubes whose ends are capped and seal welded. These tubes were made into assemblies. The LWBR has four different types of assemblies: 12 seed assemblies used the HEU to produce power, 12 blanket assemblies were used to capture neutrons and convert the thorium to uranium 233, and 9 type IV reflector assemblies and 6 type V reflector assemblies were used to reflect neutrons back into the reactor. This group does not include the nine canisters of disrupted LWBR fuel, they are in group 5.

Physical Characteristics



LWBR Assembly

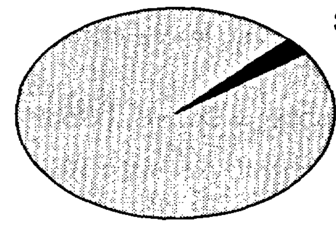
- Approximate dimensions –
  Seed assemblies: hexagonal 11 in. point to point, 136-in. long.
  Blanket assemblies: hexagonal 22 in. point to point, 142-in. long.
  Reflectors: 140-in. long
- Materials –
  Uranium and thorium oxide
  Zircaloy-4 cladding
- Uranium loading – 11.8 kg per assembly BOL
- Thorium loading – 882.6 kg per assembly BOL
- Uranium enrichment –
  94% uranium 233 + 235 EOL
- Condition – Intact assembly
- Storage configuration –
  Stored in stainless steel cans, 25.5-in. dia. x 158-in. long
- Location – Stored dry in CPP-749 2nd generation
- Storage configuration – Good

Quantity Summary

	LWBR	Group 13 Totals
Number of assemblies-	39	39
Volume, m^3 -	52	51.5
Mass, MTHM (EOL) -	39	39

Category VII, Group 14 Fuel
METALLIC SODIUM BONDED (HEU & LEU)

This fuel group consists of fuel that is an alloyed uranium metal with metallic sodium used to bond the fuel meat to the cladding. The sodium must be removed prior to storage in the repository. There are 34 fuel types in Group 14.



Group 14 volume % of INEEL total

Repository Disposal Considerations (if not treated)

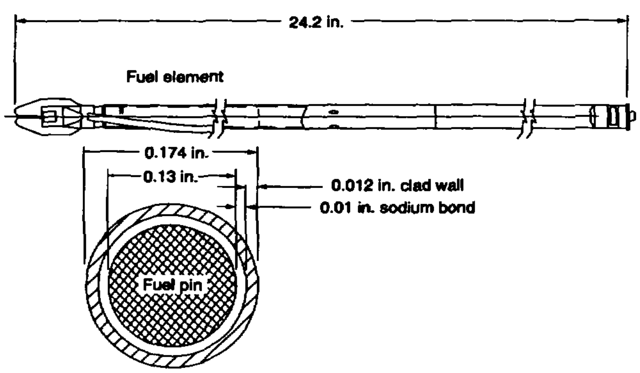
- Criticality potential of HEU in the repository.
- Chemical reactivity of metallic sodium with water.
- Potential chemical reactivity of uranium metal with water.
- Potential need to remove or passivate any uranium hydride.
- Potential drying problems with leaking cans.

Group 14 representative fuel: Experimental Breeder Reactor-II (EBR-II) Driver

Fuel Description

The EBR-II fuel is constructed of uranium metal alloyed with a mixture of metals called fissium, or alloyed with 10% zirconium. The fuel meat is made into elements and clad with stainless steel. Metallic sodium is inserted between the uranium meat and the cladding to improve heat transfer. The elements are made into assemblies. When the assemblies are removed from the reactor they are washed to remove the metallic sodium coolant from the outside of the rods and then disassembled back to rods. The rods at ICPP are in Swagelok® sealed stainless steel cans.

Physical Characteristics



EBR-II Mark-II driver-fuel element

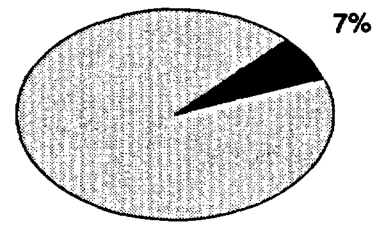
- Approximate dimensions –
  Rods: 0.2-in. dia. x 24 in.
  Can: 2-in. dia. x 25.5-in. long
- Materials –
  Uranium - 5 wt% fissium or
  Uranium - 10 wt% zirconium
  Metallic sodium
  Stainless steel clad
  Stainless steel can
- Uranium loading – 0.05 kg per pin BOL
- Uranium enrichment – 52 to 67% BOL
- Condition – Rods are intact
- Storage configuration –
  11 to 12 elements are inside SST cans
- Location –
  ICPP-603 & -666 (wet): 3,652 cans,
  ANL-W (dry): 31,577 elements
  SRS: 2 canisters
- Storage condition –
  A few cans at ICPP may be leaking

Quantity Summary

	EBR-II (driver and blanket)	Group 14 Totals
Number of units -	71,751 (rods)	72,806
Volume, m^3 -	7.75	14.55
Mass, MTHM (EOL) -	25.36	59.98

Category VII, Group 15 Fuel
ALUMINUM CLAD (HEU)

This fuel is uranium or uranium/aluminum alloy clad in aluminum. Aluminum clad fuel is planned to be sent to the Savannah River Site per the INEEL SNF EIS ROD. There are 14 fuel types in Group 15.



Group 15 volume % of INEEL total

Repository Disposal Considerations

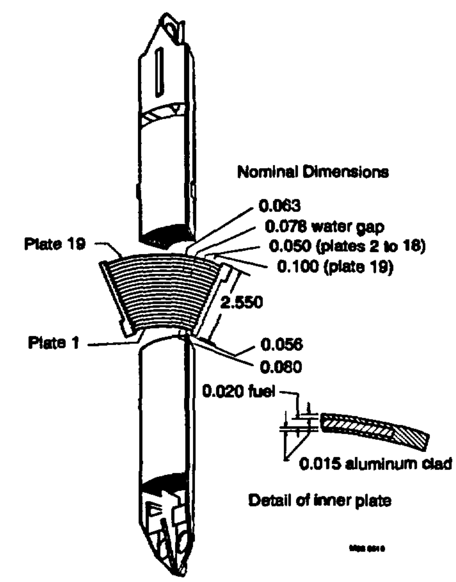
- May degrade rapidly in repository environment
- Criticality potential of HEU in the repository

Group 15 Representative Fuel: Advanced Test Reactor (ATR)

Fuel Description

The ATR fuel meat consists of UAl_x, boron carbide and aluminum particles mixed together and pressed into a 0.015-in. thick plate. The fuel plates are clad with a Type 6061 aluminum foil. The Mark-VII fuel element is made up of 19 concentric fuel plates held together with two nonfueled aluminum side plates. Boron has been added to the fuel plates to act as a burnable poison. The uranium and poison loadings are varied among the fuel plates giving a total U-235 loading of 1075 grams per fuel element.

Physical Characteristics



ATR Fuel

- Approximate dimensions –
  Assembly: 40-in. long, trapezoidal cross-section
  Total element length: 50 in.
- Materials –
  Fuel: UAl_x, boron carbide, and aluminum
  Cladding: 6061 Aluminum
- Uranium loading –
  1075 grams per fuel element
- Uranium enrichment –
  93%
- Storage configuration –
  Stored bare in ICPP water basins
- Location –
  ICPP-603: 128 assemblies
  ICPP-666: 960 assemblies
  TRA-670: 2,964 assemblies
- Storage condition –
  ICPP-603: severe corrosion and pitting, designated "heavy leakers"
  ICPP-666: good

Quantity Summary

	ATR fuel	Group 15 Totals
Number of units	4,052	4,478
Volume, m^3 -	35.75	37.54
Mass, MTHM (EOL) -	3.31	3.43

2.2 Facilities

There are facilities at the INEEL for handling, processing, packaging, and characterizing SNF. These include wet and dry storage facilities and hot cells ranging from relatively small analytical chemistry and metallurgy cells to large processing and operational cells. Table 2.2-1 summarizes the SNF facilities at the INEEL. The storage facilities are at various INEEL locations, as shown on the map in Figure 2.2-1.

Wet Storage

The wet storage facilities include several small, at-reactor, fuel storage pools and three large independent storage basins.

The various at-reactor pools have limited storage intended primarily to support operations of the reactors. These include the Advanced Test Reactor (ATR), the Power Burst Facility (PBF), and the Material Test Reactor (MTR). Each of these facilities has wet loading and unloading capability.

The majority of the wet fuel storage is in two basins at the Idaho Chemical Processing Plant (ICPP) and one basin at Test Area North (TAN) facility.

The two ICPP basins were used for the storage of fuels prior to reprocessing. CPP-603 is a first generation facility (in service since 1952) with wet loading and unloading capability and unlined storage basins. This facility is being phased out and will be shut down no later than December 2000 due to identified vulnerabilities. Fuel in storage is being moved to dry storage or into CPP-666. Two of the three pools are already empty and no additional fuel is being received there.

A second-generation stainless steel-lined storage basin utilizing high purity water is located in the Fluorinel and Storage Facility (FAST - CPP-666). This facility has been in service since 1984. The fuel is stored in racks in six interconnected pools. All of the DOE fuels stored in this facility (except for the naval fuel) are to be moved into dry storage by 2015. CPP-666 is a wet loading and unloading facility, but could utilize the associated Fluorinel process cell, with modifications, for dry loading of fuel.

The unlined fuel storage basin at TAN is connected to the TAN hot shop by an underwater transfer canal. Thus, this facility can receive fuel dry, temporarily store it wet, and then reload it into a cask, dry. All fuel at the TAN basin is to be moved in accordance with the Settlement Agreement by 2001. (See section 2.3)

Dry Storage

There are several dry storage facilities at the INEEL. The Underground Storage Facility (CPP-749) at the ICPP consists of 218 lined dry wells. Sixty-one of these are "first generation" wells, sealed at the bottom with concrete grout. The other hundred fifty-seven are "second generation" dry wells, fully sealed by steel liners. Fuel is planned to be moved from the first to the second generation dry wells.

The Irradiated Fuel Storage Facility (IFSF) was built in the mid-1970s, and currently stores about one-third of the total Fort St. Vrain SNF inventory. Metallic fuels are being moved from the CPP-603 and CPP-666 basins into IFSF. In addition, some of the fuels being received from offsite are expected to be temporarily stored in this facility. A recently installed limited capability drying and canning station in the IFSF will begin operation in May 1997.

A storage pad at TAN holds four casks containing commercial fuel as part of a dry storage demonstration program. This fuel will be relocated to ICPP, pending repository disposal.

The Radioactive Scrap and Waste Facility (RSWF) at the ANL-W site consists of carbon steel-lined dry well used for the storage of EBR-II fuel, EBR-II blanket elements and scrap materials. Also, at ANL-W is a limited amount of in-cell storage for fuels awaiting treatment in the electrometallurgical cell. Dry storage wells are also located in the floor at the Transient Reactor Test (TREAT) facility.

Facility Identification	Facility Function	Fuel Groups Stored	Planning Considerations	
			Liabilities/Strengths	Present Plans
Underwater Storage Facilities				
CPP-603	Spent Fuel Storage	3, 15, 14, 6, 7	Fuel corrosion concerns, unlined pool, seismic concerns, ventilation	Shut down by 2001
CPP-666	Spent Fuel Storage	15, 14, 9, 3, 16	Lined pool, demin water, leak detection	Shut down by 2023
TAN-607	Spent Fuel Storage	4, 1	Pool is unlined, fuel corrosion concerns, no leak detection, dry load and unload	Shut down by 2001
PBF-620	Reactor working storage	2	Only for PBF fuel storage, lined pool, small	Shut down by 2023
TRA-603 Canal	Hot cell and fuel exam storage	4	Unlined, fuel corrosion concerns	Shut down by 2000
TRA-670	Reactor working pool	15	Only ATR fuel storage	In service until reactor shutdown
Dry Storage Facilities				
CPP-603 IFSF	Above ground dry spent fuel storage	10, 11, 7, 5, 6	Seismic concerns, dry load and unload, fuel canning	Shut down by 2035
CPP-749 Dry Wells	Below ground spent fuel storage	11, 14, 13	Below grade dry storage, water present in first generation wells	Shut down by 2035
TAN-607	Above ground spent fuel dry storage	1	Dry, cask storage on concrete pad	Shut down by 2006
RSWF	Below ground spent fuel dry storage	14, 16	Below grade dry storage in stainless steel containers in carbon steel liners	Shut down by 2035
ANL-W TREAT	Reactor fuel storage	12	Dry wells in reactor building floor	Shut down by 2035
Possible Characterization Facilities				
ANL-W HFEF	Fuel examination hot cell	N/A	Modern stainless lined - extensive capability for handling highly irradiated material	Support electro-metalurgical process
TAN-607 Hot Shop	Large scale hot cell work	N/A	Large capacity unlined hot cell, large cranes, direct connection to TAN storage pool	Shut down by 2006
ICPP CPP-684	Analytical laboratories	N/A	Modern hot cell capability, full RCRA and process chemical analysis capability, stainless steel lined	Support waste processes beyond 2035
ANL-W	Analytical laboratories	N/A	Modern hot cell capability, full chemical analysis capability	Support electro-metalurgical process

Table 2.2-1 INEEL SNF storage and potential characteriazation facilities.

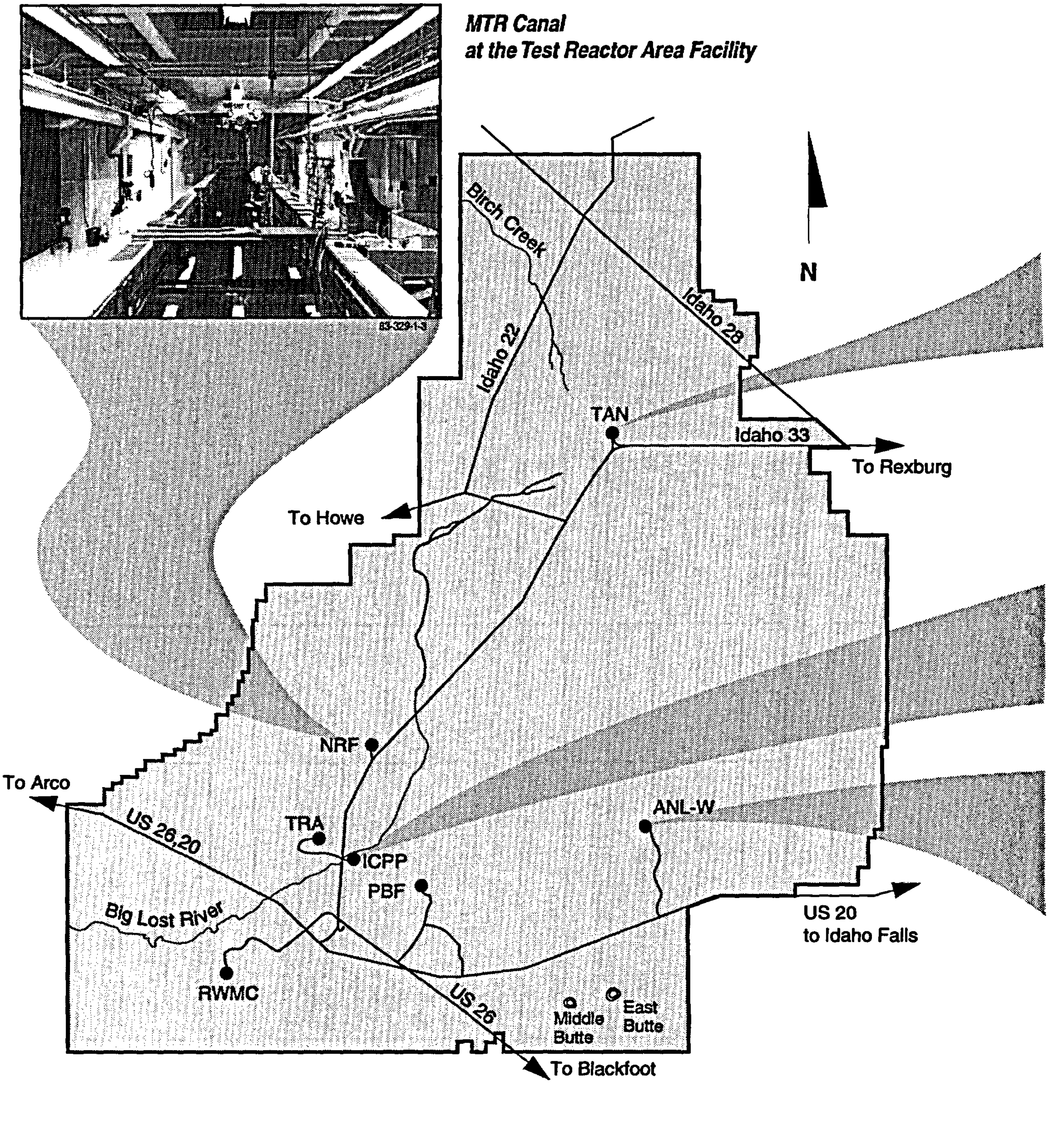
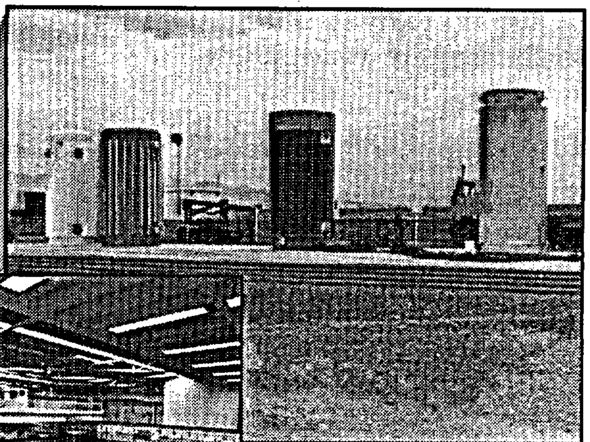
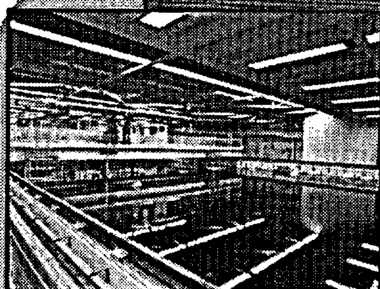


Figure 2.2-1 Map of INEEL showing SNF facilities.



90-351-1-16

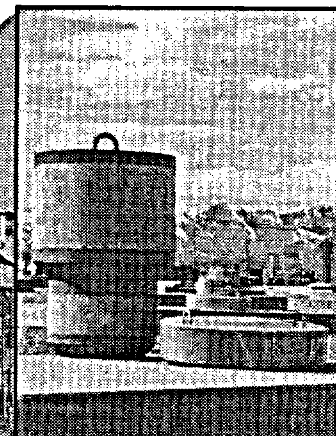
TAN Dry Storage Casks and Pad



86-253-2-4

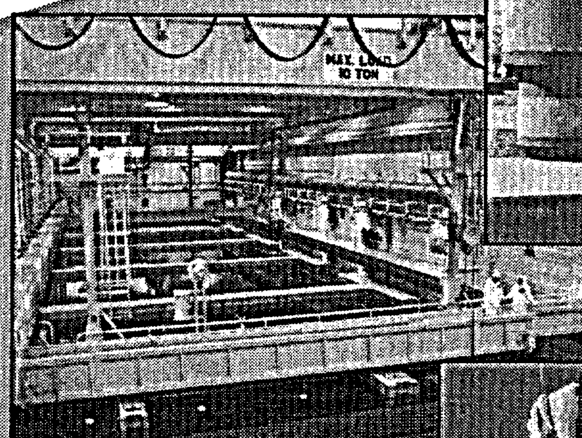
TAN 607 Pool

CPP-749
Dry Well Storage Area

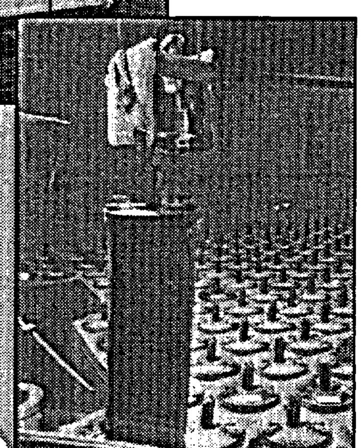


85-153-1-10

CPP-666
Fuel Storage Pools



85-268-5-6



91-140-4-8

CPP-603
Irradiated Fuel
Storage Facility

ANL-W
Radioactive Scrap
and Waste Facility

E96 0378

Two new dry storage facilities are planned for the ICPP. One of these is a NUHOMS[1] type facility, for the storage of the TMI-2 SNF. The other will be a substantially larger dry storage facility. The second facility may also include integral dry transfer, drying and packaging capability.

Support and Characterization

Several facilities at the ICPP and ANL-W have the capability for the treatment or conditioning of SNF if needed. These could be used to dissolve fuel material and convert it into diluted solid oxide. Additionally, certain types of fuels could be treated at ANL-W hot cells, (e.g., electrometallurgical treatment of sodium-bonded fuels).

The analytical laboratories at ICPP are state-of-the-art, full service hot cell facilities for analyzing highly irradiated samples. The facilities are currently being used to support high-level waste characterization and processing support. Current capability includes chemical, radiochemical, and isotopic analyses on a wide range of sample types.

There is a similar hot cell analytical laboratory at the ANL-W facility which was utilized for fuel development experiments at the EBR-II reactor and has capability for the chemical, radiochemical and isotopic analysis of SNF and irradiated materials.

The Hot Fuel Examination Facility (HFEF) is a large, modern hot-cell facility at ANL-W designed to handle and characterize SNF of all types. It's capabilities for SNF examination include visual inspection, microscopy, metrology, weighing, gamma scanning, sample preparation and neutron radiography. SNF samples at HFEF can be directly transferred to the analytical laboratory.

2.3 Constraints

There are several governmental agreements, management decisions, and commitments to stakeholders currently in place and applicable to this work. Primary among these are:

- Programmatic INEEL and SNF Environmental Impact Statement (EIS) Record of Decision

  The Record of Decision establishes jurisdiction within the DOE complex for SNF management. Accordingly, aluminum-based SNF will be managed at Savannah River, Hanford production SNF will be managed at Hanford, and the rest of the DOE inventory will be managed at INEEL.

- Settlement Agreement

  In 1995 the State of Idaho, the Department of the Navy, and the Department of Energy entered into a formal agreement regarding the disposition of SNF at INEEL. The agreement sets limits on the types of SNF, the total number of shipments, the metric Tonnes of heavy metal and the rate at which spent fuel could be brought into the State of Idaho. It specifies dates for the closure of certain facilities and the date (January 1, 2035) for final removal of all SNF from the State. By action of the U. S. District Court, the Settlement Agreement is now a legally binding court order.

- Vulnerability Assessments

  As part of its overall review of nuclear materials management within the complex, DOE has conducted assessments of the vulnerablilites (Ref. B) associated with the SNF storage facilities. The assessments of INEEL storage facilities identified aged facilities that do not meet current codes for seismic criteria and modern design criteria. These findings affect primarily the priorities for the shutdown of the wet storage facilities.

[1] NUHOMS is a trade-name for a VECTRA dry storage system

Section 3

Addressing Key Issues

Within the complex of factors which came into play in the selection of a path forward for the INEEL SNF, there are several key issues. While these are obviously interdependent (for example, criticality considerations affect the packaging design, and vice-versa), the Team examined each separately and attempted to develop a thorough understanding of the issues, its implications, and potential success factors.

It is recognized that any INEEL SNF considered for deep geologic disposal must comply with the provisions of the Nuclear Waste Policy Act (NWPA), as amended (Ref C). Spent nuclear fuel, whether it be commercial or INEEL SNF, must fall within the definition of "spent nuclear fuel" per Section 2(23) of the NWPA before it can be considered for disposal in a NWPA licensed repository.

At the time of repository disposal, the INEEL SNF will also have to comply with the licensing provisions of 10 CFR Part 60 (Ref D) and applicable DOE Office of Civilian Radioactive Waste Management (OCRWM) acceptance criteria. The OCRWM waste form criteria are in the *Waste Acceptance System Requirements Document* (Ref E). Many of the key elements discussed in this section address how the INEEL SNF will meet those criteria.

The Team's investigations of key issues are described in this section. The integration of these issues is presented in Sections 4 and 5.

3.1 Characterization

Characterization is the process of obtaining technical information about the SNF in sufficient detail to permit reasonable prediction of its behavior in storage, transportation, and repository disposal and to meet regulatory requirements. The resulting data also help to define the engineering requirements for SNF treatment (if required) and packaging, and they form the basis for regulatory acceptance of the SNF disposition actions.

Requirements

For interim storage and transportation, the characterization requirements are well understood, and in most cases, can be satisfied through existing documentation, along with analysis and/or nondestructive examination as needed.

Characterization requirements for repository acceptance, however, are not yet well defined. From a technical standpoint, the SNF must be characterized sufficiently to provide reasonable assurance of satisfactory performance in the repository, including criticality safety. Initial characterization information will be used to establish the scientific basis for repository disposal, as reflected in the repository viability assessment, environmental impact statement and license application. Subsequent characterization work may be needed to support more detailed engineering analysis (for packaging and handling systems, as examples) and for confirmation prior to receipt at the repository, that the as-packaged SNF meets the requirements of the repository license.

As a starting point for defining the characterization requirements for DOE-owned SNF, DOE has catalogued the extensive existing requirements for repository acceptance of commercial SNF. In some cases the existing commercial SNF characterization requirements seem to have been driven more by the availability of information – which is readily available for most commercial fuel – than by technical need. For some DOE fuel types these requirements may not be essential to prediction of repository performance, and may be unnecessarily expensive to apply to the DOE SNF. DOE intends to refine these data needs, and then to produce a set of guidelines for meeting repository requirements for DOE SNF, based on results of ongoing preliminary performance assessments for several INEEL SNF groups.

It is the Team's view that characterization requirements for DOE-owned SNF must be "performance based" – that is, they should prescribe the technical data which is truly necessary to demonstrate, with reasonable confidence, the behavior of the SNF in its final disposal configuration. Of course, the characterization requirements must satisfy regulatory needs, comply with IAEA standards and meet other legitimate commitments. Fundamentally, however, the requirements should be based on technical need for the specific SNF in question.

The Team considers DOE's action in establishing this set of performance driven characterization requirements to be the essential first step in the development and implementation of a strategy for the management of the INEEL SNF. It will affect engineering, operational, facility, schedule and resource requirements for the entire program. It warrants a high level of attention within DOE and early action to engage NRC in the process developing these requirement and to secure NRC concurrence at the earliest practical point.

Methods and Facilities for Characterizing INEEL SNF

The Team's preliminary assessment of characterization needs specific to the INEEL SNF, based conservatively on OCRWM's existing requirements for repository acceptance of commercial SNF, indicates that while a large body of materials performance data exists, some of the needed information is not currently available. There are several potential ways to fill this apparent gap:

- Some of the requested information may not be necessary, based on DOE's pending guidelines. (Ref F)
- Bounding analyses can be used to cost-effectively satisfy the technical need to ensure that repository performance requirements are not compromised. This may be a particularly advantageous approach for many of the INEEL SNF fuel types, for which their relatively small quantities can be shown to have little if any effect on aggregate repository performance. Also, while there may be uncertainty about the technical specifics of a given INEEL fuel type, it may be possible to demonstrate that its behavior is bounded by some other, well characterized and analyzed fuel type.
- Limited tests may be required for some DOE SNF to provide information on leaching, oxidation, hydriding and pyrophoricity characteristics under repository conditions. Some preliminary fuel grouping has been done in this regard by the EM/RW Repository Task Team (Ref. G), based conservatively on the commercial SNF requirements.

Regarding facilities, the Team's preliminary assessments suggest that existing hot cell capability at INEEL should be sufficient to support the characterization effort. Use of existing facilities would likely be more cost-effective than constructing a new characterization hot cell facility, but would require sufficient funding to maintain facility readiness and enhance (where needed) the characterization capabilities at these existing facilities. The cost benefit of utilizing existing facilities will be affected to some degree by the need date for the characterization information.

Other Considerations

The Team identified several other characterization issues, including:

- For some INEEL SNF types, such as TMI-2 SNF debris, there is extensive information about the material in aggregate, but not for individual fuel assemblies or packages within that group. For the TMI fuel, obtaining canister-by-canister characterization information would be an extremely costly endeavor with no apparent benefit in terms of repository performance. In this case, the Team recommends use of bounding analyses, based on the aggregate information.
- Minimal information is available for many unique fuel types (as examples, those of foreign origin and those received decades ago). Generally, these consist of relatively limited quantities, and can be handled via bounding analyses.
- In some cases, existing characterization information may not satisfy repository data qualification requirements imposed by the OCRWM Quality Assurance Requirements and Description document, DOE/RW-0333P. This is a regulatory compliance issue which should be resolved based on technical merits – blanket compliance with requirements initially established for different fuel (i.e., commercial utility SNF) is not practical.
- Given the current uncertainty regarding characterization requirements, it is important that SNF interim storage configurations permit retrievability of individual fuel elements or packages for subsequent examination and/or testing.

Compliance Verification

The following examinations may be required to confirm that the license requirements have been met for specific packages being readied for storage, transport or disposal:

- Visual examination of the SNF physical condition and a positive SNF identification check.
- Nondestructive assays to validate calculated fissile and radionuclide content.
- Measurement of radiation levels.
- Inspection to verify the integrity of the package closure.

It is the view of the Team that such activities can and should be limited to those driven by legitimate technical or personnel safety needs, or specifically required by regulation, and must be justified from an ALARA standpoint.

Overall Characterization Approach

The overall characterization approach recommended for finalizing characterization requirements and determining methods needed to fulfill the requirements, is depicted in Figure 3-1. This approach is intended to ensure maximum cost-effectiveness by limiting characterization work to that which has direct bearing on SNF performance in storage, transportation or disposal, and by utilizing analytical capabilities in lieu of physical testing, where practical. Finalization of the characterization approach, including regulator involvement, should be expedited to provide early closure on facility design requirements and plans.

3.2 Criticality

Many of the SNF types at the INEEL site have medium or high fissile enrichments. All of the SNF types must be disposed in a way that provides high confidence that they will remain sub-critical for many thousands of years – a design objective more challenging than for commercial reactor SNF because of the higher fissile

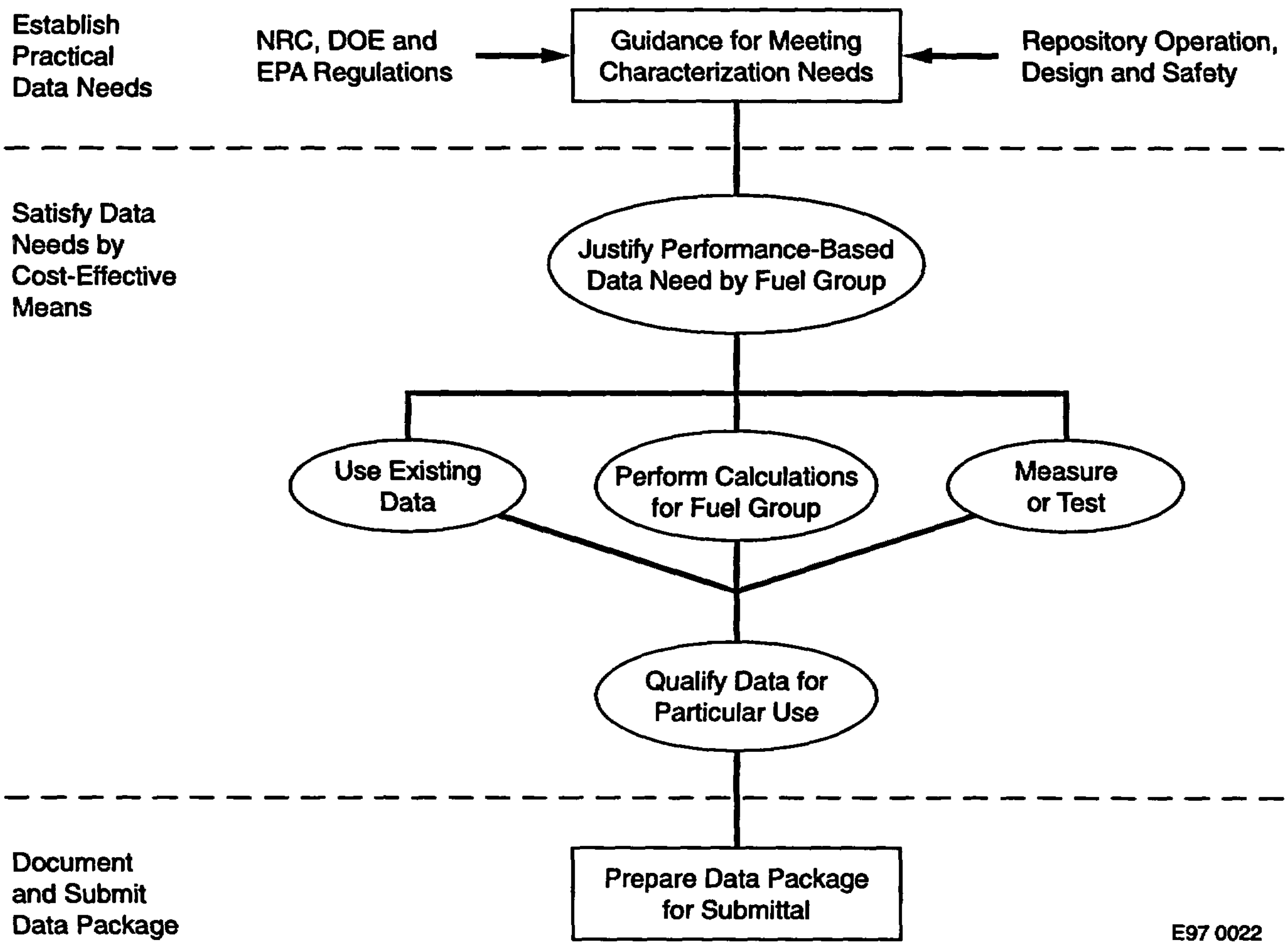


Figure 3-1 Characterization process to satisfy data needs.

content. Disposal criticality safety must consider not only the reactivity of the SNF as initially packaged and disposed but also the long-term reactivity changes that result from the degradation of the SNF assemblies and the canister basket after the waste package is breached. The analytical methods for demonstrating long-term criticality safety are still being developed.

Criticality control for transportation and storage of INEEL SNF is not addressed in this report since INEEL's internal capabilities in these areas are well in hand. Criticality control for repository disposal, long term, unmonitored and degraded states of the fuel is not as well understood. For that reason, disposal criticality control was the Team's primary focus.

Disposal Criticality Control - Analytical Approach

Presently, the NRC regulation for criticality control in a repository environment, 10 CFR 60.131 (h), (Ref D) prescribes that the calculated effective multiplication factor (keff) must be sufficiently below unity to show at least a 5% margin, after allowance for the bias in the method of calculation and the uncertainty in the experiments used to validate the method of calculation.

It is anticipated that the consequences of a criticality event during long term disposal within the repository would only be a slight increase in the overall source term characteristic of the total performance of the repository. There would be no direct hazard because the repository will be closed and sealed, and the energy release rate would be very low. For that reason, risk-based evaluations are being proposed for determining criticality risk during long time periods following closure of the repository.[1]

A conservative approach to repository design is to achieve a very low probability of a criticality regardless of the direct consequences. In the analyses described here, the intent is to show that the probability of a criticality for the INEEL SNF within the repository is no more than the criticality probability for an equivalent amount of commercial SNF, and that it meets the requirements of 10 CFR Part 60.

For commercial SNF, the current proposal in disposal criticality analysis includes credit for neutron absorbing actinides and fission products and reduced fissile content, as a result of burnup in the reactors and through use of long-lived supplemental neutron absorber materials. However, for much of the INEEL SNF, the burnup is not as well known, and therefore is not available for analytical credit in those cases. Most of the fuels will be analyzed as fresh fuel without considering the reduction in fissile content through burnup. For certain of the DOE-owned commercial SNF that have documented operational histories, conservative assumptions of burnup may be used. For SNF that is not well characterized and for which the fissile content is not well documented, available information will be used in developing conservative estimates of criticality potential.

The methodology being used to evaluate the criticality potential of commercial SNF is being documented in a series of technical/topical reports, the most recent being the *Disposal Criticality Analysis Methodology Technical Report* of August 1996 (Ref. K). The basic methodology being used for criticality analysis is expected to be applicable to a wide range of SNF, including most of the INEEL inventory. Where the methodology is not directly applicable, bounding assumptions will be made to ensure the criticality control requirements are met.

Disposal Concepts for Criticality Control

For the INEEL SNF, criticality protection will be afforded primarily by means of disposal packaging configurations. Several packaging design features were considered by the Team to be potentially effective for criticality control:

- Limitation of the amount of neutronic reactivity in a waste package
- Incorporation of long-lived neutron absorber material
- Incorporation of structural or other added material for moderator exclusion or neutron absorption capability, in the degraded state.

One or more of these methods could be used in the design of disposal packaging for any of the INEEL fuel types.

Two disposal alternatives of the INEEL HEU SNF have been evaluated to limit the amount of reactivity of fissile material in a waste package. These alternatives are direct disposal in separate, small waste packages and co-disposal with HLW canisters in large waste packages. Co-disposal makes use of otherwise unused space inside of a (HLW) waste package, and is likely to be the more cost-effective approach for highly enriched SNF. These concepts are discussed in more detail in Section 3.3.

Addition of depleted uranium to the canister was investigated. The potential benefits from adding depleted uranium to a waste package are to exclude moderator and to provide some neutron absorbing capability,

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[1] OCRWM has recommended that NRC revise the 10 CFR 60.131 (h) requirement, to allow for criticality control to be demonstrated using a risk-based approach. It has been shown that the probability of a criticality event within a repository is quite small and that the dose consequences from such an event would be negligible and the use of a risk-based approach would yield more cost-effective risk control in the repository, post-closure (Ref. H, I, J). However, risk-based methods are not currently reflected in regulatory requirements. Therefore, for purposes of this study, the Team based its evaluation on the premise that the current regulatory requirements prevail.

which is expected to have the most significant effect in external degraded mode configurations. The extent of these benefits has not yet been evaluated quantitatively, and must be balanced against the implementation difficulties. Depleted uranium is expected to be in the form of small (about one millimeter diameter) pellets or particles of uranium oxide that can be poured into the canisters to fill most of the free volume. However, this approach has some limitations. It may be difficult to demonstrate that the package void space is filled uniformly, and the addition of depleted uranium must not cause the overall weight of the loaded disposal container to exceed the capability of the transportation and repository surface and below-grade handling equipment. (Currently the heaviest disposal container for commercial SNF is projected to weigh 69 tons; assuming the figure will be the basis for handling equipment design, the amount of depleted uranium, which could be added to INEEL SNF containers would be severely limited). Also, the depleted uranium in some cases could act as a neutron reflector and increase the reactivity of an assembly.

The INEEL commercial LEU SNF is very similar to commercial SNF in its repository criticality concerns. The commercial SNF has been analyzed by DOE OCRWM, and the present plans are to place the commercial SNF into large diameter disposal containers. Pending analysis confirmation, this same disposal concept is expected to be used to address the INEEL commercial LEU[2] SNF.

Scoping analysis of a representative INEEL fuel type

The Team conducted a preliminary disposal criticality analysis of one of the INEEL SNF types, to assess both the evaluation methodology and the effectiveness of the package concepts. The analysis is provided in Appendix C.

To complement the work performed by the Research Reactor SNF Task Team (Ref. L), Shippingport PWR Core 2 Seed 2 SNF was analyzed. This fuel was chosen, like the Al-clad Research Reactor SNF, due to the higher criticality potential for HEU fuels. However, the Shippingport PWR SNF is structurally different and substantiality more robust than the Al-clad SNF. These fuels together provide a bounding picture for the HEU fuels.

For this evaluation, Shippingport PWR Core 2 Seed 2 SNF (INEEL Group 2) was analyzed to develop an acceptable criticality control strategy and to evaluate the merits of using depleted uranium oxide as a filler to isotopically dilute the U-235 content of HEU fuel types. The Shippingport PWR SNF has a beginning of life (BOL) enrichment of 93 wt% of U-235. The SNF assembly cladding, (i.e., the clusters), are made entirely of Zircaloy-4). Two clusters of the Shippingport PWR SNF are expected to exceed the current regulatory limit of $k_{eff} = 0.95$ when they are surrounded by water and no control rods or other neutron absorbing materials are present.

The Team's understanding is that the presently available commercial storage containers, such as NUHOMS, do not include long-lasting criticality control capability since they generally depend on keeping the assemblies apart and do not consider the consequences that result from the degradation of the internal structure. For the Shippingport PWR Core 2 Seed 2 SNF, supplemental poisons and/or moderator exclusion material, such as depleted uranium oxide, are expected to have to be added to provide criticality control after the internal structure has corroded and collapsed. Depleted uranium oxide could be added in the longitudinal tubes containing each cluster, but it would be difficult to add around the outside of the tubes since it would have to be placed between each of the separator plates. Additionally, although 14 Shippingport PWR clusters can be accommodated in a NUHOMS storage container, the amount of depleted uranium oxide that would be necessary to provide the required criticality safety margin in the fully degraded conditions would cause the weight of the loaded disposal container to far exceed the planned repository surface and below-grade handling equipment capability.

[2] LEU fuel has uranium 235 less than 5% of the total uranium.

Therefore, a more appropriate and more practical concept for the disposal of the Shippingport PWR Core 2 Seed 2 SNF is co-disposal with HLW canisters. Each cluster could be placed in a nominal 10" diameter canister (e.g., schedule 10 pipe) and the canister placed in the center of a disposal container containing four HLW canisters. This form of disposal will result in the 40 Shippingport PWR clusters requiring 40 disposal containers containing 160 HLW canisters. Pending confirmation, the reactivity of this configuration maintains the calculated k_{eff} to less than 0.95. Another approach would be to add supplemental neutron absorber material in the same manner as for commercial SNF.

In summary, this analysis reinforces the validity of the co-disposal packaging concept for HEU fuels. More thorough analysis, considering cost, operational impacts, and regulatory discussion with the NRC, will be required before a final selection can be made.

Recommended Path Forward for Disposal Criticality Analyses

Having completed the scoping analysis for one INEEL SNF type, the Team recommends that more extensive analyses be conducted to evaluate the long-term criticality behavior of selected DOE SNF forms in a repository environment. The expected approach, which has been subject to an informal NRC staff review (Ref. I), is similar to one adopted for commercial SNF and other DOE wastes currently being considered for repository disposal.

This methodology is based on a risk-based approach and would be implemented in three phases to cover the full range of conditions in long term disposal. The first phase is an evaluation of the criticality behavior of intact (or otherwise as-disposed) fuel assemblies and waste package configurations, in both dry and fully wet conditions. In the second phase, both chemical and physical degradation of the fuel and waste package are considered. The third phase analysis considers flow and transport in the repository far-field environment, reconcentration (if any) of the fissile material in the geosphere, and possible configurations of deposited material. The third phase also considers the probability that a critical configuration occurs, and the consequences (if any) of such criticality.

Additionally, scoping analyses similar to that performed in this report may be needed to address near-term priorities.

3.3 Packaging, Storage and Transportation

A major part of the Team's effort in formulating a path forward strategy for the INEEL SNF was conceptualizing SNF packaging systems suitable for interim SNF storage at INEEL, transportation to the repository and disposal. This section summarizes their evaluations and conclusions in this regard.

The selection of engineered packages for interim storage, transportation, and disposal of SNF is influenced by several factors. The most demanding requirements are those related to ultimate disposal in the repository, including long term performance (i.e., resistance to degradation and ultimate degradation in a predictable way), and meeting other waste acceptance criteria, particularly criticality safety for disposal of HEU and MEU. To be suitable for transportation, containers must meet established DOT and NRC requirements. And for interim storage at INEEL, the SNF package must afford extended term safety, stability, and low cost.

All of these considerations have to be accommodated within the framework of schedule and economic practicality. The Team attempted to conceptualize packaging systems which could be cost-effective and available in the relatively near term. In general, the Team considered simple standardized packages, suitable for dry storage and transportation, to be the best way to meet the requirements.

Packaging concepts

The packaging configurations envisioned by the Team for the INEEL SNF are based on current concepts under development by the OCRWM for the disposal of commercial SNF and HLW. These include large disposal packages (similar to those envisioned for commercial fuels)[3] and smaller packages which permit co-disposal of the INEEL SNF, in sealed canisters, with vitrified HLW canisters. The smaller co-disposal packages would be used primarily for the more highly enriched (i.e. HEU and MEU) fuels. The INEEL SNF, with the exception of large intact assemblies, would be loaded into standardized canisters at INEEL. These canisters and assemblies would be stored at INEEL using existing stable dry storage facilities or available licensed commercial DPC interim storage and transportation systems. At the repository these canisters and assemblies would be transferred from DPCs or other licensed transportation containers to the appropriate disposal or co-disposal container.

The suitability of large versus small containers would be dictated primarily by criticality concerns. Because of their low fissile material content, the packaging of LEU fuels is likely to be limited only by volume, and therefore multiple assembly, large container disposal configurations will best suit LEU fuel. By contrast, demonstrating criticality safety for HEU and MEU fuels will likely require packaging in smaller (and perhaps neutron-poisoned or moderator-excluded) canisters, for co-disposal or direct burial in small overpack disposal containers.[4] (Canister design features for criticality safety are discussed in greater detail in Section 3.2, above.)

For HEU and MEU, three co-disposal packaging configurations were evaluated, as shown schematically in Figure 3.3-1. These include canister designs with nominal outside diameters of 10", 17" and 24". The 17" configuration was evaluated previously for the disposal of aluminum-based SNF and the 10" and 24" configurations were included in this evaluation to provide flexibility needed for the wide variety of SNF types at the INEEL. Other package configurations may also be feasible.

In each case, the SNF canister dimensions were chosen to be compatible with anticipated HLW container designs, maximizing the use of available interior storage space. For two of the configurations, the SNF canister would be nested in the center cavity (an otherwise unused space) surrounded by glass logs. In the third, the SNF canister would be placed in one or more of the glass log storage locations. For a given fuel type, the co-disposal configuration would be chosen based on criticality requirements, on dimensional constraints (large intact assemblies such as Shippingport LWBR, will require 24" or larger canisters), and overall compatibility with the HLW disposal plans.

Based on the very large quantity of nonfissile HLW planned for repository disposal, co-disposal of SNF is likely to be economically attractive. Utilizing multiple, predesigned SNF canister configurations that are all compatible with HLW disposal containers, should provide adequate flexibility to accommodate the relatively small quantity of INEEL SNF, without disrupting the HLW disposal activities.

No physical or chemical changes would be made to the SNF placed in these canisters. Prior to interim storage, the spent fuel would be dried sufficiently to limit corrosion, and to preclude any excessive gas generation[5] during storage, and then inerted and sealed. In-canister filler material, if required for criticality control in the repository (see section 3.2), can be added at the time of initial loading or later (but prior to shipment to the repository), as preferred.

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[3]HLW glas logs with no fissile material content.

[4]The major exception to this are graphite matrix fuels such as the Fort St. Vrain fuel which is HEU but has a low relative fissile mass density – i.e., the assemblies have a significantly larger mass of nonfissile matrix than fissile matrix, resulting in a large volume required to reach the fissile limit.

[5]Even after drying, some gas buildup is possible. In some cases, this may require that the canisters be vented, re-inerted, and re-sealed prior to shipment to the repository.

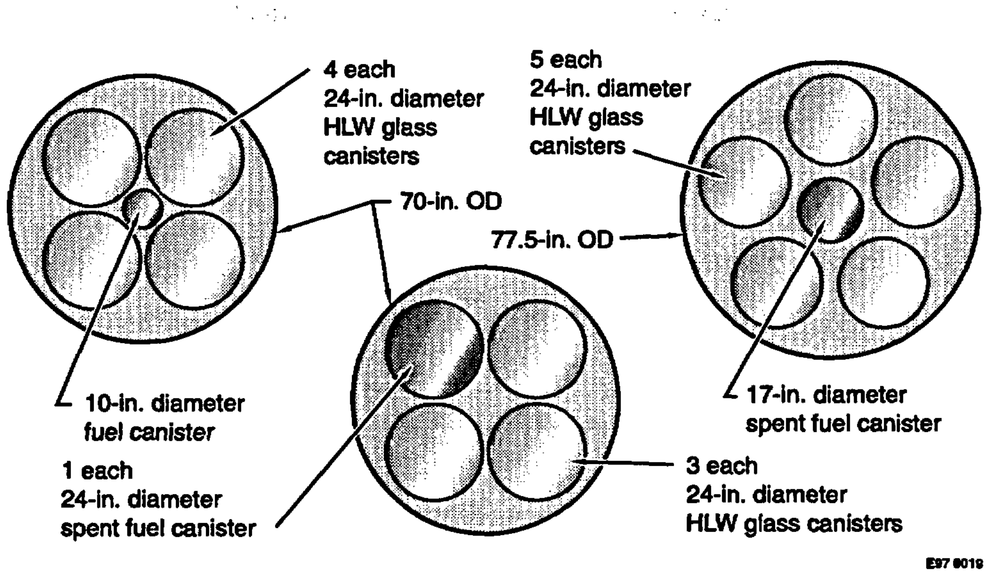

Figure 3.3-1 Co-disposal is the disposal of spent fuel with high-level waste glass logs.

The thermal and drying requirements for each fuel group has been considered. The relatively low decay heat output of these fuels is not expected to challenge thermal capability of the commercially available systems for interim storage and transportation. Considerable work has been performed at the INEEL (Ref. M), Savannah River Site (Ref. L), and at the Hanford Site (Ref. N) with respect to drying requirements. This work indicated that the vacuum drying process in use for commercial fuels, with the possible addition of an external heat source to speed up the process, will be acceptable for all except a few special fuels.[6]

Interim Storage

Regarding the application of these packaging concepts for interim storage at INEEL, several additional considerations are important:

- As a matter of policy, the SNF is to be stored in a configuration that is "road ready" – that is, essentially ready for transportation to the repository, with minimal additional preparation or repackaging. DOE intends not to require repackaging at the repository.
- Priority attention must be given to mitigation of the vulnerabilities and limitations of the current (primarily wet) storage, which in some cases will dictate moving the SNF to better storage locations in the near-term, with further actions (e.g., repackaging) delayed until later.

For most of the HEU and MEU SNF, the preferred approach is to repackage it in the near-term into the small diameter canisters meeting anticipated repository requirements, and to interim store these loaded canisters at the INEEL in dry storage facilities until shipment to the repository. These loaded canisters could be stored in dual-purpose canisters (DPCs) suitable for transportation, or they could be stored in other 10 CFR 72 compliant storage systems for later transfer to shipping casks.

For cases in which existing INEEL SNF packaging is clearly safe and sound, it may be economically advantageous to dry-store the SNF in its current configuration; the SNF would be transferred to the standardized small diameter canisters just prior to transportation. In the Team's view, this approach would meet the "road ready" criteria provided that the packaging actions at the point of transport are simple and would not entail fuel conditioning or handling breached or rubble fuel materials.

[6] In addition, the special fuels are the U-metal fuels (group 8 and 9) that may require conditioning, similar to the Hanford N-reactor fuel, and the fuels requiring treatment (groups 14 and 16).

Transportation

The transportation concept being developed by the OCRWM seems appropriate for the INEEL SNF and compatible with the packaging concepts conceptualized by the Team. Therefore, no alternative transportation concepts were developed.

3.4 Waste Form Performance

To determine the potential effect of the INEEL SNF on the repository performance, the Team decided to have a scoping Performance Assessment completed on selected SNF types that could be used to represent the total INEEL SNF inventory. The INEEL SNF performance was compared to the performance of an equivalent amount of commercial SNF, as well as its aggregate effect with the entire repository inventory.

The assessment was conducted based on a scenario from the Total System Performance Assessment-1995 (TSPA-1995.)[7] The TSPA requires the analysis of both the engineered and natural system to determine the potential long-term release of radionuclides. This assessment estimated the fuels' contribution to the dose to an individual at the accessible environment five kilometers from the repository.

To predict the performance of the repository, a series of computer models are used that includes the engineered system, i.e., waste package and the contents as well as the transport of the radionuclides through the natural barrier from the repository. The waste package performance includes the degradation of the canister and the contents. The existing computer models include a standard disposal container and radionuclide transport model, so the SNF variable is the contents of the disposal container.

The data for the performance assessment consisted of a source term of radionuclides and a release rate for the radionuclides. To perform the analysis within the scope of the Task Team, the categories developed in Section 2 were used. These categories were generally grouped to represent the important characteristics of the fuel to repository performance. Category V was split into two categories because of the difference in integrity between the fuel types. Category VII (miscellaneous) was not analyzed because of the diversity of characteristics in the group: some of the fuel has been previously addressed in analysis (Al-clad SNF Ref L) or will require some treatment prior to be disposal. A representative fuel was selected from each type and the required data for that fuel was obtained. Where the required release rate data was not available, a conservative assumption was used, for example, a multiplier of 1,000 times the commercial SNF release rate was used for the TMI-2 debris because of the increased surface area. Additionally, the radionuclide inventory for the representative fuel was scaled based on MTHM so as to represent the total category inventory. The categories, the representative fuels, and the release rate assumptions are listed in Table 3.4-1.

The dose at the accessible environment is a function of the number of waste packages and their inventory of radionuclides (such as neptunium) for which release is controlled by solubility. In order to bound the dose from INEEL SNF, two cases were analyzed. In the first case the number of packages for the fuel was based upon volume with packages loaded with the maximum amount of SNF possible on a volume basis. The second case, loading was based on criticality, which used co-disposal packages containing the SNF and HLW in vitrified glass logs, and which is based on limiting fissile material in the package. As more analyses of the potential for long-term criticality are conducted, the number of waste packages is expected to fall within the range of the two cases analyzed; however, lower fissile limits would significantly increase the number of packages required.

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[7]As noted, the above comparison is based on calculated dose history of an individual at the accessible environment. The dose to a population from the radioactive material being disposed in the repository has not been calculated. Since the EPA regulation governing repository performance is under revision and the repository has yet to be licensed, calculations of cumulative effects and effects on a population have not been performed.

	Category	Group	Quantity	Representative Fuel for PA Analysis	PA Assumptions	Performance in Repository
I	Intact oxide fuel	1 2 3	61 fuel types 41 cubic meters 92 MTHM	Typical Commercial SNF	Release rate of radionuclides is the same as commercial SNF	Dose peaks are the same as an equivalent amount of commercial SNF.
II	Disrupted oxide fuel	4 5	74 fuel types 169 cubic meters 94 MTHM	TMI-2 debris	Release rate of radionuclides is 1,000 times faster than the commercial SNF rate because of increased surface area	Dose peaks are about two orders of magnitude below an equivalent amount of commercial SNF. This result is expected due to the low burnup of the TMI fuel.
III	Uranium zirconium hydride fuel	6 7	2 fuel types 8 cubic meters 2 MTHM	TRIGA MEU	Release rate of radionuclides is 100 times slower than the commercial SNF rate	Dose peaks range from about a factor of five to more than one order of magnitude below an equivalent amount of commercial SNF, depending on dissolution assumptions.
IV	Uranium metal and uranium alloy fuel	8 9	20 fuel types 2.8 cubic meters 6 MTHM	Fermi - HEU	Release rate of radionuclides is 1,000 to 10,000 times faster than the commercial SNF rate. Metallic SNF model also used.	Dose peaks range from about one to more than two orders of magnitude below an equivalent amount of commercial SNF, depending on the number of packages and the dissolution assumptions.
V	Uranium carbide fuel	10	1 fuel type 196 cubic meters 23.4 MTHM	FSVR	Release rate of radionuclides is the same as the commercial SNF rate. Carbide and Ceramic SNF models also used.	Dose peaks range from somewhat higher to about a factor of five lower than commercial SNF, depending on dissolution assumptions.
		11 12	8 fuel types 40 cubic meters 3 MTHM	Peachbottom Core 2	Release rate of radionuclides is 1 to 10 times faster than the commercial SNF rate. Carbide SNF model also used	Dose peaks range from about a factor of five higher to an order of magnitude lower than commercial SNF, depending on the dissolution assumptions.
VI	Intact uranium and thorium oxide fuel	13	1 fuel type 52 cubic meters 39 MTHM	Shippingport LWBR	Release rate of radio- nuclides is assumed to be 100 times slower than the commercial SNF rate. Ceramic SNF model used.	Peak dose is less than a factor of five lower than commercial SNF
VII	Other	14 - Metalic sodium fuel	34 fuel types 14.6 cubic meters 60 MTHM	Category not addressed in repository performance assessment since these fuels have been either previously addressed in analysis (Al clad SNF) or will require some treatment prior to be disposal.		
		15 - Al clad fuel	14 fuel types 38 cubic meters 3.4 MTHM			
		16 - misc. other fuels	5 fuel types 4.3 cubic meters 0.2 MTHM			

Table 3.4-1 Performance assessment of categories of INEEL SNF in a repository.

The dose history calculated in each of the categories of INEEL SNF was compared analytically with that from equivalent MTHM of commercial SNF. The results of this comparison are presented in Table 3.4-1. Additionally, the dose attributed to the full INEEL SNF inventory in the categories was compared with that from an equivalent amount of commercial SNF. The peak dose attributed to the full INEEL SNF inventory was found to be about a factor of five below that of an equivalent amount of commercial SNF (See Appendix E). With the addition of the INEEL SNF to the HLW and commercial SNF in the repository, no significant change in dose at the accessible environment boundary was calculated. While preliminary, this assessment quantifies the intuitive expectation that INEEL SNF constitutes only a relatively small portion of the total quantity of SNF to be emplaced in the repository, and, therefore, should not contribute significantly to the repository dose at the accessible environment.

It should be noted that this scoping assessment showed that INEEL SNF, particularly the thorium-based fuels, contribute a different radionuclide release profile than the commercial SNF. The composite INEEL SNF peak dose is from Np-237, with U-234 being present among the top six dose producing radionuclides. The thorium-based fuels peak dose is from Th-229, with U-234 being present among the top six dose producing radionuclides. This is compared to commercial SNF where the peak dose is from Np-237, and U-234 does not appear among the top six radionuclides.

Having completed this preliminary scoping assessment, the Team recommends that work continue to study DOE SNF performance in a repository.

3.5 Dealing With Special Cases

Some of the fuels at INEEL present packaging and disposal challenges disproportionate to their quantity, because of their configuration, materials of construction, or condition. This section identifies those cases and outlines possible strategies for dealing with them.

"Special" Fuels, defined:

The "Special fuels" are those that cannot be managed like the others in their class. These fuels include small quantities of fuel compositions, fuels with fissile isotopes other than uranium-235 and sodium-bonded fuel materials. These fuels may have to be treated or packaged in unique package configurations. Examples of special fuels include:

Small Quantities of Unique SNF

Among the approximately 250 different fuel types in the INEEL inventory are numerous small quantities of one-of-a-kind material. Some examples:

- Nichrome fuel elements from the Aircraft Nuclear Propulsion (ANP) program, which was canceled in 1964.
- Ground-up ceramic fuel from the two nuclear ramjet reactor cores.
- Metallurgical mounts used for post-irradiation examination of SNF and the sectioned fuel rods from which these metallurgical mounts were prepared.
- Cans of scrap materials from the clean out of gloveboxes and hot cells.
- The disassembled clad pins from the Fermi Reactor.
- Seven cans of declad fuel pins from the Fermi Reactor.
- Two Fermi Reactor assemblies, which melted during a 1966 incident at that plant and subsequently were sectioned for examination.

Table 3.5-1 is a summary of the "special" fuels, including location, relative quantity in each SNF category, and fraction of the total INEEL inventory.

Unidentified Fuels

For some very small quantities of SNF, there is relatively little information on design, operating history, or other normally required characterization data. Many of these fuels were placed in storage at a time period when the expected endpoint was reprocessing and only minimal characterization data was required. In some cases the fuel data was classified for purposes of national security and it is not available.

It may be possible to show that the poorly characterized fuel is bounded by another well-characterized fuel that meets the requirements of DOE/RW-0333P. (Ref O)

U-233/Thorium Fuels

The INEEL SNF inventory contains 500 kg of U-233, which will be the sole contributor to the repository inventory of U-233, and thorium. It will need to be shown that these fuels meet the repository requirements.

Sodium-Bonded Fuels

There are some SNF types that contain metallic sodium as heat transfer media between the fuel meat and the fuel cladding. These are sometimes referred to as sodium-bonded fuels. Metallic sodium is very reactive with water, producing hydrogen gas and heat, the combination of which could lead to fire or explosion. It is the DOE interpretation of 10 CFR 60 that these fuels will not be allowed into the repository until the metallic sodium is removed. In addition any other DOE fuel that is shown not to meet the repository acceptance criteria will be treated so that it will meet the criteria.

Packaging Strategy for Special Fuels

A logical and cost-effective strategy for packaging and disposing many of the special fuels is to consolidate small lots of fuel together into common disposal packages. The underlying rationale for this approach is that the incremental potential for small quantities of SNF to adversely affect the repository is very limited, and the aggregate performance of any consolidated package can be conservatively projected.

This approach has not been considered (and is not needed) for commercial SNF, and will therefore need to be evaluated, developed and presented for regulator consideration. Compliance with 10 CFR 60 and RW-0333P will need to be demonstrated. This can probably be achieved by bounding analysis of all of the material based on the worst case fuel in the canister, and it may require some additional characterization of the consolidated canister.

Potential Treatment Technologies for Sodium Bonded and Special Fuels

The consolidated packaging approach is likely to be suitable only for small quantities of SNF. For other special fuels, it may be necessary or economically preferable to convert them into waste forms more suitable for disposal. Several existing or developmental treatment technologies were evaluated recently by the Research Reactor Spent Nuclear Fuel Task Team (Ref. L). These were not reevaluated for use at INEEL, but the Team considers them potentially viable based on the previous evaluation. They are summarized below:

Electro-metallurgical Process

The Electro-metallurgical process was developed specifically for fuels containing metallic sodium. The fuel elements are chopped and the fuel meat dissolved in a molten salt. The uranium is deposited by electrolysis

Classification			Quantity					
Category	Fuel Group	Description	No. Items	No. Entries	Fuel Volume m^3	Total Uranium (kg)	Mass Fissile (kg)	Other Information
I	1	Intact LEU oxide	1	1	0.19	309.4	3.5	Loose fuel rods need to be placed in a canister
	2	Intact MEU oxide	18	3	0.02	17.4	0.7	
	3	Intact HEU oxide	105	13	2.78	1396.0	110.6	5 fuels are MOX (MOX fuels are treated as HEU fuels irregardless of the U-235 enrichment), 2 are Thorium containing
II	4	Disrupted LEU oxide	33	27	1.16	356.3	24.3	19 fuels are MOX (MOX fuels are treated as HEU fuels irregardless of the U-235 enrichment)
	5	Disrupted HEU oxide	169	30	5.83	70.1	21.6	
III	6	Zirconium hydride LEU	2	1	0.04	3.0	0.4	Additional items may be found that are disrupted
	7	Zirconium hydride HEU	63	6	0.57	39.4	32.1	Additional items may be found that are disrupted
IV	8	Metal LEU	2	2	0.03	220.9	1.5	
	9	Metal HEU	7	4	0.14	4.6	3.9	
V	11	Graphite, pyrolitic carbon HEU	251	3	0.10	39.6	26.6	Fuels contain thorium and U-233
VI	16	Other	29	3	4.31	158.5	48.2	
		TOTAL	680	93	15.17	1615	273.4	
		Percentage of INEEL Inventory	1.1%	33.0%	2.7%	1.1%	2.0%	

Table 3.5-1 Inventory of small quantities of unique fuels scattered throughout INEEL Groups. Groups 10, 12, 13, and 14 have no unique fuels.

onto an electrode from which it can be diluted, converted to a storage matrix, or recycled. This process is currently undergoing hot testing in the fuel cycle facility hot cells at ANL-W. The process is well developed and the technology is mature. However, additional development would be needed to put in place a production-scale Electro-metallurgical process, on a scale suitable for the INEEL application.

Electrolytic dissolution

This process was designed for the EBR-II fuel and dissolves the fuel by using an electrical current to accelerate the dissolution of the stainless steel in nitric acid. It was used at INEEL routinely since 1972 for the recovery of uranium from the HEU EBR-II fuel. It produces a liquid product from which uranium could be extracted from and potentially diluted and a fission product containing waste stream that is the same as the HLW at the INEEL.

Custom Processing

A custom processing option has also been used to dissolve small quantities of special fuels. In this process a dissolution flow sheet is tailored to put the fuel into nitric acid solution for subsequent treatment. Its products are the same as the electrolytic process.

Glass Material Oxidation and Dissolution System (GMODS)

In this batch process, the fuel material is added to a molten glass mixture consisting of lead oxide and boric oxide which contains molten elemental lead. The lead oxide oxidizes the metal components of the fuel and dissolves them in the glass melt. Additives are included in the melt to form a more durable glass. The glass is poured into a mold suitable for disposal. The Oak Ridge National Laboratory has demonstrated this process on a laboratory scale.

Plasma Hearth

The plasma hearth processes have been developed independently by Science Applications International Corporation (SAIC) and by Pacific Northwest National Laboratory. In this process, the fuel is melted by a large plasma torch in a ceramic crucible. The product can be a ceramic, metal or slag or can be tailored to a particular waste form by means of addition of glass-forming additives and then, cast into the desired waste form. This process has been applied to various waste types but not SNF. Currently, laboratory scale experiments are being performed with mock-ups of different fuel types. A hot demonstration is planned in the next few years.

Some of the treatments outlined above could be used for the special fuels to produce waste products suitable for repository disposal. In all cases, unique flowsheets would have to be developed for each of the fuel types.

These are described in the report *Technical Strategy for the Treatment, Packaging and Disposal of Aluminum-Based Spent Nuclear Fuel,* June 1996. (Ref L)

Section 4

An Integrated Technical Strategy

The previous sections addressed various aspects of the Team's evaluations, including the technical description and categorization of INEEL SNF, facilities available onsite for SNF handling and storage, and several key technical issues: characterization, criticality, packaging and special fuels considerations. The Team first dealt with these as separate elements, and then attempted to develop a integrated technical strategy, which integrate the results of the earlier evaluations. This section outlines the Team's proposed strategy.

The integrated technical strategy is presented in tabular form, on the following pages. For each of the INEEL SNF groups, it identifies in conceptual terms a proposed characterization strategy, packaging concepts, interim storage (at INEEL) and projected repository performance. Some top-level observations are important:

- The proposed strategy is conceptual, intended by the Team to establish a starting point for more definitive and complete investigations.
- It is based on interpretations of the limited scope of evaluations of the representative fuel types, performed by the Team over the course of this evaluation. In the Team's view, it is reasonable to extrapolate these results to the full range of INEEL fuels, for early planning purposes. However, more detained evaluations, and evaluations of other specific fuel types are needed and may yield results different from those presented here.
- This strategy appears to meet the requirements of the programatic EIS, the Settlement Agreement, the INEEL Spent Fuel Management Plan (Ref P), and the recent Ten Year Plan. However, it is not unique or exclusive. Other strategies could be pursued successfully.

Other summary-level conclusions, which can be drawn are as follows:

- Regarding characterization, in all cases the determination of reasonable, performance-based requirements is vital to the development of a practical, cost-effective strategy. Actual characterization requirements may vary widely among the INEEL fuel types.
- The packaging concept outlined in Section 3.3 is broadly applicable to the INEEL fuel types, and should provide sufficient flexibility to deal with the range of enrichments, materials, configurations and fuel conditions to be encountered.
- Detailed criticality analyses will be required for all fuel types. The reference case analyses performed by the Team, along with similar evaluations of other fuels (such as the aluminum-based SNF examination) provide important insights into expected criticality behavior and form a sound basis for the conclusions in this report. However, these evaluations may not bound all cases in part because the analytical methods and acceptance criteria for repository disposal are not yet firmly established.
- Regarding interim storage, the strategy is based primarily on the Team's understanding of current priorities and constraints (Section 2.3) affecting the INEEL site facilities.

Category I – Intact Oxide SNF

Group	Description/ Representative Fuel	Characterization and Analysis Considerations	Disposal Concept Criticality/Packaging	Estimated Performance in Repository	Interim SNF Management at INEEL
1 LEU	• Like commercial utility fuel. Most are intact assemblies. Some are partial assemblies. • Group Dimensions: 31 fuel types 30 m^3 76.8 MTHM • Representative fuel: Commercial	• Characterization requirements essentially same as for commercial SNF • No major characterization issues requiring testing	Direct Disposal • Multiple assemblies or canisters per large container • Variable burnup and insufficient documentation/ assume fresh fuel for criticality analysis	Dose peaks same as an equivalent amount of commercial SNF • Representative fuel to determine source term for this group is Commercial SNF (category I) • Typical commercial SNF dissolution/leach model	• Use large dry storage dual purpose casks – existing or new • Some fuel (24%) in this group has been disassembled and/or consolidated. Canisterization of the disassembled fuel is required for ease of handling at the repository. • For Group 1, a 10 CFR part 71 exemption may be required for movement of existing TAN dry cask to ICPP
2 MEU	• Like commercial utility fuel but with higher enrichment. Some assemblies have been disassembled. • Group Dimensions: 8 fuel types 1.4 m^3 4 MTHM • Representative fuel: PBF		Direct Disposal or Co-Disposal • Few assemblies or canisters per container • Low burnup/assume fresh fuel for criticality analysis		
3 HEU	• Like commercial utility fuel but with high enriched uranium or MOX. Some assemblies have been disassembled. • Group Dimensions: 21 fuel types 9.3 m^3 8.7 MTHM • Representative fuel: Shippingport PWR		Co-Disposal • Likely limited to single assembly or canister per container • Low burnup/assume fresh fuel for criticality analysis • Supplemental criticality control material may be needed		

Category II – Disrupted Oxide SNF

Group	Description/ Representative Fuel	Characterization and Analysis Considerations	Disposal Concept Criticality/Packaging	Estimated Performance in Repository	Interim SNF Management at INEEL
4 LEU	• Like commercial utility fuel only severely disrupted. All fuel is in canisters. • Group Dimensions: 34 fuel types 145.5 m³ 87.5 MTHM • Representative fuel: TMI-2	• Inventory per canister poorly characterized • In absence of leaching data, will require use of conservative Performance Assessment approach • Dryness criteria needed for sealed canister storage.	Direct Disposal • Assumes dryness criteria can be met • Several existing canisters per container • Low burnup/assume fresh fuel for criticality analysis Issues requiring resolution: • Potential RCRA because of cadmium • Particulates • Poorly characterized individual canisters	Dose peaks about two orders of magnitude below an equivalent amount of commercial SNF • Representative fuel to determine source term for this category is TMI-2 debris (group 4) • Used typical commercial SNF dissolution/leach model with a surface area 1,000 times higher than typical commercial SNF	• Move to dry storage in existing canisters (multiple canisters within DPC) • For some of the nonrepresentative SNF, treat whenever Direct Disposal is unacceptable • Benefits of mixing small cans in individual canisters should be investigated
5 HEU	• Like commercial utility fuel but with highly enriched and severely disrupted. All fuel is in canisters. • Group Dimensions: 44 fuel types 23.6 m³ 6.2 MTHM • Representative fuel: TORY	• For individual canisters, uranium content well characterized; however, extraneous materials not well characterized • In absence of leaching data, will require use of conservative Performance Assessment approach • Dryness criteria needed	Co-Disposal • Several existing canisters per container • Low burnup/assume fresh fuel for criticality analysis • Supplemental criticality control material may be added Issues requiring resolution: • Particulates • Poorly characterized extraneous materials (non-U) in individual canisters		

Category III – Zirconium Hydride SNF

Group	Description/ Representative Fuel	Characterization and Analysis Considerations	Disposal Concept Criticality/Packaging	Estimated Performance in Repository	Interim SNF Management at INEEL
6 MEU	• Standard fuel for test reactor. Variation is uranium loading (all is low) and minor element content. Some of this group contains Al cladding that is corroding but most is intact. • Group dimensions: Only 1 type 6.6 m^3 1.8 MTHM • Representative fuel: TRIGA (MEU)	• In the absence of leaching data, will require use of conservative Performance Assessment approach	Direct Disposal • Multiple rods or canisters per large container • Variable burnup and insufficient documentation/assume fresh fuel for criticality analysis • Supplemental criticality control material may be required.	Dose peaks is one order of magnitude below an equivalent amount of commercial SNF. • Representative fuel to determine source term for this category is TRIGA LEU (group 6) • Used dissolution /leach model 0.01 times typical commercial SNF	• Move to dry storage in existing canisters (multiple rods within canisters, with multiple canisters within DPC)
7 HEU	• Standard fuel for test reactor. Variation is uranium loading (all is low) and minor element content. Some of the group contains Al cladding that is corroding but most is intact. • Group Dimensions: Only 1 type 1.3 m^3 0.2 MTHM • Representative fuel: TRIGA (HEU)		Direct Disposal or Co-Disposal • Multiple rods or canisters per container • Variable burnup and insufficient documentation/assume fresh fuel for criticality analysis •Supplemental criticality control material may be required.		

Category IV – Metal and Alloy SNF

Group	Description/ Representative Fuel	Characterization and Analysis Considerations	Disposal Concept Criticality/Packaging	Estimated Performance in Repository	Interim SNF Management at INEEL
8 LEU	• Intact Zircalloy clad uranium metal and alloy fuel. • Group dimensions: 14 fuel types 0.8 m^3 2 MTHM • Representative fuel: HWCTR	• Uranium metal fuel with intact cladding is not expected to be hydrided – however, if degraded in storage, potential hydride and chemical reactivity. • In absence of leaching data, will require use of conservative Performance Assessment approach	Direct Disposal • Multiple assemblies or canisters per large container • Low burnup/assume fresh fuel for criticality analysis	Dose peaks more than two orders of magnitude below an equivalent amount of commercial SNF. At about 800,000 years, Ac-227 and Pa-131 increases dose slightly, but overall affect is negligible. • Representative fuel to determine source term for this category is Fermi HEU (category 9) • Used N-reactor SNF dissolution/leach model from 1994 SNL PA (approximately 2 orders of magnitude higher than typical commercial SNF)	• Move to dry storage in existing canisters (multiple rods within canisters, with multiple canisters within DPC) • Small amount of fuel may require treatment if hydriding exists. May be cost-effective to treat with Na-bonded fuel, which is Group 14.
9 HEU	• Intact Zircalloy clad uranium metal and alloy fuel. • Group Dimensions: 6 fuel types 2 m^3 3.9 MTHM • Representative fuel: Fermi driver		Co-Disposal • Multiple elements or canisters per container • Low burnup/assume fresh fuel for criticality analysis • Supplemental criticality control material may be needed		

Category V – Graphite SNF

Group	Description/ Representative Fuel	Characterization and Analysis Considerations	Disposal Concept Criticality/Packaging	Estimated Performance in Repository	Interim SNF Management at INEEL
10 HEU	• Uranium and thorium carbides in a graphite block. • Group Dimensions: Only 1 type 196 m^3 23.4 MTHM • Representative fuel: Fort St. Vrain	• Potential chemical reactivity issue (flammable gasses from water and carbide reaction) • Likely to be less leachable than commercial SNF • In absence of leaching data, will require use of conservative Performance Assessment approach (such as equating to commercial SNF)	Co-Disposal • Use existing 17" canisters: will require existing lids be seal welded • Multiple assemblies (4 or 6) per canister with single canister per container • Due to high enrichment, burnup effect on criticality analysis negligible/assume fresh fuel for criticality analysis. U-233 production from thorium less than U-235 burnup • Supplemental criticality control material may be needed	Dose peaks somewhat higher than an equivalent amount of commercial SNF. Th-229 increases dose slightly, but overall affect is negligible. • Used Fort St. Vrain SNF dissolution/leach model from 1994 SNL PA (approximately the same as typical commercial SNF)	• Leave in existing dry storage • Existing storage canisters will need to be seal welded prior to transportation
11 HEU	• Uranium and thorium carbide in graphite rods. • Group Dimensions: 7 fuel types 35 m^3 3 MTHM • Representative fuel: Peachbottom	• In absence of leaching data, will require use of conservative Performance Assessment approach • Potential chemical reactivity issue (flammable gasses from water and carbide reaction)	Co-Disposal • Use existing 17" canisters that require seal-welded lids • Place existing basket (18 assemblies per basket) in a 24" canister with single canister per container • Due to high enrichment, burnup effect on criticality analysis negligible/assume fresh fuel for criticality analysis. U-233 production from thorium less than U-235 burnup • Supplemental criticality control material may be needed	Dose peaks about the same as an equivalent amount of commercial SNF Th-229 increases dose slightly, but overall affect is negligible. • Representative fuel to determine source term for these groups is Peachbottom Core 2 (category 11) • Used 10 times Fort St. Vrain SNF dissolution/leach model from 1994 SNL PA (approximately 7 orders of magnitude higher than typical commercial SNF)	• Move to dry storage in existing canister (multiple canisters within DPC) • Canisterize basket prior to transportation
12 HEU	• Uranium carbide or uranium oxide in graphite rods clad with metal. • Group Dimensions: 2 fuel types 5 m^3 0.06 MTHM • Representative fuel: SRE		Direct Disposal • Due to low fissile material density, loading full inventory in one large container is possible • Due to low burnup and high enrichment, burnup effect on criticality analysis negligible/assume fresh fuel for criticality analysis • Supplemental criticality control material may be needed		• Leave in TREAT reactor vessel until reactor termination, then move to dry storage in DPC • Move rest to dry

Category VI – Thorium Oxide SNF

Group	Description/ Representative Fuel	Characterization and Analysis Considerations	Desposal Concept Criticality/Packaging	Estimated Performance in Repository	Interim SNF Management at INEEL
13 HEU	• Uranium and thorium oxide fuel in intact assemblies. • Group dimensions: Only 1 fuel 51.5 m^3 39 MTHM • Representative fuel: Shippingport LWBR	• In absence of leaching data, will require use of conservative Performance Assessment approach (it is believed that this fuel will behave better than commercial fuel)	Direct Disposal • Probably single assembly in existing canister per container • Breeding ratio greater than 1/assume end of life enrichment for criticality analysis (good records and confirmation by destructive examination exists) • Supplemental criticality control material may be needed	Dose peaks somewhat lower than an equivalent amount of commercial SNF. Th-229 and U-233 increases dose slightly, but overall affect is negligible. • Used Ceramic form dissolution/leach model from TSPA-95 (approximately 2 orders of magnitude lower than typical commercial SNF)	• Move to dry storage in DPC

Category VII – Other SNF (10% by volume)

Group	Description/ Representative Fuel	Characterization and Analysis Considerations	Disposal Concept Criticality/Packaging	Estimated Performance in Repository	Interim SNF Management at INEEL
14 Varied uranium enrich-ment.	• Metallic Na containing fuel. • Group Dimensions: 33 fuel types 14.6 m^3 60 MTHM • Representative fuel: EBR-II	• Chemical Reactivity (RCRA) issue – not acceptable for disposal in current form	Disposal Concept pending • Waste form characteristics need further study and will be managed by appropriate classification	• Waste form will need to be shown to meet repository WAC	• Treat to remove metallic sodium (candidate treatment is electrometallurgical processing)
15 HEU	• Al-clad fuel some of which has cladding corrosion. • Group Dimensions: 14 fuel types 37.5 m^3 3.4 MTHM • Representative fuel: ATR	• Fuel similar to Savannah River Site fuel	Co-Disposal • Co-disposal option identified in the report from the Research Reactor SNF Task Team	• RR SNF task team report indicated no impact on the repository	• Move to new dry storage • Ship to Savannah River Site or store at INEL in DPCs for later direct shipment to repository
16 HEU	• Small quantities of unique fuel that does not fit into any other group. • Group Dimensions: 5 fuel types 4.3 m^3 0.2 MTHM • Representative fuel: MSRE	• Each fuel will need to be addressed individually. Some of the fuels have characteristics that are not acceptable for disposal.	Disposal Concept pending • Disposition alternative will need to be determined after final form has been determined	• Large variations in quantity and characteristics of each fuel • Benefits from treating the different fuels in this group should be investigated.	• Move to or receive in dry storage • Some of these fuels will require some treatment prior to final disposition.

Section 5

IMPLEMENTATION – The Path Forward

The technical strategy outlined in Section 4 integrates the various technical elements considered by the Team to be central to the problem. It does not, however, incorporate the programmatic factors – schedule, cost, and stakeholder commitments – which are equally important in achieving a viable solution. Although the Team did not examine these matters in depth, they are addressed here in summary fashion.

5.1 Management Considerations

The path forward must be consistent with existing political and stakeholder agreements and plans regarding the INEEL SNF, as outlined in Section 2.3 of this report. These include the Settlement Agreement between DOE and the State of Idaho, the vulnerability action plan, and the Ten-Year Plan. Other constraints that must be accommodated are the uncertainties regarding timing of the repository, availability of funding, and the finalization of regulatory requirements.

The Team has attempted to produce a strategy that takes these constraints into account, and which will not cause INEEL to miss any required actions. INEEL will need to validate this strategy, and implement in a way that provides adequate margin and flexibility to deal with the uncertainties and meet established requirements.

5.2 Integrated Path Forward

To convert the integrated strategy of Table 4-1 into a practical path forward, schedule logic is needed which identifies primary sequences, priorities and activity dependencies. The Team's initial view of such a schedule logic is shown diagrammatically as Figure 5-1. This diagram is consistent with the Table 4-1 strategy and it reflects the anticipated primary sequence of activities to package, transport, and store the INEEL SNF. (Note that this is a top-level logic diagram, and that it displays the physical sequence of activities. Critical supporting work, such as the analytical and requirements development activities are not shown.)

The Team agrees that the near-term INEEL action plans, as established in the INEEL SNF management plan (Ref P) and the vulnerability plans, are appropriate and need to be continued. These plans are necessary to meet existing agreements and they also fit well into the strategy proposed in this report.

The next steps in refining the path forward are the development of more detailed logic and the allocation of activity durations and costs. That was not done as part of this evaluation.

5.3 Privatization Opportunities

In recent years, DOE has employed privatization as a business strategy that can best leverage commercial market forces in the clean-up of the Complex. In principle, privatization gives to capable private contractors the freedom, control, and financial incentive to efficiently complete contractually authorized work; from DOE's standpoint, privatization can provide better predictability of cost, and schedule, and it shifts accountability and risk to the organization (the contractor) best able to handle them.

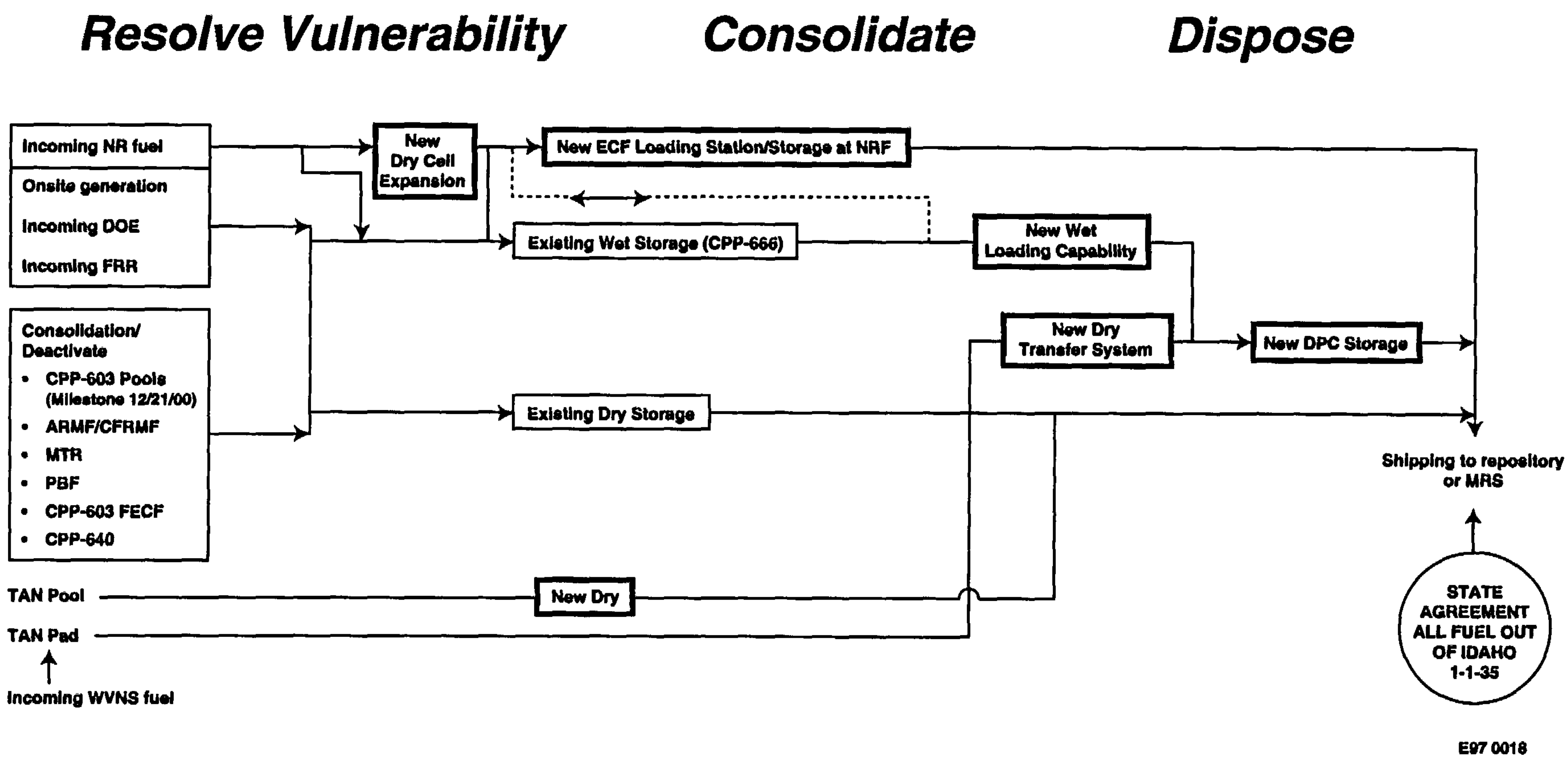

Figure 5-1 INEEL SNF management strategy.

An important benefit of establishing an overall path forward at the outset of a major program like this one, is that it provides management an early opportunity to identify and pursue imaginative contractual strategies, such as privatization, which have potential to benefit the government and the taxpayers.

While business and contractual tactics were not an explicit part of the Team's charter, some general observations in this respect are offered:

- Clearly, there are privatization opportunities in the INEEL SNF Program. An obvious example is the construction and operation of interim dry storage facilities.
- In the Team's view, privatization is a sensible business approach in circumstances in which the end product can be well defined, the requirements are clear and stable, and the work scope is isolatable from and essentially unaffected by other activities. This suggests a point of caution for the SNF work, particularly in the near term. There remains today significant uncertainty as to the ultimate technical requirements for characterization, analysis, and repository disposal of SNF. For many tasks, privatization (or other contractual arrangements that involve sharing or transfer of significant risk) should not be pursued until these uncertainties are resolved.

5.4 Findings and Recommendations

The Team reached various conclusions in the course of its evaluations, as detailed in the preceding sections of this report. The most important of these are summarized below, along with associated recommendations regarding subsequent actions.

Overall Strategy for INEEL SNF

The Team's conclusions with respect to an overall technical strategy for storage, handling, packaging, and disposal of the INEEL SNF are as presented in the tables in Section 4. Implicit in this recommended path is the conclusion that the INEEL SNF can be safely packaged, stored and transported, using methods based on current, proven technology. However, significant adaptation and analytical work will be needed to apply this proven technology to the INEEL SNF and to establish a technically sound, NRC-approved basis for implementation. Present actions onsite to resolve near-term vulnerabilities are appropriate and consistent with the proposed longer-term path forward. In most cases, processing or treatment will not be required to render the SNF suitable for repository disposal.

Based on this preliminary work, the Team recommends that DOE:

1. Continue present path to refine this conceptual strategy, including supporting criticality analysis, performance assessments, and further engineering work.
2. Continue the working interfaces between EM and RW.
3. Engage the NRC in the envisioned process for the qualification of the INEEL SNF for the repository, as soon and as directly as possible.

Characterization

The Team evaluated the technical need for characterization, anticipated regulations and guidelines, availability of characterization data, methods and facilities for acquiring data, and potential ways to improve the cost-effectiveness of characterization activities. The Team finds that:

- Characterization requirements for the DOE SNF are not yet well defined.

- SNF should be characterized to the degree necessary to permit reasonable prediction of its performance in storage, transport and repository disposal. DOE SNF characterization requirements need not be the same as those currently prescribed for commercial SNF.
- Based on technical need (in the Team's view), sufficient characterization information is already available for more than 90% of the INEEL inventory of SNF, without additional examination. However, the data must be demonstrated to meet the Quality Assurance requirements as defined in RW-0333P.
- For that SNF which is to be chemically treated, characterization should be limited to that necessary to ensure treatment effectiveness. (A waste suitable for repository disposal will be produced during the subsequent treatment.)
- Even for conservative projections of characterization requirements (i.e., based on current requirements for commercial fuel), existing INEEL facilities should be adequate to meet program characterization needs.

The Team recommends the following DOE actions, regarding characterization:

1. Continue to refine characterization requirements for SNF, based strictly on the need to determine SNF performance. (This is currently in progress, and a report is scheduled to be issued by the National Spent Nuclear Fuel program, in March 1997.)
2. Continue to collect and qualify data that has been determined to be necessary for disposal.
3. Engage NRC in the development of SNF characterization and analysis strategy. Secure NRC concurrence to the degree possible.

Criticality

The Team performed preliminary evaluations of in-repository criticality performance for one of the INEEL SNF groups, and inferred conclusions regarding criticality potential for several others. Based on this limited analysis, the team concludes that:

- Repository criticality safety (particularly for SNF with higher enrichments) can be achieved through proper package design. Design features for criticality control can include limitations on the amount of neutronic reactivity contributed by fissile content, and/or incorporation of neutron poison or moderator exclusion materials. See Section 3.2.
- Co-disposal of packaged HEU or MEU with high level waste is a simple and conservative way to achieve long term repository criticality safety.
- Criticality safety will not constrain LEU waste packaging. Large, dedicated (SNF only) packages can be used. See section 3.2.

The Team recommends that DOE:

1. Proceed with more extensive disposal criticality evaluations, as described in Section 3.2.
2. Engage NRC in the DOE work to develop and refine criticality analysis methods, and in the development of safe packaging concepts; secure NRC review and comment, to the degree practical.
3. Continue to pursue with NRC the proposed change to 10CFR60 to permit use of risk-based analyses to demonstrate criticality safety.

Performance Assessment

A preliminary performance assessment scoping analysis was performed for several INEEL SNF types, using methods currently employed for evaluation of commercial SNF performance in the repository. Based on that work, the Team finds that Repository disposal of INEEL SNF in the OCRWMS repository would contribute only a very small increment to the overall projected peak annual dose to persons in the accessible environment. See Section 3.4

The Team recommends that:

1. Conduct more detailed performance assessments for the INEEL SNF using refined inputs.

Special Fuels

The Team finds that:

- For the small quantities of SNF (see Table 2.1-1), the most cost-effective repository disposal approach will likely be to package multiple fuel types together. Performance assessments for these combined fuel packages should be based on conservative bounding assumptions, and characterization requirements should be limited accordingly. See Section 3.3. It may also be cost-effective to process some of these small-quantity fuels, particularly in cases when characterization costs are likely to be high.
- Sodium-bonded fuels (approximately 3% by volume of the INEEL inventory) are not suitable for repository disposal and therefore must be treated. See Section 2.3.

The Team recommends that DOE:

1. Conduct repository evaluations for combined packaging of selected small quantity INEEL SNF.
2. Evaluate whether it is cost-effective to treat or process these SNF types.
3. Engage the NRC early in developing suitable packaging and analysis approaches for small quantities of SNF.
4. Proceed with the technical work needed to qualify the Electro-metallurgical process, or an alternative process, for treatment of the sodium-bonded fuel.

Packaging and Transportation

The Team findings regarding packaging and transportation are:

- Current dual-purpose container (DPC) designs do not address long-term criticality control in the degraded condition. As a result, they are not currently considered appropriate for repository disposal. Their potential use for disposal would depend on meeting the repository design criteria, when available, as constructed or modified.
- Simple, standardized and relatively small cylindrical canisters (nominal diameters of 10, 17, and 24 inches have been evaluated) appear to provide adequate criticality safety and optimal packaging flexibility for the INEEL HEU and MEU SNF.
- In many cases, it may be acceptable (and consistent with the "road ready" requirements) to utilize the existing SNF canisters for on-site staging, provided they meet repository design requirement, and that the preparations for transportation (e.g., over packing) are reasonably simple and can be accomplished in a short time and with available facilities. See Section 3.3.

The Team recommends that DOE:

1. Begin the development of standard canister designs suitable for disposal of HEU and MEU fuel.
2. Begin integration of the INEEL SNF in the interfaces between DOE and NRC to introduce the approach and conclusions of these analyses.

Section 6

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HUGH BENTON

B.S. Electrical Engineering, United States Naval Academy
U. S. Naval Nuclear Power Training Program

Mr. Benton served with the U. S. Navy for 29 years, mostly in the nuclear power program. His assignments included service on the new constructions and operating crews of four nuclear powered submarines, command of groups of up to ten nuclear powered and support ships, and command of the Charleston, S.C. Naval Base which included a nuclear qualified shipyard. He was also in charge of the team responsible for the inspection of nuclear powered ships in the Pacific Fleet. He completed his naval service as Deputy Chief of Naval Personnel.

Mr. Benton was employed by the Babcock & Wilcox Company for 10 years. As Manager and later Vice President for Ventures and Licensing of Babcock & Wilcox International, he established and supervised companies including manufacturing plants in China, India, and Indonesia, and engineering and marketing companies in Mexico and Turkey. In 1991, Mr. Benton joined the B&W Fuel Company (now Framatome Cogema Fuels), then a subsidiary of the Babcock & Wilcox Company, as a Manager of Waste Package Development for the OCRWM M&O contractor. The group currently consists of 24 scientists and engineers, and is developing the technical basis and designing the containers for commercial spent nuclear fuel and defense high-level waste, as well as the other engineered barriers in the potential repository.

Mr. Benton was a member of the core team of the Research Reactor Spent Nuclear Fuel Task Team.

TERRY L. BRADLEY

B. S. Civil Engineering, North Carolina State University
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Mr. Bradley has more than 25 years of experience in the commercial nuclear industry and is currently a senior project engineer at Duke Engineering and Services Inc. Mr. Bradley has architectural and engineering experience with projects related to spent fuel storage, and is a member of the American Nuclear Society committee 57.9 (spent fuel storage) and the American Society for Testing Materials Committee E10.11.06 (spent fuel storage and transportation).

Mr. Bradley has provided support in the area of spent fuel storage to commercial utilities and the Department of Energy. He participated as a member of the Independent Technical Assessment Team for Dry Storage of N Reactor Fuel at Hanford, evaluating the feasibility of stabilization and dry storage of N Reactor spent fuel at the Hanford K basins. In addition, he is a member of the Technical Assistance Group for providing support for the design, licensing, construction, and operation of a dry spent fuel storage facility for N Reactor.

Mr. Bradely was a member of the core team for the Research Reactor Spent Nuclear Fuel Task Team.

EDWARD M. BURNS

J.D. Law, Duquesne University

M.S. Nuclear Engineering, University of Wisconsin

M.S. Management Information Systems, University of Southern California

B.S. Mechanical Engineering, Milwaukee School of Engineering

An attorney and engineer, Mr. Burns has over 20 years experience in the nuclear field, from nuclear weapons in the Army to nuclear licensing at Westinghouse. As nuclear plant licensing manager in the 1980's, he managed NRC, ACRS, and ASLB licensing actions associated with the startup of the Diablo Canyon, Comanche Peak, and Vogtle nuclear plant in the U.S. and Krisko (Yugoslavia), Maanshan (Taiwan), Korea 5-8 nuclear plants overseas. Since 1988, he has developed environmental and regulatory strategies for advanced plant designs including the Westinghouse Advanced Passive (AP-600) nuclear plant, a hardened 10 Mwe plant for the Air Force, and a new heavy water reactor for the Department of Energy. Mr. Burns has worked as a senior technical and regulatory advisor in DOE's Office of Spent Fuel Management since its inception in 1993.

Mr. Burns was a member of the technical support team of the Research Reactor Spent Nuclear Fuel Task Team.

CARL DETRICK

Ph.D. Nuclear Science and Engineering (Neutron Physics), Carnegie-Mellon University

M.S. Nuclear Science and Engineering (Reactor Physics), University of Virginia

B.S. Physics, Western Illinois University

Dr. Detrick has more than 29 years of experience in the design, analysis, manufacture and construction, testing, and operation of naval nuclear propulsion plants. He also has in-depth experience in emergency planning, radiological and hazardous waste management, and environmental evaluations for the Naval Nuclear Propulsion Program and in the development of models, methods, and computer programs for neutron transport, kinetics, and shielding calculations for nuclear reactors.

Dr. Detrick has been employed by the Westinghouse Electric Corporation at Bettis Laboratory since 1967. He is currently Manager, Environmental Affairs Activity at Bettis, reporting to the Laboratory General Manager. He has qualified as a Test Physicist, shipyard Joint Test Group member and Joint Refueling Group member, Engineering Officer of the Watch, and Prototype Plant Manager with the Naval Nuclear Propulsion Program. He has been associated at one time or another with the design, operation, or analysis of every naval nuclear core design under Bettis cognizance. He has served on the Bettis Reactor Operating Safety Committee and is currently serving for the second time on the Bettis Advisory Safety Committee. He is also a member of the Naval Reactors Senior Panel for Human Reliability Assessment.

Dr. Detrick was a member of the technical support team of the Research Reactor Spent Nuclear Fuel Task Team.

JOHN (JACK) C. DEVINE, JR.

B.S. Mathematics, United States Naval Academy
U.S. Navy Nuclear Power Training Program

Mr. DeVine has more than 30 years of nuclear power experience including service with the U.S. Navy as a commissioned officer in the nuclear submarine force. Mr. DeVine also is experienced in the recovery of degraded nuclear fuel, including work as Recovery Engineering Manager at Three Mile Island Unit 2 following the March 1979 nuclear accident.

Mr. DeVine was employed with General Public Utilities Corporation (GPU) for 22 years, most recently as Vice Present and Director, Technical Functions, responsible for engineering support of the corporation's nuclear stations. He was a member of the GPU Nuclear Board of Directors, the Project Management Board of the Advanced Reactor Corporation, and the Executive Board of the Edison Electric Institute Utility Waste Management Group.

Mr. DeVine is co-founder of Polestar Applied Technology, Inc., a management and engineering services firm. He is currently providing technical support to the Department of Energy (DOE) in the evaluation of spent nuclear fuel options at the Savannah River Site. He also functions as an Independent Technical Expert providing review and assistance to the DOE (EM-63) for the deactivation of the PUREX/UO3 facilities at the Hanford Site, and deactivation of ROVER fuel processing facility at the Idaho National Engineering and Environmental Laboratory. Previously, he served as Chairman of the Independent Technical Assessment Team for Dry Storage of N Reactor Fuel at Hanford. This team established the technical feasibility and developed a conceptual engineering approach for packaging, transport, stabilization and dry storage of the spent nuclear fuel at the Hanford K Basins

Mr. DeVine was the Team Leader of the Research Reactor Spent Nuclear Fuel Task Team.

DENZEL L. FILLMORE

Ph.D. Inorganic Chemistry, Brigham Young University

B.S. Inorganic Chemistry, Brigham Young University

Dr. Fillmore has over 25 years experience in a diversity of nuclear industry positions. He has extensive knowledge in nuclear fuel design, characteristics, material properties, potential hazardous material, and the Department of Energy (DOE) fuel inventory.

Dr. Fillmore began his career with the Bettis Atomic Power Laboratory at the Naval Reactor Facility by qualifying on a nuclear submarine prototype. He then qualified Westinghouse and Naval personnel to operate Naval Nuclear Reactors. For the last eight years, Dr. Fillmore has worked at the Idaho National Engineering and Environmental Laboratory (INEEL) in the Spent Nuclear Fuel programs. He aided in the establishment of both the INEEL and National SNF programs. He participated in the development of the DOE SNF Technology Integration Plan, and developed the SNF inventory and characteristics information used in the preparation of the INEEL Environmental Impact Statement. Presently Dr. Fillmore is supporting the INEEL and National programs in the areas of SNF characterization, data collection and analysis, and program development.

Dr. Fillmore was a member of the core team of the Research Reactor Spent Nuclear Fuel Task Team.

ALAN P. HOSKINS

B.S. Chemical Engineering, Montana State University

Mr. Hoskins has over 23 years of experience in the nuclear field at the Idaho National Engineering and Environmental Laboratory (INEEL) working in the areas of spent nuclear fuel management, radioactive waste storage and treatment, nuclear facility design, and facility start-up and operations.

Mr. Hoskins began his career at the INEEL as a process support engineer during the Decontamination and Decommissioning (D&D) of the EBR-I reactor. He worked in the high-level waste programs and was responsible for preparing and conducting the system operational tests for the start-up of the New Waste Calcination Facility (NWCF). He was the first NWCF facility manager responsible for initial "hot" start-up. He has been the Idaho Chemical Processing Plant (ICPP) Plant Shift Manager and Shift Operations Manager, responsible for all ICPP operations. He was the Technical Process Analysis Manager, responsible for SNF reprocessing analysis improvement, and the Calcine Immobilization Manager, directing technology development and demonstration on high-level waste dispositioning methods. Presently Mr. Hoskins is the manager of the INEEL SNF program, responsible for the dispositioning of all of the INEEL SNF.

Mr. Hoskins has served on the DOE-HQ National SNF Program, the DOE-HQ High-Level Waste Tank Safety Task Force, national and international technical review panels for high-level waste and consulting assignments to Japan's high-level waste solidification facilities.

DANNY R. INGLE, PE

B.S.E. Urban and Environmental Engineering, University of North Carolina at Charlotte

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Mr. Ingle is a Civil Engineer with 20 years in the nuclear and non-nuclear power utility industry. He is currently a Senior Engineer at Duke Engineering and Services and has broad engineering experience in nuclear projects including spent nuclear fuel storage.

At Duke Engineering and Services, Mr. Ingle has participated in DOE and commercial spent fuel storage projects such as the Hanford N Reactor fuel storage project and Commonwealth Edison's Dresden Station Independent Spent Fuel Storage Installation (ISFSI). Mr. Ingle is thoroughly familiar with the current dry spent fuel storage technologies and suppliers, and has had lead responsibilities for developing technical specifications and providing engineering support for procurement, project management, licensing, design, construction, plant modifications, fuel handling, and operations of ISFSIs. He has performed structural analysis and design of spent fuel pool structures. Mr. Ingle is currently the lead civil engineer responsible for design of a low level nuclear waste processing facility.

LEROY LEWIS

Ph.D. Physical Chemistry, Oregon State University

B.S. Chemistry and Mathematics, College of Idaho

Dr. Lewis has more than 29 years of experience in nuclear fuel management. This includes: nuclear fuel reprocessing development and processing support, nuclear waste processing, spent fuel storage, decontamination and decommissioning (D&D) of nuclear facilities, and remote analytical chemistry laboratory design and operation.

Dr. Lewis has been employed at the Idaho Chemical Processing Plant since 1968 and is currently a Science and Engineering Fellow for Lockheed Martin Idaho Technologies Company. He has been involved in all facets of spent nuclear fuel management. He has three patents and approximately 50 publications in the areas of process development, fuel and waste processing, D&D of facilities, and Chemical History. He is a member of the American Society for Testing Materials Committee C-26 on the Nuclear Fuel Cycle, and Institute of Nuclear Materials Management Standards Committees 5.1 on Quality Assurance for Chemical Measurements on Nuclear Materials

HENRY H. LOO

B. S. Chemical Engineering, University of California Berkeley

Mr. Loo has more than 20 years experience in areas such as performance assessment, and waste form criteria and requirements for disposal of spent nuclear fuel and high-level waste. His background includes process design, procurement, and installation of a radioactive fluid bed waste calcination and a fuel reprocessing process; and process design of a vitrification facility for radioactive waste.

Mr. Loo has functioned as a contractor at the Idaho National Engineering and Environmental (INEEL) for 15 years. In this capacity, Mr. Loo has managed work assignments aimed at determining the performance of various spent fuel and waste forms destined for permanent disposal in geologic repositories. He has also been involved in the development of the waste form criteria and requirements document for repository disposal. In addition, he spent three years as a process engineer for the Defense Waste Processing Facility. Mr. Loo is currently employed at the INEEL with Lockheed-Martin Idaho Technologies Company, working on the Department of Energy Spent Nuclear Fuel programs.

Mr. Loo was a member of the support team of the Research Reactor Spent Nuclear Fuel Task Team.

PHYLLIS M. LOVETT

M. S. Nuclear Engineering, Massachusetts Institute of Technology

B. S. Nuclear Engineering. Georgia Institute of Technology

Ms. Lovett has over 10 years experience in the nuclear industry, in both the commercial and government arenas. Ms. Lovett earned her Masters of Science degree from the Massachusetts Institute of Technology (MIT) in the area of experimental thermal hydraulics, and was selected as an Institute of Nuclear Power Operations Fellow.

Ms. Lovett began her career with Virginia Power, as a licensing engineer working with the Nuclear Regulatory Commission on projects related to the Surry Power Station independent spent fuel storage installation, power plant physical protection, emergency planning, and decommissioning. She then became Shift Technical Advisor working on shift as part of the operations team for the Surry nuclear unit.

Ms. Lovett spent five years with TRW Environmental Safety Systems, Inc., the Management and Operating contractor for the Department of Energy (DOE) Office of Civilian Radioactive Waste Management (OCRWM). Ms. Lovett was the lead technical contractor on DOE spent nuclear fuel and other high-level waste for the OCRWM. She authored the baseline requirements for both waste acceptance and transportation by U.S. high-level waste programs, as well as other waste form criteria documents.

Ms. Lovett is presently associated with Shaw, Pittman, Potts, and Trowbridge.

Ms. Lovett was a member of the core team of the Research Reactor Spent Nuclear Fuel Task Team.

ROGER McCORMACK

B.S. Chemical Engineering, Oregon State University

Mr. McCormack has more than 16 years of experience in the nuclear field and is currently a project manager for Duke Engineering and Services Hanford, Inc. Mr. McCormack is responsible for activities to implement long-term interim storage of DOE's non-defense production reactor spent nuclear fuel at the Hanford Site. In addition, he interfaces with the National Spent Nuclear Fuel Program to develop final disposition plans for all DOE-owned spent nuclear fuel at the Hanford Site.

As an employee of Westinghouse Hanford Company in the 1980's and early 1990's, Mr. McCormack completed various engineering and management assignments. He managed engineering organizations responsible for completion of feasibility studies for spent nuclear fuel management, process design and processing technology support for metallic and mixed oxide spent fuel processing, and processing engineering support. As a principal engineer, he was responsible for various process system improvements required for restart, operation, and life extension of a nuclear processing facility.

THOMAS P. McLAUGHLIN

Ph.D. Nuclear Engineering, University of Arizona

Dr. McLaughlin has over 25 years experience in nuclear criticality safety, reactor safety research, reactor design, and the performance of critical experiments. He consults the Department of Energy (DOE), DOE contractors, and private sector firms on nuclear criticality safety and related matters. He is chairman of the American Nuclear Society Standards Subcommittee 8, which is responsible for the development of all U.S. consensus standards in criticality safety.

Dr. McLaughlin has worked at Los Alamos National Laboratory (LANL) since 1972, and is currently the Group Leader for Nuclear Criticality Safety. He is responsible for criticality safety of all weapons and on weapons activities involving fissile material within LANL and throughout the Department of Defense/military complex. He also designs and instructs courses in criticality safety.

Dr. McLaughlin was a member of the core team of the Research Reactor Spent Nuclear Fuel Task Team.

J. RICHARD MURPHY

M. S. Mechanical Engineering, University of Texas at Austin

B. S. Mechanical Engineering, University of Texas at Austin

Mr. Murphy has been employed at the Savannah River Site (SRS) by Westinghouse Savannah River Company (WSRC), and its predecessor, I. E. DuPont, for 14 years. Assignments have included a variety of engineering and management positions. Technical positions have been in the areas of reactor system engineering, reactor safety engineering, project management, and economic development. Management positions have included reactor systems engineering and reactor fuel engineering teams. Other advisory and evaluation activities have included Technical Lead for the SRS Reactor Restart Joint Test Group, and being primary participant in the development for the DOE/EM National Spent Nuclear Fuels Technology Integration Plan.

Mr. Murphy coordinated technology development activities for Spent Nuclear Fuels as part of the WSRC Spent Nuclear Fuel Program Management Group. His activities include coordinating the funding for SNF Research and Development, monitoring the progress of specific R & D tasks, integrating the R&D with the site engineering projects and interfacing with other SNF technology development programs throughout the DOE complex.

Mr. Murphy was a member of the core team of the Research Reactor Spent Nuclear Fuel Task Team.

ROBERT PAHL

Ph.D. Materials Science and Engineering, Northwestern University

B.S. Materials Science and Engineering, Wilkes College

Dr. Pahl has 15 years of research experience in nuclear fuels performance, testing, and characterization. He has worked at Argonne National Laboratory-West since 1982 in the area of liquid metal cooled fast reactor fuel performance. Dr. Pahl has been responsible for developing driver fuel for the EBR-II reactor in Idaho and was Section Manager for the Fuel Performance Section. His section developed and tested the prototype metallic fuel for the Integral Fast Reactor Program.

Dr. Pahl's current research interests center on spent fuel treatment and performance testing. He currently coordinates work in spent fuel program development for Argonne National Laboratory-West and interfaces with local and national spent fuel programs throughout the DOE complex.

D. KENT PARSONS

Ph.D. Nuclear Engineering, Massachusetts Institute of Technology

M.S. Nuclear Engineering, Massachusetts Institute of Technology

B.S. Nuclear Engineering, Texas A&M University

Dr. Parsons has over 12 years of experience in computational neutronics, reactor physics, radiation shielding, and nuclear criticality safety. He began his career at the Idaho National Engineering and Environmental Laboratory working in the area of reactor design (e.g., the early design of the Advanced Neutron Source reactor), radiation shielding (Boron Neutron Capture Therapy at PBF), and neutronics code development (ANISN/PC).

Dr. Parsons has been with the Los Alamos National Laboratory (LANL) for the past seven years. He has applied many different neutronics codes to the analysis of practical problems - to classified applications, to pressure vessel embrittlement analysis, and to nuclear criticality safety. Dr. Parsons currently works in the Nuclear Criticality Safety Group, where he has been responsible for criticality safety analysis at the various LANL facilities which handle and process special nuclear material. He has also been an active participant in the ongoing LANL underground Supercriticality Review.

ELWOOD P. STROUPE

J.D. Law, University of Santa Clara

M.S. Chemical Engineering, Purdue University

B.S. Chemical Engineering, Rose Hulman Institute of Technology

Mr. Stroupe has over 30 years of management experience in spent nuclear fuel, environmental compliance, technical consulting, and nuclear power plant licensing. His extensive experience with managing technical and consulting services spans several different companies. The types of organizations he has managed include business systems, software development, engineering, nuclear licensing, financial, and project management.

Mr. Stroupe is currently the manager of the National Spent Nuclear Fuel Department at the Idaho National Engineering and Environmental Laboratory which is managed by Lockheed-Martin Idaho Technologies. In this position, Mr. Stroupe routinely interfaces with numerous governmental entities, commissions, laboratories, and subcontractors.

Mr. Stroupe was a member of the core team of the Research Reactor Spent Nuclear Fuel Task Team.

Acronyms

ALARA	As Low as Reasonably Achievable
ANL-W	Argonne National Laboratory - West
ANL-E	Argonne National Laboratory - East
ATR	Advanced Test Reactor
BOL	Beginning of Life
CPP	Chemical Processing Plant
CPP-603	Spent Fuel Wet Storage Building at CPP
CPP-666	Modern Spent Fuel Wet Storage Building at CPP
CPP-684	Remote Analytical Laboratory at CPP
CPP-749	Dry, Below Grade Fuel Storage at CPP
DOE	Department of Energy
DOE/RW/0333P	QA Guidance Document
DOT	Department of Transportation
DPC	Dual Purpose Cask
DTC	Dry Transfer Cell
EBR-II	Experimental Breeder Reactor, No. 2
EBWR	Experimental Boiling Water Reactor
EIS	Environmental Impact Statement
EM	Office of the Assistant Secretary for Environmental Management
EM-Process	Electro Metallurgical Process
EOL	End of Life
EPA	Environmental Protection Agency
FAST	Fluorinel and Storage Facility at CPP (CPP-666)
FCF	Fuel Conditioning Facility at ANL-W
FERMI	Liquid Metal Cooled Reactor near Detroit, Michigan *(named for Enrico Fermi)*
FSVR	Fort St. Vrain Reactor
GMODS	Glass Material Oxidation and Dissolution System
HEU	High Enriched Uranium
HFEF	Hot Fuel Examination Facility at ANL-W
HLW	High Level Waste
HWCTR	Heavy Water Cooled Test Reactor
IAEA	International Atomic Energy Agency
ICPP	Idaho Chemical Processing Plant
IFSF	Irradiated Fuel Storage Facility
INEEL	Idaho National Engineering and Environmental Laboratory
ISFSI	Independent Spent Fuel Storage Installation
Keff	Neutron Multiplication Coefficient
LEU	Low Enriched Uranium (<5% U-235)
LMITCO	Lockheed Martin Idaho Technologies Company
LWBR	Light Water Breeder Reactor

MEU	Medium Enriched Uranium 5% <U-235 < 20%
MPC	Multi Purpose Cask
MTHM	Metric Tonnes Heavy Metal
MSRE	Molten Salt Reactor Experiment
MTR	Material Test Reactor (at INEEL)
NEPA	National Environmental Policy Act
NISF	National Interim Storage Facility
NRC	Nuclear Regulatory Commission
NRF	Naval Reactor Facility
NUHOMS®	Nutech Horizontal Modular Storage System
NUPAC	Nuclear Pacific (manufacturer of casks)
OCRWM	Office of Civilian Radioactive Waste Management
ORNL	Oak Ridge National Laboratory
PA	Performance Assessment
PBF	Power Burst Facility
PBF-620	Power Burst Reactor Building with Storage Pool at INEEL
PNNL	Pacific Northwest National Laboratory
PWR	Pressurized Water Reactor (commercial reactors)
QA	Quality Assurance
RAL	Remote Analytical Laboratory (at ICPP)
RCRA	Resource Conservation and Recovery Act
ROD	Record of Decision
RSWF	Radioactive Scrap and Waste Facility (at ANL-W)
RW	Office of Civilian Radioactive Waste
SAIC	Science Applications International Company
SNF	Spent Nuclear Fuel
SNL	Sandia National Laboratory
SRE	Sodium Reactor Experiment
SRS	Savannah River Site
TAN	Test Area North (at the INEEL)
TAN-607	Hot Shop and Fuel Storage Basin (at Test Area North)
TMI-2	Three Mile Island Reactor No. 2
TORY	Nuclear Ramjet Reactor
TRA	Test Reactor Area (at the INEEL)
TRA-603	Material Test Reactor Building containing Storage Canal
TRA-660	Zero Power Reactor Pool at the Test Reactor Area
TRA-670	Advanced Test Reactor Building including Working Canal
TREAT	Transient Reactor Experiment and Test
TRIGA	Training, Research, Irradiation Reactors from General Atomics
TSPA	Total System Performance Assessment
WAC	Waste Acceptance Criteria
10 CFR 60	Section of the Code of Federal Regulations Governing the Repository
10 CFR 71	Section of the Code Governing Interim Storage of SNF
10 CFR 72	Section of the Code Governing Transportation of SNF