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Characterization Plan for Fort St. Vrain and Peach Bottom Graphite Fuels

S. C. Marschman
F. M. Berting
R. G. Clemmer
E. R. Gilbert
R. J. Guenther
W. C. Morgan
P. Sliva

September 1993

Prepared for
Westinghouse Idaho Nuclear Company, Inc.
under a Related Services Agreement
with the U.S. Department of Energy
under Contract DE-AC06-76RLO 1830

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PREFACE

The work described in this report was completed September 1993, with no additional research occurring between September 1993 and October 1996. Funding constraints in 1993 prevented publication at that time. Publication of this report was funded by the Pacific Northwest National Laboratory, operated for the U.S. Department of Energy under Contract DE-AC06-76RLO 1830. Due to the original report being written prior to 1996, style conventions for this report reflect those in use at that time.

SUMMARY

Portions of the Fort St. Vrain (FSV) and most of the Peach Bottom (PB) reactor spent nuclear fuels are currently stored at the Idaho National Engineering Laboratory (INEL) site. Westinghouse Idaho Nuclear Company (WINCO) has the responsibility for developing a strategy for the safe storage and ultimate disposal of these fuels in a repository. These fuels may remain in storage for many years before disposal.

Although these fuels may already exist in a form suitable for disposal, it may be necessary to condition these fuels prior to disposal. WINCO has proposed three basic pathways for disposal; intact disposal of the fuels, fuels partially disassembled and the high-level waste fraction conditioned prior to disposal, and fuels completely disassembled and conditioned prior to disposal. Many options exist within each of these pathways and in order to select an appropriate pathway for storage and disposal, each needs to be evaluated for feasibility, practicality, effectiveness, risk, safety, and cost.

In support of the WINCO activities, Pacific Northwest Laboratory (PNL) was contracted to perform an evaluation of the literature and other references to develop a fuels characterization plan for the PB and FSV graphite fuels stored at the INEL. This plan provides guidance for the characteristics of the fuel which will be needed to pursue any of the storage or disposal pathways. It also provides a suggested fuels monitoring program for the current storage facilities.

Based on a technical review of the regulations and available literature, this report recommends a minimum of seven fuel elements be characterized.

1) PB Core 1 fuel. One Type II non-failed element, one Type II failed element, and one Type III non-failed element.

2) PB Core 2 fuel. Two Type II non-failed fuel elements should be examined.

3) FSV fuel. At least two fuel blocks from regions of high temperature and fluence and long in-reactor performance (preferably blocks removed at reactor end-of-life (EOL)).

Selection of PB fuel elements for characterization should focus on fuel elements between radial core position 8 and 14 and fuel compacts between compact numbers 10 and 20. Selection of FSV fuel elements should focus on fuel elements from Fuel Zones II and III, located in Core Layers 6, 7, and possibly 8.

Acronyms and Abbreviations

ADM	Activity Description Memorandum
ANL	Argonne National Laboratory (-W, west, -E, east)
ANSI	American National Standards Institute
ASTM	American Society for Testing and Materials
BCL	Battelle Columbus Laboratories
BISO	Buffered, isotropic (coating process used on PB Core 2 fuels)
CAA	Clean Air Act
CERCLA	Comprehensive Environmental Response and Compensation and Liability Act
CFR	Code of Federal Regulations
CHA	Cask Handling Area (324 Building at Pacific Northwest Laboratory)
DOE	US Department of Energy
DOT	US Department of Transportation
EFPD	Effective Full Power Days
EOL	End-of-life
EPA	Environmental Protection Agency
EPMA	Electron probe microanalysis
FSV	Fort St. Vrain
GA	General Atomic Company
HLW	High-Level Waste
HTGR	High-Temperature, Gas-Cooled Reactor
ICP	Inductively-Coupled Plasma Spectroscopy
IFSF	Irradiated Fuel Storage Facility (at INEL)
INEL	Idaho National Engineering Laboratory
ISFSI	Independent Spent Fuel Storage Installation
LLW	Low-Level Waste
LWBR	Light Water Breeder Reactor
LWR	Light Water Reactor
MRS	Monitored Retrievable Storage
MS	Mass Spectroscopy
MTIHM	Metric Ton Initial Heavy Metal
MW(e)	Megawatts electric
MW(t)	Megawatts thermal
MWd	Megawatt-day
NEPA	National Environmental Policy Act
NESHAP	National Emission Standards for Hazardous Air Pollutants
NRC	Nuclear Regulatory Commission
NWPA	Nuclear Waste Policy Act
ORNL	Oak Ridge National Laboratory
PB	Peach Bottom
PIE	Post-Irradiation examination
PNL	Pacific Northwest Laboratory
PSC	Public Service Company of Colorado

PTL	Postirradiation Testing Laboratory (at Pacific Northwest Laboratory)
RCRA	Resource Conservation and Recovery Act
SAL	Shielded Analytical Laboratory (at Pacific Northwest Laboratory)
SAR	Safety Analysis Report
SARA	Superfund Amendments and Reauthorization Act
SEM	Scanning Electron Microscopy
SERF	Special Environmental Radiometallurgical Facility (at PNL)
SMF	Shielded Materials Facility (at Pacific Northwest Laboratory)
TEM	Transmission Electron Microscopy
TRISO	Tricoating, Isotropic (coating process used for FSV fuels and some PB test fuels)
TRU	Transuranic (waste)
WINCO	Westinghouse Idaho Nuclear Company, Inc.
WIPP	Waste Isolation Pilot Plant

Contents

1.0 INTRODUCTION

1.1 BACKGROUND

The FSV (FSV) and Peach Bottom (PB) reactors were high-temperature gas-cooled (HTGR) commercial reactors built for the purpose of electrical power generation. These reactors are unique in the United States for several reasons; the reactors used a gas coolant rather than water and operated at much higher temperatures than water-cooled reactors, the uranium carbide-based fuels used in these reactors differ from the more common Light Water Reactor (LWR) fuels which are uranium oxide-based, and finally, the fuels are contained in graphitic structures rather than metal claddings.

The FSV spent fuel inventory comprises 2,208 graphite-block elements. The PB spent fuel inventory comprises 1,639 graphite-based assemblies. Of these elements and assemblies, 744 FSV blocks, and nearly all of the PB assemblies are stored at the U.S. Department of Energy's (DOE) INEL. WINCO has the responsibility for developing a strategy for the safe storage and ultimate disposal of these fuels. Although not currently responsible for the storage and disposal of the remaining FSV fuel blocks which remain in Colorado, the DOE has a contractual agreement with the Public Service Company of Colorado to accept the additional fuel blocks for storage and ultimately, disposal.

Two issues will impact the storage and disposal of these fuels. First, since reprocessing of fuels for recovery of fissionable materials will no longer be conducted by the DOE, disposal options must be capable of handling all of the radionuclides contained in the fuels. Second, since the DOE has faced delays in siting a geologic repository, storage of these fuels at the INEL site may have to extend between 15 and 50 years.

Some of the graphite fuels have already been stored at the INEL site for about 20 years. Of the fuels stored at the INEL site; PB Core 1 is stored in open-field drywells, PB Core 2 and FSV fuel blocks are in dry storage in the Irradiated Fuel Storage Facility (IFSF). Because the fuels may need to be stored for longer periods of time prior to disposal, there is a need to determine whether the fuels can continue to be stored in their current configuration. An evaluation of the current condition of the fuel will be needed to make this determination. It is important that the fuels not be degraded or negatively impacted by the storage environment, which makes it important to evaluate the fuel storage conditions.

At the end of the storage period, it is proposed that the fuels be disposed in a geologic repository. Although these fuels may already exist in a form suitable for disposal, it may be necessary to condition[1] these fuels prior to disposal. WINCO has proposed three basic pathways for disposal; intact disposal of the fuels, fuels partially disassembled and the high-level waste (HLW) fraction conditioned prior to disposal, and fuels completely disassembled and conditioned prior to disposal. Within these three basic pathways, many options and variations for conditioning and disposal exist. To select an appropriate

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[1]Conditioning is a term used to denote processes by which a fuel is altered from its as-discharged state to another state or form in preparation for disposal. For example, conditioning might represent the process of disassembling a fuel element to produce high-level and low-level fractions to reduce the total quantity of high-level waste needing to be disposed.

pathway for storage and disposal, each needs to be evaluated for feasibility, practicality, effectiveness, safety, cost, and possibly other factors.

1.2 CHARACTERIZATION PLAN

In support of the WINCO activities, PNL[2] has undertaken the contract to prepare a post-irradiation fuels characterization plan for the PB Unit 1 and FSV graphite fuels stored at the INEL. The purpose of this plan is to provide guidance for the characterization of the fuel which will be needed to pursue any of the storage or disposal pathways.

First, a review of the literature is necessary to gather as much information as possible on the fuel design, the manufacturing processes, the irradiation history (including fuel configuration within the reactor core), the post-irradiation examinations that have been reported, the storage conditions, and the effects of the storage conditions. Design characteristics of the fuel of both PB Unit 1 and FSV, with the test elements and subsequent improvements made during the lifetime of the reactors are summarized. The irradiation history of the reactors, and of the individual fuel elements, are provided where available. A review of the data from post-irradiation studies, where available, are incorporated for both the PB Unit I and the FSV reactors. This information, covered in Section 2.0, provides the basis for preliminary inspection of the fuel condition and an overview of the information gaps that should be filled in.

The activities required to obtain an adequate characterization are both guided by and in some cases dictated by regulatory laws and rules. An important precursor to moving the fuel and carrying out any examinations and tests is an understanding of the regulatory framework in which the work must be performed. The major regulations covering all phases of possible characterization work are reviewed briefly in Section 3.0, in relation to the options being considered for disposal of the fuel, which are also described.

On the basis of the information available from the literature and the regulatory constraints and drivers, this plan attempts to define the information that is required for moving and disposing of the graphite fuel under each of the various proposed options. The required data and the methods of generating that data are determined. The identification of the samples that will provide a statistically adequate and encompassing set of data is then attempted. These samples must provide answers to questions about the extent of particle coating failure after the various fuel histories and the potential technical problems these failures might have on the disposal process options. A detailed characterization plan is set forth in Sections 3.5 and 4.0.

Extensive references are listed in Section 5.0.

The appendices contain a suggested fuels monitoring program for the current storage facilities, a description of the hot cell requirements for performing the characterization work and excerpts from the most pertinent regulatory requirements.

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[2]Pacific Northwest Laboratory is operated for the U.S. Department of Energy by Battelle Memorial Institute under contract DE-AC06-76RLO 1830. This project was funded by a Related Services Agreement with WINCO, Inc., under P.O. Number 228116.

2.0 REVIEW OF THE LITERATURE

The literature and other sources[3] were reviewed to gather information regarding:

- a description of the fuel
- variability of the fuel
- storage and handling conditions of the fuel.

This information provides the basis for the development of the characterization plan. This section provides a summary of what information is known about the FSV and PB fuels. The evaluation of the adequacy of this information to support the storage/disposal options is discussed in the next section.

2.1 PEACH BOTTOM UNIT 1 REACTOR

Peach Bottom (PB) Unit 1 was a graphite-moderated, helium-cooled nuclear reactor classified as an HTGR. It used the ^{235}U fuel cycle, was designed by Gulf General Atomic as a prototype HTGR, and was operated by Philadelphia Electric from March 1966 (initial criticality) to October 1974 at Peach Bottom, Pennsylvania. This 40 MW(e), 115 MW(t) reactor is depicted in Figure 2.1.

Two fuel cores were loaded into the reactor. Commercial operation of Core 1 was from June 1967 to October 1969 for a total of 451.5 Effective Full Power Days (EFPD) which is equivalent to 30,795 Megawatt days/metric ton of initial heavy metal (MWd/MTIHM). Core 2 ran from July 1970 until October 1974 for a total of 897.4 EFPD or 72,717 MWd/MTIHM. The heavy metal loadings in Core 1 and Core 2 were 1,686.14 kg and 1,418.6 kg, respectively. Core 1 was run for approximately half of the expected time because of the failure of the fuel to contain the fission products; this problem was addressed by using a different fuel particle design for Core 2.

The following summary describes the fuel elements used in the PB reactor. Information is provided on the various subcomponents, their dimensions, materials, and other pertinent information.

2.1.1 PB Fuel Element Description

Each core contained a total of 804 fuel elements. The fuel elements were of three general types: 1) standard (also called driver fuels), 2) instrumented, and 3) test. A typical PB fuel element is shown in Figure 2.2. Table 2.1 provides a summary of each number of these element types.

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[3]Other sources consisted primarily of discussions with persons associated with or having knowledge of storage facilities, operations, or the history of the FSV and PB fuels.

The fuel elements are further divided into four types (subcategories), depending on the amount of uranium (fissile material), thorium (fertile material), and ^{103}Rh and zirconium diboride (burnable poisons) as shown in Table 2.2. These four types of fuel elements were placed in one of the three regions of the core as shown in Figures 2.3 and 2.4. Placement of the fuel elements was used to shape the power profile and control the reactivity in the core. Detailed descriptions are provided below for the standard and test elements.

Standard Element Design

The primary components of the fuel elements are: bottom connector, sleeve, screen, internal fission product trap assembly, lower reflector piece, fuel compacts, spine, burnable poison compacts (in selected elements), fuel cap, and an upper reflector assembly (Figures 2.2 and 2.3). All of these components are made of graphite except for the stainless steel screen. The fission product trap also includes graphite granules with a silver coating to getter the cesium and iodine fission products. A list of the various components as well as their basic material of construction is provided in Table 2.3. The weight of each component is given in Table 2.4. The instrumented fuel elements are very similar to the standard elements except for a few changes to the bottom connector and internal changes to accommodate thermocouple leads.

Within each fuel element, there are three fuel compact assemblies; each assembly contains ten donut-shaped compacts along a spine down the central hole as shown in Figure 2.5. Core 1 fuel compacts had axial grooves while the Core 2 compacts were smooth with slots on the ends but similar dimensions otherwise. The fuel was a mix of uranium and thorium carbide fuel particles in a graphite matrix that was pressed and sintered during a final processing at 1400°C. The uranium was enriched to 93.15% ^{235}U.

The fuel particles were distinctly different in Core 1 and Core 2. Core 1 fuel particles were 150 to 400 μm (Simnad, 1971) in diameter including a 50 to 60 μm thick layer of pyrocarbon. Some discrepancies exist; another literature source lists the particle sizes between 210 and 595 μm (Morissette, 1986) with coating thicknesses of 55 ±10 μm. The particle loading in the compact was 22 to 28 vol%, and not greater than 30 vol%. The pyrocarbon was added primarily to protect the fuel particle during fabrication, but was expected to afford some protection from recoil damage and fission product release. The particles were pressed along with a graphite flour using a variety of components including a pitch binder. Failure of the fuel particles to retain fission products during irradiation of Core 1 fuel led to the replacement of the core at about half the projected burnup with an improved fuel design.

Core 2 fuel particles were improved to include fuel particles coated with an inner, low density, pyrolytic carbon coating surrounded by an outer isotropic pyrolytic carbon coating called BISO fuel particles (Simnad, 1971) (Figure 2.6). The coated particles are between 340 and 630 μm in diameter with a total coating thickness of 90 to 130 μm. While this fuel performed much better than the monolithic coated particles of Core 1, data on the examination of the fuel indicates that there was still considerable release of fission products (Cs, Sr, Ba, ^{3}H, and ^{14}C) from Core 2 fuel to the surrounding graphite sleeves and spines.

Test Fuel Elements

There were 33 test fuel elements irradiated in Core 2, including one assembly (named PTE-2) which had prior irradiation in Core 1. These assemblies were externally similar to the standard PB fuel elements with respect to basic material of construction (graphite) upper/lower end fittings, overall length, and diameter. However, the fuel sections were of three different designs, none of which were similar to the normal PB annular fuel compact design. These three fuel sections are shown in Figure 2.7. One assembly, PTE-2, had the hexagonal, internally-cooled design similar to that which would be used for the FSV fuels. Thirty assemblies were graphite bodies with 8, 6, or 3 holes each, that held test fuel in rod form. Two assemblies had fuel compacts, but different in design from PB fuel assemblies.

Typically, the test fuel assemblies contained six separate "fuel bodies" (segments) each 396 mm long, consisting of a machined graphite cylindrical body (H-327 graphite) with holes for the test rods surrounding the central spine hole. The fuel holes were capped by counter-sunk threaded plugs, and the upper end of the spine hole was similarly capped. The axial layout is sketched in Figure 2.8 and the arrangement of the plugs and holes is shown in Figure 2.9. Table 2.5 lists the three separate fuel body designs, the corresponding fuel element type designations, and the figures that depict them.

The fuel contained in the fuel sections was of varied types. It was all standard HTGR fuel, in the sense of consisting of pyrolytic carbon-coated high-enriched uranium and natural thorium (or mixed uranium-thorium) carbide or oxide fuel microspheres, dispersed in a baked graphite matrix. Both BISO and a newer TRISO[4] fuel kernal coating processes were represented. One element, FTE-13, also contained plutonium oxide and mixed plutonia-thoria microspheres. Two others, FTE-1 and FTE-3, contained ^{238}U-oxide microspheres. The average atomic ratios of uranium to thorium, and heavy metals to carbon, also varied in these test fuels. All the test fuel materials, however, differed from PB fuel compact material in that none contained rhodium. The test elements and the fuel types contained therein are listed in Table 2.6. Irradiation times for the test elements varied from a low of 133 EFPD to a maximum of 897 EFPD. The peak fuel temperatures varied from 1167 to 1640°C, with the majoritiy being in the 1100 to 1400°C range. A typical radial temperature profile across a fuel body is shown in Figure 2.10.

2.1.2 Fuel Element Status

According to DOE/RW 1992 and Morissette 1986a, PB Core 1 fuels at the INEL were packaged at the reactor in sealed aluminum canisters with stainless steel liners; however, according to the safety analysis report (SAR), these canisters have mild-steel liners. (Currently we do not have information to determine which is correct.) In either case, seal failures have made it necessary to repackage some of the canisters at INEL.

Core 2 fuel was packaged at the reactor in the same type of canisters that were used for Core 1. However, to place the fuel in the IFSF, the fuel had to be removed from the canisters and the reflectors had to be cut off the elements so that they would fit into the IFSF storage canisters.

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[4]The TRISO fuels were developed for use in the FSV reactor which are described in Section 2.2.

2.1.3 Fuel Content and Burnup

Fuel element contents varied between fuel type and core. In general, the elements contained about 300 g enriched U and about 1.5 kg Th. The average burnups are given in Section 2.1, and fuel element loadings in Table 2.2.

2.1.4 Variation of Fluence and Burnup

Little information exists for PB Core 1. No axial power data were found for Core 1. While power histories are available for this core, fluxes, power, or relative power versus time information was not available from which to calculate burnup or fuel inventories for individual fuel elements.

The average relative radial power density at 452 EFPD for Core 1 (which was EOL, General Atomic [GA] 1970) is shown in Figure 1.1. The relative power within a sector varies from about 0.8 near the center of the core to a high of 1.18 in the row next to the reflector. The reduction in power around core position 3 is due to the influence of control rods inserted in the core. Variation in the power density from the top to the bottom of the core could not be determined.

The axial power distribution in Core 2 is shown in Figure 2.11. This figure shows both the time-averaged distribution over the life of the core and the distribution at EOL. The measured EOL profile exhibits a slight shift toward the top of the core. The Core 2 radial power distributions, both calculated (using GAUGE and BUG R-Z) and as measured, are shown in Figure 2.12.

The time averaged radial power distribution of Core 2 is similar to that of Core 1. Decreases in power between core positions 3 through 6 can be attributed to the insertion of control rods near those locations. Beginning-of-Life (BOL) and EOL radial power curves are also shown for Core 2. Although data was not found for BOL and EOL Core 1 power curves, the trends might be expected to be similar because of the similarities in the time averaged curves.

Fluence profiles were reported differently for Core 1 and Core 2. The Core 1 fast fluence distribution was reported as a function of radial positions. Core 2 fast fluence distributions were reported as axial distributions within individual elements.

An example of the Core 1 radial distribution is shown in Figure 2.13. This profile is similar to the Core 1 power profile. The radial profile was noted to be symmetrical about the central vertical axis within the reactor core.

The axial fast fluence profiles of several Core 2 assemblies are shown in Figure 2.14. The curves indicate that the maximum fluence is encountered near the middle of the stack of fuel compacts (between compacts 10 and 20).

By assuming that the Core 1 and Core 2 fluence profiles are similar, it is possible to postulate the location of fuel elements and fuel compacts which will have the highest fluence exposure. These locations are radially from core positions 10 to 16 and axially from compact locations 10 through 20. This represents approximately 40% of the total fuel compacts in a core load. The remaining 60% of the fuel compacts would have been exposed to lower fluences.

The total burnup for Core 1 and Core 2 are given in Section 2.1. Burnup information for specific fuel elements was found for Core 2. Six Core 2 elements were subjected to destructive examinations as part of an examination of PB Driver fuel elements. Fuel burnup was measured and calculated as part of these examinations. Additionally, following the final shutdown of the reactor, 55 fuel driver elements, 21 fuel test elements, and three reflector elements were examined as part of the PB EOL Program. Much of the examinations conducted were to determine fission produce distributions.

The method used to calculate burnup for two of the six elements destructively examined relied on measurement of elemental zirconium, uranium, and thorium concentrations. Five fuel compacts were destructively examined (three from element E11-07, two from element E06-01). Zirconium was used to calculate the number of fissions that occurred in the fuel particles. However, this method did not work well because of analytical difficulties in quantifying the small quantities of zirconium generated in the fuel particles (1 to 3 μg typically). Burnup analyses were not conducted on the remaining four fuel elements examined during this post-irradiation examination (PIE) program.

The fuel elements examined in the EOL program were scanned using gamma spectroscopy. The burnup for these elements was calculated from the ^{137}Cs content in the fuels. These EOL examinations showed that on a core-average basis the measured burnup values agreed quite well with those calculated by using the GAUGE computer code.

However, on an element-to-element basis, the measured and computed values often did not agree. This was attributed to Cs migration within the elements. The BISO fuel particles could not adequately retrain fission products. Cs was shown to migrate from the top of an element towards the bottom due to migration in the direction of the element purge flow.

Although the measured and calculated burnups could vary by as much as 15%, the agreement is close enough to allow identification of which areas of the fuel experienced the highest number of fissions. A calculated burnup profile for one element, E11-07, is shown in Figure 2.15. Three curves are shown: 1) fissile burnup is the mole fraction of initial fissile material, 2) fertile burnup is the mole fraction of initial ^{232}Th, and 3) mixed burnup encompasses the mole fraction of initial mixed U-Th-carbide. The profiles indicate that the highest burnup is achieved between compact numbers 10 and 20. This is not unexpected as the burnup profile should follow closely that of the fluence profiles.

Several observations can be made from this review. First, the information existing on power profiles, fluence profiles, and burnup profiles is limited. Second, fission product migration introduced errors which made correlation of the measured and calculated fuel burnup difficult. Fission product migration may introduce errors in future modeling or calculational efforts, but the significance of this problem cannot be predicted at this time.

Even though the information is limited, enough exists to assist in the selection of fuels to characterize. Typically these would be fuel elements between radial core position 8 and 14 and between compact numbers 10 and 20. Also, the fuel elements should not be selected from areas near control rod insertion points due to the decrease in fluence at those locations. Fuels from these locations will have seen the highest fluence, have the highest power density, and have the highest burnup. These fuels should have the greatest damage and therefore, any information gained from these fuels should be conservative from a fuel population standpoint.

2.1.5 Variation of Temperature in the PB Core

There is practically no temperature data available for Core 1. Core 2 has been reasonably well characterized in spite of experimental difficulties and calculational dependence on other reactor parameters. Table 2.7 summarizes the operating temperatures for two elements from Core 2. Despite individual elements being monitored, no data were found from which radial or axial temperature profiles across the reactor could be determined. It can only be assumed that these distributions hold across the reactor core.

2.1.6 Postirradiation Examination of PB Elements

Only a few elements from each of the PB reactor cores have been postirradiatively examined. The fuel design used in Core 1 did not perform to expectations, and in general little effort was expended on detailed examinations. The Core 2 fuel design performed more satisfactorily than Core 1, and several elements were examined.

Distribution of Radionuclides Throughout Reactor Core Components, Failures of Particles, Fuel Elements, and Releases of Radionuclides to Coolant

The fuel elements and fuel particles of Core 1 experienced significantly more failures than Core 2. There were a total of 90 elements cracked during irradiation in Core 1 plus two broken during subsequent handling. About 80% of the fuel particles are believed to have breached coatings in Core 1. These failures were identified by in-core gas sampling and measurement of the purge gas flow rate. No detailed examinations of Core 1 fuel elements were found which described the redistribution of fission products in the elements.

There were no physical failures of Core 2 fuel elements. Actual fuel particle failure percentages were measured during PIE examinations of eight fuel compacts taken from fuel elements E11-07 and F03-01. The physical failure rate was less than 1%.

Although the BISO-coated fuel particles used in Core 2 fuel did not physically fail, they proved inadequate for the retention of fission products. The purged fuel element design used in PB successfully kept the fission products separate from the primary coolant loop and from the other major reactor components. However, within the fuel elements fission products migrated from the fuel compacts to the spines, element casings, and other components.

Detailed examinations were conducted on six PB Core 2 fuel elements, E01-01, E03-02, E06-03, E09-01, E11-01, and E14-01. These analyses were conducted to confirm fuel performance and to assess the likelihood of fission products migrating to the reactor primary coolant circuit. Radial distributions of fission products in the spines and sleeves were determined at various fuel compact positions. This was done using a remote-operated lathe to take "peelings" from element sections which were subsequently analyzed. A typical radial profile is shown in Figure 2.16. Axial distributions of fission products were determined by gamma scanning. Bottom connectors, top reflectors, and fission product traps were also analyzed for fission products. These analyses are well documented in Oak Ridge National Laboratory (ORNL) reports *ORNL-5126, ORNL-5214, ORNL/TM-5730, ORNL/TM-5996, ORNL/TM-6455, ORNL-6353*, and General Atomic Company report *GA-A-13453*.

The fission product distributions can likely be updated by recalculation to account for decay losses. New radiochemical analyses will be needed to confirm and verify the ORNL work (which was conducted in the 1970's).

PB Fuels Summary

Because of the difference in fuel design and fuel failures, it may be necessary to treat PB Core 1 and Core 2 as separate entities for the source of the disposal options. The lack of information on Core 1 makes it the most difficult case. No data exists which can be used to determine the suitability of Core 1 fuel element compacts for consideration as LLW. The high percentage of fuel particle failures suggests that no portion of Core 1 may qualify as LLW without dilution of the waste stream (resulting in a volume increase for disposal). No information exists on the distribution of TRU elements within Core 1 fuel elements. This may also complicate disposal of any Core 1 components as LLW if for any reason TRU elements have migrated from the fuel compacts.

Much more information has been gathered on Core 2. It may be possible to show that some or all fuel element components (excluding fuel compacts) can be disposed as LLW by recalculating measured radionuclide inventories to account for radioactive decay. These calculations will require validation through additional radiochemical analyses. Also, like Core 1, confirmation of the location of TRU elements will also be required.

Selection of components to analyze can be deduced from the information presented in Section 2.14. As summarized in that section, fuel elements between radial core position 8 and 14 and fuel compacts between numbers 10 and 20 would be likely candidates for characterization based on fluence, power, and burnup. It would also be desirable to select elements with the longest in reactor exposure.

For comparison purposes, fuel element E11-07 from Core 2 (previously examined by ORNL) has fuel compacts which fall within this criteria. Information from this element may prove useful for performing radionuclide decay calculations which can be verified by analyzing similar elements from near this element's core location.

2.2 Fort St. Vrain Reactor

A literature search was conducted to compile the available information and to define the characteristics of Fort St. Vrain (FSV) spent fuel. The search includes direct information on FSV elements as well as other information from testing of FSV-type fuels. Most of the documents reviewed were reports from GA and other laboratories, including ORNL. It is clear that information exists which may be useful to the characterization activities, but this information has not been released to the public. In the time period allocated to the preparation of this plan, it was not possible to gain access to unpublished information. In particular, there is generally less detailed information available at the end of operation of FSV, although this information may exist at GA.

The FSV reactor, like the PB reactor, was a HTGR. The reactor was started in 1974 and operated until August 1989. The reactor was rated at 300 MW(e), 842 MW(t), and a schematic, cross-sectional view is shown in Figure 2.17.

2.2.1 FSV Fuel Element Description

The FSV fuel is described in some detail in two reports compiled by ORNL and DOE/RW-0184-R1, Volumes 2 and 4, and references therein. The fuel particles are microspheres, spherical TRISO-coated carbide kernels. The kernels are prepared by a drop-melt process. Fissile kernels consist of MC_2, where M is 81% Th and 19% U (93% U-235 enriched). Fertile kernels are ThC_2. The kernels are coated with the TRISO coating process, as shown in Figure 2.18. The coating consists of successive layers of a low-density carbon buffer layer, a high-density pyrolytic carbon inner layer, a silicon carbide (SiC) layer, and an outer high-density pyrolytic carbon layer. If intact, this TRISO coating is a very good barrier for retaining the radionuclides inside the microsphere. The fissile microspheres are about 400 μm in diameter and the fertile microspheres are about 700 μm in diameter. Primarily because of the SiC layer, the TRISO coating retains high pressures and acts as a barrier to diffusion of radionuclides. The TRISO coating is much more effective than the BISO coating used in the PB reactor fuel design. However, the carbide kernel does not retain lanthanide fission products. Eventually, these products release from the carbide kernels and react with the SiC. This mechanism is the principal cause of FSV fuel particle failure. It has been found that if the carbide is at least 15% converted to oxide, the lanthanides are retained in the kernels. Thus, newer HTGR fuel designs are oxide- or oxycarbide-based, not carbide-based as used in FSV.

The coated microspheres look like poppyseeds (Figure 2.19). The microspheres were compacted into rods ½" OD X 2" long (Figure 2.19).[5] The rods are inserted into a fuel channel drilled into the fuel element. Each channel contains about 15 rods and there are 210 channels in a fuel element, with a maximum total of 3,132 fuel rods in an element. The element (Figure 2.20) is a hexagonal prism of type H-451 or H-327 graphite 14.1" across the flats and 31.2" high. The fuel channels are drilled to a depth within one-half inch of the bottom in a hexagonal array. Coolant channels are arranged in a trigonal array, each located in the center of six fuel channels. Fuel elements are stacked six deep in the core, and the core has a total of 1,482 elements. The He coolant flowed downward through the core. The core (Figure 2.21) is divided into 37 fuel regions. Six of the regions on the outside edge of the core contain only five elements per layer.

2.2.2 Fuel Element Status

The discharge schedule of FSV fuel elements is shown in Table 2.8. Of the first 726 elements, one from region 17 from the first discharge (Element 1-0743, Sauerwein, 1982) was sent to GA for destructive examination. One other element, 1-2415 was also destructively examined near the end of the FSV program. The remaining elements from the first three fuel discharges are at INEL. Most of the remaining 1,482 elements remain in Colorado at the FSV site and are from later reactor fuel discharges. Eighteen elements from the final discharge of the 1,482 elements were shipped to INEL in 1992, leaving 1,464 in Colorado at the FSV site.

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[5]Some of the earlier rods used in FSV were 3" long, but the length was shortened to alleviate warpage problems

2.2.3 Fuel Content and Burnup

Each fuel element weighs 128 Kg (certain special elements weigh up to 20 Kg less). The graphite body weighs from 85 to 94 Kg. Each element contains an initial loading of approximately 450 g enriched U and 11 kg Th. The average burnup was 16.5% for the fissile material and 0.8% for the fertile material, or 30,000 to 35,000 MWd/MTIHM. The average fast fluence (E > 29 fJ) was approximately 2.5×10^{25} n/m^2. Maximum burnup and fluence values were approximately 60% higher than the average values.

Detailed fuel accountability records are apparently available on microfiche; however, these could not be obtained prior to completion of this report. Also, a complete power history is needed. To date, an example for 1981 was found in Burnette, 1982 (Figure 2.22).

2.2.4 Variation of Fluence and Burnup

Neutronics codes can calculate fluence to an accuracy of 10 to 20% (see example comparison of measured data with calculations, DOE/RW-0184-R1, Volume 2). Calculated fluence profiles as a function of core position are shown in Figure 2.23. Radial zones II, III, and IV, have similar fluence values. Only Zone V, with elements at the outermost edge of the core having significantly different values, about half the average. There is also a variation of about a factor of two as a function of core layer, with layer 9 at the bottom of the core and layer 4 at the top of the core having the lowest values. One element is in layer 3 because the central element in each group of seven in a region is displaced upward about 8" and it is considered to be in the next level. Variations among the seven positions within a region is shown in Figure 2.24. There is a range of up to a factor of two with the highest values being in the position closest to the core center, position 4. Overall, the variations in fluence and burnup may be expected to vary up to a factor of two, with the highest values being near the core center.

2.2.5 Variation of Temperature in the FSV Core

Average temperatures have been calculated for the first FSV fueling cycle (Figure 2.25). Note the center position of the seven elements in the region is displaced one layer. Because the coolant is flowing downward, highest temperatures are at the bottom, layer 9. Position 4, closest to the core center, is about 200°C hotter than position 7, farthest from the core center. These temperatures are for early operation at less than 100% full power. Maximum temperatures at EOL will be several hundred degrees higher. A detailed temperature history over the life of the reactor might be needed, but was not available.

2.2.6 Postirradiation Examination of FSV Elements

All elements from the first three discharges have been nondestructively examined. These elements show slight shrinkage and warping, and there are a few elements with small cracks, but all were in good condition. Only one element has received a thorough destructive examination, element 1-0743 (Sauerwein). This element had the temperature and fluences described above, and had a burnup of 6.2% fissile and 0.3% fertile. About 0.3% of the fissile and 0.2% of the fertile microspheres were failed. These failures were ascribed to manufacturing defects such as no coating, cracks, thin coatings, etc. About 3% of the rods were broken, most of them were believed to have been broken by the disassembly process involving pushing them out from the bottom. Note that temperature and fluence conditions were

modest. It may be anticipated that failure rates of coated particles will be higher at EOL and at higher temperatures.

Distribution of Radionuclides Throughout Reactor Core Components, Failures of Particles, Fuel Elements, and Releases of Radionuclides to Coolant

For the first two FSV fuel cycles, measured and calculated quantities of ^{85m}Kr in the He coolant were used to determine fuel performance (Stansfield, et al., 1983). The apparent failure rate for the entire core was only 1×10^{-5}, about a factor of five lower than model predictions. It was noted that fuels removed at the end of the first cycle had higher contamination levels, and the release rates were lower for the second cycle. There are verbal indications that failure rates were higher near the EOL, but no documents have yet been found to substantiate those claims.

Variation in Fuel Types

There were several variations of fuel and test elements, with similarities being that they were all hexagonal prismatic elements of very similar dimensions and content. The standard fuel elements contained 3,132 fuel rods distributed among 210 fuel holes. These elements had 108 gas coolant passages. Control fuel elements were similar to the standard fuel element except for two, four-inch diameter control rod channels and one, 3.75 inch reserve shutdown channel. The control elements contained 1,782 fuel rods in a total of 120 fuel holes. The control rods placed at the bottom of the control rod columns extended 7.5 inches below the core. The fuel holes were drilled to a depth which would place the bottom fuel rods at the same levels as all the other fuel elements.

Burnable poison rods could be inserted, as needed, in holes provided in each element. Standard elements have holes in each of the six corners while control rods only have holes on four of the corners.

Conclusions Regarding FSV Fuels

The literature data are somewhat limited for FSV fuels. Calculated burnup and fluence data may be available from GA, but were unaccessible during research time on this report. As expected, neutron flux is highest in the center of the core, declining by a factor of two at the edge of the core. Temperatures are highest at the bottom of the core because of the downflow of the He coolant and are also highest in the regions of highest flux. Very little direct information on the release of fission products from the fuel microspheres is available. Based upon data for FSV-type fuels, failure rates can be estimated as a function of temperature and fluence. Failure rates increase with both temperature and fluence, so the highest failure rates would be anticipated in the elements that were not removed in the first three cycles, at the lower levels (e.g., level 8), and nearest the center of the core. More information in certain areas is needed. If the levels of radionuclides in the coolant at the late stages of FSV operation are known, this information needs to be documented. Detailed fluence, temperature, and power history for elements, particularly those elements with the highest fluences and with temperatures above 1400°C is also needed.

SECTION 2.0

FIGURES AND TABLES

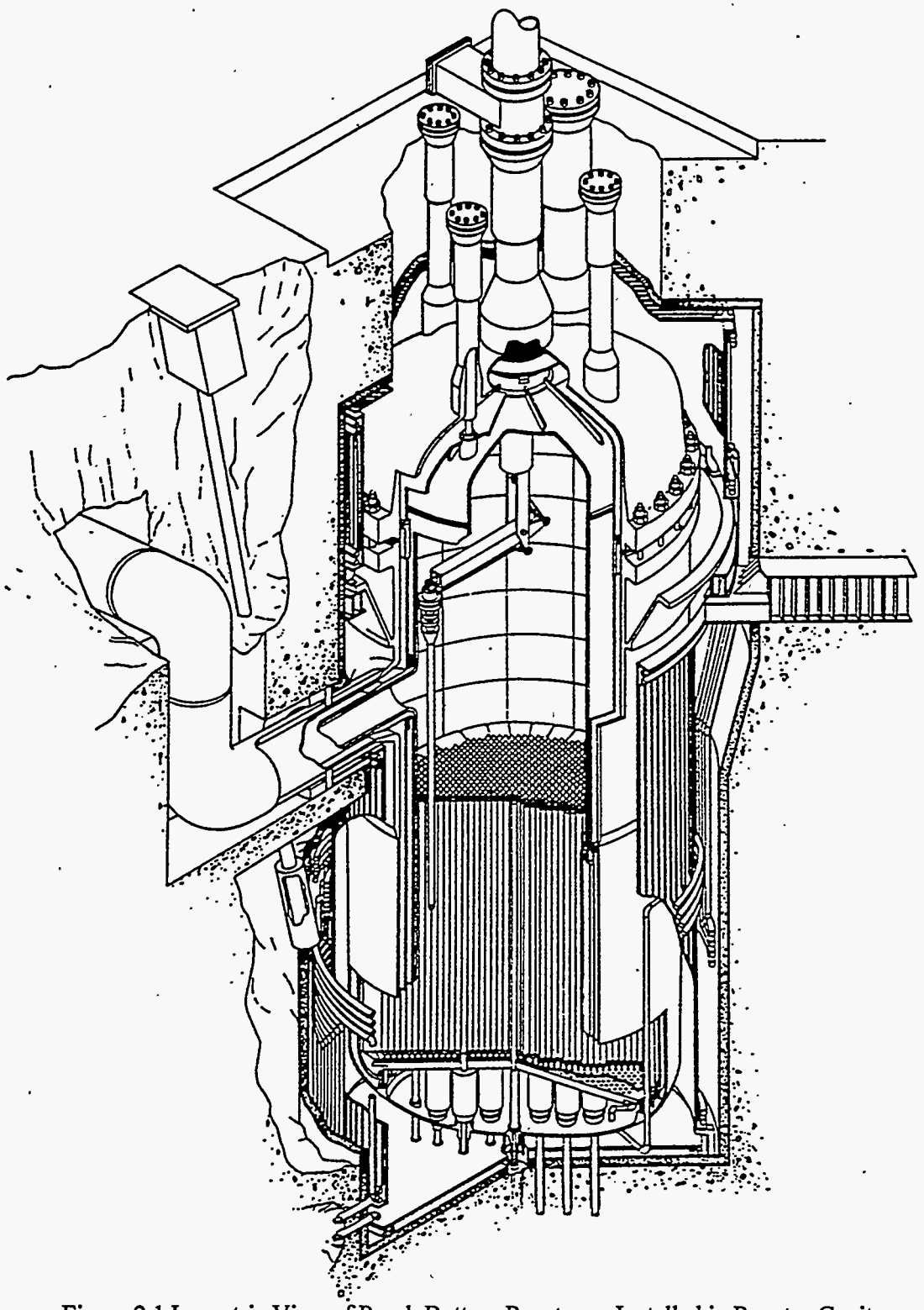


Figure 2.1 Isometric View of Peach Bottom Reactor as Installed in Reactor Cavity

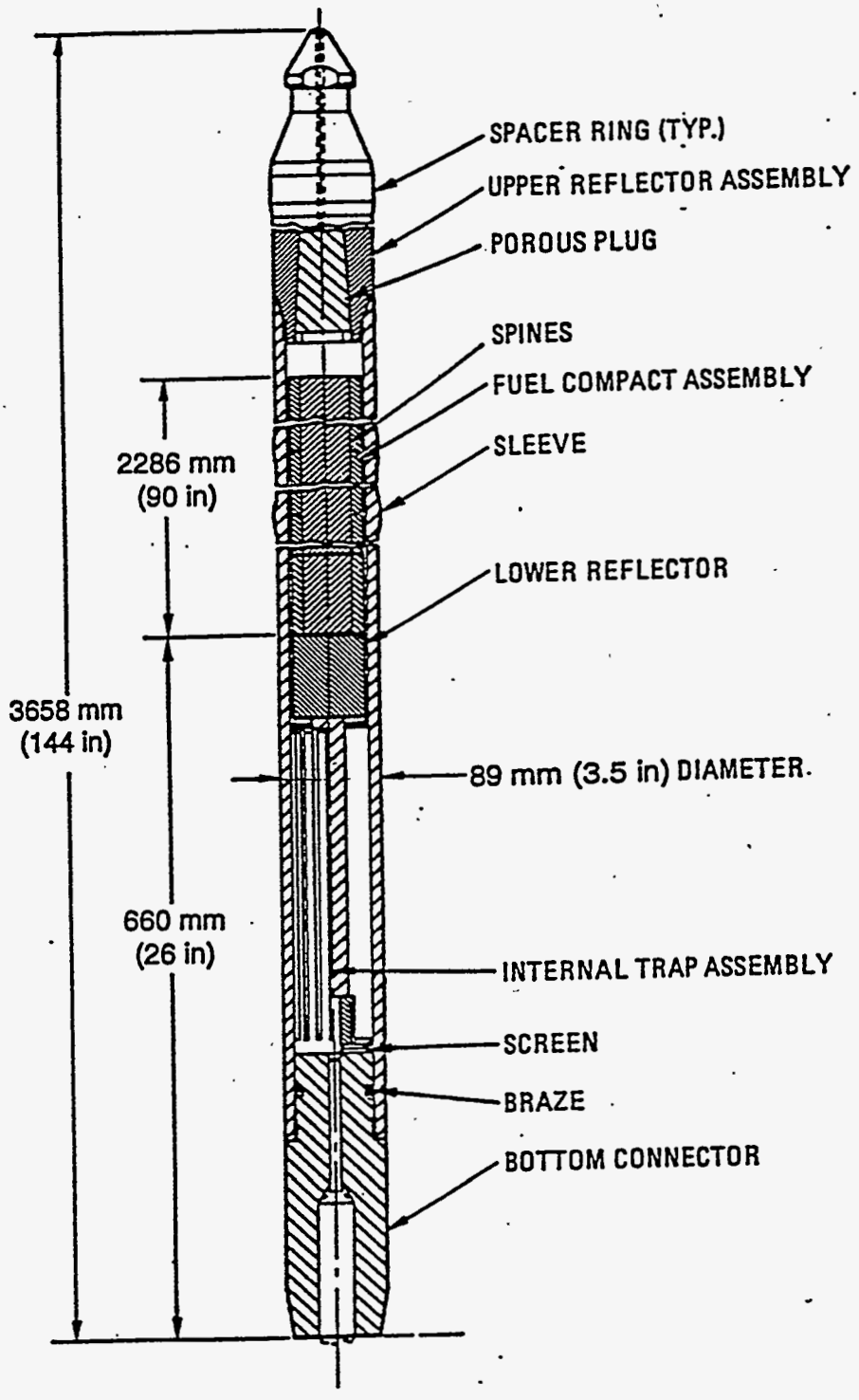


Figure 2.2 Typical Peach Bottom Reactor Fuel Element

○, ●, ⊙, and ○ = 804 Fuel Elements

▨ = 36 Control Rods

▩ = 19 Emergency Control Rods

Region 1 contains 54 heavy rhodium elements (Type I) = ○
Region 2 contains 564 light rhodium elements (Type II) = ○
Region 2 contains 84 light rhodium elements with poisoned spines (Type III) = ●
Region 3 contains 102 light uranium, heavy thorium elements (Type IV) = ○

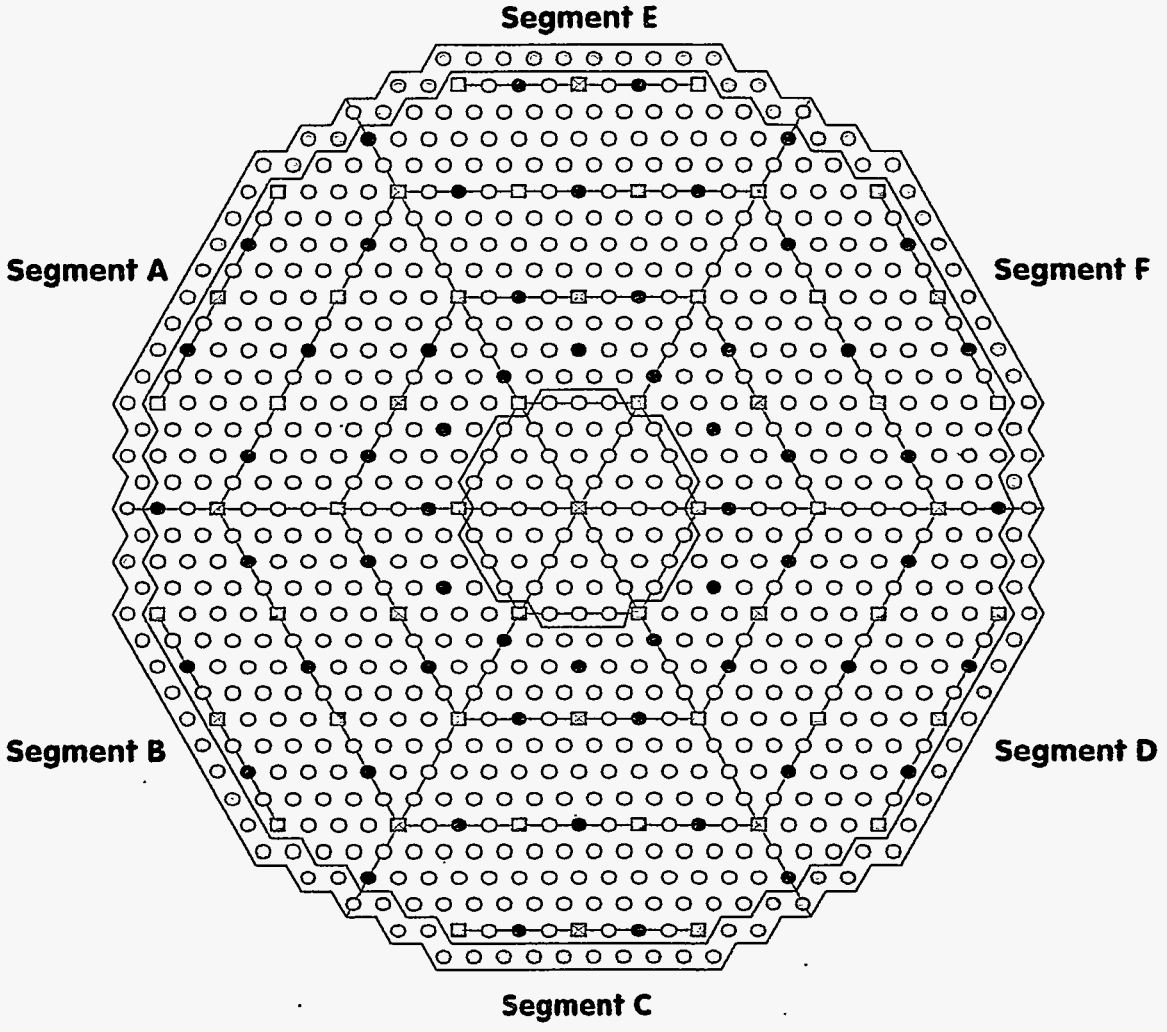
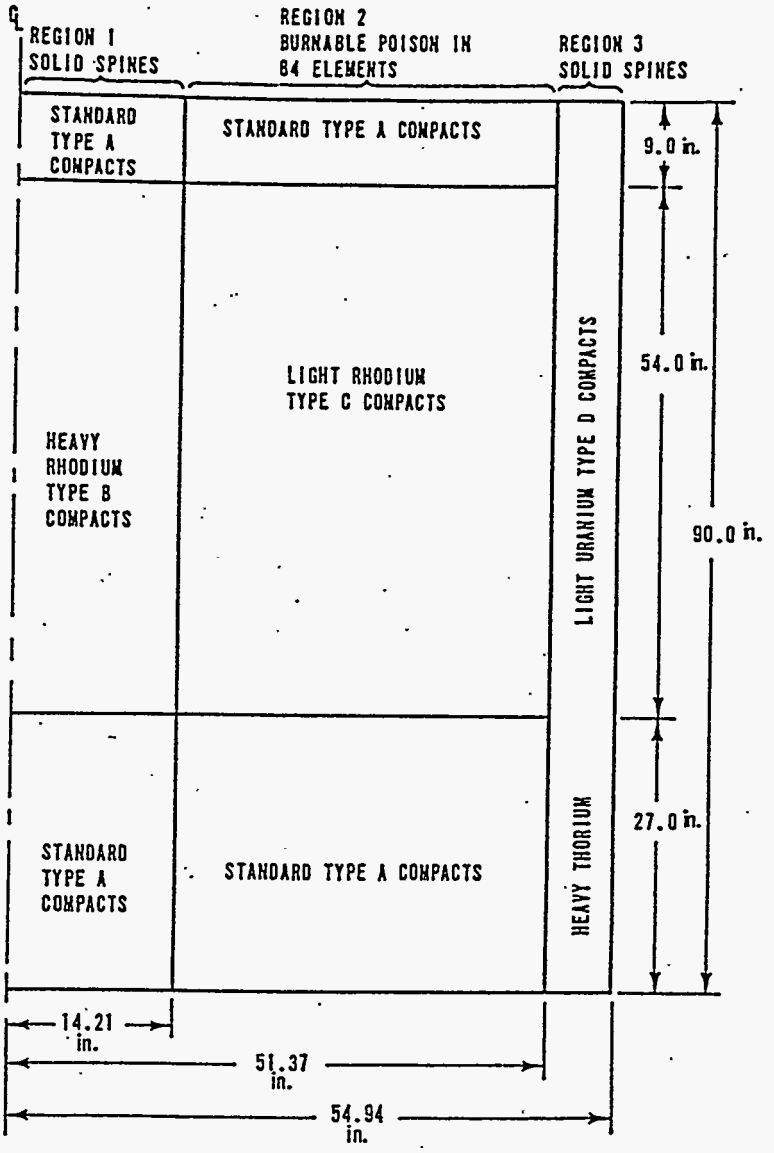


Figure 2.3 Peach Bottom Reactor Core Region Diagram



CORE ZONING

Figure 2.4 Peach Bottom Reactor Core Zone Diagram

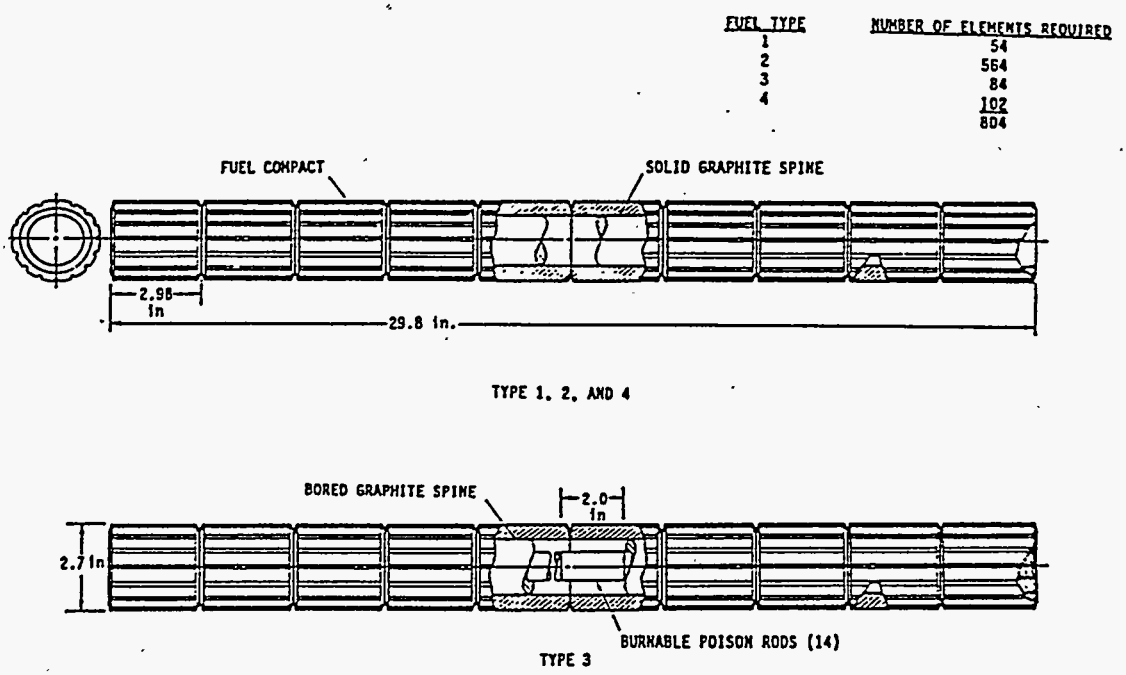


Figure 2.5 Peach Bottom Reactor Fuel Compact Placement on Spine

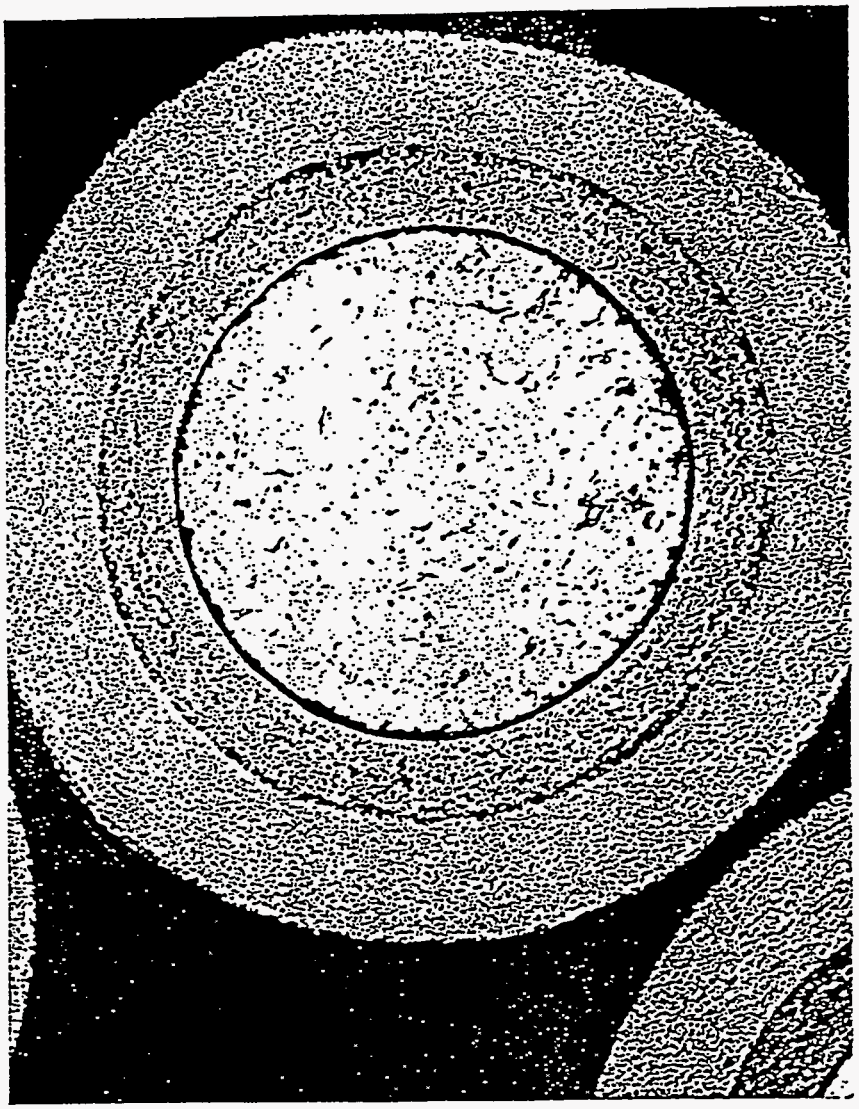
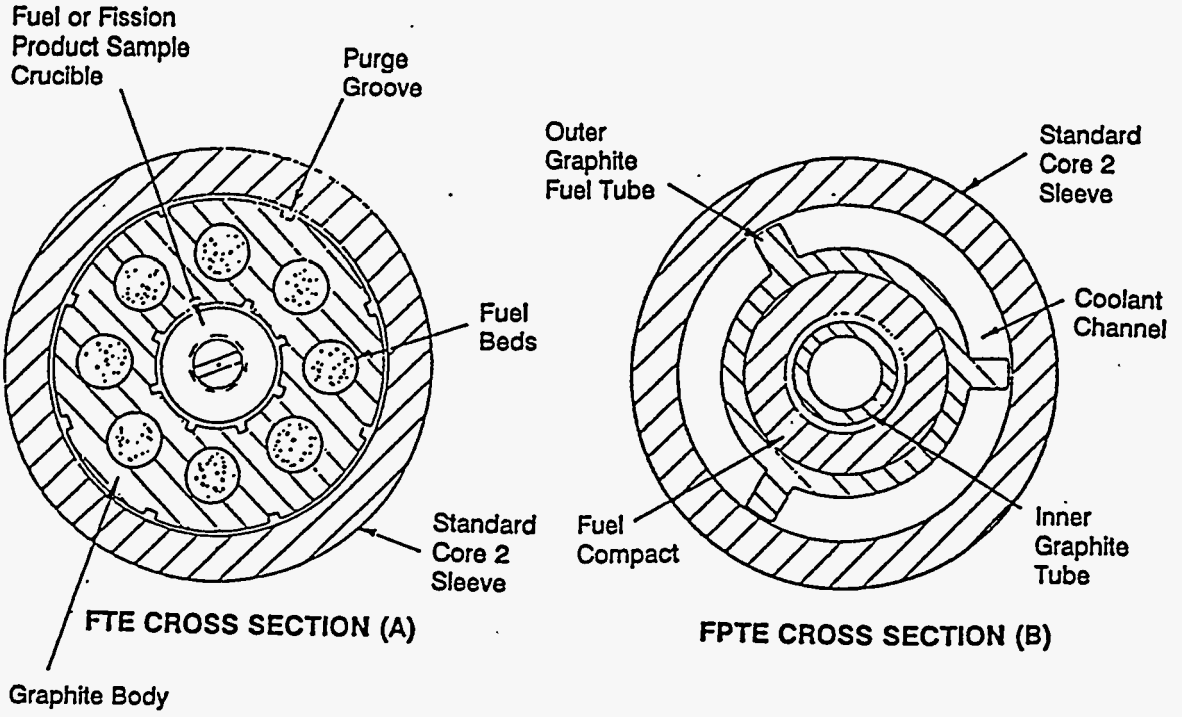


Figure 2.6 Peach Bottom Core 1 BISO Fuel Particle Microstructure



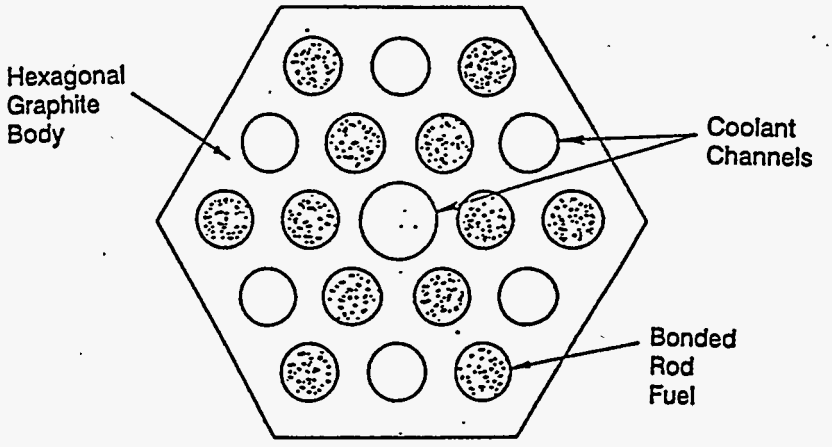


Figure 2.7 Peach Bottom Reactor Test Element Sections

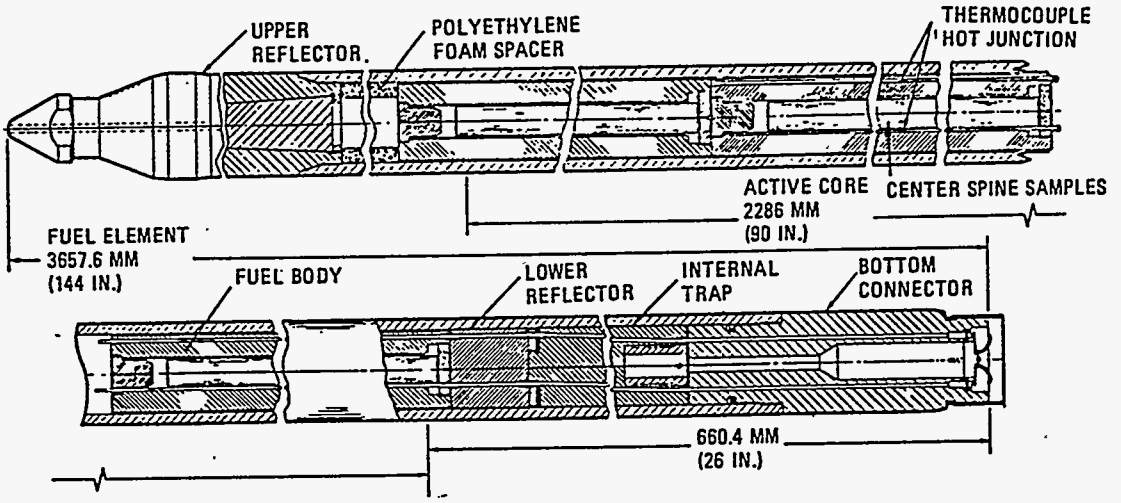


Figure 2.8 Typical Peach Bottom Test Element Axial Layout

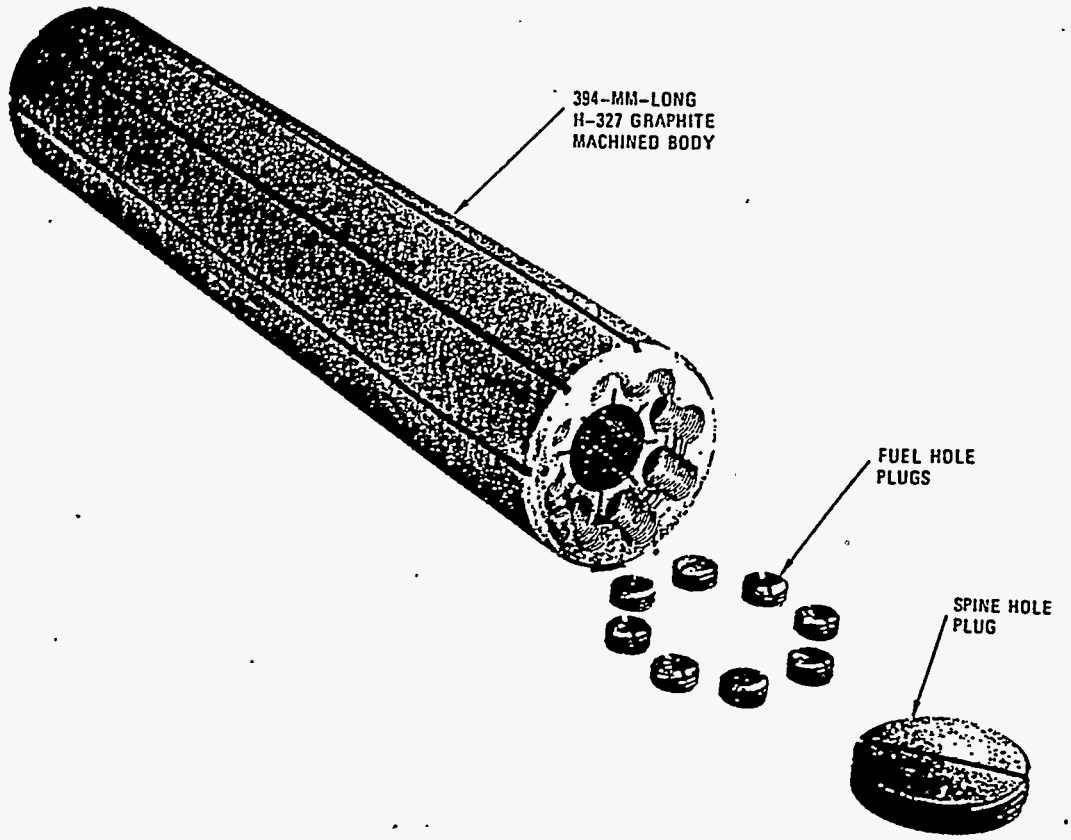


Figure 2.9 Arrangement of Holes and Plugs for PB Test Elements

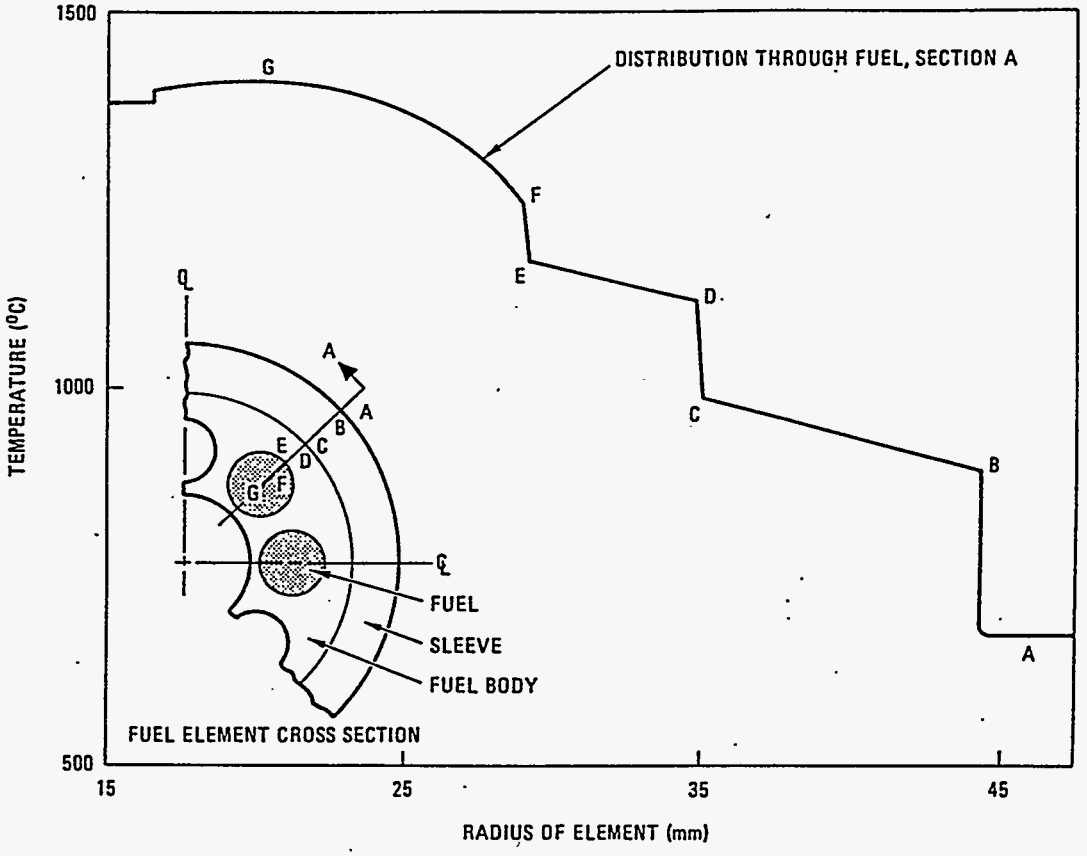


Figure 2.10 Typical PB Radial Temperature Profile

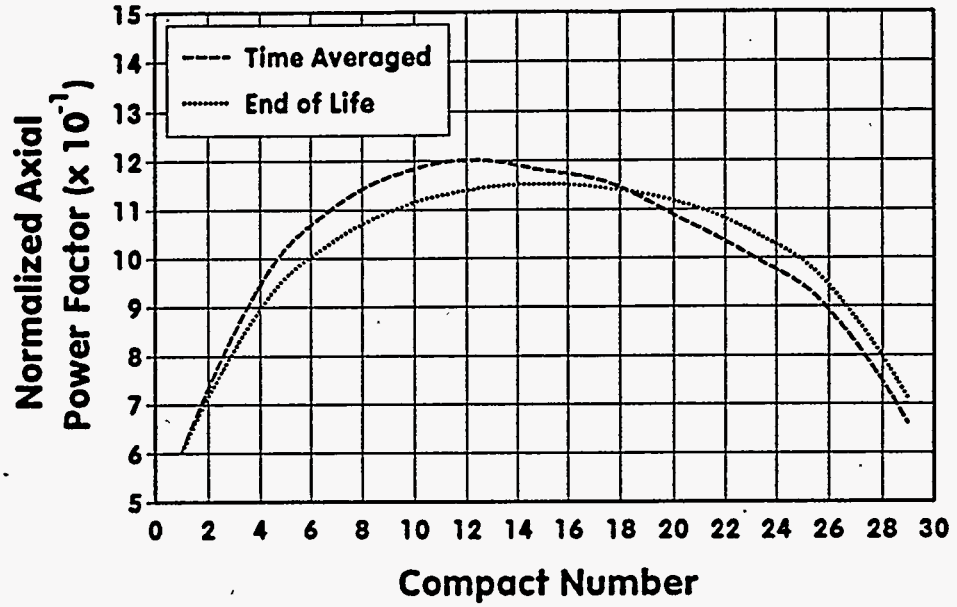


Figure 2.11 Axial Power Distribution in PB Core 2

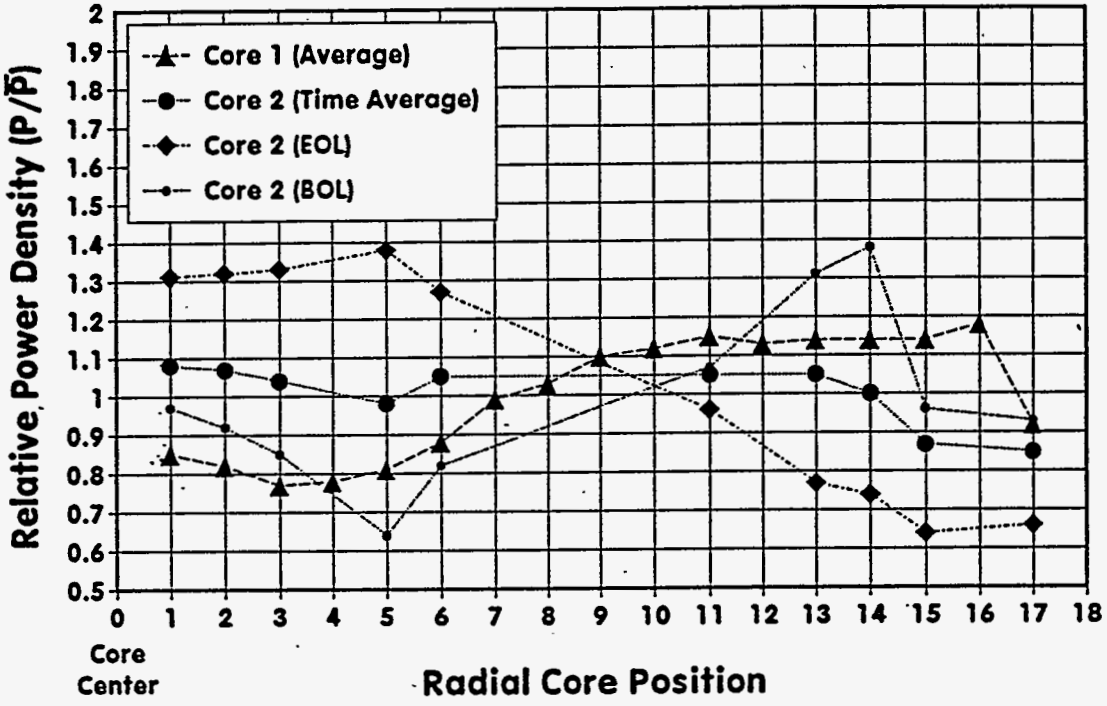


Figure 2.12 Peach Bottom Core 2 Radial Power Distribution

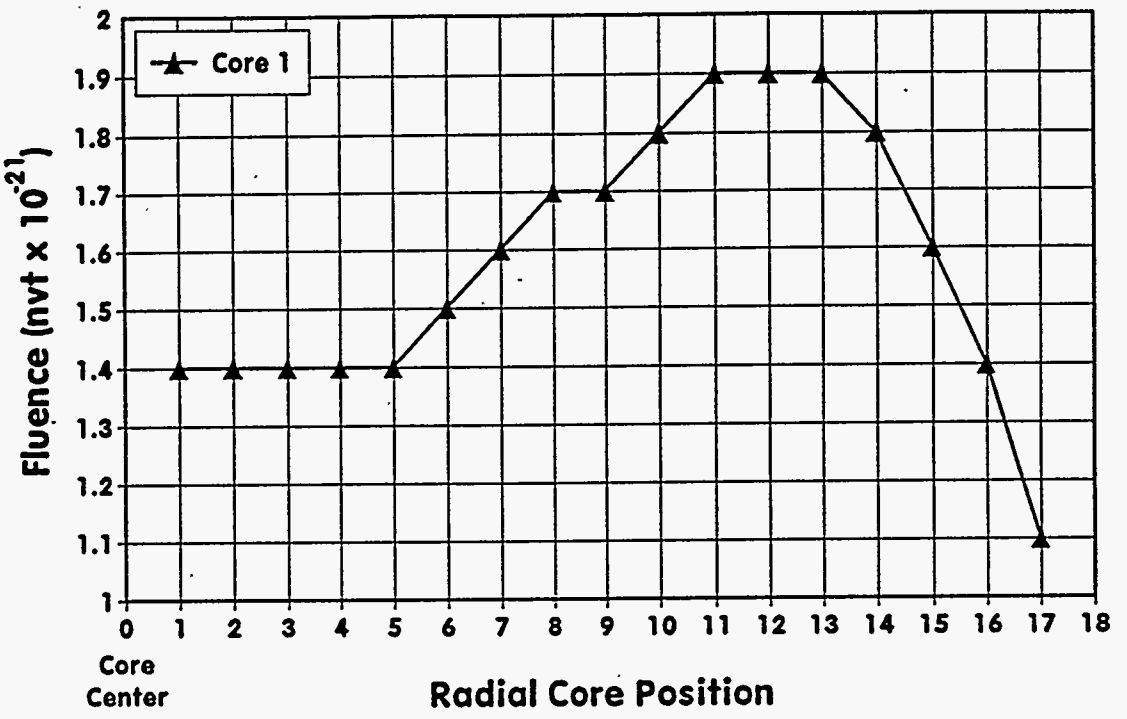


Figure 2.13 Fast Fluence Profile for PB Core 1

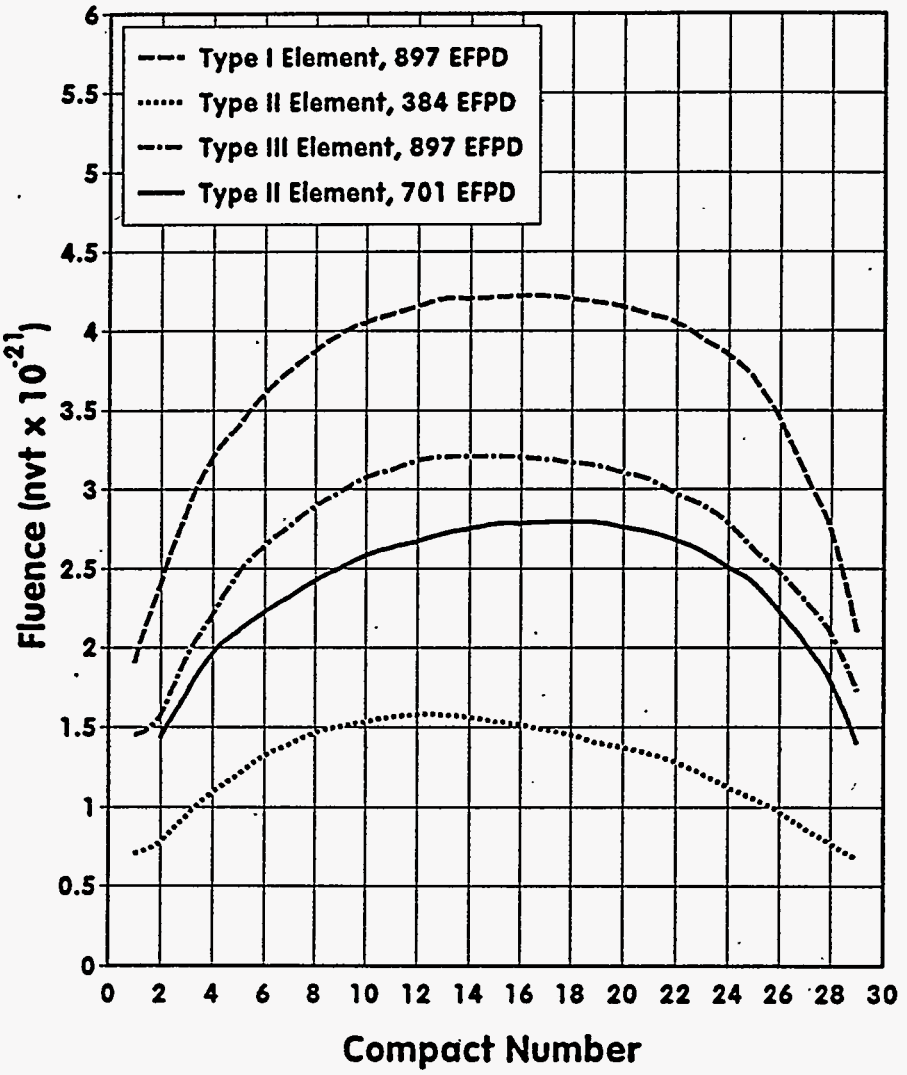


Figure 2.14 Fast Fluence Distributions for Typical PB Core 2 Elements

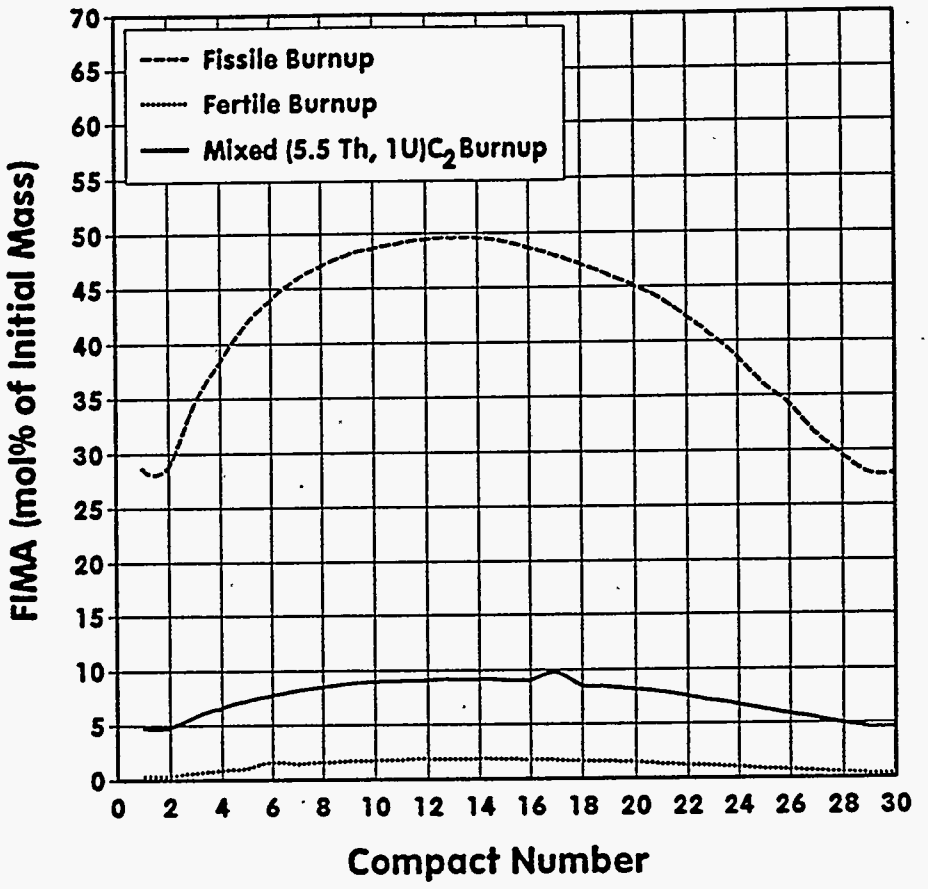


Figure 2.15 Calculated Axial Burnup for PB Core 2 Element E11-07

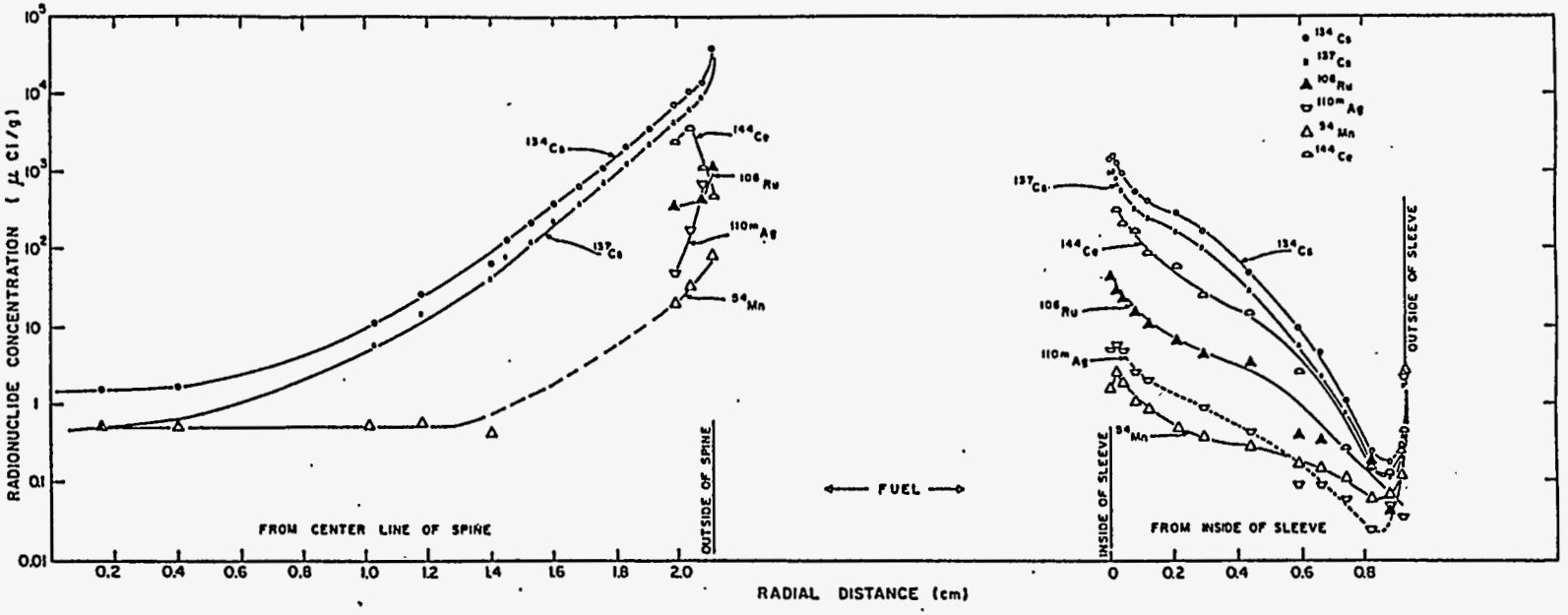


Figure 2.16 Radial Distribution of Gamma Emitters from a Typical PB Fuel Element

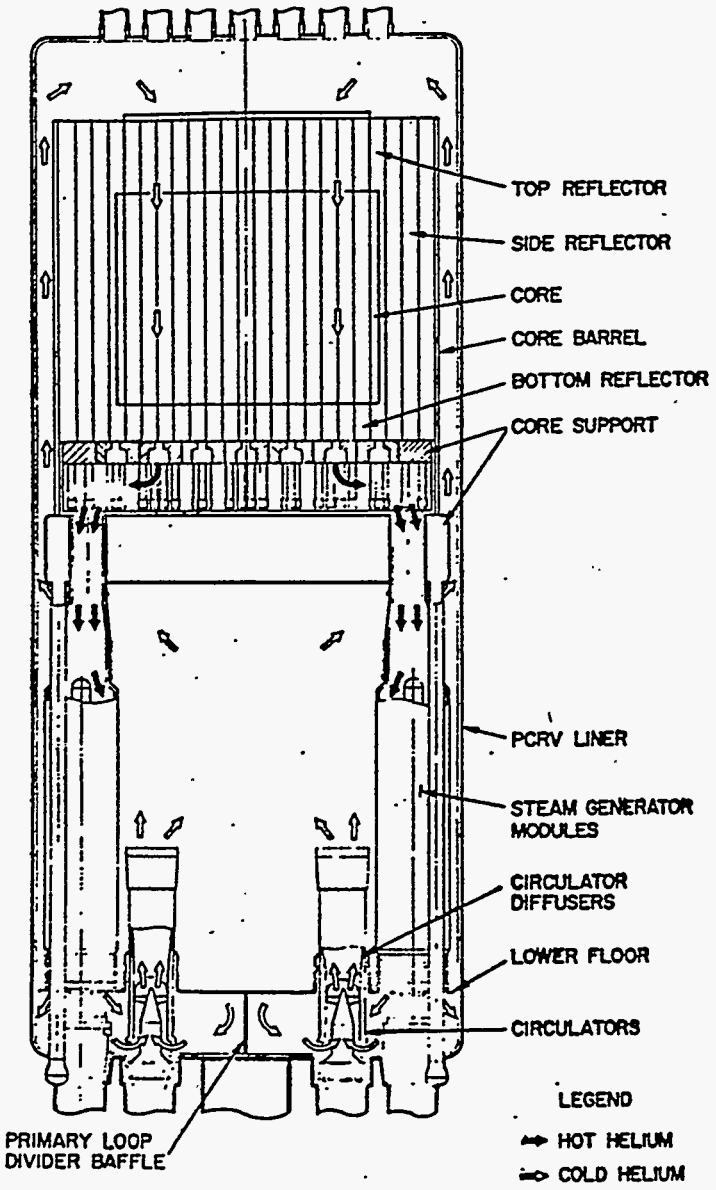


Figure 2.17 Fort St. Vrain Reactor Cross-Section Diagram

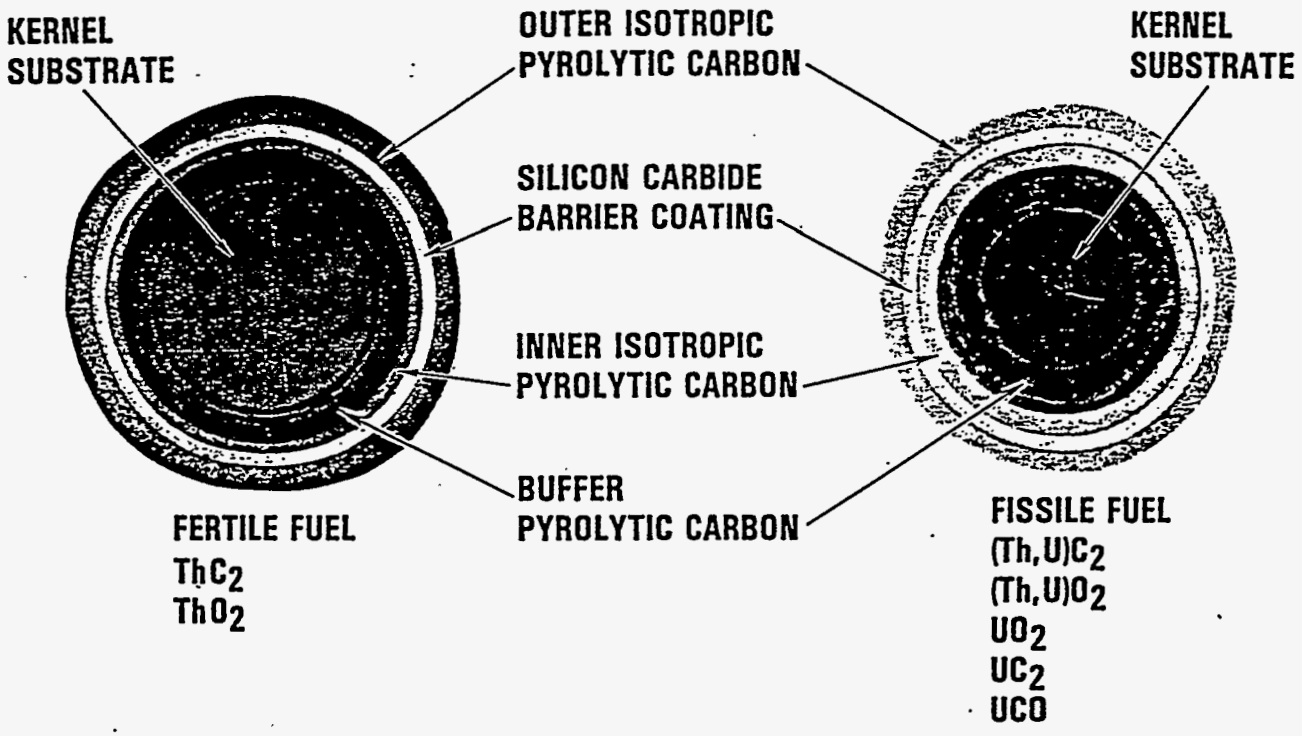


Figure 2.18 Photomicrographs of TRISO Coated FSV Fuel Particles

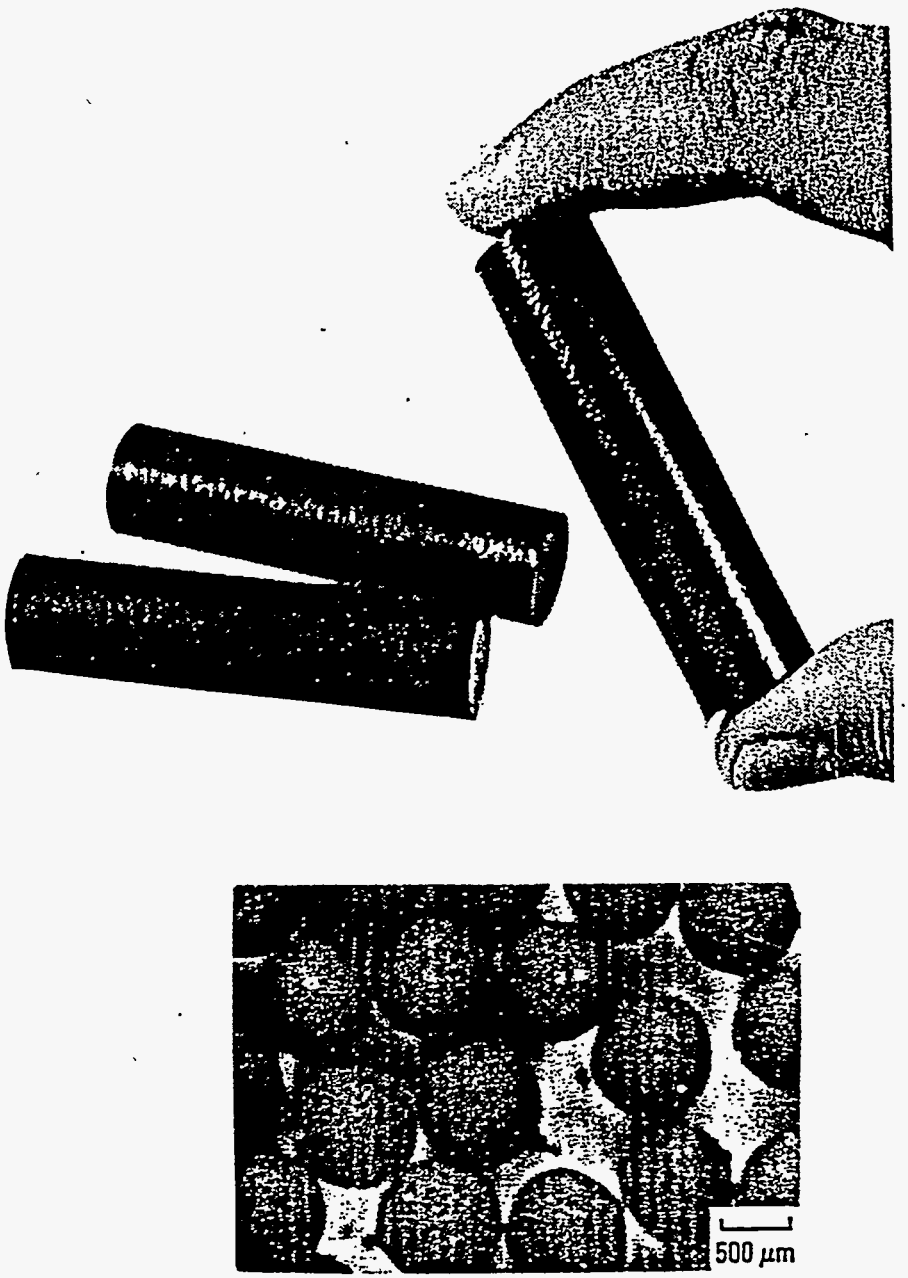


Figure 2.19 FSV Fuel Sticks and Micrograph of Fuel Particles

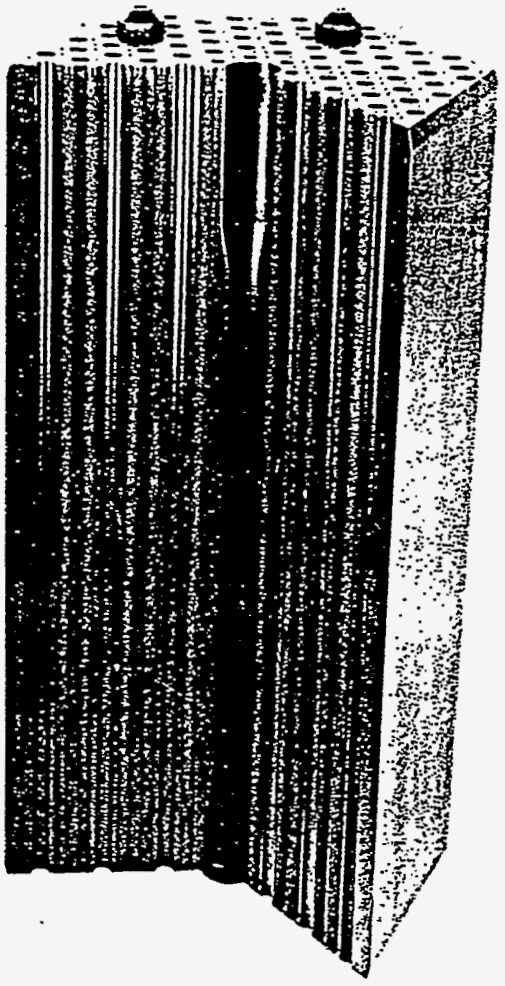


Figure 2.20 Cross-Section of a FSV Fuel Block

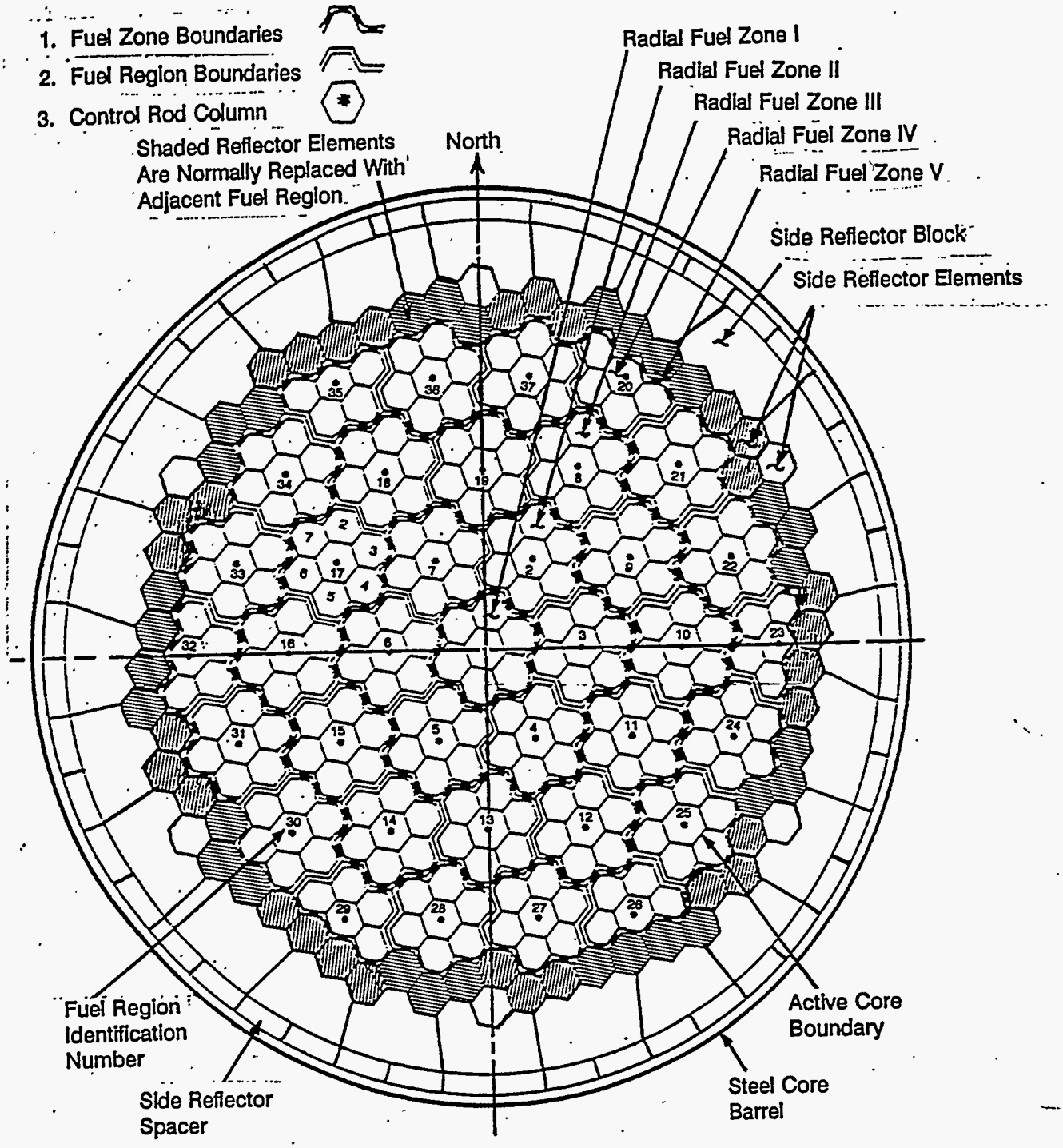


Figure 2.21 Schematic Diagram of FSV Reactor Core

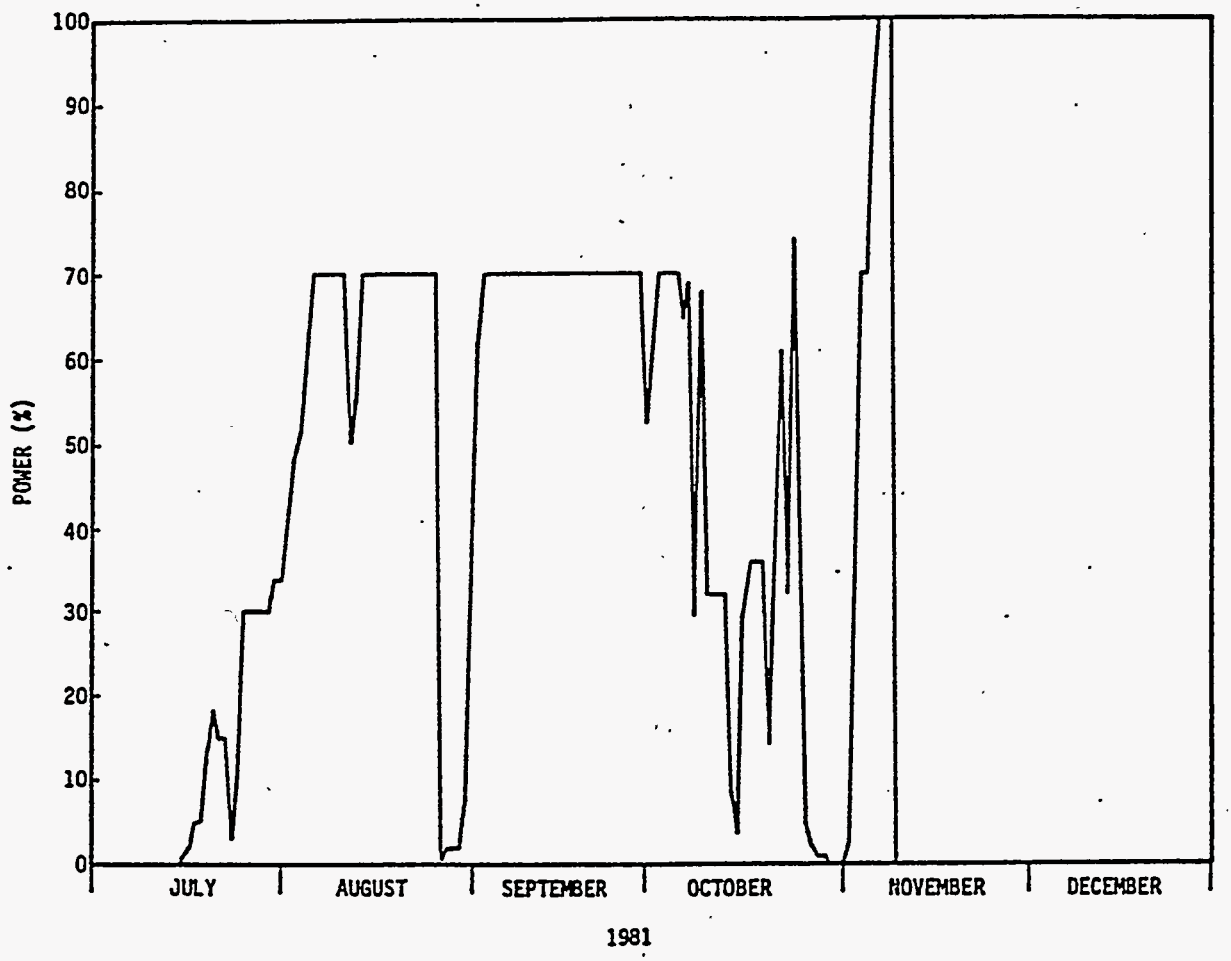
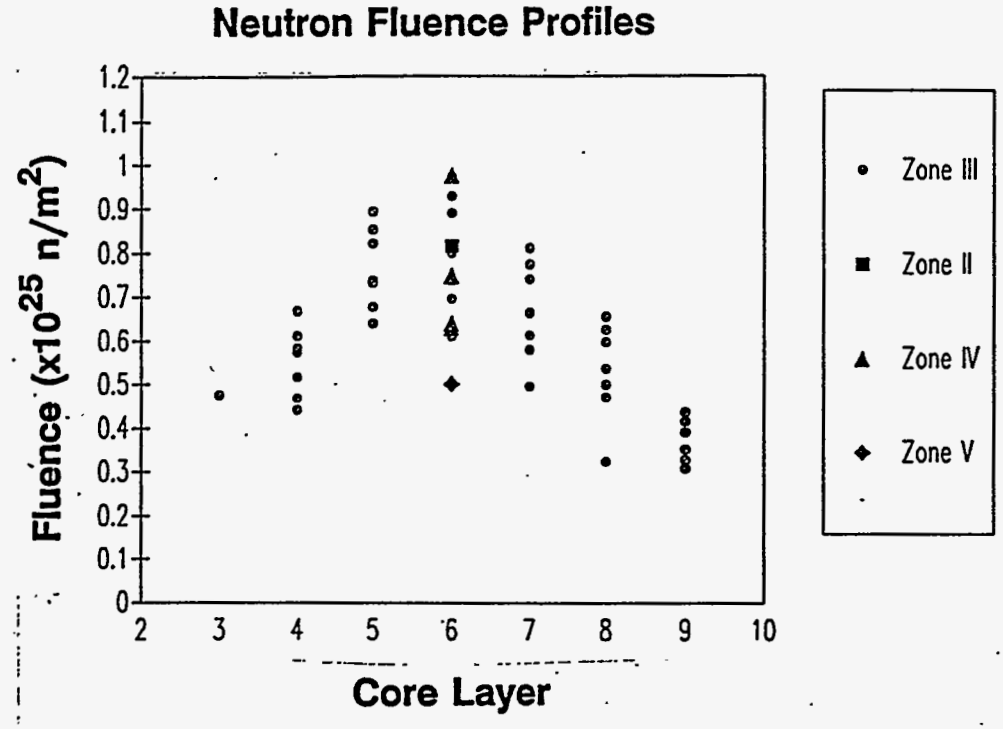
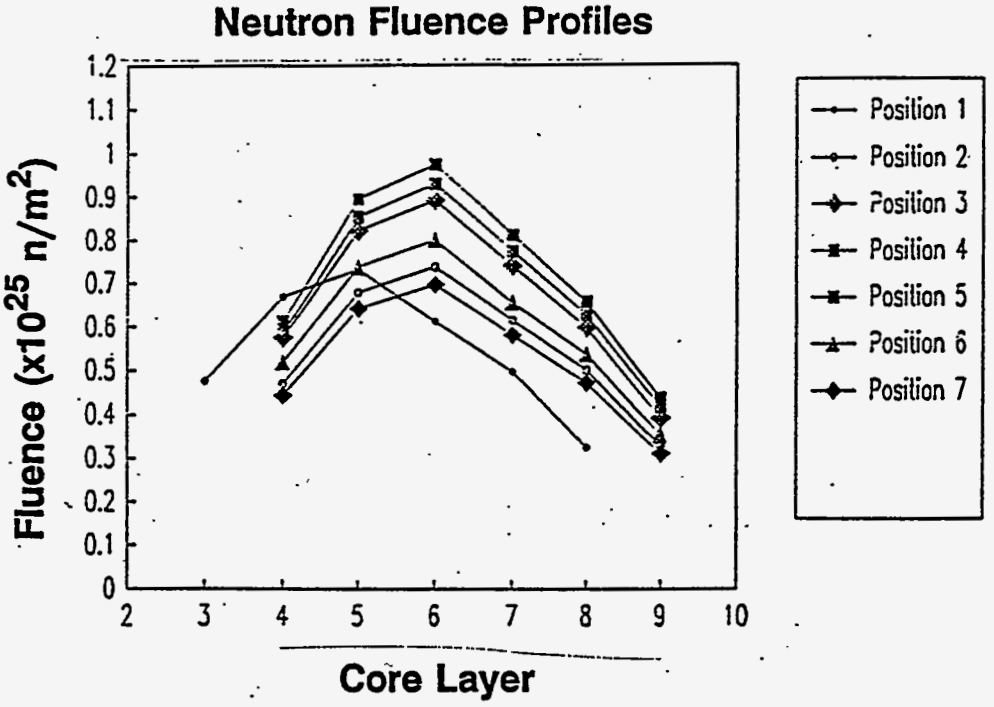


Figure 2.22 FSV Reactor Power History for 1981



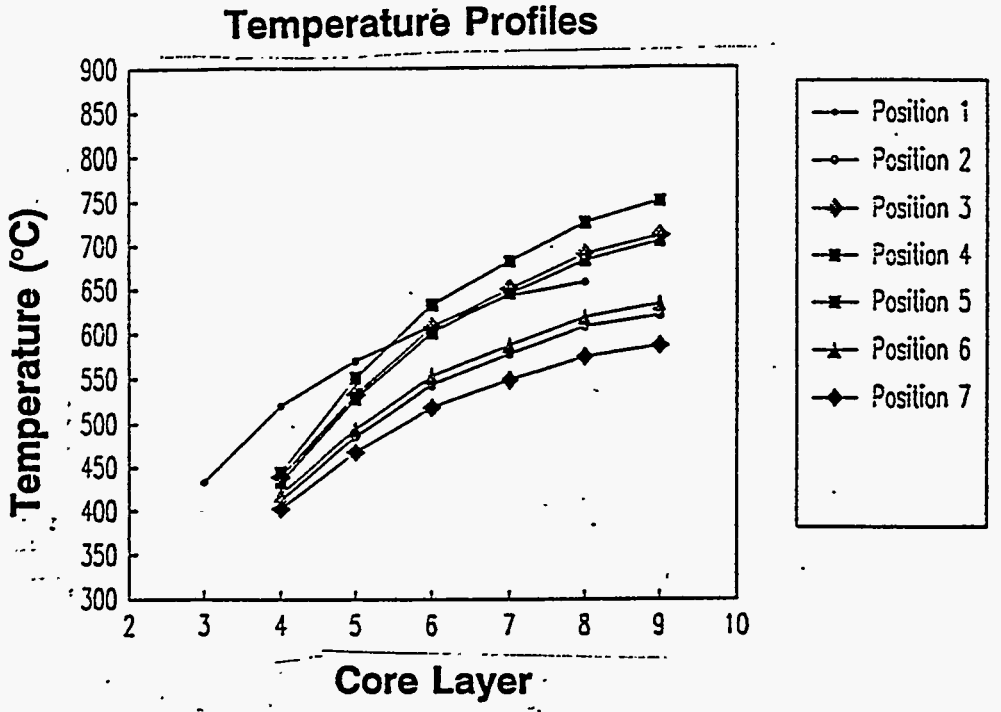
The volume averaged fast fluences (E > 29 fJ, 0.18MeV) as calculated by SURVEY code for FSV cycle 1. Estimated uncertainty is 10%. RMS variation in volumetric fluence within an element is 2×10^{24} n/m^2. From Miller and Saurwein, "Nondestructive Examination of 51 Fuel and Reflector Elements from Fort St. Vrain Core Segment 1," General Atomic Company, GA-A-16000, December, 1980.

Figure 2.23 FSV Calculated Fluence Profiles as a Function of Zone



The volume averaged fast fluences (E > 29 fJ, 0.18MeV) as calculated by SURVEY code for FSV cycle 1. Position #4 is nearest the reactor core center and #7 is the farthest for fuel region 17. Estimated uncertainty is 10%. RMS ariation in volumetric fluence within an element is 2×10^{24} n/m^2. From Miller and Saurwein, "Nondestructive Examination of 51 Fuel and Reflector Elements from Fort St. Vrain Core Segment 1," General Atomic Company, GA-A-16000, December, 1980.

Figure 2.24 FSV Calculated Fluence Profiles as a Function of Reactor Position



The volume averaged temperatures as calculated by SURVEY code for FSV cycle 1. Position #4 is nearest the reactor core center and #7 is the farthest for fuel region 17. Estimated uncertainty is given by the equation, s = 0.1x[T(°C)-330]. RMS variation in temperature is about 10% of the absolute temperature (45 to 90°C). From Miller and Saurwein, "Nondestructive Examination of 51 Fuel and Reflector Elements from Fort St. Vrain Core Segment 1," General Atomic Company, GA-A-16000, December, 1980.

Figure 2.25 Variations in Temperature as a Function of Reactor Position (FSV)

Table 2.1. Number of Fuel Element Types in Peach Bottom Cores

Fuel Element Type	Number for Core 1	Number for Core 2
Standard	782	751
Instrumented	36	36
Test	1	33
Total Number of Elements	819	820

Table 2.2. Peach Bottom Fuel Element Initial Metal Loadings (grams)

	Type I (A)		Type II (B)		Type III (C)		Type IV (D)	
Material	Core 1	Core 2	Core 1	Core 2	Core 1	Core 2	Core 1	Core 2
Uranium, 93%	313	250	313	250	313	250	166	141
Thorium	1563	1374	1563	1374	1563	1374	3468	2598
Thodium 103	18.5	18.5	6.16	6.16	6.16	6.16	0.00	0.00
Boron	0.00		0.00		18.3	18.3	0.00	0.00

Table 2.3. Fuel Element Component Compositions

Component	Material
Fuel Compact Assemblies	
Fuel compacts	Pyrolitic carbon coated UC_2/ThC_2 particles in graphite matrix
Solid or bored spines	Graphite
Burnable poison compacts	ZrB_2 in graphite matrix
Non-fuel Components	
Upper reflector	Graphite
Porous plug	Graphite
Fuel cap	Graphite
Sleeve	Graphite
Lower reflector	Graphite
Internal trap	Graphite
Screen	Stainless steel
Brazing ring	Silicon
Bottom connector	Graphite
Instrumented bottom connector (instrumented elements only)	Graphite, Stainless steel, Inconel
Thermocouples (instrumented elements only)	Inconel sheath, tungsten-rhenium, chromelalumel Nb-1% Zr sheath
Test samples	Niobium canned, fission product release samples

Table 2.4. Fuel Element Component Weights

Item	Approximate wt, kg
Fuel Elements	
Standard fuel element	41
Instrumented fuel element	41
Fuel test element (PTE designs)	45
Fuel test element (others)	41
Core 2 cut-off fuel element	38
Core 2 cut-off instrumented fuel element	38
Storage Apparatus	
Core 1 fuel element with storage canister	68
Storage basket with Core 1 fuel	1642
Fuel Element Components	
Upper reflector	6
Sleeve	13
Lower reflector	0.6
Internal trap	2
Bottom connector	3
Fuel compact assembly (each)	5[b]
Fuel compact	0.4

Table 2.5. Summary of Test Assembly Designs

Design Type	Assembly Types/Numbers	Description, Figure Number
Fort St. Vrain Proof Test	PTE-2	Hex-shape with fuel in small holes. Figure 2.7c
Teledial Design	FTE, RTE, FBTE, Type Assemblies (30 total)	Cylindrical shape, with 8, 6, or 3 holes per assembly. Figure 2.7a
Fuel Compact Design	FPTE-1, FPTE-3	Cylindrical shape, cylindrical fuel compact. Figure 2.7b

Table 2.6. Phase 1, 2, and 3 Peach Bottom Test Elements

Type	Phase[a]	Ident. Number	Core Position[b]	Thermocouples	Purge Sampling[c]	Fuel Bed	Fuel Particles	
							Fissile	Fertile
Fuel Test Elements								
Fuel bed test	1	FBTE-1	CO5-04	Yes	No	Rods[d]	$(Th,U)C_2$ BISO	ThC_2 BISO
Fuel bed test	1	FBTE-2	All-1	Yes	Yes	Rods	UC_2 BISO	ThC_2 BISO
Fuel bed test	1	FBTE-3	B14-08	Yes	Yes	Rods	UC_2 TRISO	ThC_2 BISO
Fuel bed test	1	FBTE-4	F06-01[e]	Yes	Yes	Rods	$(Th,U)C_2$ TRISO	ThC_2 TRISO
Fuel bed test	1	FBTE-5	D09-04	No	No	Blended[f]	$(Th,U)C_2$ BISO $(Th,U)C_2$ TRISO UC_2 TRISO	ThC_2 BISO ThC_2 BISO
Fuel bed test	1	FBTE-6	F14-08[e]	Yes	Yes	Blended	$(Th,U)C_2$ BSIO	ThC_2 BISO
Fuel test	1	FTE-1	A14-08[e]	Yes	No	Blended	$(Th,U)C_2$ BISO $(Th,U)C_2$ TRISO UC_2 BISO UC_2 BISO	ThC_2 BISO ThC_2 TRISO
Fuel test	1	FTE-1	A14-08[e]	Yes	No	Blended	$(Th,U)C_2$ BISO $(Th,U)C_2$ TRISO UC, BISO	ThC_2 BISO ThC_2 TRISO
Fuel test	2	FTE-3	A03-03[g]	Yes	No	Rods	$(Th,U)C_2$ TRISO UC_2 TRISO UO_2 TRISO $(Th,U)C_2$ TRISO	ThC_2 BISO ThC_2 BISO ThO_2 BISO ThC_2 TRISO

							Fuel Particles	
Type	Phase[a]	Ident. Number	Core Position[b]	Thermocouples	Purge Sampling[c]	Fuel Bed	Fissile	Fertile
Fuel test		FTE-4	?	?	?	Rods	UC_2 TRISO UO_2 TRISO $(Th,U)C_2$ TRISO	ThC_2 BISO ThO_2 BISO ThC_2 TRISO
Fuel test	1	FTE-5	C14-08	Yes	No	Rods	UC_2 BISO $(Th,U)C_2$ BISO UC_2 TRISO $(Th,U)C_2$ TRISO	ThC_2BISO ThC_2BISO ThC_2BISO ThC_2 TRISO
Fuel test	2	FTE-6	C02-01	Yes	No	Rods	UC_2 TRISO UO_2 TRISO $(Th,U)C_2$ TRISO $(Th,U)C_2$ TRISO	ThC_2 BISO ThO_2 BISO ThC_2 TRISO ThC_2 TRISO
Fuel test	2	FTE-7	F14-08	Yes	Yes	Rods	UO_2 TRISO	ThO_2 BISO
Fuel test	2	FTE-8	D14-08	Yes	Yes	Rods	UO_2 TRISO	ThC_2 BISO
Fuel test	2	FTE-9	D06-01	Yes	Yes	Rods	$(Th,U)C_2$ TRISO	ThC_2 BISO
Fuel test	2	FTE-12	B06-01	Yes	Yes	Rods	$(Th,U)O_2$ BISO	ThO_2 BISO
Fuel test	3	FTE-14	B03-02[c]	Yes	No	Rods	UO_2 TRISO UC_2 TRISO UO_2 TRISO UC_2 TRISO $(Th,U)O_2$ TRISO UO_2 TRISO UO_2 TRISO	ThO_2 TRISO ThC_2 BISO ThO_2 BISO ThC_2 TRISO ThC_2 BISO ThC_2 BISO ThC_2 TRISO
Fuel test	3	FTE-15	A14-08	Yes	No	Rods	UO_2 TRISO UC_2 TRISO UO_2 TRISO UC_2 TRISO $(Th,U)O_2$ TRISO UO_2 TRISO UO_2 TRISO	ThO_2 TRISO ThC_2 BISO ThO_2 BISO ThC_2 TRISO ThC_2 BISO ThC_2 BISO ThC_2 TRISO
Plutonium fuel test	3	FTE-13	E10-01	Yes	No	Rods	PuO_2 TRISO $(Th,Pu)O_2$ TRISO $(Th,U)C_2$ TRISO	ThO_2 BISO ThO_2 BISO ThC_2 TRISO

Type	Phase[a]	Ident. Number	Core Position[b]	Thermocouples	Purge Sampling[c]	Fuel Bed	Fuel Particles	
							Fissile	Fertile
FSV proof test (FTE type)	2	FTE-10 (PTE-3)	E02-01	Yes	Yes	Rods	(Th,U)C_2 TRISO	ThC_2 TRISO
FSV proof test (FTE type)	3	FTE-16 (PTE-4)	F06-01	Yes	Yes	Rods	(Th,U)C_2 TRISO	ThC_2 TRISO
FSV proof test (FTE type)	3	FTE-17 (PTE-5)	A02-01	Yes	Yes	Rods	(Th,U)C_2 TRISO	ThC_2 TRISO
Monolithic fuel test	3	FTE-18	E06-01	Yes	No	Monolithic	(Th,U)O_2 BISO	None
Recyle test	2	FTE-11 (RTE-1)	E10-06	No	No	Rods	UO_2 TRISO (Th,U)O_2 BISO UC_2 BISO UC_2 TRISO UC_2 BISO (Th,U)O_2 BISO (Th,U)O_2 BISO UO_2 BISO UC_2 TRISO	ThO_2 BISO ThC_2 BISO ThC_2 BISO ThC_2 BISO ThO_2 BISO ThC_2 BISO ThO_2 BISO ThO_2 BISO ThC_2 BISO
Recycle test	1	RTE-2	F07-06[h]	No	No	Mixed (3 rods, 3 blended)	UC_2 TRISO UC_2 BISO (Th,U)O_2 BISO (Th,U)O_2 BISO UC_2 TRISO UC_2 BISO	ThC_2 BISO ThC_2 BISO ThC_2 BISO ThC_2 BISO ThC_2 BISO ThC_2 BISO
Recycle test	1	RTE-4	B10-06[g]	No	No	Same as RTE-2	Same as RTE-2	Same as RTE-2
Recycle test	1	RTE-5	C10-06	No	No	Rods	UO_2 BISO (Th,U)O_2 BISO UC_2 BISO UC_2 TRISO UC_2 BISO (Th,U)O_2 BISO (Th,U)O_2 BISO UC_2 TRISO UC_2 BISO	ThC_2 BISO ThC_2 BISO ThC_2 BISO ThC_2 TRISO ThO_2 BISO ThC_2 BISO ThO_2 BISO ThC_2 BISO ThO_2 BISO

Type	Phase[a]	Ident. Number	Core Position[b]	Thermocouples	Purge Sampling[c]	Fuel Bed	Fuel Particles	
							Fissile	Fertile
Recycle test	1	RTE-6	D10-06	No	No	Rods	UO_2 BISO UC_2 BISO $(Th,U)O_2$ BISO UC_2 BISO	ThO_2 BISO ThC_2 BISO ThC_2 BISO ThC_2 BISO
Recycle test	1	RTE-7	E10-06[e]	No	No	Rods	$(Th,U)O_2$ BISO $(Th,U)O_2$ BISO UC_2 TRISO UO_2 BISO UO_2 BISO UC_2 BISO $(Th,U)O_2$ BISO UC_2 TRISO	ThC_2 BISO ThO_2 BISO ThC_2 BISO ThO_2 BISO ThC_2 BISO ThO_2 BISO ThC_2 BISO ThC_2 BISO
Recycle test	1	RTE-8	F10-06	No	No	Rods	UC_2 TRISO UC_2 TRISO $(Th,U)O_2$ BISO UC_2 BISO	ThC_2 BISO ThC_2 TRISO ThC_2 BISO ThC_2 BISO
Proof Test Elements								
	1[i]	PTE-2	A07-07[e]	Yes	No	Rods	$(Th,U)C_2$ TRISO	ThC_2 TRISO
	1	FPTE-1	E14-08[e]	Yes	No	Compacts	$(U\text{-}238)O_2$ TRISO	None
	2	FPTE-3	E14-08	Yes	No	Compacts	$(U\text{-}238)O_2$ TRISO	None

[a]Phase 1 loaded at O EFPD of Core 2, Phase 2 loaded at 252 EFPD of Core 2, and Phase 3 loaded at 385 of Core 2.
[b]The core position shown is the last position in which the test element resided in the Peach Bottom core.
[c]An indication of purge sampling is relative only to the last position of the test element in the core.
[d]A fuel rod, as used here, is a close-packed assembly of coated fuel particles bonded together with a carbonaceous matrix.
[e]Removed during 252-EFPD Core 2 shutdown.
[f]A blended bed, as used here, is a close-packed assembly of unbonded, coated fuel particles.
[g]Removed during 385-EFPD Core 2 shutdown.
[h]Removed during 701-EFPD Core 2 shutdown.
[i]PTE-2 was irradiated 152 EFPD in Core 1 prior to irradiation of Core 2.

Table 2.7 Comparison of Operation Temperatures in Type II Elements E06-01 (384 EFPD) and E11-07 (701 EFPD) with Low Rhodium (°C)

Component	Bottom	Top	Peak	Variability
Helium Coolant	375	750	750	~linear increase from bottom to top
Sleeve OD	425	800	800	Skewed cosine peaks towards top of element
Sleeve ID	475	850	875	Skewed cosine peaks towards top of element
Fuel OD	550	1000	1300	Skewed chopped cosine cure with minimum at bottom; apparent effect of flux peaking at elements ends
Fuel ID	600	1100	1200	Same as OD

Note: Type II elements represent the majority of the fuel elements, but variability from element to element can be expected depending upon location in core.

Table 2.8. FSV Fuel Element Discharge Schedule

Discharge	Fuel Regions	Number of Elements
2/1/79	5,10,17,21,28,35	246
5/13/79	4,8,15,25,32,36	240
1/2/84	3,13,18,22,29,33	240
Final 8/18/89	All	1482

3.0 DISPOSAL OPTIONS

Figures 3.1 and 3.2 show the three primary options that are being considered for disposal of the HTGR spent fuel. The three options are to 1) dispose of the intact fuel elements with a minimum of conditioning, or 2) separate the fuel compacts or rods from other components, and dispose of them separately, or 3) burn the fuel elements and dispose of the residue. For each of these options, there are various suboptions that must be considered before a final decision can be made as to the preferred method for disposal. At each step of the decision process, consideration must be given to such factors as costs, personnel exposures, and risk to the public. In turn, these factors are impacted by regulatory and technical issues, availability of information needed to address those issues, and the level of effort required to adequately address the issues.

The three disposal options, and their associated suboptions are described in Section 3.2. Sections 3.3 and 3.4 provide overviews of the regulations and technical issues, respectively, that may affect the choices. The fuel characteristics that will be needed to support each option are identified in Section 3.5, and the matrix of tests necessary to obtain those characteristics is described in Section 3.6.

3.1 BACKGROUND INFORMATION

Chapter 2 contains detailed descriptions of 1) the fuel elements irradiated in both cores of the PB Reactor and the fuel blocks irradiated in the FSV Reactor, 2) the PB fuel compacts and the FSV fuel rods, and 3) the variations of fissile and fertile fuel particles contained within the fuel compacts and rods. Therefore, the following brief description of similarities and differences is intended only to highlight those aspects which may influence the choice(s) of disposal options.

Both PB and FSV are HTGRs, designed to operate on the ^{235}U, ^{232}Th, ^{233}U fuel cycle using replaceable graphite fuel elements that also serve as the neutron moderator. However, the PB fuel elements are long cylinders with the fuel contained in circular compacts, while the FSV elements are hexagonal blocks that contain small rods of extruded fuel composites. There are also significant differences in the coatings applied to the fuel kernels to form the fuel particles.

The fuel kernels used in PB Core 1 consists of a mixture of thorium- and uranium-carbides, with the uranium being enriched to slightly more than 93% ^{235}U. The kernels were given a single coating of a laminar pyrolytic carbon; the resultant fuel particles were mixed with graphite particles and a resin-binder, and the mixture was formed into hollow-cylinders (compacts). After pyrolyses of the resin-binder, and heating to higher temperatures, the PBR compacts were assembled around a graphite rod, and enclosed in a graphite sleeve. These assemblies comprised the active portion of the standard PB fuel elements.

When Core 1 was discharged (after about 50% of its design burnup), many of the coatings had split and blossomed, causing swelling and distortion of the compacts. In turn, failure of the coatings had released fission products from the coated particles. Moreover, swelling and distortion of the compacts had produced cracks in over 10% of the graphite sleeves.

For PB Core 2, the ThC_2 content of the fuel kernels was reduced and larger diameter "fertile" kernels, containing only ThC_2, were produced. Both types of kernels were given a low-density pyrolytic carbon (buffer) coating before a dense-isotropic pyrolytic carbon coating was applied; these were called BISO particles. In all other aspects, the fuel compacts and standard fuel elements are similar to those in Core 1. Fewer of the coatings and none of the sleeves failed during operation to essentially 100% of design burnup.

The "TRISO" particles for the FSV fuel blocks are essentially BISO particles that have been given two additional layers of pyrolytic coating. A thin layer of dense silicon carbide was deposited on the BISO particle, and then overlaid with an additional dense-isotropic carbon coating (Dahlberg, Turner, and Goeddel 1969). A mixture of graphite, coated particles, and resin was extruded as rods, and baked at high temperature; these fuel rods were then inserted in small-diameter holes drilled in the FSV fuel blocks.

The fertile particles in the FSV fuel rods are larger in diameter than the fuel (fissile) particles. The different diameters were chosen to facilitate reprocessing of the FSV fuel. The original plan was to crush and burn the entire fuel block. During incineration, the silicon carbide layer would serve as a protective barrier around the inner coatings and carbide kernels. Dissolution of the ash, and screening of the residue, would separate the two sizes of particles so that they could be processed separately.

During irradiation, the ^{235}U content of the fissile particles was reduced; however, significant amounts of ^{235}U are still present, even after the highest exposures experienced by any of the fuel particles. At the same time, some Pu was generated in the fissile particles, and ^{233}U was bred in the fertile particles.

3.2 DESCRIPTION OF OPTIONS

The three options for consideration are to 1) dispose of the intact fuel elements, with a minimum of conditioning, or 2) separate the fuel compacts or rods from other components, and dispose of them separately, or 3) burn the fuel elements and dispose of the residue. Each of these options, and the associated suboptions, are discussed in the following subsections.

3.2.1 Option 1: Dispose of Intact Fuel Elements

It is envisioned that this option would require 1) design and qualification of one or more suitable waste packages, 2) retrieval and packaging (this could also entail conditioning)[6] of the elements, and 3) transportation of the waste packages to a repository. Suboptions that should be considered are 1a) interim storage in a surface, or near-surface, storage facility,[7] 1b) deep-geologic disposal, and 1c) deep-sea disposal.

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[6]Such as "siliconizing" the fuel element surfaces; this process results in partial conversion of the graphite to silicon carbide and filling of the surface pores with silicon, thereby producing an impermeable surface layer on the graphite.

[7]Because of the long half-lives of ^{233}U (1.59×10^5 years) and ^{235}U (7.04×10^8 years), surface or near-surface storage should be considered only as a temporary expedient.

- A decision that the fuel would not be reprocessed, or the determination that the concentration of radionuclides in the graphite components exceeds that of Class C waste, are events that would favor intact disposal of the fuel elements.

- Concerns regarding the long half-lives of the fissionable isotopes, and the potential for diversion to nonpeaceful uses, tend to diminish the appeal of this option.

3.2.2 Option 2: Separate Fuel from Other Components

Disassembly of the PB fuel elements and separation of the compacts is a rather straight forward operation. Pressing the fuel pins from the FSV fuel blocks has been demonstrated on one block from the first segment of discharged fuel; if the fuel pins cannot be pressed out of the high-burnup blocks, the fuel pins could be removed using a hollow core-drill. Those components that do not have radionuclide concentrations in excess of the Class C criteria could then be disposed of as low-level waste (LLW). Components that have radionuclide concentrations in excess of the Class C criteria would still need to be treated as HLW, although it might be possible to temporarily store some until they met the Class C criteria. Consideration should also be given to the potential for burning the graphite components that exceed the Class C criteria in order to reduce the HLW volume and to avoid the cost of designing and qualifying the waste package.

Suboptions for disposal of the fuel compacts and rods include 2a) packaging them for disposal as HLW, with a minimum of conditioning, 2b) conditioning the fuel shapes or coated particles and incorporating them in a waste form prior to packaging,[8] 2c) crushing the fuel particles and dissolving the fuel for reprocessing or vitrification, and disposing of the residue as HLW, or 2d) incinerating the fuel compacts and rods (TRISO particles would need to be crushed and reincinerated), and reprocessing or vitrifying the ash. Several other variations are possible.

- Disposal of unconditioned compacts, especially those from Core 1, could raise concerns regarding the potential for eventual exposure to moisture, with the attendant potential for reaction of the carbides with water, resulting in the generation of flammable gas mixtures, and leaching of the fissile nuclides.

- Disposal of the fuel as HLW, even after conditioning, could raise concerns regarding the long half-lives of the fissionable isotopes, and the potential for diversion to nonpeaceful uses.

- Suboptions 2c and 2d would remove the potential for diversion, either by separation of the fissionable isotopes, or by their dilution in a waste glass. This advantage is somewhat offset by the necessity to deal with fission products that are either gases or would become volatile under some processing conditions.

- Reprocessing to separate the fissionable isotopes offers another advantage, in that it would considerably reduce the total activity of the HLW, especially after thousands of years.

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[8]For example, combustion of the TRISO particles would remove the outer pyrolytic-carbon layer, leaving a layer of SiO_2 that might facilitate incorporation of the particles in a vitreous waste form.

3.2.3 Option 3: Burn the Fuel Elements

Under this option, the entire fuel element, along with the fuel and other components of the elements, would be burned. The single-coated and BISO-coated fuel particles would be reduced to oxides, as would impurities in the graphite. Unbroken TRISO-coated particles would remain as particles with a layer of SiO_2 protecting the inner coatings; but, they could be separated from the ash, crushed, and reincinerated.

There are two ways to proceed with burning of the fuel elements: whole or after size reduction. The original plan for reprocessing of FSV fuel blocks envisioned that the blocks and fuel rods would be reduced to small particles by crushing, and the particles burned in a fluidized-bed furnace (Brooks et al. 1972).

- Reduction of the FSV fuel blocks to small particles requires use of a large multistage crusher, and provision must be made to confine the dusts produced during crushing and handling.
- Provision must be made to contain the ash, volatile fission products, and fines that are entrained in the combustion gases.
- Commercially available equipment can be successfully employed to meet most needs, with only minor modifications.

A much smaller facility is required to burn entire fuel elements, without first reducing them to small particles (Haas 1974, Xien et al. 1992).

- Combustion control requires a specially designed furnace, using a mixture of O_2 and CO.
- The reduced flow rates of the gases reduces problems with containment of ash, volatile fission products, and fines, compared to a fluidized-bed burner.

3.3 OVERVIEW OF PERTINENT REGULATIONS

Appendix C, Regulatory Requirements for Disposal, contains a detailed review of pertinent regulations, and a discussion of regulatory issues, that could impact the choice of a disposal option. It should be noted that, although identical regulatory issues must be addressed for all of the HTGR spent fuel, there are differences in the fuels that may impact the degree of concern, the relative importance of the technical issues, and the viability of specific disposal options. Table C.2 sites specific sections of the regulations under three general categories: those that 1) are pertinent to all three options, 2) contain requirements that are especially applicable to Option 1, Disposal of Intact Fuel Elements, and 3) contain additional requirements that must be considered if Options 2 or 3 are chosen.

As a result of the review of pertinent regulations, it has been concluded that:

1) For any specific activity, it will be necessary to consult the referenced regulations directly, to determine the fine details of exemptions, applicability, radioactivity limits, exact non-radioactive hazardous materials, etc.

2) One of the most costly stipulations (*10 CFR 20.2002*; *10 CFR 60.43*; and *40 CFR 265.13*) is the requirement that there be full chemical and physical characterization of waste material before it can be shipped, stored, or processed.

3) The most rigorous and controlling regulations appear to be:

a) *10 CFR 71, 49 CFR 173* and *DOE 5480.3 - Transportation Regulations*, which require strict limits on hazardous materials and radioactivity content, and

b) *40 CFR 191, 10 CFR 60* and *10 CFR 61 - Detailing Permissible Emissions to Ground, Groundwater, Air and the Entire Environment from Stored Waste.*

Some of the regulatory issues that may impact the choice of disposal option are summarized below.

One of the primary regulatory issues is the limit placed on ^{14}C release from a deep-geologic repository. This issue is currently being reconsidered, as a direct result of the Energy Policy Act of 1992. Another issue, which is partly regulatory and partly technical, is the exclusion of combustible waste materials from the repository: "unless it can be demonstrated that a fire involving the waste packages containing combustibles will not compromise the integrity of other waste packages, adversely affect any structures, systems, or components important to safety, or compromise the ability of the underground facility to contribute to waste isolation." (10 CFR 60.135(c)(3)).

Under *DOE Order 5633.3A*, the fuel meets the conditions for classification as Nuclear Materials Safeguards Category I, Attractiveness Level C; at issue is whether safeguards can be terminated on this material if it is disposed of as HLW. For disposal of FSV fuel blocks as LLW, *10 CFR 61.56(b)(3)* requires that "void spaces within the waste and between the waste and its package must be reduced to the extent practicable;" this requirement needs clarification.

The differences between regulatory limits on release, maximum permissible individual exposure, and population exposures is an issue that can impact disposal options, and needs to be addressed in general as well as within the context of this decision-making process.

3.4 OVERVIEW OF TECHNICAL ISSUES

Deterioration of the fuel elements in the present storage facilities, especially elements from the PB Core 1, is a question that should be addressed, irrespective of the choice of disposal option. Other technical issues that may impact the choice of disposal options are discussed in the following sections.

3.4.1 Option 1: Dispose of Intact Fuel Elements

- Diversion - Is it feasible to provide barriers that will ensure that diversion of the fissile isotopes for illicit uses will not occur?

- Criticality - What are the best means of assuring against inadvertently creating a critical configuration in an interim storage facility or in the repository (or in the sea), during movement of fuel?

- Isolation - Qualification of the waste package needs to provide assurance that deterioration of the package will not result in the generation of combustible gases (in storage or disposal facility), or the release of excessive amounts of radioisotopes to the environment.

- Repository Interface - If the choice is deep-geologic disposal, the waste package should be designed such that it does not require special handling equipment, and does not adversely impact performance of the repository.

3.4.2 Option 2: Separate Fuel from Other Components

Technical issues that should be addressed during consideration of Option 2 include those considered for Option 1, as well as the following.

- Disassembly - Can the fuel rods be pressed out of the FSV fuel blocks, or must they be removed by other means?

- Disposal of Components - What fraction of the non-fuel components qualify for the less-than Class C designation, and can be disposed of as LLW?

- Volatile Fission Products - If the fuel is incinerated or dissolved, how does one ensure that volatile fission products are not released to the environment?

- Economics - What is the cost of reprocessing, versus the costs of other options?

- Volume of Waste - If the fuel is vitrified, what volume of waste glass is required to dilute the fuel?

3.4.3 Option 3: Burn the Fuel Elements

In addition to many of the above issues, the following technical issues should be addressed during consideration of Option 3.

- ^{14}C - Can it be safely released to the atmosphere?

- ^{36}Cl - Can it be separated from the combustion gases, and disposed of safely?

- Other Radioisotopes - What measures are needed to separate the volatile fission products and activation products from the combustion gases?

- Waste Generation - What is the volume of waste that is generated by disposal of the equipment used to process and burn the fuel elements?

3.5 INFORMATION ON FUEL CHARACTERISTICS

Consideration of the regulatory and technical issues that can impact the choice of disposal option(s) resulted in a listing of criteria, and of the information needed to meet those criteria, for each of the options and main sub-options. These information needs are summarized in Table 3.1. Table 3.2 gives information regarding the availability of information from literature sources. It should be noted that this evaluation could change because information may exist which was not available during the time this plan was developed. For example, we were unable to gather business sensitive data from GA, which might fill some gaps in the literature data.

The drivers which will impact the information needs for FSV and PB fuels characterization are both regulatory and technical. Often, the level of detail required for one driver is less than for the others. For example, the radiochemical information necessary to evaluate the release of radionuclides from the fuel is generally less than that required to determine a full radiochemical inventory as required by regulations. In general, U.S. regulations require more detailed documentation of all aspects of waste disposal than may actually be useful for making technical decisions.

In the sections that follow, the technical factors that make it necessary to obtain the information shown in Table 3.1 are discussed and the regulations requiring compliance are quoted and referenced.

3.5.1 Chemical and Isotopic Inventories

Although the chemical and isotopic composition and distribution of elements within the fuels and components are partially known, the data are limited. Much of the data is calculated using computer codes written prior to National Quality Assurance-1 (NQA-1) specifications and may not meet storage or disposal licensing requirements. Even data calculated using ORIGIN-II may be of limited use, as this code has evolved with an emphasis on LWR fuels and may not adequately model graphite fuels.

It will be necessary to verify the chemical and isotopic composition and distribution for these fuels. Data gathered from the fuels can be used to feed computer codes for development, validation, and confirmation purposes. If the codes can be "adequately" validated, only additional confirmation work on fuels with "outlying" properties (cooler regions of the reactors, lower fluence, etc.) will need to be performed. Adequate validation is subjective, but agreement to within 5 to 10% on the major radionuclides is generally acceptable, though the definition can change. If the codes cannot be validated, a more exhaustive characterization program will have to be developed, but the authors believe this scenario is only remotely possible.

It is recommended that a determination be made of what elements and radionuclides could affect the safety and effectiveness of the pre-disposal and disposal procedures for each option (see *40 CFR 265.13* below) and that elemental and radiochemical analyses for these elements be conducted on the fuel components. This will provide documentation of the inventory as required by Federal regulations, information to confirm the limited data already available, and information for model validation.

The regulations which govern the information requirements indicated in Table 3.1 are referenced briefly in the sections in the same order as they appear. Citations from the regulations are given in Appendix C, where the regulations appear in numerical order.

Total Elemental Inventory

The definitive regulation requiring physical and chemical analyses is *40 CFR 264*, which includes all phases of treatment, storing and disposal. Regulation *40 CFR 264* does not appear to mention radioactive waste, by either including or excluding it. However, by extensions via reference in and to other CFRs, *40 CFR 264* will be used to apply to radioactive as well as ordinary hazardous waste, unless additional search of the labyrinth of cross-references in chapter 40 and chapter 10 reveals an exclusion or a substitute regulation.

- Before an owner or operator treats, stores, or disposes of any hazardous wastes, he must obtain a detailed chemical and physical analysis of a representative sample of the waste. At a minimum, the analysis must contain all the information which must be known to treat, store, or dispose of the waste safely. **(40 CFR 264.13(a)(1))**

- The analysis must be repeated as necessary to ensure that it is accurate and up to date. **(40 CFR 264.13 (a)(3))**

- A waste analysis plan must be written and followed. **(40 CFR 264.13 (b))**

- See Isotopic Inventory (see section 3.5.1.2 below for requirements for radionuclide analyses.)

- LLW must be characterized with sufficient accuracy to permit proper segregation, treatment, storage, and disposal. This characterization must ensure that, upon generation and after processing, the actual physical and chemical characteristics and major radionuclide content are recorded and known during all stages of the waste management process. **(DOE 5820.2 Chap III 3.d.(1),(2) and (3))**

Isotopic Inventory

The isotopic inventory is needed for Option I for storage or disposal of the fuel elements "as is"; for Options II and III, the isotopic inventory is needed for transportation, for development of procedures and processes for the dismantling and burning of the fuel components, and for interim storage and final disposal of the resulting waste forms. The regulations driving these needs are *10 CFR 20.2002* and *10 CFR 60.43* for disposal, and *49 CFR 172.403* and *172.404* for packaging (shipping).

- Each application for disposal of licensed material must contain a description of the material to be disposed of, including the physical and chemical properties important to risk evaluation... and analyses and procedures to ensure that doses are maintained ALARA and within the dose limits in this Part. **(10 CFR 20.2002)**

- The repository licensee must conform to restrictions regarding radioisotope content of the waste to be deposited, which is interpreted to mean (based on *40 CFR 264.13* above) that the final step of the spent fuel processing must provide an assay of the product in representative containers to be deposited. **(10 CFR 60.43)**

- For shipping, the label of each package must list contents, giving the names of all the radionuclides as they appear in *40 CFR 173.435*. **(40 CFR 172.403)**

- For mixed waste of different hazard classes in the same package, the label must conform to the regulations for each class. **(49 CFR 172.404(a) and (b))** This will become a consideration for transportation directly to storage as in Option I, or for transport of LLW and HLW to or from processing facilities and to the disposal site for Options II and III.

Inventory Distribution

The requirement to know where the hazardous elements are within the fuel element components is not driven specifically by regulations, but rather by the need to devise safe procedures for, and to deal with conditions that arise in handling and treating the waste.

3.5.2 Release of Radionuclides and Other Hazardous Materials

Little data regarding the release of radionuclides from the fuel elements to the environment are available in the literature. Although the distribution of ^{137}Cs within the PB fuel elements has been documented in six ORNL reports; *ORNL/TM-6455, ORNL/TM-6353, ORNL/TM-5996, ORNL-5126, ORNL-5214, and ORNL/TM-5730*, little is known regarding release to the atmosphere other than to the reactor helium coolant. Data are available regarding the inventories and distribution of ^{3}H and ^{14}C isotopes in PB Core 2 fuel elements (Wichner and Dyer, 1979; ORNL-5497 and ORNL-5597, 1980)[9]; however, these data are not cast in terms of release to the atmosphere. No data was found for ^{3}H and ^{14}C releases from PB Core 1 fuel elements, nor was significant data found for FSV. Little is known about the inventory of the water soluble radionuclides, ^{99}Tc and ^{129}I. These inventories and their distribution throughout the fuel element components will be important in selecting conditioning options and in classifying wastes.

Possible release of air- and water-borne radioactive pollution from disposal sites, and means to prevent it, must be determined before a site can be licensed or a shipment of waste to a site can be approved. To be considered are:

- For Option I - Possible release from the fuel elements themselves into their canisters or from the canisters - either the original or new - during interim storage and transportation, and eventually from the disposal site, where it will be classed as Spent Nuclear Fuel (SNF).

- For Option II - Possible release from the process of separating the fuel from the surrounding graphite blocks (FSV) or the supporting graphite sleeves and spines (PB) and repackaging the components, and then potential release from the separate fuel packages (TRU) and graphite packages (either HLW or LLW).

- For Option III - Possible release from the crushing, burning and separating processes, and subsequent release from the resulting wastes in their packages, the fuel residue (TRU) and the graphite ash (either HLW or LLW).

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[9]Wichner and Dyer claimed the estimates of tritium release from fuel particles was based on too few data to yield a good indication of whole-core performance. They did feel their estimated core-wide inventory and distributions of ^{14}C for the various core components was reasonably accurate.

Release limits for the radionuclides of major concern are given in Appendix C, Table C.3.

Release of Radionuclides from Disposal Facilities

EPA regulations for releases of radionuclides from the disposal containers and disposal sites for SNF, HLW and TRU, are given in *40 CFR 191.13-16*. The NRC regulations for HLW disposal in geologic repositories are given in *10 CFR 60.102, .113, and .135*. NRC licensing requirements for land disposal of radioactive waste are found in *10 CFR 61.41*. The fuel elements of Option I in their current canisters or in new containers, the fuel compacts and fuel rods (and possibly some sections of radionuclide-permeated graphite) from Option II in their containers, and the separated and processed fuel and graphite (and possibly some of the residual ash) from Option III in their containers - all will be subject to the limitations on radionuclide releases to the air and to the ground and ground-water. Following are some of the major stipulations.

- As part of the "engineered barrier system," the waste containers must be designed to release essentially no radiation or radionuclides between 300 and 1,000 years after closure of the repository (the containment period). If small releases do occur, they must be very slow and containable by the repository itself. Following the containment period, releases per year from the engineered barrier system (containers plus repository) must not exceed 1 part in 100,000 of the original inventory of radioactive waste. **(10 CFR 60.113(a)(1))**

- Performance assessments need not provide complete assurance that the requirements of 40 CFR 191.13(a) will be met. What is required is a reasonable expectation, on the basis of the record before the implementing agency, that compliance will be achieved. **(40 CFR 191.13)**

- Disposal systems shall not cause exposure to any member of the public to exceed 25 mrems to the whole body or 75 mrems to any critical organ, during 1,000 years after disposal. **(40 CFR 191.15)** All potential pathways from the disposal site to people must be considered. **(10 CFR 61.41)**

- Performance assessments of isolation of the wastes from the accessible environment shall not consider any contributions from active controls for more than 100 years after disposal. **(40 CFR 191.14)**

Release of Resource Conservation and Recovery Act (RCRA) (Non-Radioactive Hazardous Materials) from Disposal Facilities

Possible release of non-radioactive RCRA hazardous materials must also be considered if the analyses find evidence of significant quantities of the elements listed in Table C.4 in Appendix C of this report (table taken from *40 CFR 261.3*).

If any of these RCRA hazardous elements are present, the radioactive waste becomes mixed waste and is subject to both the RCRA and the NWPA regulations.

The EPA Standards for Owners and Operators of Hazardous Waste Treatment, Storage, and Disposal Facilities (*40 CFR 264*) and its precursor "Interim Status Standards for Owners ..." (*40 CFR 265*) govern releases of RCRA (non-radioactive) hazardous materials from all activities.

- Release of RCRA hazardous wastes from disposal facilities of various types is governed by the following sections, to which the reader is directed for details (from *40 CFR 264* and *265*):

Containers	Subpart I
Tanks	Subpart J
Surface impoundments	Subpart K
Waste piles	Subpart L
Land treatment	Subpart M
Landfills	Subpart N
Incinerators	Subpart O
Chemical/physical/biological treatment	Subpart Q

As an example of these Subparts, Subpart I on Containers requires the owners to maintain the containers in good condition, to make them of materials that are compatible with the wastes they are to contain, to inspect them at least weekly, and to decontaminate them between uses for different wastes. Provisions must be made in the facility to drain away precipitation and to contain spills and leaks if the waste contains liquid. Containers holding ignitable waste must be stored at least 50 feet from the facility's property line. Incompatible wastes must not be placed in the same container. Wastes incompatible with wastes in any other container must be stored at a safe distance or behind a wall or dike. **(40 CFR 264 and 265)**

Gaseous Release from Disposal Facilities or from Processing Systems

Releases from disposal facilities regulated by the documents cited in the section *Release of Radionuclides from Disposal Facilities*, include gaseous radionuclides and toxic chemicals. A list of specific air pollutants to guard against and some of the specific upper limits permitted are given in the following references.

- One of the main set of regulations stemming from the Clean Air Act is *40 CFR 61*. Its application is limited to radionuclides not covered in *40 CFR 191*; i.e., radionuclides other than those in SNF, HLW and TRU under NRC or Agreement States or the DOE.

- A very extensive list of hazardous air pollutants is given in *40 CFR 61.01*. The reader is referred to that set of lists for further information.

- Regulations concerning various specific hazardous materials, such as beryllium, mercury, asbestos, arsenic from glass manufacturing plants, and benzene are given in several subparts of *40 CFR 61*. Of particular possible interest to spent fuel are:

  Subpart H - Radionuclides other than Radon from DOE Facilities
  Subpart Q - Radon Emissions from DOE facilities

- The standard release limit imposed by *40 CFR 61.92* is that "Emissions of radionuclides to the ambient air from Department of Energy facilities shall not exceed those amounts that would cause any member of the public to receive in any year an effective dose equivalent of 10 mrem."

- Containment requirements for SNF, HLW and TRU from *40 CFR 191.13* are that disposal systems for these wastes should be designed to provide a reasonable expectation (proof is not expected) that the cumulative releases in 10,000 years after disposal will have a likelihood of less than one chance in 10 of exceeding the allowed release quantities (Appendix C, Table C.3) and have a likelihood of less than one chance in 1,000 of exceeding ten times those quantities. For monitoring and active institutional controls, markers and barriers, etc., see the document itself. These requirements apply to air but also to other forms of release.

- "Waste must not contain, or be capable of generating, quantities of toxic gases, vapors, or fumes harmful to persons transporting, handling, or disposing of the waste," (except for special provisions for radioactive gaseous waste). **(10 CFR 61.56(a)(5))**

Aqueous

The primary concern regarding ground water is to keep the source of drinking water pure. Hence management of all aspects of handling SNF, HLW, TRU and LLW as well as non-radioactive waste are geared to keeping pollutants out of the water chain.

Note: *40 CFR 191* currently does not apply to the potential repository at Yucca Mountain. Replacement regulations are scheduled for release in December 1994. In the interim, 40 CFR 191 is being used as a guideline.

- In addition to the containment requirements given above in *40 CFR 191.13*, individual protection requirements stipulate that disposal systems must provide reasonable assurance that, for 1,000 years after disposal, the disposal system shall not cause exposure to any member of the public in the accessible environment of more than 25 mrem to the whole body or 75 mrem to any critical organ. Included is the assumption that a person might drink 2 liters of the ground water from the area. **(40 CFR 191.15)**

- The Ground water protection requirements section, *40 CFR 191.16*, which has been remanded but remains the best guidance available, provides limits on beta-, gamma- and alpha-emitting radionuclides for drinking water. (See the section itself for details.)

- For owners and operators of facilities that treat, store or dispose of hazardous constituents, *40 CFR 264.94* sets concentration limits for leaking into the ground-water for arsenic, barium, cadmium, chromium, lead, mercury, selenium, silver and six organic compounds. *40 CFR 264.97* sets general ground-water monitoring requirements.

Respirable Powder

Concerns regarding respirable powders arise in connection with handling and conditioning activities. It will be important to know if the dry storage canisters or the fuel elements that may have to be removed from their canisters have dusts on their surfaces which contain radionuclides, and if they have, in what quantities; with what physical characteristics; and how easily they fall or rub off.

- Regulatory requirements are derived from *40 CFR 61.92* as quoted in *Release of RCRA (Non-Radioactive Hazardous Materials) from Disposal Facilities* above; it applies in addition to the normal precautions taken to protect radiation workers.

3.5.3 Chemical and Phase Stability

Oxidation - Hydrolysis

Little information exists regarding the oxidation and/or hydrolysis stability of the fuel elements, compacts, or fuel kernels. When considering direct disposal or disposal of fuel compacts or kernels, these factors are important for developing performance assessment predictions. If direct disposal of any of these components is desired (as HLW) it will be necessary to develop the time-temperature-oxidation relationships for those materials in terms of oxidation. Degradation in the presence of water is also poorly understood. These mechanisms will need to be evaluated to understand the release of radionuclides for LLW and for HLW disposal. Specific types of tests to evaluate these phenomena are not proposed at this time and may require development.

Ultimately the chemical and phase stability for any waste forms developed as a result of conditioning steps and waste form containers will need to be tested. If the waste forms include glasses or ceramics, a testing program similar to that being implemented by defense waste glass producers can be adopted. This involves comparing the short-term durability of the glass product against a known glass. If the glass product performs better than the reference glass by some measure, the glass is deemed consistent and acceptable. These tests, however, can tell nothing about the long-term durability of the glass product, and a controversy continues to be waged regarding the usefulness of these tests. It is expected that in the next 5 to 10 years improved tests and strategies will evolve for verifying glass waste form performance. The development of these new tests will most likely coincide nicely with the start-up of any storage/disposal programs involving these graphite fuels.

Waste containers must be evaluated for corrosion resistance. Typically, standard tests exist for testing the corrosion of metals in aqueous or moisture laden atmospheres. Short-term tests which can predict long-term performance are generally untested. Specific tests will be required once the waste form container material is selected. In general, selection of waste container materials is relatively insensitive to waste form unless the waste form is a corrosive liquid or can create oxygen-differential corrosion cells.

For regulations regarding release from conditioning procedures, or from storage, see *40 CFR 61.92, 40 CFR 191.15 and .16,* above.

3.5.4 Combustibility (non-Combustibility)

The conditions that foster self-sustained burning in solid graphite are dependent on a number of factors (Nightingale, 1962) so that combustibility of solid graphite is not a high-risk factor in handling, treating, or storing graphite fuel elements or their components. However, disposal Option III involves crushing the graphite fuel elements, followed by burning the resulting powdered graphite. There exists the potential for release of this combustible/ignitable graphite powder into processing equipment surroundings and the potential for disposing of some powdered graphite unburned.

- "All combustible radioactive wastes shall be reduced to a noncombustible form unless it can be demonstrated that a fire involving the waste packages containing combustibles will not compromise the integrity of other waste packages,..." **(10 CFR 60.135 (c)(3))**

- A solid waste exhibits the characteristic of ignitability if a representative sample of the waste has, among other properties, the capability, "under standard temperature and pressure, of causing fire through friction, absorption of moisture or spontaneous chemical changes, and when ignited, burns so vigorously and persistently that it creates a hazard." or if it "is an ignitable compressed gas as defined in *49 CFR 173.300*;" or if it "is an oxidizer as defined in *49 CFR 173.151*." **(40 CFR 261.21(a)(2) and (3))**

- The general facility requirements for ignitable waste are found in *40 CFR 264.17*, in which the owner is required to prevent accidental ignition by protecting it from all potential sources that could cause it to ignite, and, in specific storage facilities, by separating it from other waste. (See *40 CFR 264.176, 264.198, 264.229, 264.256, 264.281, and 264.312* for Special Requirements for ignitable or reactive waste in containers, tank systems, surface impoundments, waste piles, land treatment and landfills, respectively).

3.5.5 Criticality

The search of the literature did not yield a validated model for performing criticality calculations. A code development activity will likely be needed to predict criticalities for specific configurations. This will be needed for safety considerations during handling and for use in developing conditioning options (for example, how much burner ash can be allowed to accumulate prior to reaching a critical configuration). GA may have such a model developed, but this was not available during the time this report was prepared.

- Typical regulations concerning criticality require that "all systems for processing, transporting, handling, storage, retrieval, emplacement, and isolation of radioactive waste shall be designed to ensure that a nuclear criticality accident is not possible unless at least two unlikely, independent, and concurrent or sequential changes have occurred in the conditions essential to nuclear criticality safety. Each system shall be designed for criticality safety under normal and accident conditions." **(10 CFR 60.131 (b)(7))**

- In addition to the criteria stated in *10 CFR 60.131* above, when possible, the design of an ISFSI or MRS must "be based on favorable geometry, permanently fixed neutron absorbing materials (poisons), or both..." with means for continuing verification of its efficacy. **(10 CFR 72.124(b))**

3.5.6 Pyrophoricity, Corrosivity, Reactivity and Incompatibility

The characteristics of explosiveness and pyrophoricity are extreme cases of combustibility or ignitability, one of the defining characteristics of hazardous waste. Information about the potential for a given waste form to explode or ignite spontaneously is essential for safety at all stages of handling, processing, and storing (see Section 3.5.4).

The characteristics of chemical reactivity and corrosivity are also defining characteristics of hazardous waste. Information about the potential for a given waste form to react with other components or to

corrode in potential handling, treatment or storage situations, is essential to worker and environmental safety. Specific regulations regarding these characteristics are given in the following citations.

General Requirements for Ignitable, Reactive, or Incompatible Wastes

- The owner is required to take precautions to prevent accidental reaction; to separate the waste from potential sources that would trigger a reaction; and to prevent reactions which cause extreme heat, pressure, fire, explosions, or damage to the facility, or produce uncontrolled toxic mists, fumes, dusts, or gases in sufficient quantity to be a hazard to humans or the environment or by some other like means threaten human health or the environment. **(40 CFR 264.17 and 265.17)**

Pyrophoricity

- The waste must not be pyrophoric or capable of detonation, explosive reaction with water, or explosive decomposition or reaction at normal pressures and temperatures. Pyrophoric materials contained in waste must be treated, prepared, and packaged to be nonflammable. **(10 CFR 61.56 (4) and (6))**

Corrosivity

- Solid waste containing liquid must contain as little free standing and noncorrosive liquid as possible, in no case more than 1% liquid by volume. **(10 CFR 61.56 (3))**

- If a material in contact with SAE 1020 steel corrodes the steel at a rate greater than 0.250 inch per year at a test temperature of 55°C, as determined by the standard or equivalent test, it is defined as corrosive (as stated for liquids). **(40 CFR 261.22)**

Reactivity

- A solid waste exhibits the characteristic of reactivity if a representative sample of the waste has any of the following properties.
  1) Normally unstable and readily undergoes violent change without detonating.
  2) Reacts violently with water.
  3) Forms potentially explosive mixtures with water.
  4) When mixed with water, generates toxic gases, vapors or fumes in a quantity sufficient to present a danger to human health or the environment.
  5) Is capable of detonation or explosive reaction if it is subjected to a strong initiating source or if heated under confinement.
  6) Is readily capable of detonation or explosive decomposition or reaction at standard temperature and pressure. **(40 CFR 261.23)**

Incompatibility

- Regulation *40 CFR 264* contains special requirements for incompatible wastes for containers, tanks, surface impoundments, waste piles, land treatment, and landfills (*264.177, .199, .231, .257, .282, and .313*). Typical of these is *40 CFR 264.177* for containers, which states that

incompatible wastes must not be placed in the same container and must not be placed in an unwashed container in which there had been an incompatible material. Furthermore, a storage container holding hazardous waste that is incompatible with other materials stored nearby must be separated from the other materials or protected with a barricade.

- Several groups of incompatible materials are listed in *40 CFR 264 Appendix V.*

3.5.7 Toxicity

Toxicity of these wastes is important for RCRA considerations. Typically the toxic characteristic leaching procedure (TCLP) test is used to determine toxicity of a material. Currently, spent fuel is considered to be exempt from RCRA toxicity considerations, but this issue is being revisited at this time. Determination would be necessary for materials scheduled for LLW disposal.

- A solid waste exhibits the characteristic of toxicity if, using the prescribed test methods, the extract from the waste contains any of the contaminants listed in Appendix C, Table C.5 of this report. **(40 CFR 261.24)**

- Several lists of hazardous and potentially toxic wastes are provided in *40 CFR 261 Subpart D (EPA)* and *49 CFR 172.101 (DOT).*

3.5.8 Physical Size and Weight

The physical size and weight of the fuel elements and the waste forms derived from any pre-disposal procedure must be known for a variety of reasons:

1) developing procedures for lifting, transporting, treating;
2) designing containers for transporting or storing;
3) designating storage space required;
4) providing required information for licensing (*10 CFR 60.43 (b)(1)&(2)* on form, size and shape;
5) assuring worker safety.

3.5.9 Physical Properties

The physical properties of friability and compressive strength are important for two reasons. First, direct disposal requirements may require the fuel elements to have some compressive strength to resist stacking loads. It is known that there were water leaks into the FSV reactor on several occasions. The effect of those limited exposures to water on the physical properties of the fuel block graphite has not been determined. Water leakage into the PB storage drywells over the course of 20 years is probable, and the effect on the graphite has not been determined.

Friability is important as a measure of how well the fuels will survive handling and transportation. If the fuels are friable, procedures for handling and transportation may have to be developed or modified. These properties are also important to conditioning steps. These properties may affect fuel compact or pin extraction, element crushing, etc. The regulations requiring this information are those safeguarding worker and environmental safety.

3.5.10 Physical Condition

The quantity of fuel elements and blocks that are intact or broken is generally well known; however, the quantity of elements, blocks, or other components that are cracked is not well known. This information is most useful for the direct disposal options. These properties could potentially affect the overall release of radionuclides and may be important if there are significant numbers of cracked components, etc. The fuel elements that are selected for characterization should be examined for these attributes. This information, coupled with literature information should be sufficient to describe these properties.

The number of kernel failures appears to be reasonably well documented for PB Core 2; physical examinations indicated that predictive failure models were conservative. Because PB Core 1 was judged to have failed due to fuel failure, the percentage of kernel failures was only estimated. The estimated value may need to be confirmed. The percentage of failed kernels from FSV is documented for element 1-0743 which was removed from the core early in life. The percentage of failures is lower than model predictions. However, no data exist for FSV fuel elements which were irradiated to higher burnups. Though the available data may be satisfactory in most cases, it would be safest to verify the percentage of fuel failures to fully understand the release characteristics of these fuels. There was no information available to document the percentage of kernels which may be near failure, but this information might be obtained or inferred from potential corrosion or leach testing of the fuel compacts or rods.

The temperature reached by these fuels is only partially documented. Knowledge of the temperature of each fuel element, together with specific fluence exposure and/or burnup would be useful for determining the radionuclide content and the probable condition (cracking, swelling, etc.) of each element. Temperature profiles for the reactor cores are generally known, but the temperature extremes experienced by individual fuel elements is undocumented. In a few cases (PB Core 2 monitoring program), temperature data are known for one or two vertical and radial locations in the core. No information like this exists for PB Core 1, and although data may exist for FSV, it was not found in the open literature. Therefore, for consideration of handling, transportation, and direct disposal procedures for any given fuel element, these data, in general, must be estimated.

Again, regulations requiring this information are those for the safety of the workers and the environment, which are generic for any facility handling radioactive materials and are therefore not discussed in this report.

- In addition, "waste must have structural stability." A structurally stable waste form will generally maintain its physical dimensions and its form, under expected disposal conditions, such as weight of overburden and compaction equipment, the presence of moisture, and microbial activity, and internal factors such as radiation effects and chemical changes. Structural stability can be provided by the waste form itself, processing the waste to a stable form or placing the waste in a disposal container or structure that provides stability after disposal. **(10 CFR 61,56 (b)(1))**

- Void spaces within the waste and between the waste and its package must be reduced to the extent practicable. **(10 CFR 61.56(b)(3))**

SECTION 3.0

FIGURES AND TABLES

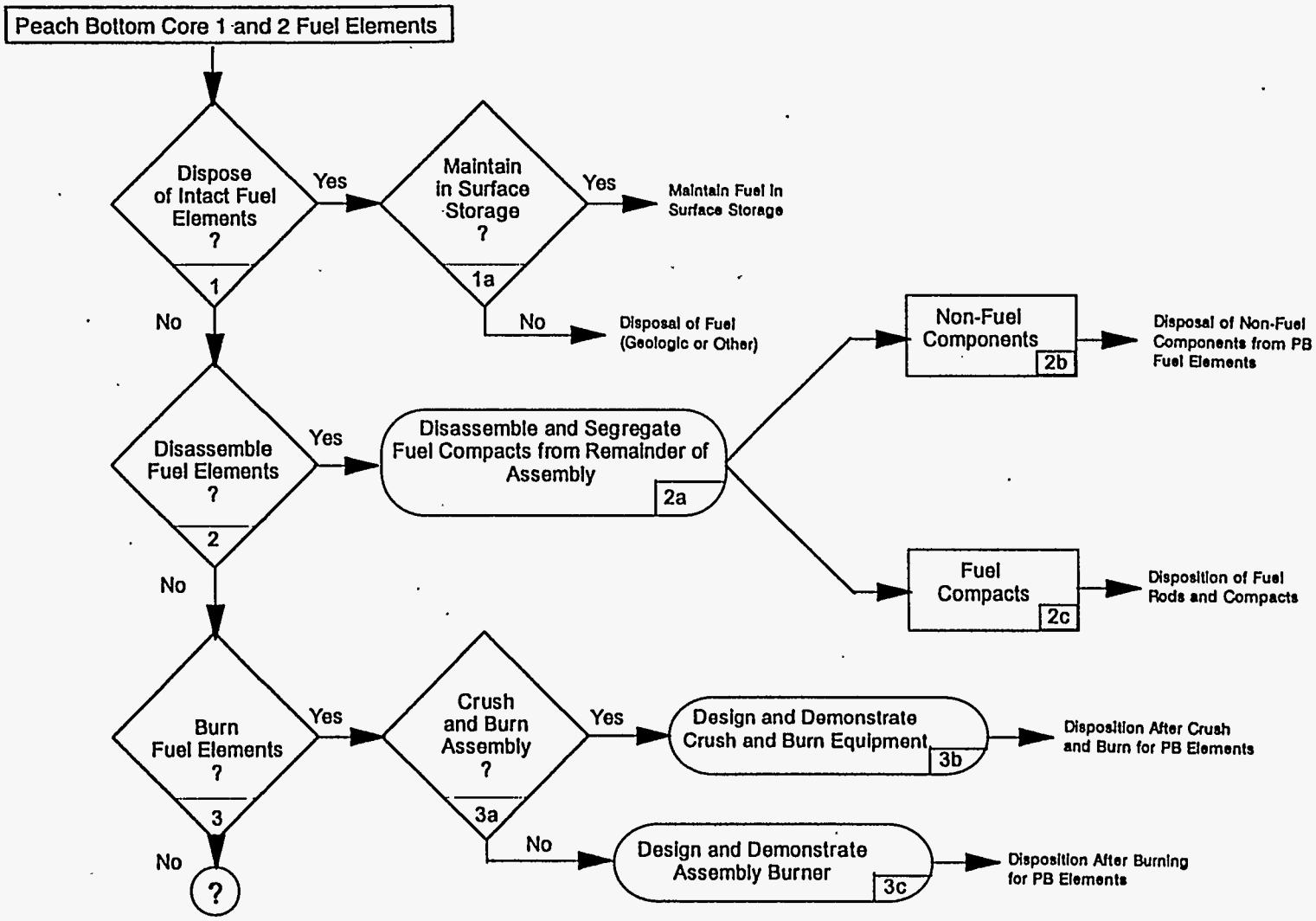

Figure 3.1 Disposal Options for Peach Bottom 1 Fuel Elements

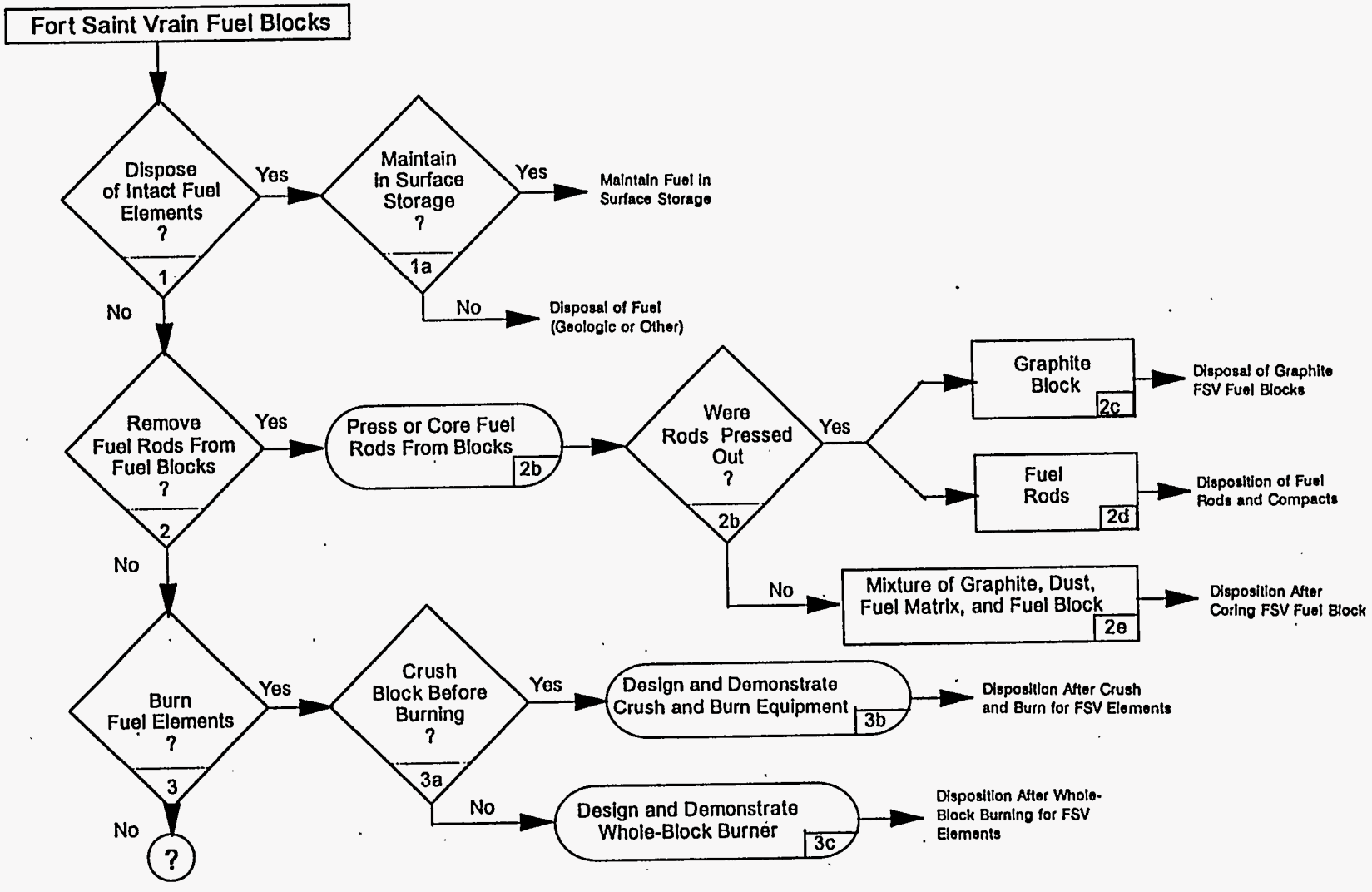

Figure 3.2 Disposal Options for Fort St. Vrain Fuel Elements

Table 3.1. Information Needs to Meet the Criteria

Criteria	Dry Storage		Handling	Transportation	As Is Option 1	Partial Dissassembly Option 2		Total Dissassembly Option 3
	Unlicensed	Licensed				LLW	HLW	
1. Chemical/Isotopic Composition & Distribution								
a) Total Elemental	yes	yes	no	yes	yes	yes (b)	yes	yes
b) Isotopic Inv.	yes	yes	no	yes	yes	yes	yes	yes
c) Inv. Distribution	no	no	no	no	yes (a)	yes	yes	yes
2. Release								
a) Gaseous	yes	yes	yes	yes	yes	no	yes	yes
b) Aqueous	yes	no	no	yes	yes	no	yes	yes
c) Respirable Powder	no	no	yes	yes	no	no	no	no
3. Chemical & Phase Stability/Degradation								
a) Oxidation - Hydrolysis								
i) Elements	yes	yes	yes (m)	no	yes	no	N/A	N/A
ii) Compacts	yes	yes	yes (m)	no	yes	N/A	yes	N/A
iii) Kernels	yes	yes	yes (m)	no	yes	N/A	yes	N/A
iv) Wasteform	N/A	N/A	no	no	N/A	N/A	yes	yes
v) Containers	yes	yes	yes (m)	no	yes (m)	N/A		N/A
4. Combustibility (n)	yes	yes	yes	yes	yes	yes	yes	yes
5. Criticality								
a) Adequate Model	yes	yes	yes	yes	yes	yes	yes	yes
b) Isotopic Inventory & Distrib.	yes	yes	yes	yes	yes	yes	yes	yes
c) Planned Geometry	yes	yes	yes	yes	yes	yes	yes	yes
6. Explosive, Pyrophoric, Chemical Reactivity, Corrosive	yes	yes	yes	yes	yes	yes	yes (l)	yes (l)

	Dry Storage					Partial Dissassembly Option 2		
7. Toxicity	yes	yes	no	no	yes	yes	yes	yes
Criteria	Unlicensed	Licensed	Handling	Transportation	As Is Option 1	LLW	HLW	Total Dissassembly Option 3
8. Physical Size & Weight								
a) Fuel Element	no	yes	yes	yes	yes	yes	yes	yes
b) Waste Form	N/A	yes	yes	yes	yes	yes	yes	yes
9. Physical Properties								
a) Friability	no	no	yes	yes	no	no	no	no
b) Compressive Strength	no	no	yes	yes	yes (j)	yes (k)	no	yes (k)
10. Physical Condition								
a) Intact/Broken	no	yes (c)	yes	no (d)	yes	yes	yes (e)	no
b) Cracks -								no
1. Donut	no	no	no	no	yes (f)	yes (e)		no
2. Fuel Stick	no	no	no	no	yes (f)	yes (e)		no
3. Other Component	no	no	yes	yes	yes (f)	yes (e)		no
c) PB Canisters	yes	yes	yes	yes (g)	yes	no (yes		yes
d) Kernal Failure	no	yes	no	no	yes	LLW)		no
e) Fuel Stick/Donut Swelling	no	no	no	no	no	yes		
f) Temperature	no	no	no	yes	yes	yes (h)		no (i)
						no (c)		

(a) meet reg. - meet secondary
(b) RCRA requirements
(c) for criticality calc.
(d) If broken, assume actions taken prior to transport to take care of problem.
(e) For disassembly of FSV.
(f) For release considerations.
(g) Must consider impact of fire on A1 canister.
(h) for disassembly
(i) Temperature must be addressed in waste form development.
(j) For repository accident scenario.
(k) for disassembly
(l) This assumes waster form will be developed with this criteria as requirement.
(m) Need to know whether problem exists before handling fuels.
(n) The issues of critical temp., atmosphere, configuration, heat transfer, and external heat source must be considered for each scenario.

Table 3.2. Information Availability for FSV and PB Fuels

	FSV	PB (Core 1)	PB (Core 2)
1. Chem/Isotopic Comp & Distribution			
a) Total elemental inventory	partially	partially	partially
b) RCRA requirements	no	no	no
c) Isotopic inventory	Limited Calc.	partially	partially
d) Inventory distribution	No	no	partially
2. Release			
a) Gaseous	no	no	no
b) Aqueous	no	no	no
c) Respirable powder	no	no	no
3. Chemical & Phase Stability/Degradation			
a) Oxidation Hydrolysis			
i) Elements	no (g)	no (g)	no (g)
ii) Compacts	no (g)	no (g)	no (g)
iii) Kernels	no (g)	no (g)	no (g)
iv) Waste form	no	no	no
v) Containers	N/A	no	N/A
5. Combustibility (h)	yes (f)	yes (f)	yes (f)
6. Criticality			
a) Adequate Model	?	?	?
b) Istopic Inventory & Distribution	no	no	no
c) Planned Geometry	?	?	?
7. Explosive, Chemical Reactivity, Corrosive, Pyrophoric	yes	yes	yes
8. Toxicity	no	no	no
9. Physical Size & Weight			
a) Fuel Elements	yes (e)	yes	yes
b) Waste Forms	no (d)	no (d)	no (d)
10. Physical Properties			
a) Friability	no	no	no
b) Compressive Strength	partial (c)	no (c)	no (c)
11. Physical Condition			
a) Intact/Broken	yes (a)	yes (a)	yes (a)
b) Cracking			
Fuel Rod	no	n/a	n/a
Other Components	no	partial	partial
PB Compacts	N/A	no	no
c) PB Canisters	N/A	no	no
d) Kernal Failure	yes (b)	yes (b)	yes (b)
e) Fuel Stick/Donut Swelling	no	no	no
f) Temperature	no	no	no

(a) As discharged, but not as is after storage.
(b) Needs confirmation.
(c) Have some information, need confirmation.
(d) Except for disposal as-is.
(e) More exact dimensions may be needed for fuel disassembly.
(f) General literature is adequate for most scenarios.
(g) However generic information will be adequate for some scenarios.
(h) The issues of fuel compacts, kernels, waste form, and elements/components must be considered for each scenario

4.0 FUELS NEEDED FOR CHARACTERIZATION

The drivers which will impact the information needs for FSV and PB fuels characterization are both regulatory and technical as discussed in section 3.5. Often, the level of detail required for one driver is less than others. For example, the radiochemical information necessary to evaluate the release of radionuclides from the fuel is generally less than that required to determine a full radiochemical inventory as required by regulations. In general, U.S. regulations require more detailed documentation of all aspects of waste disposal than what may actually be useful for making technical decisions.

The following sections contain recommendations for characterization activities to be conducted prior to choosing a storage/disposal pathway for these graphite fuels. These recommendations are based on a technical assessment of the information summarized in Tables 3.1 and 3.2, and on information gathered from the review of the literature and regulations.

4.1 General Recommendations

The following general recommendations emerged during preparation of this document.

1) PB Core 1, PB Core 2, and FSV reactor fuels are recommended for treatment as separate entities. Reasons for this recommendation include variation in storage environment, significant performance differences in-reactor, fuel element design, fuel particle type, and different degrees of previous PIE examinations.

2) It is recommended that FSV fuels be characterized first. Generally it is accepted and understood that the FSV fuels represent the best performers of the PB Core 1, PB Core 2, and FSV. Treatment of FSV fuels first allows for development of operator/handler proficiency working with the safest fuels. The monolayer and BISO coatings used in the PB fuels did not sufficiently contain fission products; disassembly of PB fuels may result in higher releases of fission products from element components.

3) Test elements from PB Core 2 and FSV represent only a small fraction of the total inventory of fuel. They should not be expected to present complications for any storage or disposal pathway selected.

4.2 Potential Fuels for Characterization

To reduce the number of fuel elements that will have to be evaluated, fuels with "worst case" properties should be selected for the first examinations. Selection of fuels with worst case properties generally means selecting fuels from the areas of the reactor cores where fluence and temperature are highest. It also means selecting fuels which saw the longest exposures in the reactors. If a successful storage/disposal strategy can be developed based on the information gathered from these fuels, the number of examinations, time, safety risk, and cost are reduced.

The literature supports the position that both reactors appear to be radially symmetrical in properties (fluence, temperature) about the vertical central axis. This means that areas of highest fluence, burnup, and temperature within the reactors can be established with reasonable certainty. Selection of fuels with long in-reactor performance from the PB inventory is not a problem as nearly all of the PB elements are stored at INEL. However, obtaining representative fuels from FSV may be complicated by the ability to obtain similar fuels with higher in-reactor exposure. It is hoped that the few fuel blocks shipped from Colorado to INEL in

1992 will contain representative samples useful for the needs of any characterization activities. If not, representative samples would have to be obtained from Colorado.

Selection of PB fuel elements to analyze can be deduced from the information presented in Section 2.1.4. As summarized in that section, fuel elements between radial core position 8 and 14 and fuel compacts between compact numbers 10 and 20 would be likely candidates for characterization based on fluence, power, and burnup. It would also be desirable to select elements with the longest in-reactor exposure.

Selection of FSV fuel elements can be determined from the information presented throughout Section 2.1 and its subsections. Based on fluence and temperature, fuel elements from fuel zones II and III, located in core layers 6, 7, and possibly 8 would be likely candidates for characterization. Although the fluence is lower in core layer 8 (than in layer 6), the temperature is higher; this warrants layer 8's inclusion for consideration.

Fuels from these regions of the two reactors should provide sufficient material for developing an understanding of the "worst case" characteristics of these fuels. The authors believe that these fuels will be sufficient to encompass the inventories of these fuels if the worst case characteristics are acceptable within the storage/disposal pathway(s) selected. If these characteristics fall outside "acceptable" performance boundaries, more examinations on a wider number of fuels will have to be conducted. Using the scenario of selecting worst case fuels, the following fuels are recommended for examination.

1) PB Core 1. At least one Type II non-failed element, one Type II failed element, and one Type III non-failed element. These elements can be selected from the core regions described above. Listed below are the reasons for these selections.

- A Type II non-failed element would represent the largest number of elements in Core 1. Since non-failed elements would be selected, these elements may be the easiest to examine. Elements should be able to be obtained with suitably high exposure to temperature and fluence to represent a "worst case" fuel.

- A Type II failed element probably represents the worst case in terms fission product release and redistribution in the fuel element components. It will also allow comparative tests between the different types of fuel compacts.

- A Type III non-failed element would provide data on the ZrB_2 burnable poison contained in some of the elements.

2) PB Core 2. At least two Type II non-failed fuel elements should be examined.

- The Type II fuel represents the largest inventory within this coreload. Elements should be able to be obtained with suitably high exposure to temperature and fluence to represent a "worst case" fuel.

3) FSV fuel. At least two fuel blocks from regions of high temperature and fluence and long in-reactor performance (preferably blocks removed at reactor EOL).

Long-term in-reactor performance may present problems for acquisition unless examples of these fuels are present at INEL as part the fuels shipped from Colorado in 1992. If the best fuels for these attributes are in storage at Colorado and are unavailable, the INEL inventory should be searched for representative samples (which may exist in a few blocks). Fuel blocks from fuel zone II or III from core layers 6, or 7 are likely candidates for samples to examine. These will have been exposed to the highest fluence and experienced some of the highest temperatures.

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APPENDIX A

MONITORING PROGRAM FOR FSV AND PB FUELS STORED AT INEL

Spent fuels from the PB, Unit 1, and FSV Reactors are stored at INEL. The following spent fuel monitoring program[10] was developed as a tool to enhance the ability of WINCO staff to maintain cognizance of the fuel condition (effects of irradiation, shipping, handling, and past storage) and to detect changes in the condition of the stored fuel (effects of future storage, handling, and shipping).

A.1 STRATEGY

The following strategy was used to develop the monitoring plan:

- Interact with WINCO staff to develop a spent graphite fuel monitoring program that defines the monitoring tasks.
- Secure historical data and details of the graphite fuels and storage facilities from published reports and WINCO staff.
- Develop a monitoring plan that 1) is compatible with system/facility/site capabilities, 2) provides answers to questions regarding fuel condition, subcriticality, storability, and retrievability, and 3) has cost and schedule requirements that are consistent with the benefits.

A.2 BACKGROUND

Sections A2.1, A2.2, and A2.3 contain information that is pertinent to the development and implementation of the monitoring plan. Issues that need to be considered are discussed in Section A.3 and nondestructive and destructive monitoring activities are discussed in Section A.4. Additional information regarding the fuel is presented in more detail in Chapter 2 of this report.

A.2.1 PB Core-1 Fuel

The fuel kernels in Core 1 of PB fuel consists of a mixture of thorium- and uranium-carbides, with the uranium being enriched to slightly more than 93% ^{235}U. The kernels were given a single coating of a laminar pyrolytic carbon, before being formed into hollow cylinders (compacts). The compacts were assembled around a graphite spine, and enclosed in 3.5-in. diameter graphite sleeve, to become the fueled portion of the 12-ft long PB fuel elements.

In-reactor failure information and predictions for a helium coolant temperature range of 375 to 750°C and a maximum calculated irradiation fuel temperature of 1200 to 1300°C indicate that the PB Core 1 coated fuel particle failure rate was 80%. Cracking and shape change of coatings resulted in release of fission products and cracking of the fuel element sleeves; two more sleeves were broken during subsequent handling.

The Core 1 elements are stored in 4.48-in. diameter canisters; each is 153-in. long. Each canister is constructed from aluminum with stainless steel liners and weighs 150 lbs fully loaded with spent fuel. The aluminum sleeves were roll pressed into o-ring grooves on the canister end-caps, providing a vacuum-quality tested seal. In cases where leakage of the internal helium gas was detected, the element and canister were

[10]The scope does not include several dozen (maybe about 50) canisters of nonirradiated Rover graphite matrix, UC_2, fuel in storage at INEL in the ICPP 603 air vault; there are 15 Rover fuel tubes {1.9 cm outer diameter by 132 cm (51 in.) long} per canister.

sealed in a second canister before shipping to the INEL. The Core 1 elements were shipped to the INEL in 46 baskets; each basket was 25.5-in. in diameter, contained up to 18 canistered Core 1 elements, and weighed 3,400 lbs when fully loaded. Canisters of fuel remain in these forty-six baskets and are situated in dry wells. The ICPP Fermi I Blanket Storage Facility 749 at INEL consists of an open field of dry wells that contain 813 Core 1 elements.

A.2.2 PB Core 2 Fuel

The fuel kernels in the standard Core 2 fuel elements were given a low-density pyrolytic carbon (buffer) coating before a dense-isotropic pyrolytic carbon coating was applied; these were called BISO particles. In all other aspects, the fuel compacts and standard fuel elements are similar to those in Core 1. Fewer of the coatings failed, and none of the sleeves failed, during operation of Core 2.

The IFSF in Building 603 at INEL contains 785 Core 2 elements. Up to 12 Core 2 elements are stored in an 18-in. diameter canister that is 129-in. long. Each canister is constructed with carbon steel and has a lid latched by 3 toggles. The top 18 in. of each Core 2 element were cut off prior to its emplacement into canisters. ORNL is storing 10 Core 2 fuel elements; two additional Core 2 fuel elements were destructively examined at ORNL and then transported to England.

A.2.3 FSV Reactor Fuel

The "TRISO" particles for the FSV fuel blocks[11] are essentially BISO particles that have been given two additional layers of pyrolytic coating. A thin layer of dense silicon carbide was deposited on the BISO particle, and then overlaid with an additional dense-isotropic pyrolytic carbon coating. The TRISO coating is an effective barrier that retains radionuclides up to high internal fuel particle pressures; however, lanthanide fission products can migrate during high temperature irradiation and react with the SiC layer.

The FSV fuel elements are hexagonal graphite blocks drilled with a multiplicity of fuel holes and coolant channels. Each hexagonal block is 14.172 in. across flats and 31.22 in. in length. INEL stores 725 of the 726 elements from the first three FSV discharges. Element 1-0743 was destructively examined at GA. Most of the remaining 1482 elements are stored in a modular vault dry store system at FSV in Colorado (several fuel blocks were shipped to INEL in 1992). In-reactor failure information and predictions for a helium coolant temperature range of 375 to 750°C and a maximum calculated irradiation fuel temperature of 1260°C indicate that the FSV fuel failure rate was <1%.

The FSV spent fuel elements are stored in 1/4-in.-thick carbon steel canisters (18-in. diameter by 11-ft long) in an ICPP air vault (hot cell with a single overhead bridge crane, and one window that provides limited viewing) in Building 603. The canister lids do not have gaskets and are held in place by remotely operable DE-STA-CO clamps. The storage facility air temperature is approximately 80°F. Heating by the spent fuel increases the exhausted air to approximately 91°F; whereas the facility is designed for heating exhausted air up to 180°F. Air flow is single pass and exhausts through a high efficiency particulate air (HEPA) filter.

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[11]Some of the PBR test elements also contained TRISO coated particles.

A.3 POTENTIAL FACTORS IN THE DEGRADATION OF SPENT FUEL IN STORAGE

Following are several factors that have potential to degrade spent fuel and containers.

1) Water Ingress into vault and fuel can, resulting in:

- galvanic corrosion, further compromising liner/canister integrity
- carbide-water reactions; higher molar volume (lower density) of oxide products may cause internal stresses that may result in loss of noncriticality configuration and affect retrievability.

2) Air ingress into fuel, resulting in oxidation of uranium and thorium carbides to the oxides; may cause internal stresses that may result in degradation of noncriticality configuration and affect retrievability.

3) Fuel condition:

- kernel failure fraction
- graphite strength, as a consequence of pressure, thermal, and chemical stresses.

A.4 NONDESTRUCTIVE MONITORING ACTIVITIES[12]

Nondestructive monitoring activities may determine 1) condition of fuel degradation, 2) liner and canister conditions, 3) releases to storage systems and environment during dry storage, and 4) future retrievability.

A.4.1 PB Core 1 Spent Fuel

A summary assessment of monitoring PB Core 1 Spent fuel in the 749 Dry Wells is provided in Table A.1. Current monitoring activities are introduced and identified by the word "CONTINUE:" and should be continued. Some of these monitoring activities should be increased in frequency to establish rates of degradation. Because the Core 1 PB spent fuel was degraded during irradiation, its monitoring requirements have been more extensive than for other fuels with less reactor-induced degradation. The monitoring results provide an opportunity to identify degradation that may result during dry storage. Increased surveillance for changes in the spent fuel/canister conditions may be provided by adding some monitoring activities that are not now being performed. These are introduced and identified by the word "NEW:" in Table A.1.

Monitoring results include gas sampling that revealed the appearance of hydrogen primarily during the first few years of storage, no acetylene, oxygen depletion or dilution, and the recent appearance of krypton and helium. Video photography showed evidence of localized canister and basket corrosion.

Temperature

The measurement of the temperature of the gas during gas sampling should be continued. This temperature should be useful for updating heat transfer analyses of fuel temperatures.

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[12]Some of these monitoring activities may require the design of a special lid to accommodate conductors for probes.

Chemical Environment

Gas samples have been taken for several years by inserting a tube connected to a vacuum pump through a pipe cap and pulling gas into a collector. Wet and dry bulb temperatures of the gas are taken. Gas analyses include helium, hydrogen, acetylene, oxygen, carbon dioxide, krypton, and xenon. Levels of 0.2 to 4,000 pCi/cm^3 ^{85}Kr have been detected in the gas samples and tend to be associated with helium. Analysis for ^{14}C has not been requested; however, data on ^{14}C releases may be required for eventual repository disposal. The gas composition data have not been evaluated for trends; however, a quick look indicates that hydrogen contents were highest shortly after the fuel was loaded into the dry wells and krypton appeared after several years of storage. There appears to be no correlation between humidity and hydrogen, or krypton.

Hydrogen appeared in gas samples early in the storage period; it may have been generated by corrosion of the steel dry well caisson. However, after a few years the hydrogen content was much lower; when evidence of corrosion pitting of the inner surface of the caisson appeared in videotapes, no hydrogen was detectable. This apparent contradiction may be due to the fact that hydrogen content is likely to be seasonal. Water migrating around and through cracks in the grout causes corrosion of the carbon steel caisson. Depending on the year and time of year, the gas concentrations would be expected to vary. H_2 gas will diffuse readily out of the well; an active source of production would be required to maintain a detectable quantity of hydrogen in the well.

The tendency for the coappearance of helium and krypton in gas samples implies leakage from the fuel through breaches in the aluminum canister hermetic seals. The absence of acetylene suggests that the reaction of moisture with the fuel carbides was not rapid and verifies that the canister atmosphere is nonexplosive.

The interpretation of corrosion phenomena that has taken place in the dry wells with PB Core 1 fuel could be impacted by effects of operations. Further attempts to refine the interpretation should take into account operational records.

Video photographs of some canisters show a crusty appearing deposit. Smear samples of this deposit should be analyzed to determine the composition of this deposit. Fluorine or chlorine released by radiolytic decomposition of the canister seals would be very reactive and could generate reaction products on the canisters. Although there appears to be no correlation between humidity of the gas samples and krypton or hydrogen, hydrogen and water from radiolytic decomposition of the epoxy paint could participate in corrosive reactions. Analyses of these deposits could help to answer questions about possible mechanisms and reactions taking place in the dry well systems.

As indicated in Table A.1, water samples should be drawn from empty drywells containing water to verify an absence of radionuclides. Sampling of the ground water should be continued.

Mechanical Strength

Opening of a dry well requires cutting a weld. The top must be rewelded to close a dry well. The welds are not tested for leak tightness and are not considered to be vacuum tight seals. Consequently the wells were not backfilled with nitrogen. In contrast, dry wells with Light Water Breeder Reactor (LWBR) fuel are backfilled with nitrogen. Because the spent fuel is encapsulated in an aluminum sleeve, it does not appear to be feasible to measure the graphite mechanical strength without penetrating the sleeve, even if a fuel element were removed from a dry well. Measurement of mechanical strength is therefore beyond the scope of monitoring.

Dimensional Stability

It may be feasible to measure the diameters of the aluminum canister, particularly for cases where the canisters are being handled for other reasons or are being recanistered. For these cases, this information may be easy to obtain and may have potential value. It may not be cost effective to extract fueled canisters from dry wells only to perform this measurement because of the cost of cutting the top weld, extracting the basket of 18 canisters, removing them to a test cave, returning them to the dry well, and rewelding the lid to the dry well. It would not be feasible to repeat this measurement on a routine schedule.

Detection of Degradation

Video tapes have documented (from within) dry well contents from different positions, including viewing up under a lifted basket, within dry wells. Discernable differences between canisters containing PB Core 1 fuel are documented on video tapes. Localized areas of the aluminum canister appear to be corroded. The corrosion appears to first form in the vicinity of the top closure where the aluminum canisters had been roll pressed onto the fuel elements. The canister walls were roll pressed into an o-ring in a groove and were found to be leak tight upon vacuum leak testing. All canisters are still considered to be retrievable. A water-like reflective surface was visible from the bottom of at least one dry well.

The Aluminum Company of America (ALCOA) engineering drawings for the canister cap of the regular can (C-601363-NK) and for the canister cap of the salvage can (C-601703-NK) specify aluminum alloy 6061-T6 and both show two grooves in the upper aluminum cap. Apparently two o-rings were placed on these grooves prior to crimping the tube onto the upper aluminum cap. The o-ring material is not specified, but if it were of a copolymer of vinylidine fluoride and hexafluoropropylene type, radiation-induced radiolytic decomposition could release fluorine gas that would be reactive with the aluminum canister components and with moisture to form hydrofluoric acid. The location of the initial onset of corrosion at this closure is consistent with the release of a corrosive agent from within the canister.

Although its exact composition is proprietary, VITON™ is a copolymer of vinylidine fluoride and hexafluoropropylene (Dixon, Rexford, and Rugg 1957). According to information given in a manufacturer's brochure (Dupont product information, VT-515.1, E37758) total combustion of this product in excess oxygen gives 251 mg of HF and 1836 mg of CO_2; thus, VITON-A™ is approximately 24% fluorine by weight. Data by Byrne (1953) show halide release from fluorothene as a function of gamma dose beginning at doses less than 10^8 R. This information does not define a threshold for radiation damage. Some vendor information is stated to rate VITON™ gaskets for approximately 10^5 R total dose without experiencing physical damage. Therefore, in a radiation field of approximately 100 R/h, the vendor rating could be reached in 1000 h or in less than 2 months. Although loss of mechanical integrity of a VITON™ gasket in this field is not anticipated, released fluorine gas would react with the storage system surfaces and could potentially enhance rates of corrosion and stress corrosion cracking.

Some degraded canisters are being placed in stainless steel canisters with a steel lid. The lid is bolted and has a VITON™ o-ring on the outer lip. The o-ring is not crushed and the closure is not considered to provide a vacuum seal. Reactive halogens, such as might be released from the inner aluminum canister, are known to enhance stress corrosion cracking of stainless steel. The potential for the VITON™ to degrade in the radiation field of the canister and release fluorine gas should be evaluated.

A.4.2 PB Core 2 Spent Fuel

A summary assessment of monitoring for the PB Core 2 spent fuel is provided in Table A.2. None of the potential monitoring activities identified in Table A.2 are now being performed. These are introduced and identified by the word "NEW:" in Table A.2.

A.4.2.1 Temperature

Temperature measurement of the external canister should be feasible. Direct measurement of fuel assembly temperatures would require removing the canister lid; this operation may cause some cooling of the fuel elements because of mixing with the cool cave atmosphere.

A.4.2.2 Chemical Environment

Gas samples have not been taken in the PB Core 2 canisters because 1) lid removal for gas sampling would cause mixing with the facility atmosphere, 2) shielding and accessibility is limited, and 3) degradation in the dry atmosphere is not anticipated. Removal of fuel elements would require development of a friction lifting tool. Lifting of fuel elements would not provide a radiological risk, but there would be a risk of damaging the elements if they were dropped. There are three bridge cranes and two master slave manipulators (Power Actuated Remote, PAR, manipulators) available for operations with the Core 2 canisters.

Gas samples from the interior of PB Core 2 canisters would identify the emission rate of gaseous radionuclides such as ^{85}Kr and ^{14}C. This information may be required for repository disposal.

A.4.2.3 Mechanical Strength

The mechanical strength of graphite can be measured with an eddy current probe. The graphite strength is determined using the relationships between eddy-current conductivity and density and between density and compressive strength for oxidized graphite (Morgan, Prince, and Posakony 1982). The availability of this technology at PNL is discussed in Attachment A. It would be difficult to position a mechanical strength probe (eddy current or hardness) on the graphite through the top of a canister. Removal of fuel elements for mechanical strength measurements would require development of a friction lifting tool. Lifting of fuel elements would not provide a radiological risk, but there would be some risk of damaging the elements if they were dropped. There are three bridge cranes and two master slave manipulators (Power Actuated Remote, PAR, manipulators) available for operations with the Core 2 canisters. This test does not appear to be readily feasible.

A.4.2.4 Dimensional Stability

Dimensional measurements would require removal of fuel elements from a canister after a friction lifting tool were developed. Lifting of fuel elements would not provide a radiological risk, but there would be some risk of damaging the elements if they were dropped.

A.4.2.5 Detection of Degradation

Insertion of a probe for video taping of the internals of a canister should be feasible. However, because of the dry atmosphere, visibly detectible degradation is not expected.

A.4.3 FSV Spent Fuel

A summary evaluation of monitoring activities for the FSV Spent Fuel is provided in Table A.3. Current monitoring activities are introduced and identified by the word "CONTINUE:" and should be continued; monitoring activities not currently being performed are introduced and identified by the word "NEW:."

A.4.3.1 Temperature

With the current configuration, it would be very difficult to add instrument cables through the hot cell for temperature measurement or monitoring equipment. Some preliminary concepts are to provide for temperature measurement by lowering a metal-stem or a battery-powered thermometer with the overhead crane to the surface of a canister for attachment with a magnetic latch. The temperature indication could be read through a telescope. If possible, attachment of a thermocouple to a fuel element or the exterior of canister containing the highest burnup FSV fuel elements could provide temperature measurements that would be useful for upgrading heat transfer analyses to identify the peak fuel temperature.

The measurement of the FSV temperature is considered low priority because the initial temperature calculations for the technical bases showed the temperatures to be acceptably low. However, there is some interest in verifying the actual peak fuel temperatures by measurement. If a thermocouple or other device is used for temperature measurement, special attention to interpretation of the measurement device with respect to the radial temperature gradient and the peak fuel temperature is needed.

Some graphite blocks are stored without canisters and therefore provide the potential for visual examination. The edges of these appear to be sharp[13]. However, the edges of undisturbed friable material may still appear to be sharp. The material may become friable if internal oxidation under the high temperature conditions associated with irradiation had been significant (Morgan and Thomas, 1982). Visual appearance does not provide assurance that the high strength of the graphite is retained.

Blocks of FSV fuel could be removed from the canisters to an adjacent handling cave and tested, using manipulators, for properties described in Table A.3. Measurements may include fuel temperature;[14] however, the temperature would likely be reduced by the effect of removing the fuel from the storage population and by opening the canister.

A.4.3.2 Chemical Environment

Gas samples could be collected from a canister after it was opened in the adjacent cave and analyzed for ^{85}Kr, ^{14}C, ^{129}I, ^{3}H, He, CO, CO_2, H_2, N_2, and O_2. However, mixing with the cell air after the lid was removed may limit the ability to measure absolute concentrations of the gases representative of the canister before the lid was disturbed. As indicated in Table A.3, special lid designs or methods may be required to obtain suitable gas samples.

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[13]Observations made by WINCO Staff as described to S. C. Marschman (PNL) during 1993.

[14]Spacial temperature gradients are not expected to be important because of the high value of thermal conductivity of graphite (5 to 10 times more conductive than stainless steel). If temperature measurement is not feasible, temperature estimates from heat transfer calculations may be substituted.

A.4.3.3 Mechanical Strength

Manipulators in a handling cave adjacent to the storage cell could be used to remove blocks of FSV fuel from the canisters for testing of mechanical strength with an eddy current probe or a hardness probe. The graphite strength is determined using the relationships between eddy-current conductivity and density and between density and compressive strength for oxidized graphite (Morgan, Prince, and Posakony 1982). The availability of this technology at PNL is discussed in Attachment A. There are some FSV graphite blocks in the storage cell that are not canistered.

A.4.3.4 Dimensional Stability

Manipulators in a handling cave adjacent to the storage cell can be used to remove blocks of FSV fuel from the canisters for testing with air gauges or other dimensional gauges for dimensional changes.

A.4.3.5 Detection of Degradation

Mechanical measurements can be used to detect degradation; graphite has the ability to degrade to a state of being extremely friable without changes in visual appearance. Visual examination of graphite may be misleading as an indicator of residual strength (Morgan and Thomas, 1982). Small quantities of H_2, N_2, CO, and CO_2 in the coolant affect graphite oxidation of impurities and strength changes during irradiation. During storage, oxidation is expected to be shallow because of the lower temperatures and lower diffusion rates. Ionization effects may be more limited to the surfaces; however, the surface area includes surfaces of flow channels, etc.

APPENDIX A

TABLES

Table A.1. Monitoring PB Core-I Spent Fuel and the CPP 749 Dry Wells

Property	Monitoring Activity	Priority	Frequency	Driver
Temperature	CONTINUE: Measure gas temperatures	Medium: Heat transfer analyses indicate fuel oxidation rates are very low below the temperature limit of 1100°F; calculate temperature-dependent corrosion rates of canisters and oxidation rates of fuel	Annual for five years; reevaluate biannually if property changes greatly within two years	canister corrosion and fuel degradation increases with increasing temperature; refine temperature calculations with gas temperature measurement; calculate temperature dependent canister corrosion and fuel oxidation rates
Chemical Environment	CONTINUE: Collect gas samples; analyze for He, ^{85}Kr, H_2, O_2, N_2, acetylene, dew point NEW: Add analyses for ^{14}C, ^{129}I, $^3H_2$, and CH_4	High: evaluate oxygen depletion, hydrogen and hydrocarbon generation, and release rates	Annual for five years; reevaluate biannually if property changes greatly within two years	measure release; degradation; verify nonexplosive/nonflammable environment; monitor corrosion reactants and products
Water samples	NEW: Collect water samples from empty wells; analyze for Cs, I, U, Th, halides, pH CONTINUE: Monitor ground water	High: verify low radionuclide release High: radionuclide release	each time water is found annual	radionuclide release; corrosive reactants radionuclide release
Dimensional Stability	NEW: Measure diameters of aluminum canisters before recanistering or during other periods of accessibility	High: evaluate corrosion and swelling from dimensional changes	each time canister is accessible	canister corrosion and fuel degradation effects

Table A.1. (Continued)

Canister Thickness	NEW: Measure canister wall thickness with eddy current or ultrasonics	High: evaluate canister wastage from corrosion	each time canister is accessible	canister corrosion
Smear	NEW: Take canister smear samples and analyze for radionuclides, halogens, etc. during operations to recanister	High: determine corrosion products and contaminants on canister	two years	radionuclide release; canister corrosion
	NEW: Take dry well smear samples; evaluate radionuclides, composition and structure of epoxy paint	High: evaluate source of hydrogen gas; evaluate radionuclide contamination	two years	explosive and corrosive environment; radionuclide release
Visual	CONTINUE: Document appearance with remote camera	High: continue photographing condition of fuel canisters and dry wells	two years	dry well condition

Table A.2. Monitoring PB Core-2 Spent Fuel in ICPP 603 Vault

Property	Monitoring Activity	Priority	Frequency	Driver
Temperature	NEW: Attach a thermocouple or temperature indicator to a canister containing highest burnup fuel; measure temperature	Low: heat transfer analyses indicate a peak fuel temperature will be below the temperature limit of 1100°F	once	degradation increases with increasing fuel temperature; refine peak fuel storage temperature calculation with measured fuel temperature; calculate temperature-dependent oxidation rate
Chemical Environment	NEW: Collect gas samples from within the canister; analyze for ^{85}Kr, ^{14}C, ^{129}I, $^{3}H_2$, H_2, O_2, N_2, He, CH_4, acetylene	Low: expect low fuel oxidation and release rates	once	radionuclide release and degradation
Dimensional Stability	NEW: Measure canister dimensions with profilometry	Low: expect negligible dimensional changes from degradation	once	degradation, handling, packaging, shipping
Canister Thickness	NEW: Measure canister wall thickness with eddy current or ultrasonics	Low: evaluate canister wastage from corrosion	once	canister corrosion
Visual	NEW: Document canister appearance with camera	Low: appearance not expected to change	once	degradation, corrosion
Smearable Contamination	NEW: Perform standard wipe tests and analyses of canisters	Low: verify low smearable contamination levels; identify smearable contaminants	once	baseline contamination data

Table A.3. Monitoring FSV Spent Fuel in the ICPP 603 Vault

Property	Monitoring Activity	Priority	Frequency	Driver
Temperature	NEW: Attach a thermocouple or temperature indicator onto a fuel element or the exterior of a canister containing highest burnup FSV elements; measure temperature.	Low: heat transfer analyses indicate a peak fuel temperature <365°F; oxidation rates are very low below the temperature limit of 1100°F	once	degradation increases with increasing fuel temperature; refine peak fuel storage temperature calculation with measured fuel element or canister temperature; calculate temperature-dependent oxidation rate
Chemical Environment	CONTINUE: Continuously monitor exhaust air for radionuclides	High: verify low radionuclide release from ICPP 603 Vault	continuous	measure radionuclide release detect fuel degradation
	NEW: Lift or tilt canister lid, or design lid with sampling port, or laser drill canister for sampling; collect gas samples; analyze for ^{85}Kr, ^{14}C, ^{129}I, $^{3}H_2$, H_2, O_2, N_2, He, CH_4, acetylene	Medium: verify low oxygen depletion and release rates; positive indications from air diluted gas samples would be significant	five years	
Mechanical Strength	NEW: Move a canister with FSV fuel elements to a handling area; remove a fuel element; measure the strength with an NDE eddy current probe	Medium: verify no strength reduction from oxidation	once	post storage retrieval, friability, handling, and shipping develop baseline hardness data
	NEW: Measure hardness of FSV fuel elements	Medium: indicates strength and friability	once	
Dimensional Stability	NEW: Move a FSV canister to a handling area; remove a fuel element; measure the canister and fuel element dimensions with a profilometer	Low: verify negligible dimensional changes from oxidation during storage	once	post storage retrieval, friability, handling, and shipping
Visual	NEW: Document the appearance of FSV canisters and fuel elements with a remote camera	Low: develop a baseline database on canister and fuel element appearance; performed in conjunction with other operations	once	baseline appearance is needed for future visual examinations
Smearable Contamination	NEW: Perform standard swipe tests and analyses of FSV canisters; perform same on FSV fuel elements	Low: verify low smearable contamination levels; identify smearable contaminants	once	develop baseline smearability of radionuclides from FSV canisters and fuel elements

APPENDIX B

HOT CELL REQUIREMENTS

The hot cell requirements at the PNL were reviewed to determine how the PB and FSV spent fuels might be shipped from their current locations, received at PNL, disassembled, sectioned, and distributed for a variety of examinations. The general flow of activities anticipated for examining these fuels is shown in Figure B.1.

There are two major routes by which the PB and FSV fuels may arrive at PNL. This report places emphasis on shipment of the fuel from INEL storage locations and the FSV reactor site directly to the 324 Building at Hanford. Alternative options are shown as yellow lines in Figure B.1 and may involve shipments from the INEL storage locations and the Ft. St. Vrain site to hot cells at INEL, such as those at the Argonne National Laboratory-West (ANL-W), for disassembly and sectioning prior to shipment of fuel components to the 324 or 327 Buildings at Hanford. There is also a possibility that some useful capabilities exist at ORNL which may ultimately be needed for characterization activities.

Once the materials have arrived at PNL, either as intact fuel elements or as sectioned components, they will probably be received at the 324 Building where a variety of operations will be conducted as indicated in Figure B.1. A possible variation would include direct shipment to the 327 Building if an acceptable cask can be used so that only waste disposal operations will occur in the 324 Building. In either event, fuel element samples will be distributed to the 327 Building for further preparation or examination. Some samples will then be sent to the 325 Building for radiochemical analyses, shielded electron probe microanalyses (EPMA), scanning electron microscopy (SEM), or possibly to the 326 Building for transmission electron microscopy (TEM).

The number of shipments will depend on the availability and kind of cask(s) used. Based on evaluations thus far, the FSV-1 cask is the most likely choice for shipping both FSV and PB spent fuels because this cask is the only one currently licensed for shipping highly enriched (U,Th)C fuel particles in a graphite matrix. Public Service Company of Colorado (PSC) is presently building new casks and plans to sell the existing FSV-1 casks in 1995 or later to any interested parties. PSC holds the license for two FSV-1 casks. The two casks and three licensed liners are at the reactor site: each cask has an available internal cavity 0.45 m in diameter by 4.76 m long (17.7" x 187.6") and usually holds six 136 kg (300 lb) FSV blocks. No currently licensed cask has been identified that is specific to the PB fuel, although the old cask is believed to still be at INEL. An amendment may be required for the FSV-1 cask license to specifically cover shipment of the PB spent fuel elements; staff at PSC have indicated that variances to the shipping cask license have not been lengthy processes in the past, but it would probably take six months to conduct any analyses needed to include the PB design and an additional six months for review by the Nuclear Regulatory Agency (NRC). Apparently, the FSV-1 cask will not be available for use until 1994 because of current leasing arrangements; daily leasing charges are $1,000 per day or more.

The assumed number and types of fuel elements that will be shipped are listed in Table B.1. These assumptions facilitate this review as well as the estimation of waste volumes and disposal options. The actual types of elements may change, but the number of elements and the impact on shipping requirements should be reasonably close to what might occur. This schedule was also assumed to minimize the activity in the hot cells from the potentially higher burnup in the EOL fuel at the FSV reactor site. While the PB fuel elements will have about 1500 Ci per element, and the FSV elements

stored at INEL will have about 5200 Ci maximum[15] in 1994, the EOL elements at the reactor site could each have about 22,000 Ci in 1996 or 19,100 in 1997. Further details on the possible work within each building, limitations, and preparation requirements are discussed below.

B.1.1 FACILITY REQUIREMENTS

There are three facilities at Hanford that would be used to examine the PB and FSV fuels. Fuel elements can be received in the 324 Building and sectioned to sizes more suitable for examinations in other facilities. Gamma scanning may also be conducted in the Shielded Material Facility (SMF) or in D Cell of the 324 Building, although upgrades would be required for gamma scanning in D Cell. The metallography/ceramography and fuel sample preparation for radiochemistry, SEM, EPMA, TEM, and/or other analyses will be conducted in the Postirradiation Testing Laboratory (PTL) in the 327 Building. Radiochemistry, SEM, EPMA, and analyses for ^{14}C and ^{3}H will be conducted in the 325 Building in a variety of hot cells designed for such work. It is also possible that TEM will be conducted on very small samples in the 326 Building. All of these facilities are in the 300 Area within a few minutes walking distance of each other.

While the hot cell requirements at INEL are not part of this review, it will be important to ensure that the decisions on casks and container requirements do not impair operations at INEL, such as loading and unloading. Details on the specific facilities at PNL are given below.

B1.1.1 324 Building

The activities planned or possible in the 324 Building include the following:

- receipt of shipping casks, unloading and loading operations
- retrieval and initial sectioning of PB and FSV elements
- potential gamma scanning of fuel components
- shipment of samples to the 327 Building for further preparation and examination
- packaging of spent fuel wastes for shipment to INEL.

The 324 Building is located near the East side of the 300 Area of the Hanford Site. Highly radioactive material in shipping casks weighing up to 30 tons can be received and unloaded in this facility. Casks are brought into the building by way of the truck loading dock and the Cask Handling Area (CHA). Options exist for loading and unloading casks either horizontally or vertically. Dimensions for the CHA and other Radiochemical Engineering Cells in the 324 Building are shown in Figure B.2. The air lock

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[15]The FSV fuel elements stored at INEL had a maximum burnup of <47,000 megawatt days per metric ton of initial heavy metal (Mwd/MTIHM). The elements had 0.011254 MTIHM on average and a decay time of 10 years or more. The estimated activity is based on calculations for FSV fuel with 100,000 Mwd/MTIHM. The activities in Table 1 are not converted to ^{90}Sr equivalents.

has a total height of 10.4 m (34.12 ft); the effective working space is 9.9 m (32.5 ft) after accounting for the crane and cask rail locations.

C Cell is a 3.6 m long x 6 m wide x 4.6 m high room with concrete shielding rated for 10^6 R/h and contains two viewing windows and manipulators for remote operation. This cell is equipped with a periscope for detailed examination of materials.

D Cell is above C Cell and is slightly larger; both cells have doors opening to the air lock. D Cell has been used extensively to characterize full length commercial spent fuel. Examinations conducted in D Cell have included gamma scanning, laser puncturing and gas sampling, and sectioning.

The general scenario envisioned for unloading the FSV-1 cask consists of moving the shipping cask with the spent fuel into the air lock, pulling out the spent fuel container inside the shipping cask, unloading the spent fuel container, and moving the PB fuel element and its storage container into C Cell. At present it is envisioned that the FSV blocks would be removed from the spent fuel container and loaded individually onto an existing tray for transfer into C Cell. Operations are expected to be done vertically, but could be performed horizontally with some modifications to the existing transfer port in the air lock door.

Procedures and equipment are in place to move samples of the fuel element components from C or D Cells to the Shielding Material Facility (SMF) for gamma scanning within the 324 Building. The SMF is an alpha-free complex of three concrete cells in the shape of an L surrounded by an operating gallery. The SMF gallery is adjacent to the CHA that adjoins the airlock to C Cell described above. Testing and services take place in the South and East Cells. Materials in large shielded containers are introduced through the Airlock Cell of the SMF and positioned by a dolly on tracks to enter the South or East Cells. Three-ton bridge cranes move material in these cells. The East Cell, where the gamma scanning has been conducted extensively on fuel rods from the Fast Flux Test Facility and commercial reactors, waste tank cores, and irradiated materials, has a 4.8 m x 7 m (16 ft x 23 ft) floor space and has shielding rated for 10^6 R/h.

Preparations necessary to receive and unload the PB and FSV fuel elements in the 324 Building will include preparing, loading and unloading procedures for the FSV cask. Currently, the 324 Building routinely handles casks such as the T2, T3, and NLI ½. It is expected that the NLI ½ procedures will be adapted using the unloading procedures provided for the FSV cask. Some designing and fabrication of handling equipment will be required. Horizontal unloading of the cask would require enlargement of the transfer port in the air lock door, although it is expected that vertical unloading will be conducted. A saw exists in the C Cell that could be used to cut the PB elements in their containers, but a saw would have to be installed to cut the FSV blocks. Training would be required specific to both operations, but sawing has been a routine activity in several of the hot cells. In as much as all of the cells in the 324 Building have air atmospheres, some provisions may be required to cut and store samples under inert atmospheres to preclude degradation, particularly for the FSV blocks that have not been exposed to air already. A similar approach was used to inert spent fuel samples for irradiated UO_2 fuel examined in support of the U.S. repository program; existing procedures for this inerting process could be adapted for the PB and FSV fuels.

There are some special requirements dictated by the current draft revision to the SAR for the 324 Building, PNL-7989. First, all isotopic inventories need to be converted into ^{90}Sr equivalents. The

remaining requirements relate to the quantity and type of dispersable radioactive materials that can be in the cells to preclude potential release to the environment during a postulated seismic event. Certain assumptions are made for this event, including the number of fuel rods/assemblies that fail and what materials are released. At present, the SAR limits the highly dispersables in the 324 Building to 23,000 Ci in ^{90}Sr equivalents, excluding the current inventory in B Cell. There is a limit of 1.5 million Ci of mobile dispersables if contained in unrated storage containers; this limit may be revised significantly upwards. Dispersables stored in rated containers, such as Department of Transportation (DOT) approved containers (i.e., they meet seismic requirements), do not count against the dispersable inventory. Releases of the following are assumed during this postulated seismic event:

- 100% of tritium from a worst case fuel rod
- 10% of volatiles (halogens, which are F, Cl, Br, I, At) are released, of which half (i.e., 5%) become surface contamination
- all noble gases (Xe/Kr, He, Ne, Ar) are released.

Fuel in powder form that is loose, such as sawing fines, in the cell is considered highly dispersable. Fuel in rods is not considered dispersable, but must meet current 1.5 million Ci limit for the facility. A safety evaluation document will most likely have to be written for this work. A controlled inventory of several PB and FSV elements would not pose a major issue due to dispersability if cuttings are collected and contained.

B.1.1.2 325 Building

The 325 Building is located in the 300 Area intermediate between the 324 and 327 Buildings. The initial receipt of fuel samples in the 325 Building would occur in Cell 6 of the Shielded Analytical Laboratory (SAL) located at the West side of the building. Cell 6 is expected to be used for dissolution of the fuel samples crushed in the 327 Building. Aliquots of the dissolved materials would be sent from the SAL to a variety of existing small labs in the 325 Building for specific analyses. All of these labs have radiation and fuel limits significantly lower than for the 324 or 327 Buildings. There is a 240 g limit on fissile materials for all six cells in the SAL. The probable analyses and the status of the procedures, staff training, and development requirements are summarized in Table B.2.

The 325 Building facilities have ongoing activities in support of other programs that will benefit the proposed analyses of graphite fuels. Trained staff in the 325 Building are operating an Environmental Protection Agency (EPA) approved laboratory for chemical analyses. These analyses have involved use of some of the latest analytical equipment, including a hot inductively coupled plasma (ICP) mass spectrometer (MS) for analyzing low concentrations of radionuclides. Recently, upgrades were made to the procedures and equipment to analyze ^{14}C in nonfuel bearing components of commercial spent fuel in support of computer code verification, which should specifically benefit examination of the graphite fuels.

There is some development required to conduct radiochemistry on the graphite fuels. Probably the most significant radiochemistry development required is a procedure for dissolution of the fuel. Typically, 10 g samples of oxide or metal fuel are dissolved in 250 mL of acid; small aliquots (1 mL) are usually taken and sent to the various labs for analyses. Dissolution of oxide and plutonium bearing fuels (e.g.,

liquid metal fast breeder fuel) is conducted under American National Standards Institute (ANSI)/American Society for Testing and Materials (ASTM) Standard Test Procedure E321-79. However, there is no current national standard for burnup analyses of graphite fuels. W. Matsumoto of PNL, who has been on the national standard's committee and was a principle contributor to the standard, has indicated that there was a procedure used by General Atomic (GA) and that it involves dissolution of the fuel with perchloric acid, a potentially explosive acid under certain circumstances. While GA used to do this work, they have not for years according to Matsumoto. Development of this procedure is feasible.

The burnup analyses will require some adaptation because of the inclusion of ^{233}U in the (U,TH)C fuel. This will require changing the spike from the current E321-79 procedure.

Waste disposal is of particular concern to operations in the 325 Building because of limits on fissile material inventories and potential impacts on other programs for the Hanford waste tanks. Thus, it will be required that all wastes be removed from the building and returned to the 327 Building for final disposal. Because only milligrams of fuel will be in samples taken to the labs, that material should be disposable as TRU waste. The remaining solution(s) in the SAL will be neutralized, turned into a salt cake, crushed, packaged and returned to the 327 Building.

Lab 317 in the 325 building would probably have to be refurbished to conduct the burnup analyses. These analyses have not been conducted for several years in this 10 x 10 ft lab. The major equipment for this analysis is a mass spectrometer which is in place.

B.1.1.3 327 Building

The activities planned for the 327 Building include the following:

- receipt of spent fuel components, unloading and loading operations
- preparation of samples (spines, sleeves, fuel compacts, graphite matrix) for metallography and ceramography
- dimensional analyses of components
- preparations of samples for radiochemistry, SEM, and EPMA to be conducted in the 325 Building
- preparation of samples for TEM to be conducted in the 326 Building
- waste form testing and evaluation, such as oxidation and leach testing
- repackaging for return to the 324 Building for final disposition.

The 327 Building is located in the center of the 300 Area of the Hanford Site. Highly radioactive material in shipping casks weighing up to 20 tons can be received and unloaded in this facility. The largest cask used in this facility is the T2 cask, with internal dimensions of 0.12 m in diameter by 2.23 m long (4.65" x 88"). Casks must be loaded and unloaded horizontally because there is no air lock in this

facility. Typically, casks such as the PRTR graphite cask and the BCL-3 cask are used to move fuel samples between the buildings.

The PTL facility (327 Building) consists of a series of hot cells located in the main operating area (Canyon) as shown in Figure B.3. Ten high density iron or steel shielded cells are located in this area for examination of irradiated materials. Fissile material is currently limited to 194 g per cell with a total of 600,000 Ci of mixed fission products per cell. One of the ten cells is the Special Environmental Radiometallurgy Facility (SERF), which has an inert nitrogen gas atmosphere. Either pool or dry storage systems are available in the 327 Building. Containers transferred into the 327 Building hot cells have a diametral limit of about 0.17 m (7 in.) with lengths up to about 2.4 m (8 ft). Larger containers could be accommodated with alterations.

The 327 Building is in good working order and is presently providing support to several programs. Several of the cells are available for additional work. It is anticipated that fuel components will be shipped to the 327 Building from the 324 Building and initially loaded into F or the SERF Cells. Selected examinations or preparations will be conducted in one of several cells, such as SERF, C or E. All but the SERF cell presently have air atmospheres, and each of Cells A-F and the SERF cell have water supply lines to provide small amounts of process water that may be required. Each cell is exhausted through a filtered and monitored exhaust system. Brief descriptions of selected cells are provided below. F Cell has interior dimensions of 2.4 m long x 1.5 m wide x 2.5 m high (8 ft x 5 ft x 8.17 ft) with 0.46 m (18 in.) mechanite shielding. F Cell is equipped with three viewing windows, three manipulators (one at each window), a wall mounted milling machine and lathe, and sliding blocks for access. Depending on the size of the fuel components, a cut off saw could be moved to this location.

C and E Cells have interior dimensions of 1.8 m long x 1.3 m wide x 1.3 m high (6 ft x 4.3 ft x 4.3 ft) with 0.26 m (10.5 in.) of mechanite shielding. These cells each have one viewing window with other port-size windows, two manipulators (one at each window), a metallograph blister, and several access ports.

The SERF cell is specially equipped to provide an inert atmosphere and is continuously monitored for water vapor and oxygen. This cell has interior dimensions of 3.7 m long x 1.8 m wide x 2.4 m high (12 ft x 6 ft x 8 ft) with 0.46 m (18 in.) steel shielding. The cell has five viewing windows, port-size windows, a large air lock for access, 4 manipulators (one at each window), a metallograph blister, and a small glove box on the South end of the cell. This cell would provide an ideal arrangement for cutting samples under an inert atmosphere with an ability to conduct photomicrography and sample viewing at high magnification. The SERF Storage Cell in the basement below the SERF Cell has a total of 460 locations for storage in three storage racks that accommodate sample cans 0.064 m diameter (2.5 in.) by 0.1 m (4 in.).

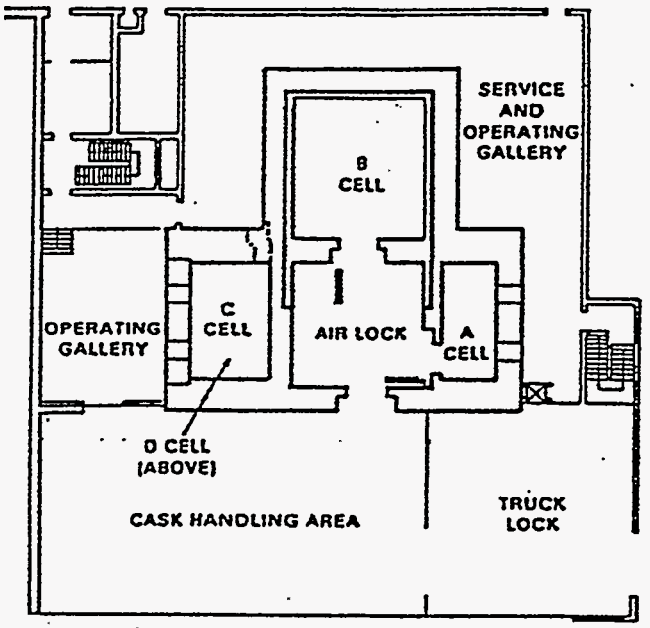
The PTL is operating under the current SAR approved in 1987. It is anticipated that most of the planned activities will fall within current work practices. Only a few requirements need to be met to support preparation and examination of the PB and FSV graphite fuels. Metallographs, grinders, and polishers that will be used will need to be examined and reconditioned by outside vendors as required. Any special techniques desired, such as particle sizing, will have to be evaluated and established. Preparation of samples for radiochemistry will probably be prepared in a manner similar to some current activities,

but will have to be reviewed and defined to meet radiochemical requirements. Repackaging and return of wastes will have to be evaluated to determine specific requirements for these fuels.

APPENDIX B

FIGURES AND TABLES

Figure B.1 Draft Hot Cell/Characterization Activities



CELL	LENGTH, m	WIDTH, m	HEIGHT, m	SHIELD CONCRETE TYPE AND THICKNESS, m
AIRLOCK	6.7	6.5	10.4	NORMAL, 1.4
A	2.8	6.4	10.4	NORMAL, 1.4
B	6.7	7.6	9 3	DENSE, 1.2(a)
C	3.6	6.0	4.6	DENSE, 1.2(a)
D	4.0	6.4	5.2	DENSE, 1.2

Figure B.2 Arrangement of Radiochemical Engineering Cells

Figure B.3 Arrangement of Hot Cells in the 327 Postirradiation Testing Laboratory

Postirradiation Testing Laboratory

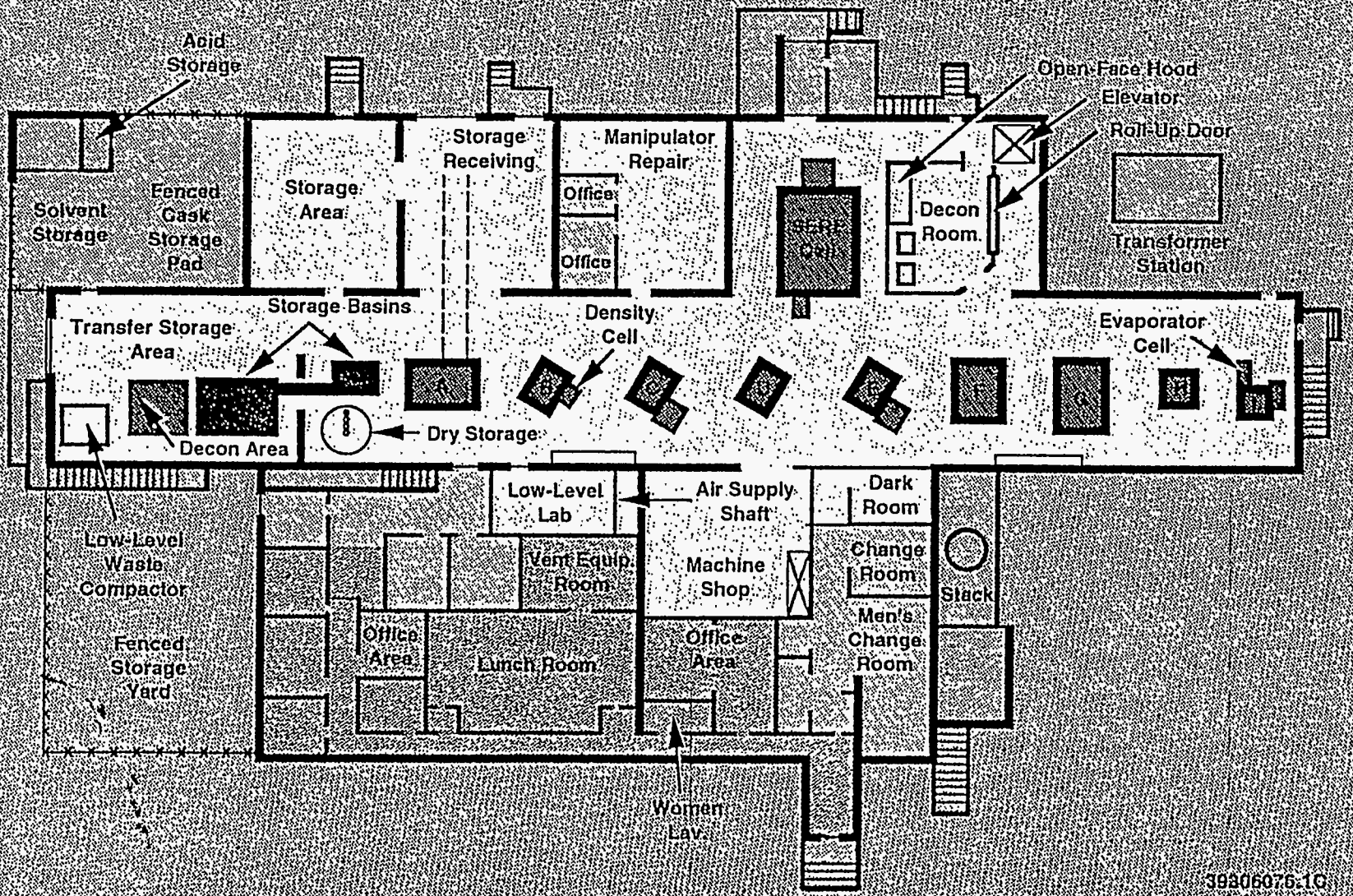


Table B.1. Potential Fuel Elements Shipped for Examination

Fuel Source	Number of Shipments	Estimated Curies/Element	Fuel Element Description
Ft. St. Vrain at INEL	1	5200	1 or 2 discharged 1984 or earlier
Ft. St. Vrain, Colorado	1	19100	1 or 2 EOL discharge
PB	1	1500	1 Core 1, Type II Non Failed
PB	2	1500	1 Core 1, Type I failed and 1 Core 2, Type II non-failed
PB	1	1500	1 Core 1, Type III Non-failed

Table B.2. Radionuclides to be Evaluated for Graphite Fuel

Radionuclide	Comments
	Actinide Analyses
Fuel Burnup	Burnup not done for awhile; no existing procedure for HTGR fuel. Uncertainty for UO_2: ±2.5% Atom% Burnup, ±1.6% Pu, ± 1.6% U.
^{228}Th ^{230}Th ^{232}Th	∝ spectroscopy
^{233}U ^{234}U ^{235}U ^{236}U ^{238}U	Total Uranium by laser fluorimetry, mass spectroscopy for isotopes.
^{237}Np	∝ spectroscopy. Uncertainty for UO_2: ±1.9%.
^{238}Pu ^{239}Pu ^{240}Pu ^{241}Pu ^{242}Pu	Total Pu, mass spectroscopy for isotopes (could use ∝ spectroscopy for 238Pu and 239Pu + 240Pu).
^{241}Am ^{242}Am ^{243}Am	∝ spectroscopy for 241Am and 242Am + 243Am. Uncertainty for UO_2: ±4.9% 241Am, ±4.1% 242Am + 243Am.
^{243}Cm+^{244}Cm	Separated by cation and anion exchange and counted by ∝ spectroscopy.
	Fission Product Analyses
^{3}H	Liquid scintillation.
^{14}C	^{14}C evolved as CO_2 is measured by liquid scintillation counting. Conducted for fuel, metals, and gases. Uncertainty: for UO_2 fuel and cladding: ±5.6%.
^{36}Cl	36Cl in graphite is analyzed in other graphite materials, use liquid scintillation.
^{59}Ni ^{63}Ni	ICP MS, get both at same time.
^{60}Co	Gamma Energy Analysis (GEA), potentially all gamma emitters available from one sample.
^{79}Se	Series of extractions followed by liquid scintillation counting. Uncertainty for UO_2: ±4.9%.
^{90}Sr ^{90}Y	Selective elution and beta counting. Uncertainty for UO_2: ±5.7%.
^{94}Nb	GEA, though detection can be difficult.

Table B.2 Continued

Radionuclide	Comments
	Actinide Analyses
^{99}Tc	Separation by cation exchange and beta counting. Uncertainty for UO_2: ±3.5%
^{106}Ru-^{106}Rh	GEA, get both 106Ru + 106Rh.
^{107}Pd	Beta emitter, may get by ICP MS or scintillation counting.
^{126}Sn	Separated by cation and anion exchange and counted by GeLi spectrometer. Uncertainty for UO_2: ±10.2%.
^{125}Sb	GEA
^{129}I	Improvement in technique desirable; development and training reqd. Current procedure separates iodine as AgI and counts in GeLi detector. Uncertainty for UO_2: ±2.2%.
^{134}Cs ^{135}Cs ^{137}Cs	135Cs by chromatographic elution and mass spectroscopy, others by GEA. Uncertainty for UO_2: ±14% 135Cs, ±3.5% 137Cs.
^{144}Ce	GEA
^{148}Nd	Measured in oxide and metal fuels during burnup analyses, see above.
^{151}Sm	Possibly by ICP MS.
^{154}Eu	GEA

APPENDIX C

REGULATORY REQUIREMENTS FOR DISPOSAL

The purpose of this review is to provide brief summaries of the major current regulations applicable to the treatment, storage and disposal of PB Unit 1 and FSV fuel for each of the three disposal options. Briefly, the three options are: 1) dispose of intact fuel elements; 2) separate fuel from other components; and 3) burn fuel elements.

C.1 INTRODUCTION

The disposal options and the subsets associated with them are described in Chapter 3. The scope of this review is limited to the regulations that directly impact the options for handling, transporting, and processing the spent fuel elements and designing storage and disposal containers for the fuel elements and the resulting waste forms. The required regulations for these specific tasks, over and above routine laboratory regulatory requirements, will be used to help determine the disposal path involving least cost and least total radiation and toxic material exposure to the public and the environment for both the short and the long terms.

Generic topics (such as radiation safety for personnel, personnel training, record-keeping, reporting, and other administrative requirements), that are applicable at any facility that contains radioactive materials, are not discussed here. Some of the disposal site licensing requirements are included if there is potential for impact on the type of storage container required.

The regulations are those required by Public Law, as interpreted by the NRC, the EPA, the DOE and other agencies. The major governing laws are:

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

- The *National Environmental Policy Act of 1969* (NEPA, Public Law 91-190 and amendments).

- The *Resource Conservation and Recovery Act of 1976* (RCRA, Public Law 94-580 and amendments), regulating waste, which must meet two criteria: 1) it must be solid, and 2) it must exhibit certain hazardous characteristics. (Interpreted in EPA's 40 CFR 261.)

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

- The *Nuclear Waste Policy Act of 1982* (NWPA, Public Law 97-425 and amendments).

- The *Comprehensive Environmental Response, Compensation and Liability Act* (CERCLA, Public Law 96-510) of 1980, and the *Superfund Amendments and Reauthorization Act of 1986* (SARA) and amendments.

The primary interpretation of the governing legislation is provided in the Code of Federal Regulations (CFR) by the Nuclear Regulatory Commission (NRC) and the Environmental Protection Agency (EPA). DOE Orders are supplementary to the CFR, generally written for specific application to DOE tasks.

Table C.1 lists the regulations that are reviewed herein. Section C.2 contains a summary table that lists the sections of the regulations that are pertinent to the various disposal options. Pertinent parts of these regulations are summarized in Section C.3. Fundamental terms and their definitions, as given in the various regulations, are listed in Section C.4.

Hazardous waste regulations 40 CFR 261, 264, 265 and 268 were written for non-radioactive hazardous materials or wastes. However, they do apply to any listed hazardous materials when the hazardous materials are mixed with the radioactive waste; in that case, the regulations on "radioactive mixed waste" apply also. The possible presence of barium, silver and processing chemicals in the PB and FSV spent fuel and spent fuel processed waste will mean that these regulations are pertinent to the disposal of these graphite fuel elements.

Major lists of hazardous waste or materials are found in *40 CFR 61* (Air pollutants), *40 CFR 261 Subpart B* (wastes from manufacturing and other sources), and *49 CFR 172* (materials that must be marked for shipping).

This review is not exhaustive, in that there may be special-case exemptions and modifying clauses that have escaped notice or have been judged not to be pertinent. Furthermore, the summaries and excerpts given in what follows should be used as general guidelines only; many qualifying statements have been omitted in the interests of brevity. Detailed evaluation of applicability and interpretation of the fine points of these required regulations can be made more effectively after decisions have been reached on the specific disposal procedures to be used.

C.2 PERTINENT REGULATIONS

Table C.2 lists the sections of pertinent regulations which could impact the choice of disposal option. The listing is divided into three categories:

1) generally applicable to all three options
2) especially applicable to Option I, Disposal of Intact Assemblies
3) applicable if Option II (Disassembly) or III (Burn) is chosen (i.e., applicable for the processing that may be required for some of the sub-options in options II and III).

There appears to be no appreciable difference between PB and FSV with regard to the applicability of the regulations, given the uncertainty as to disposal-preparation processes to be used. Therefore, there has been no attempt to differentiate between them in Table C.2.

Within each category, like topics are grouped together where possible; e.g., the several documents treating packaging and transportation are presented in parallel. Only the title and number of each regulation subsection is given in the table. The pertinent excerpt from the text or, where there is too much detail to include, an indication that there is detail that should be read, is then found in Section 3.0, where the regulations are arranged in the same order as they appear in Table C.1.

C.3 PERTINENT EXCERPTS FROM THE REGULATIONS

C.3.1 10 CFR Part 20 - Standards for Protection Against Radiation

Of specific interest to the PB and FSV spent fuel disposal option determination is Subpart K-Waste Disposal, Sections .2001 to .2007.

- **10 CFR 20.2001 (a) General Requirements**
  A licensee shall dispose of licensed material only by:
  1) transfer to an authorized recipient,
  2) decay in storage,
  3) release in effluents within the limits in 20.1301, or
  4) disposal by incineration. (20.2001)

- **10 CFR 20.2001 (b) General Requirements**
  A licensee must be specifically licensed to receive waste containing licensed material for:
  1) treatment prior to disposal,
  2) treatment or disposal by incineration,
  3) decay in storage,
  4) disposal in a land disposal facility licensed under part 61 of CFR Chapter 10, or
  5) disposal at a geologic repository under part 60 of CFR Chapter 10.

- **10 CFR 20.2002 Method for Obtaining Approval for Proposed Disposal Procedures**
  "...for approval...to dispose of licensed material generated in the licensee's activities. Each application shall include: (a) A description of the waste containing licensed material to be disposed of, including the physical and chemical properties important to risk evaluation, and the proposed manner and conditions of waste disposal;..."

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

C.3.2 10 CFR Part 60 - Disposal of High-level Radioactive Wastes in Geologic Repositories

This regulation governs the siting, licensing, operating and closing of a geologic repository for high level radioactive wastes. It stipulates the characteristics of the engineered barrier system, which includes the waste packages, needed to protect the environment and humans in the vicinity of the repository from radiation exposure and releases, for the containment period (the first several hundred years in the repository).

- "High-level radioactive waste" or "HLW" means: 1) irradiated reactor fuel; 2) wastes from solvent extraction processes; and 3) solids into which liquid wastes have been converted. (60.2)

10 CFR 60 may apply to the disposition in a geologic repository of PB and FSV spent fuel for the intact-disposal option or to the residual fuel after separation from the other components, or to the ash after burning the total element, if the overall activity is greater than a level not specified or defined in this document. Indications are that the intact fuel elements are too radioactive for classification as LLW.

Specific pertinent excerpts follow:

- **10 CFR 60.43 License Specification**
  (b) "License conditions shall include items in the following categories:"
  (1) "Restrictions as to the physical and chemical form and radioisotopic content of radioactive waste."

- **10 CFR 60.101(a)(2) (Section E - Technical Criteria)**
  "Proof of future performance of engineered barrier systems over time periods of many hundreds...of years is not to be had in the ordinary sense of the word... What is required is reasonable assurance..."

- **10 CFR 60.102(e)(1) Isolation of Waste**
  During the first several hundred years following the closure of the repository (the "containment period"), emphasis is on the ability of the waste packages to contain the waste. (Thereafter emphasis is on the engineered barrier system of the repository itself to achieve isolation of the waste.)

- **10 CFR 60.113(a)(1)(ii)(A) Engineered Barrier System**
  "Containment of HLW within the waste packages will be substantially complete for a period...not less than 300 years nor more than 1,000 years after permanent closure of the geologic repository;..." For further details, the reader should consult the document itself.

- **10 CFR 60.131(b)(7) Criticality Control**
  "All systems for processing, transporting, handling, storage, retrieval, emplacement, and isolation of radioactive waste shall be designed to ensure that a nuclear criticality accident is not possible unless at least two unlikely, independent, and concurrent or sequential changes have occurred in the conditions essential to nuclear criticality safety. Each system shall be designed for criticality safety under normal and accident conditions."

- **10 CFR 60.135(b) Specific criteria for HLW Package Design**
  (1) "The waste package shall not contain explosive or pyrophoric materials or chemically reactive materials in an amount that could compromise the ability of the underground facility to contribute to waste isolation..."
  (2) "The waste packages shall not contain free liquids in an amount that could compromise the...containment of HLW...or result in spillage..."
  (3) Waste packages shall be designed to maintain waste containment during transportation, emplacement and retrieval.

- **10 CFR 60.135(c) Waste Form Criteria**
  (1) All HLW must "be in solid form and placed in sealed containers."
  (2) "Particulate waste forms shall be consolidated (for example by incorporation into an encapsulating matrix) to limit the availability and generation of particulates."
  (3) "All combustible radioactive wastes shall be reduced to a noncombustible form unless it can be demonstrated that a fire involving the waste packages containing combustibles will not compromise the integrity of other waste packages, adversely affect any structures, systems or components important to safety,..."

C3.3 10 CFR PART 61 - Licensing Requirements for Land Disposal of Radioactive Waste

This regulation contains much of the information needed to obtain a reactor operating license for operation of near-surface land disposal of LLW, including the specifics of the disposal site, its construction, and all the administrative details. It pertains to the disposal of the graphite fuel only insofar as it provides radioactivity limits with which waste packages must comply if they are to be buried in a near-surface site. Portions of the graphite blocks of FSV and the sleeves and spines of PB may qualify as LLW, after removal of the fuel.

The following provisions of *10 CFR 61* are pertinent to the fuel disposal option determination.

- **10 CFR 61.7(a) The Disposal Facility**
  Near-surface means no deeper than 30 meters, i.e., a trench.

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

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

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

- **10 CFR 61.41 Protection of the General Population from Releases of Radioactivity**
  "Concentrations of radioactive material which may be released to the general environment (from a land disposal facility) in ground water, surface water, air, soil, plants, or animals must not result in an annual dose exceeding an equivalent of 25 millirems to the whole body, 75 millirems to the thyroid, and 25 millirems to any other organ of any member of the public. Reasonable

effort should be made to maintain releases of radioactivity in effluents to the general environment as low as reasonably achievable."

- **10 CFR 61.52 Land Disposal Facility Operation and Disposal Site Closure**
  For near-surface disposal requirements are laid out for placement of Class A and for Class C Wastes.

- **10 CFR 61.56 Waste Characteristics**
  Minimum requirements for waste characteristics are listed here and include:
  (1) Structural stability (to withstand the weight of overburden and compaction equipment),
  (2) Elimination of void spaces within the waste and between the waste and its package,
  (3) Minimization of capability for detonation or explosive decomposition at normal temperatures and pressures,
  (4) Minimization of capability for generating toxic gases (except for radioactive gases, which may be packed at a pressure not to exceed 1.5 atmospheres at 20°C, with total activity not to exceed 100 Ci per container).
  (5) Content of no more than 1% by volume of free liquid in a container designed to ensure stability or 5% of the volume of waste processed to a stable form.

C.3.4 10 CFR PART 71 Packaging and Transportation of Radioactive Material

Part 71 is the key authority on requirements for licensing shipments, with definitions of categories that determine package types and license requirements. The tables of radioactivity limits for most radionuclides for shipment under the "Type A" category are given in Appendix A of Part 71.

- **10 CFR 71.3 Requirement for License**
  A license to ship radioactive materials is required, with few exceptions. The details and the exemptions are covered in this regulation and in 49 CFR 173.

- **10 CFR 71.4 Definitions**
  The following are definitions basic to an understanding of the regulations on packaging and transporting.

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

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

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

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

- **10 CFR 71.31-39 Subpart D - Application for Package Approval**
  Details of acceptable, licensable packaging are given.

- **10 CFR 71.33 Package Description**
  (b) "With respect to the contents of the package:
  (1) Identification and maximum radioactivity of radioactive constituents;...
  (3) Chemical and physical form;...
  (5) Maximum normal operating pressure;
  (6) Maximum weight;
  (7) Maximum amount of decay heat;..."

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

- **10 CFR 47 External Radiation Standards for all Packages**
  A package must be designed and prepared for shipment so that the radiation level does not exceed 200 millirem per hour at any point in the external surface of the package and the transport index does not exceed 10. (See 71.4 "Definitions".)

- **10 CFR 71.71-77 Subpart F - Package and Special Form Tests**
  Tests for packages and for special form materials are given in detail.

- **10 CFR 71.81-89, 91, 95, 97 Subpart G - Operating Controls and Procedures**
  Requirements for package inspections (71.81-89), records (71.91), reports (71.95), and advance notification of shipment (71.97) are provided.

- **10 CFR 71.101-137 Subpart H - Quality Assurance**
  Quality Assurance requirements are detailed.

C.3.5 10 CFR PART 72 - Licensing Requirements/independent Storage of SNF and High-level Radioactive Waste

This order would apply if the current storage arrangements at INEL were to be licensed so that the fuel elements would remain intact (Option I) and in storage at INEL. This regulation establishes requirements, procedures, and criteria for the issuance of licenses to receive, transfer, and possess power reactor spent fuel and other radioactive materials associated with spent fuel storage in an ISFSI or a MRS facility. Details are concerned with all aspects of a storage operation. Of technical concern are *Subpart F - General Design Criteria, and Subpart L - Approval of Spent Fuel Storage Casks.*

This regulation applies to disposal Option I for PB and FSV spent fuel in that disposal of intact fuel elements would be feasible in an ISFSI or an MRS if compliance with the following requirements could be assured.

- **10 CFR 72.104 Criteria for Radioactive Materials in Effluents and Direct Radiation from an ISFSI and MRS**
  The waste at the disposal site must not cause a dose equivalent to any individual beyond the controlled area of more than 25 mrem to the whole body, 75 mrem to the thyroid and 25 mrem to any other organ as a result of any emissions to the general environment, direct radiation from the ISFSI or MRS operations or any other source (72.104). This is the same restriction as is found in 10 CFR 61.7.

- **10 CFR 72.120-130 Subpart F - General Design Criteria**
  General, non-detailed requirements applicable to PB and FSV spent fuel disposal Option I are:
  71.124 Criteria for Nuclear Criticality Safety;
  72.126 Criteria for Radiological Protection; and
  72.128 Criteria for Spent Fuel, High-level Radioactive Waste, and Other Radioactive Waste Storage and Handling.

C.3.6 40 CFR PART 61 - National Emission Standards for Hazardous Air Pollutants

This document is one of the defining regulations for the CAA. It contains extensive lists of pollutants, to which the reader is referred (*40 CFR 61.01*). Specific regulations apply for beryllium, mercury, benzene, asbestos, and arsenic from glass manufacturing plants among others. Of particular interest to PB and FSV fuel element disposal are the sections on radionuclide emissions which are discussed below.

- **10 CFR 61.90**
  Subpart H National Emission Standards for Emissions of Radionuclides Other Than Radon from Department of Energy Facilities.

  (Subpart H does not apply to the disposal facilities subject to 40 CFR 191 Waste Isolation Pilot Plant (WIPP) or 40 CFR 192.)

  Subpart H could be important for PB or FSV disposal Option III, in which burning the graphite (and possibly then extracting the Uranium and Thorium by chemical methods) could release fumes containing radionuclides. Similarly, in Option II, if the separated fuel compacts are burned or the fuel is chemically extracted from them, the air emissions would require monitoring to ensure compliance with the limit below.

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

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

- **40 CFR 61**
  Subpart I — National Emission Standards for Radionuclide Emission from Facilities Licensed by the Nuclear Regulatory Commission and Federal Facilities not covered by Subpart H.

  The applicability of this Subpart is the same as for Subpart H.

- **40 CFR 61.102 Standard**
  Same as in Subpart H but include iodine.

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

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

- **40 CFR 61**
  Subpart Q — National Emission Standards for Radon Emissions from Department of Energy Facilities.

- **40 CFR 61.192 Standard**
  The limit for emission of radon into the air from all DOE storage and disposal facilities is 20 Pci/m2-s of radon-222 as an average for the entire source. (61.190)

C.3.7 40 CFR PART 191 - Environmental Radiation Protection Standards for Management and Disposal of SNF, High-level and Transuranic Wastes

(This regulation does not apply to the Yucca Mountain site,[16] but it can be used as guidance while the regulations for Yucca Mountain are being written.)

- **40 CFR 191.13 Containment Requirements**
  Releases to the environment for 10,000 years after disposal shall have a likelihood of less one chance in 10 of exceeding the listed quantities for the listed radionuclides or one chance in 1000 of exceeding 10 times the quantities listed in Table C.3. (191.13)

- **40 CFR 191.15 Individual Protection Requirements**
  "Disposal systems for spent nuclear fuel or high-level or transuranic radioactive wastes shall be designed to provide a reasonable expectation that, for 1,000 years after disposal, undisturbed performance of the disposal system shall not cause the annual dose equivalent from the disposal system to any member of the public in the accessible environment to exceed 25 mrem to the whole body or 75 mrem to any critical organ. All potential pathways (associated with undisturbed performance) from the disposal system to people shall be considered, including the

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

assumption that individuals consume 2 liters per day of drinking water from any significant source of ground water outside of the controlled area."

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

C.3.8 40 CFR PART 261 - Identification and Listing of Hazardous Waste (Non-radioactive)

This key regulation contains the fundamental information required for determining the acceptable non-radioactive hazard level of any waste.

- **40 CFR 261.3 Definition of Hazardous Waste**
  Basic definition of hazardous waste - exhibits any of the characteristics listed in Subpart C below. For extensive conditions and discussion, the reader is referred to the extensive discussion in the document itself.

  A table of limits in mg/l of thirteen elements considered hazardous is shown in Table C.4.

- **30 CFR 261.10 Criteria for Identifying the Characteristics of Hazardous Waste**
  Criteria for Identifying the Characteristics of Hazardous Waste - The waste poses a substantial present or potential hazard to human health or the environment when it is improperly treated, stored, transported, disposed of or otherwise managed.

- **40 CFR 261.11 Criteria for Listing Hazardous Waste**
  (a) Characteristics of Hazardous Waste:
  (1) It exhibits any of these characteristics (See **261.21-24** for details.)
  ignitability
  corrosivity
  reactivity
  toxicity
  (2) It has been found to be fatal to human beings in low doses...
  (3) It contains any of the constituents listed in 40 CFR 261 Appendix VIII and can pose a substantial present or potential hazard...

- **40 CFR 261 Subpart D Lists of Hazardous Wastes**
  The primary lists of non-radioactive hazardous materials are presented.

- **40 CFR 261.24 Toxicity Characteristic**
  Table C.5 shows the maximum concentration of contaminants for the toxicity characteristics.

C.3.9 40 CFR PART 264 - Standards for Owners and Operators of Hazardous Waste Treatment, Storage, and Disposal Facilities (Non-radioactive)

This regulation differs from *40 CFR 265*, "Interim Status Standards..." primarily in that the latter covers the owner/operator while waiting for a RCRA permit or until certification of final closure or, if the

facility is subject to post-closure requirements, until post-closure responsibilities are fulfilled (265.1). It applies to all owners/operators of facilities that treat, store, or dispose of hazardous wastes referred to in *40 CRT 268, Land Disposal Restrictions*, after receiving the appropriate permits.

Most of the sections of *40 CFR 264* are similar; some are identical to *40 CFR 265.* (See 40 CFR 265 for the applicable sections.)

C.3.10 40 CFR PART 265 - Interim Status Standards for Owners and Operators of Hazardous Waste Treatment, Storage, and Disposal Facilities (Non-Radioactive)

- **40 CFR 265 Subpart A - General**
  This regulation establishes national standards for handling and interim storage of hazardous (non-radioactive) waste. The fine points of applicability of 40 CFR 265 are detailed in this first Subpart.

- **40 CFR 264.340 and 265.340 Applicability**
  Check the list of hazardous waste in 40 CFR 261 to ensure compliance with any specific conditions.

  The following are fundamental requirements of this regulation.

- **40 CFR 264.13 and 265.13** - (Major regulation)
  "Before an owner treats, stores, or disposes of any hazardous wastes, or nonhazardous wastes if applicable under §264.113(d), he must obtain a detailed chemical and physical analysis of a representative sample of the wastes. At a minimum, the analysis must contain all the information which must be known to treat, store, or dispose of the waste in accordance with this part and part 268 (Land Disposal Restrictions) of this chapter."

- **40 CFR 264.17 and 265.17 General Requirements for Ignitable, Reactive, or Incompatible Wastes**
  Ignitable or reactive material is implied to be that which ignites easily - from open flame, hot surfaces, frictional heat, sparks, spontaneous ignition from chemical reactions, etc. From this it may also be inferred that graphite is not ignitable, even though it can be made to burn. Ignitable or reactive material must not be placed in a landfill or burned unless special precautions are taken to avoid conditions that would cause it to ignite, (precautions given in *40 CFR 268*). For requirements governing these characteristics in waste piles see *265.312* and *264.256*; in surface impoundments see *265.299* and *264.229*. Also, see *10 CFR 60.135* with respect to "combustible" material, which appears to imply that burning that may have to be sustained by an outside heat source, may still be included as hazardous under the ignitability category.

- **40 CFR 264.97 and 265.91 - Ground-Water Monitoring**
  A ground water monitoring system must be devised unless it can be demonstrated that there is a low potential of seepage of hazardous waste. The hazardous parameters are chloride, iron, manganese, phenols, sodium and sulphate. Plans for ground-water sampling and analysis are required by 265.92.

- **40 CFR 264.170-172 & 265.170-174 Subpart I - Use and Management of Containers**
  Containers must be in good condition, must be inspected weekly, and must not interact with the contents (compatibility requirement).

- **40 CFR 264.177 & 265.177 Special Requirements for Incompatible Waste**
  (a) Incompatible wastes (and/or materials) must not be placed in the same container unless § 264.17 or 265.17 is complied with.
  (b) Hazardous waste must not be placed in an unwashed container that previously held an incompatible waste or material.
  (c) A storage container holding a hazardous waste that is incompatible with any waste or other materials stored nearby in other containers, piles, open tanks, or surface impoundments must be separated from the other materials or protected from them by means of a dike, berm, wall, or other device."

- **40 CFR 264 and 265 Subpart K - Surface Impoundments**
  This subpart appears to apply to MRS installations and drywells (See 265.220-229 and 264.220-231). There are some differences between 265 and 264. *40 CFR 265.225* requires additional analyses if the surface impoundment is used for processing waste.

- **40 CFR 264. And 264 Subpart O - Incinerators**
  The burning of the FSV graphite blocks, with or without the fuel sticks in place, and the burning of the PB components, with or without the fuel compacts, will come under these regulations regarding incinerators. If the burning facility used is designated as an industrial furnace, Subpart P, Thermal Treatment (265 only), also applies. The operator of an incinerator must:

- **40 CFR 264.341 and 265.341 Waste Analysis**
  (1) Analyze any material not previously burned in the facility to establish steady state conditions for that material and to determine the type of pollutants that will be generated; at the minimum,
  (2) determinations must be made of (265.341):
  (3) heating value of the waste;
  (4) halogen and sulfur content of the waste;
  (5) lead & mercury concentrations, unless the waste can be certified not to contain any of either.

- **40 CFR 264.345 and 265.345 General Operating Requirements**
  Not feed material into the incinerator until the incinerator has reached steady state conditions of temperature, air flow, etc.

- **40 CFR 264.347 and 265.347 Monitoring and Inspections**
  (a) Monitor incinerator conditions relating to combustion and emissions at least every 15 minutes and make corrections immediately. Included measurements would be waste feed, auxiliary fuel feed, air flow, incinerator temperature, scrubber flow, scrubber Ph, and relevant level controls.
  (b) Inspect incinerator and associated equipment daily for leaks, spills, fugitive emissions, and all emergency shutdown controls, etc.

- **40 CFR 264.351 and 265.351 Closure**
  At closure, the ash must be removed. Unless shown not to be hazardous, the ash must be handled as hazardous waste.

- **40 CFR 265 (only) Subpart P - Thermal Treatment**
  The operator of a thermal treatment facility (other than an incinerator) or waste burner must:

- **40 CFR 265.370 Other Thermal Treatment**
  (b)(1) Determine whether the process is a controlled flame combustion in an incinerator or is thermal treatment in an industrial furnace (see 40 CFR 260.1 for definition of industrial furnace). If the latter, then Subpart P pertains, requiring the operator to demonstrate that the equipment can meet the same performance standards as an incinerator.

- **40 CFR 265.377 Monitoring and Inspections**
  Monitor the stack emissions visually at least once an hour and make corrections immediately if the appearance is not normal. (265.377)

- **40 CFR 264 and 265 Subpart AA - Air Emission Standards for Process Vents**
  This section applies to process vents associated with solvent extraction, air or steam stripping operations, etc., which may be part of the fuel disposal process in the second stage of Options 2 or 3.

C.3.11 40 CFR PART 268 - Land Disposal Restrictions

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

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

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

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

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

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

C.3.12 49 CFR 172 - Hazardous Materials Table, Special Provisions, Hazardous Materials Communications, Emergency Response Information, and Training Requirements

- **49 CFR 172.101 and 102 Table of Hazardous Materials and Special Provisions**

- **49 CFR 172.403(a)&(g)**
  "The following applicable items of information must be entered in the blank spaces on the RADIOACTIVE label..." (for shipment):
  (1) "Contents. The name of the radionuclides as taken from the listing of radionuclides in §173.435 of this subchapter..."
  (2) "Activity..."
  (3) "Transport index..."

C.3.13 49 CFR PART 173 - Shippers - General Requirements For Shipments and Packagings

All packages to be shipped must be prepared according to this chapter in order to be accepted for any mode of public transportation. The Hazardous Materials Regulations (HMRs) in this document are consistent with most U.S. and international standard codes. Extensive tables show the hazardous materials class and division numbers of the materials that are classified for special handling for transportation. The order of precedence is given for cases in which there are several hazardous materials present.

- **49 CFR 173 Subpart I Subpart I - Radioactive Materials**
  The regulations for radioactive materials must be met in addition to the requirements for hazardous materials listed in the tables.

- **49 CFR 173.403(cc)(dd)(gg)(hh)**
  Definitions of Type A and B packages and packaging are determined by radioactivity levels A_1 and A_2.

- **49 CFR 173.469**
  Special form radioactive material is defined as radioactive material which satisfies the following conditions:
  (1) it is either a single solid piece or is contained in a sealed capsule that can be opened only by destroying the capsule,
  (2) the piece has at least one dimension not less than 5 millimeters, and
  (3) it satisfies the test requirements of this section.

Note: If PB fuel in canisters can withstand these test requirements, it should qualify as special form Radioactive material.

- **49 CFR 173.403(y)**
  Radioactive material in this context means any material having a specific activity (activity per unit mass) greater than 0.002 microcurie per gram (μCi/g).

  (Note that in *10 CFR 20.2005*, a licensee can dispose of licensed material as if it were not radioactive if its radioactivity is 0.05 μCi/g (scintillation counter liquid), 25 times higher than the 0.002 μCi/g definition of radioactive material.)

- **49 CFR 173.411-478 Subpart I Radioactive Materials**
  The reader will find extensive directions and requirements for packaging, for determining packaging classifications based on radioactivity level, for specimen testing, for radiation level and thermal limitations, for contamination control, and many other facets for which requirements must be met before transporting radioactive materials.

C.3.14 DOE-5400.3 Hazardous and Radioactive Mixed Waste Program

This Order establishes the policies and requirements to implement RCRA within the framework of the Environmental Protection program. It lists the major references that apply to hazardous waste under RCRA and to radioactive mixed waste and provides working definitions of radioactive waste, radioactive mixed waste, and byproduct material.

- **DOE5400.3 (4a)**
  Hazardous waste is as defined in 40 CFR 261, and does not include radioactive waste (radionuclides of source material, special nuclear material, and byproduct material).

- **DOE5400.3 (4c)**
  Radioactive waste is solid, liquid, or gaseous material that contains radionuclides regulated under the Atomic Energy Act of 1954, as amended, and that is of negligible economic value considering costs of recovery.

- **DOE5400.3 (5b)**
  Whenever any hazardous waste listed in Title 40 CFR 261 is inadvertently mixed with any source material, special nuclear material or byproduct material, (i.e., radioactive waste) the hazardous waste component is subject to regulation under Subtitle C of RCRA.

C.3.15 DOE ORDER 5480.3 - Safety Requirements for Packaging and Transportation Hazardous Materials, Hazardous Substances, and Hazardous Wastes

This Order is a DOE rewriting of *10 CFR 71*, the NRC requirements document for packaging, preparation for shipment, and transportation of radioactive material. In some cases, the emphasis is somewhat different from that in *10 CFR 71*, or the wording is clarified, or the original document is referenced for additional details.

- **DOE5480.3 Preamble 4.a.-n.**
  This order lists other references pertaining to packaging and transportation.

- **DOE5480.3 5.d.**
  This order provides fissile classification as the classification of a package or shipment of fissile materials according to the controls needed to provide nuclear criticality safety during transportation (see the Definitions Section, following, for these classifications, definition and identification of fissile materials, and definition of transport index).

- **DOE5480.3 7.a**
  This order specifies that shipments of hazardous materials or waste must be in compliance with this order, the applicable safety regulations of the Department of Transportation and the applicable packaging standards set forth by the NRC in 10 CFR 71.

- **DOE5480.3 7.b.(4)**
  This order exempts reactor fuel elements and wastes and contaminated equipment from the requirements for special packaging for Pu in packages in excess of 20 Ci.

- **DOE5480.3 7.c.**
  This order specifies package standards for radioactive materials in amounts in excess of Type A quantities.

C.3.16 DOE 5633.3A Control and Accountability of Nuclear Materials

This order prescribes the minimum DOE requirements and procedures for control and accountability of nuclear materials at DOE-owned and -leased facilities and DOE-owned nuclear materials at other facilities which are exempt from licensing by the NRC.

Criteria for determining the Nuclear Material Safeguards Category, and Attractiveness Level (attractiveness in terms of desirability or ease of diversion), of quantities of special nuclear material.

C.3.17 DOE 5820.2A - Radioactive Waste Management

This order "does not apply to the management by the Department of commercially generated spent nuclear fuel or high-level radioactive waste, nor to the geologic disposal of high-level waste produced by the Department's activities and operations." Such operations come under the Nuclear Waste Policy Act of 1982 as amended (Public Law 97-425). The provisions of this Order that would seem to be pertinent to the options for disposition of PB and FSV fuel will be listed anyway below, for use until better guidelines can be found. It is written specifically to apply to WIPP.

Chapters I - High-level Waste, II - Transuranic Waste, and III - Low-level Waste provide generic requirements on waste package leakage, segregation of waste types, ventilation and filtration systems, criticality, waste treatment and minimization, waste characterization and certification, waste packaging, shipping to WIPP, acceptance criteria, and alternate burial sites.

C.4 FUNDAMENTAL DEFINITIONS

- **Classes A, B, and C (10 CFR 61 - for Near-Surface Disposal)**

  Class A - Meets certain radioactivity limits but is unstable in the sense of the decomposition of normal trash.

  Class B - Meets certain radioactivity limits and is considered stable.

  Class C - Meets certain even more rigorous conditions of stability and requires additional measures against inadvertent intrusion.

- **Fissile Classification**

  10 CFR 71.4 - for Transportation
  Classification of a package or shipment of fissile materials according to the controls needed to provide nuclear criticality safety during transportation, as follows:

  Fissile Class I - Packages that may be transported in unlimited numbers and in any arrangement, require no nuclear criticality safety controls during transportation, and no "transport index" number.

  Fissile Class II - Packages that may be transported in any arrangement but in numbers that do not exceed a transport index of 50. For nuclear criticality safety control, individual packages may have a transport index of not less than 0.1 nor more than 10.

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

  49 CFR 173.417 - for Transportation
  Fissile Class I, II, or III Classes of packaging for fissile materials with radioactive content exceeding A_1 or A_2 limits. Radioactivity and decay heat limits for each class are given.

- **Fissile Material (10 CFR 71.4 - for Transportation)**

  Any material consisting of or containing one or more fissile radionuclides. Fissile materials are classified according to the controls needed to provide nuclear criticality safety during transportation.

- **Fissile Radionuclides (10 CFR 71.4 - for Transportation)**

  ^{233}U, ^{235}U, ^{238}Pu, ^{239}Pu, and ^{241}Pu.
  Note: DOE Order 5480.3/P also includes ^{237}Np and ^{244}Cm.

- **Hazardous Waste**

  10 CFR 61 - for disposal in Near-Surface Land site
  Those wastes designated as hazardous by EPA 40 CFR 261.

  40 CFR 261 - for Listing/Identification
  Meets criteria in 261.20-24 of Ignitability, Corrosivity, Reactivity or Toxicity.

  DOE 5820.2A - for Management
  Those wastes that are designated hazardous by EPA 40 CFR 261.

- **High-Level Radioactive Waste or HLW**

  Nuclear Waste Policy Act of 1982
  1.The highly radioactive material resulting from the reprocessing of spent nuclear fuel, including liquid waste produced directly in reprocessing and any solid material derived from such liquid waste that contains fission products in sufficient concentrations; and
  2. Other highly radioactive material that the Commission, consistent with existing law, determines by rule requires permanent isolation.

  10 CFR 60 - for Disposal in a Geologic Repository
  1. Irradiated reactor fuel
  2. Liquid wastes resulting from the operation of the first cycle solvent extraction system and the concentrated wastes from subsequent extraction cycles, in a facility for reprocessing irradiated reactor fuel, (and by assumption, extended to processing for disposal, if using the same processes as for reprocessing).
  3. Solids into which such liquid wastes have been converted.

  10 CFR 72 - for Independent Storage
  Same as in the Nuclear Waste Policy Act of 1982.

  40 CFR 191 - for Management and Disposal of Spent Nuclear Fuel, HLW and Transuranic Waste
  As "defined in the Nuclear Waste Policy Act of 1982."

  DOE 5820.2A - for Management
  Same as definition 1 in the Nuclear Waste Policy Act of 1982.

- **Low-Level Radioactive Waste or LLW**

  10 CFR 61 - for disposal in Near-Surface Land site
  Low-Level Radioactive Wastes containing source, special nuclear, or byproduct material that are acceptable for disposal in a land disposal facility. Radioactive waste not classified as high-level radioactive waste, transuranic waste, spent nuclear fuel, or byproduct material.

DOE 5820.2A - for Management
Waste that contains radioactivity and is not classified as HLW, TRU, spent nuclear fuel, or byproduct material. (May include non-power production test specimens.)

- **Normal Form Radioactive Material**

49 CFR 173- for Transportation
Not Special Form Radioactive Material

- **Radioactive Waste**

10 CFR 60 - for disposal in a Geologic Repository
HLW and radioactive materials other than HLW that are received for emplacement in a geologic repository.

40 CFR 191 - for Management and Disposal of Spent Nuclear Fuel, HLW and Transuranic Waste
High-level and transuranic radioactive waste as covered by this Part. See "High-level radioactive waste" as given below for the Nuclear Waste Policy Act of 1982, and "Transuranic Radioactive Waste" definition below for 40 CFR 191.

DOE 5820.2A - for Management
Solid, liquid, or gaseous material that contains radionuclides regulated under the Atomic Energy Act of 1954, as amended, and of negligible economic value considering costs of recovery.

- **Source Material**

10 CFR 72 - for Independent Storage
1. Uranium or thorium, or any combination thereof, in any physical or chemical form or
2. Ores that contain by weight one-twentieth of one percent (0.05%) or more of:
a. Uranium
b. Thorium
c. Any combination thereof.

Source material does not include special nuclear material.

- **Special Form Radioactive Material**

49 CFR 173-for Transportation
Radioactive material which satisfies the following conditions:
1. Is either a single solid piece or is contained in a sealed capsule that can be opened only by destroying the capsule;
2. The piece or capsule has at least one dimension not less than 5 millimeters (0.197"); and
3. It satisfies the test requirements of 137.469 (Normal form) or 173.389 (Special form).

Note: PB or FSV fuel elements in their canisters appear to be examples.

- **Special Nuclear Material**

  10 CFR 20.3 - for Management; and 10 CFR 72 - for Independent Storage
  1. Plutonium, Uranium-233, Uranium enriched in the isotope 233 or 235, and any other material the Commission determines to be special nuclear material, but not source material;
  2. Any material artificially enriched by any of the foregoing but which does not include source material.

- **Spent Nuclear Fuel**

  40 CFR 191 - for Management and Disposal of Spent Nuclear Fuel, HLW and Transuranic Waste
  Fuel that has been withdrawn from a nuclear reactor following irradiation, the constituent elements of which have not been separated by reprocessing.

  DOE 5820.2A - for Management
  Essentially the same as in 40 CFR 191.

- **Transport Index**

  10 CFR 71.4 - for Transportation
  1. The maximum radiation dose rate in mrem/hour at 1 meter from any accessible external surface of the package.
  2. For Fissile Class II, the maximum radiation level in mrem/hour at 1 meter from the external surface of the package, or the number obtained by dividing 50 by the allowable number of packages which may be transported together as determined under 71.59 (Specific Standards for Fissile Class II Package).

- **Transuranic Radioactive Waste**

  40 CFR 191 - for Management and Disposal of Spent Nuclear Fuel, HLW and Transuranic Waste
  Waste containing more than 100 3nCi of alpha-emitting transuranic isotopes, with half-lives greater than twenty years, per gram of waste, except for:
  1. High-Level Radioactive Wastes;
  2. Wastes that the DOE has determined...do not need the degree of isolation required by this part; or
  3. Wastes that the Commission has approved for disposal on a case-by-case basis in accordance with 10 CFR 61.

  DOE 5820.2A - for Management
  Without regard to source or form, waste that is contaminated with alpha-emitting transuranium radionuclides with half-lives >20 years and concentration >100 nCi/g at the time of assay. Heads of Field Elements can determine that other alpha contaminated wastes, peculiar to a specific site, must be managed as transuranic waste.

- **Type A Package**

49 CFR 173 Subpart I - for Transportation
Type A Packaging together with its limited radioactive contents. Does not require competent authority approval, since its contents are limited to A_1 and A_2. Type A Packaging is designed to retain the integrity of containment and shielding under normal conditions of transport as demonstrated by tests 173.465 (water spray, free drop, compression, penetration) and 173.466 (for liquids and gases) as appropriate (173.403).
A_1 - maximum activity permitted for special form radioactive material in a Type A package (173.431(a)).
A_1 - maximum activity permitted for radioactive material other than special form or low specific activity material in a Type A package (173.431).

See 173.433 for the method to calculate the A's for X-ray and gamma emitters, beta emitters, and mixed radionuclides; and 173.434 for a table of A_1 and A_2 values for specific radionuclides. See also 173.415 Authorized Type A Packages and 173.417 Authorized Packaging - Fissile Materials.

10 CFR 71.4 - for Packaging and Transportation
A_1 - See 10 CFR 71.4 under "Type A quantity" and "Type B Quantity" relative to A_1 and A_2

- **Type B Package**

49 CFR 173.403- for Transportation
Type B packaging together with its radioactive contents. Type B packaging is designed to retain its integrity of containment and shielding under normal conditions of transport and hypothetical accident test conditions set forth in 10 CFR 71 (173.403). See also 173.416 - Authorized Type B Packages.

10 CFR 71 - for Packaging and Transportation
See 10 CFR 71.4 under "Package (2) Type B Package" and under "Packaging", where the explanation is essentially the same as in 49 CFR 173 above in slightly different form.

- **Waste**

10 CFR 61 - for disposal in Near-Surface Land site
Those Low-Level Radioactive Wastes containing source, special nuclear, or byproduct material that are acceptable for disposal in a land disposal facility.

40 CFR 191 - for Management and Disposal of Spent Nuclear Fuel, HLW and Transuranic Waste
Any spent nuclear fuel or radioactive waste isolated in a disposal system.

APPENDIX C

TABLES

Table C.1. List of Regulations Reviewed

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

Table C.2. Listing of Applicable Regulations

Peach Bottom and Fort St. Vrain	
Required for All Three Options	
Packaging and Land Disposal	
10 CFR 60	Disposal/HL Radioactive Wastes in Geologic Repositories
10 CFR 60.43	License Specification
10 CFR 60.102(e)	Isolation of Waste (the containment period)
10 CFR 60.111(b)	Retrievability of Waste
10 CFR 60.113(a)(1)	Engineered Barrier System
10 CFR 60.131(b)(7)	General Design Criteria for Geologic Repository Operations (criticality control)
10 CFR 60.135	Criteria for the Waste Package & its Components (HLW Design, specific criteria)
10 CFR 61	Licensing Requirements for Land Disposal of Radioactive Waste
10 CFR 61.41	Protection of the General Public from Releases of Radioactivity (limits)
10 CFR 61.52	Land Disposal Facility Operation & Disposal Site Closure (Class C depth)
10 CFR 61.55	Waste Classification
10 CFR 61.56	Waste Characteristics
Packaging and Transportation	
10 CFR 71	Packaging & Transportation of Radioactive Material
49 CFR 173	Shippers - General Requirements for Shipment & Packagings
DOE Order 5480.3	Safety Requirements/Packaging & Transportation/Hazardous Materials
10 CFR 71.4	Definitions, classifications
49 CFR 173.2	Hazardous materials classes & index to hazard class definitions.
173.403 Subpart I	Radioactive Materials: Definitions
DOE 5480.3	7.d. DOE certificates of compliance/in excess to Type A
10 CFR 71.7-10	Subpart B Exemptions
49 CFR 173.3-4	Packaging & exceptions; Exceptions for small quantities
173.12	Exceptions for shipment of waste materials
DOE 5480.3	7.f Exemptions (49 CFR 107.103)
10 CFR 71.31-39	Subpart D Application for Package Approval;
71.41-65	Subpart E Package Approval Standards
49 CFR 173.412-417	Design Requirements.., Authorized Type A Packages, Type B Packages, Packaging Fissile Materials
DOE 5480.3	8. Packaging Standards
10 CFR 71.41-65	General & specific package standards & radioactivity limits
173.441-443	Radiation Level Limitations, Thermal Limitations, Contamination Control
DOE 5480.3	7.b. Special packaging for Plutonium Pu-bearing conditions
10 CFR 71.71-77	Normal conditions of transport & hypothetical conditions
DOE 5480.3	11. Normal conditions & 12. Hypothetical accident conditions

Table C.2 Continued

Peach Bottom and Fort St. Vrain	
10 CFR 71.81-97 DOE 5480.3	Operating Procedures/Inspections, records, reports, etc. 10. Operating Procedures
10 CFR 71.101-137 DOE 5480.3	Quality Assurance requirements 9. Quality Assurance Procedures for Offsite Containers
Radiation Standards & Exposure Limits	
10 CFR 20	Standards for Protection Against Radiation
10 CFR 20.2001(b)	Must be Specifically Licensed to Receive Licensed Material for any treatment
10 CFR 20.2002	Method for obtaining approval of proposed disposal procedures (chemical & physical properties)
10 CFR 20.2005(1)	Disposal as if Not Radioactive (limits for scintillation counting liquid for H-3 & C-14)
40 CFR 191	Radiation Protection Standards/Disposal/HLW & Tru Wastes
40 CFR 191.03(a)(2)	Standards (millirem exposure limits to body)
40 CFR 191.13(a)	Containment (release limits per year for 10,000 years)
40 CFR 191.14(a)	Active institutional controls not needed after 100 years
40 CFR 191.15	Individual protection (same as 191.03(a)(2)
40 CFR 191.16	Ground water protection requirements - picocuries/liter of drinking water (Currently remanded)
Treatment, Storage and Disposal Standards	
40 CFR 264 40 CFR 265	Standards for Owners & Operators of Hazardous Waste Treatment, Storage & Disposal Facilities Interim Status Standards for Owners & Operators of Hazardous Waste Treatment, Storage, and Disposal Facilities
40 CFR 264.13 40 CFR 265.13	Same as 265.13 - requires full and updated analyses General Waste Analysis
40 CFR 264.17 40 CFR 265.17	Same as 265.17 - adds the need to document the analyses General Requirements for Ignitable, Reactive or Incompatible Wastes
40 CFR 264.170-178 40 CFR 265.170-177	See 264.175 details on containment; 264.178 on Closure Subpart I Use & Management of Containers

Table C.2 Continued

Peach Bottom and Fort St. Vrain	
40 CFR 264.220-231	See 264.220 detail on liners; 264.221 info on monofill deleted; no trial test -.225; 264.226 monitoring different; adds 264.229 & .230 special requirements for ignitable and reactive waste and incompatible wastes.
40 CFR 265.220-229	Subpart K Surface Impoundments
40 CFR 268	Land Disposal Restrictions
40 CFR 268.1(a)(b) & 268.30-35	Restrictions/Materials Disposed in Landfill
40 CFR 268.2(c)	Land Disposal Regulations Cover Vaults & Bunkers
40 CFR 268.40-45	Standards for Treatment Expressed as Concentration, etc.
DOE Order 5400.3/P	Hazardous & Radioactive Mixed Waste
DOE 5400.3 4.a	Hazardous Waste
DOE 5400.3 4.c	Radioactive Waste
DOE 5400.3 5.b	Nonradioactive Hazardous Waste under RCRA
DOE Order 5820.2A	Safety Req'ts/Packaging & Transportation/Hazardous Wastes
DOE 5820.2A	Generic Requirements, HLW, LLW & Transuranic Waste
Required Especially for Option 1 - Disposal of Intact Fuel Elements - Licensed	
10 CFR 72	Licensing Requirements/Independent Storage of Spent Nuclear Fuel & High Level Radioactive Waste
10 CFR 72.2 Scope	Independent Spent Fuel Storage Installation (ISFSI) or MRS
10 CFR 72.6	License requires; Types of Licenses
10 CFR 72.16-34	Subpart B License Application, Form & Contents
10 CFR 72.40-62	Subpart C Issuance & Conditions of License
10 CFR 72.70-86	Subpart D Records, Reports, Inspections, & Enforcement
10 CFR 72.90-108	Subpart E Siting Evaluation Factors
10 CFR 72.104	Criteria/radioactive materials in effluents & direct radiation for ISFSI or MRS
10 CFR 72.120-130	Subpart F General Design Criteria for Storage Installation
10 CFR 72.140-220	Subparts G-K Quality Assurance, Administrative Requirements
10 CFR 72.230-240	Subpart L Approval of Spent Fuel Storage Casks

Table C.2 Continued

Peach Bottom and Fort St. Vrain	
Required for Options II - Separate Fuel from other Components & III - Burn Fuel Elements	
40 CFR 265	Interim Standards/Hazardous Waste Treatment, Storage, Disposal
10 CFR 20.2004 40 CFR 265.312	Treatment or disposal by incineration Incinerator Restrictions - Ignitable & Reactive Wastes
40 CFR 264.341 40 CFR 265.341	Incinerators - Waste Analysis - not as stringent as 265.341 Incinerators - Waste Analysis Required of Waste to be Burned
40 CFR 264.343	Incinerators - Performance Standards - detailed
40 CFR 264.345 40 CFR 265.345	Incinerators - Operating Requirements - much more detailed than 265.345 Incinerators - Feed only under Steady State Conditions
40 CFR 264.347 40 CFR 265.347	Incinerators - Monitoring & Inspections; different details Incinerators - Monitoring & Inspections; monitor conditions frequently
40 CFR 264.351 40 CFR 265.351	Incinerators - Closure - same as 265.351 Incinerators - Handle ash as hazardous waste, or justify
40 CFR 265.370	Thermal Treatment - Incinerator or Industrial Furnace?
40 CFR 265.377	Thermal Treatment - Monitor stack emissions
40 CFR 264.1030 40 CFR 265.1030	Air Emission Standards for Process Vents Air Emission Standards for Process Vents

Table C.3. Release Limits for Containment Requirements
[Cumulative releases to the accessible environment for 10,000 years after disposal]
from *40 CFR 191, Subpart B, Appendix A*

Radionuclide	Release Limit per 1,000 MTHM, Curies
Americium-241 or -243	100
Carbon-14	100
Cesium-125 or -137	1,000
Iodine-129	100
Neptunium-237	100
Plutonium-238,-239,-240,-or -242	100
Radium-226	100
Strontium-90	1,000
Technetium-99	10,000
Thorium-230, or -232	10
Tin-126	1,000
Uranium-233,-234,-235,-236, or -238	100
Any other alpha-emitting radionuclide with a half-life >20 years	100
Any other radionuclide with a half-life >20 years that does not emit alpha particles	1,000

Table C.4. RCRA Toxic Elements (from *40 CFR 261.3*)

RCRA Toxic Element	Max. For Any Single Composite Sample, in mg/L
Antimony	0.063
Arsenic	0.055
Barium	6.3
Beryllium	0.0063
Cadmium	0.032
Chromium (total)	0.33
Lead	0.095
Mercurey	0.009
Nickel	0.63
Selenium	0.16
Silver	0.30
Thallium	0.013
Vanadium	1.26

Table C.5. Maximum Concentration of Contaminants for the Toxicity Characteristic from *40 CFR 261.24*

EPA HW NO.[1]	Contaminant	CAS NO.[2]	Regulatory Level (mg/L)
D004	Arsenic	7440-38-2	5.0
D005	Barium	7440-39-3	100.0
D018	Benzene	71-43-2	0.5
D006	Cadmium	7440-43-9	1.0
D019	Carbon Tetrachloride	56-23-5	0.5
D020	Chlordane	57-74-9	0.03
D021	Chlorobenzene	108-90-7	100.0
D022	Chloroform	67-66-3	6.0
D007	Chromium	7440-47-3	5.0
D023	o-Cresol	95-48-7	[4]200.0
D024	m-Cresol	108-39-4	[4]200.0
D025	p-Cresol	106-44-5	[4]200.0
D026	Cresol		[4]200.0
D016	2,4-D	94-75-7	10.0
D027	1,4-Dichlorobenzene	106-46-7	7.5
D028	1,2-Dichloroethane	107-06-2	0.5
D029	1,1-Dichlorethylene	75-35-4	0.7
D030	2,4-Dinitrotoluene	121-14-2	[3]0.13
D012	Endrin	72-20-8	0.02
D031	Heptachlor (and its epoxide)	76-44-8	0.008
D032	Hexachlorobenzene	118-74-1	[3]0.13
D033	Hexachlorobutadiene	87-68-3	0.5
D034	Hexachloroethane	67-72-1	3.0
D008	Lead	7439-92-1	5.0
D013	Lindane	58-89-9	0.4
D009	Mercure	7439-97-6	0.2

EPA HW NO.[1]	Contaminant	CAS NO.[2]	Regulatory Level (mg/L)
D014	Mtehoxychlor	72-43-5	10.00
D035	Methyl ethyl ketone	78-93-3	200.0
D036	Nitrobenzene	98-95-3	2.0
D037	Pentachlorophenol	87-86-5	100.0
D038	Pyridine	110-86-1	[3]5.0
D010	Selenium	7782-49-2	1.0
D011	Silver	7440-22-4	5.0
D039	Tetrachlorethylene	127-18-4	0.7
D015	Toxaphene	8001-35-2	0.5
D040	Trichlorethylene	79-01-6	0.5
D041	2,4,5-Trichlorophenol	95-95-4	400.0
D042	2,4,6-Trichlorophenol	88-06-2	2.0
D017	2,4,5-TP (Silvex)	93-72-1	1.0
D043	Vinyl chloride	75-01-4	0.2

[1] Hazardous waste number.
[2] Chemical abstracts service number.
[3] Quantitation limit is greater than the calculated regulatory level. The quantitation limit therefore becomes the regulatory level.
[4] If o-, m-, and p-Cresol concentrations cannot be differentiated, the total cresol (D026) concentration is used. The regulatory level of total cresol is 200 mg/l.

ATTACHMENT A

Availability of Eddy Current Technology

at PNL

for Measuring Graphite Strength

The objective of Attachment A is to provide information on the availability of a nondestructive Eddy Current measurement at PNL for measuring graphite strength. The method was developed at PNL for measuring potential changes in FSV structural graphite caused by oxidation. This attachment provides information on the potential application of this method for measuring the strength of graphite spent fuels being stored at INEL. The equipment for doing this work has now become commercially available and has been significantly improved.

The mechanical strength of graphite can be measured with an eddy current probe. The graphite strength is determined using the relationships between eddy-current conductivity and density and between density and compressive strength for oxidized graphite (Morgan, Prince, and Posakony 1982). The following describes some features of this technology and its availability at PNL.

The footprint of the eddy current (EC) round cylinder probe that was developed for testing graphite at the FSV plant was approximately 1-in^2 (Morgan, Prince, and Posakony 1982). A similar size probe was developed for ultrasound testing (UT) of the FSV graphite. Characterization of graphite fuel types at INEL would require probes to be custom fitted to the fuel geometry. Surface contact defines the coil proximity to the test article. Since the results of the studies for FSV were completed in 1982, commercial EC systems have become available. Bob Ferris[17] and Ron Hockey are PNL authorities on these systems. The UT probe is less developed than the EC probe and probably should not be considered for testing of INEL graphite fuels.

Because a wireless probe does not appear to be feasible, the logistics for positioning and operating the probe must be engineered to interface with the facility constraints. PNL could provide the equipment, operating procedures, and staff to perform the measurements or to train INEL staff to perform the measurements.

The measurements may not be sensitive to irradiation hardening that may have resulted from point defect configurations generated during irradiation, but should be conservative because irradiation strengthens the material and reduces the conductivity. In contrast degradation by oxidation and the formation of porosity weakens the material and reduces the conductivity.

Oxidation can cause graphite to become very fragile, without changes in visual appearance. Visual examination of graphite may be misleading as an indicator of residual strength. The presence of H_2, N_2, CO, CO_2 in the reactor coolant affects oxidation of graphite and impurities and strength changes during irradiation. During storage, oxidation is expected to be limited to the surfaces; because, at low temperatures, ionized gases are the primary oxidizing species. However, the surface area includes surfaces of flow channels, etc.

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[17]Bob Ferris indicated that PNL has systems, probes, and staff that could be applied to this application.

ATTACHMENT B

10 CFR 61.55, WASTE CLASSIFICATION

(a) Classification of waste for near surface disposal.

(1) Considerations. Determination of the classification of radioactive waste involves two considerations. First, consideration must be given to the concentration of long-lived radionuclides (and their shorter-lived precursors) whose potential hazard will persist long after such precautions as institutional controls, improved waste form, and deeper disposal have ceased to be effective. These precautions delay the time when long-lived radionuclides could cause exposures. In addition, the magnitude of the potential dose is limited by the concentration and availability of the radionuclide at the time of exposure. Second, consideration must be given to the concentration of shorter-lived radionuclides for which requirements on institutional controls, waste form, and disposal methods are effective.

(2) Classes of waste.

(i) Class A waste is waste that is usually segregated from other waste classes at the disposal site. The physical form and characteristics of Class A waste must meet the minimum requirements set forth in Section 61.56(a). If Class A waste also meets the stability requirements set forth in Section 61.56(b), it is not necessary to segregate the waste for disposal.

(ii) Class B waste is waste that must meet more rigorous requirements on waste form to ensure stability after disposal. The physical form and characteristics of Class B waste must meet both the minimum and stability requirements set forth in Section 61.56.

(iii) Class C waste is waste that not only must meet more rigorous requirements on waste form to ensure stability but also requires additional measures at the disposal facility to protect against inadvertent intrusion. The physical form and characteristics of Class C waste must meet both the minimum and stability requirements set forth in Section 61.56.

(iv) Waste that is not generally acceptable for near-surface disposal is waste for which waste form and disposal methods must be different, and in general more stringent, than those specified for Class C waste. In the absence of specific requirements in this part, such waste must be disposed of in a geologic repository as defined in Part 60 of this chapter unless proposals for disposal of such waste in a disposal site licensed pursuant to this part are approved by the Commission.

(3) Classification determined by long-lived radionuclides. If radioactive waste contains only radionuclides listed in Table 1, classification shall be determined as follows:

(i) If the concentration does not exceed 0.1 times the value in Table 1, the waste is Class A.

(ii) If the concentration exceeds 0.1 times the value in Table 1, but does not exceed the value in Table 1, the waste is Class C.

(iii) If the concentration exceeds the value in Table 1, the waste is not generally acceptable for near surface disposal.

(iv) For wastes containing mixtures of radionuclides listed in Table 1, the total concentration shall be determined by the sum of fractions rule described in paragraph (a)(7) of this section.

(4) Classification determined by short-lived radionuclides. If radioactive waste does not contain any of the radionuclides listed in Table 1, classification shall be determined based on the concentrations shown in Table 2. However, as specified in paragraph (a)(6) of this section, if radioactive waste does not contain any nuclides listed in either Table 1 or 2, it is Class A.

(i) If the concentration does not exceed the value in Column 1, the waste is Class A.

(ii) If the concentration exceeds the value in Column 1, but does not exceed the value in Column 2, the waste is Class B.

(iii) If the concentration exceeds the value in Column 2, but does not exceed the value in Column 3, the waste is Class C.

(iv) If the concentration exceeds the value in Column 3, the waste is not generally acceptable for near-surface disposal.

(v) For wastes containing mixtures of the nuclides listed in Table 2, the total concentration shall be determined by the sum of fractions rule described in paragraph (a)(7) of this section.

(i) If the concentration does not exceed 0.1 times the value in Table 1, the waste is Class A.

(ii) If the concentration exceeds 0.1 times the value in Table 1, but does not exceed the value in Table 1, the waste is Class C.

(iii) If the concentration exceeds the value in Table 1, the waste is not generally acceptable for near surface disposal.

(iv) For wastes containing mixtures of radionuclides listed in Table 1, the total concentration shall be determined by the sum of fractions rule described in paragraph (a)(7) of this section.

ATTACHMENT B

TABLES

Table 1

Radionuclide	Concentration, Ci/m^3
C-14	8
C-14 in activated metal	80
Ni-59 in activated metal	220
Nb-94 in activated metal	0.2
Tc-99	3
I-129	0.08
Alpha emitting transuranic nuclides with half-life greater than five years	100[a]
Pu-241	3,500[a]
Cm-242	20,000[a]

[a]Units are nCi/g.

Table 2

Radionuclide	Concentration, curies per cubic meter		
	Col. 1	Col. 2	Col. 3
Total of all nuclides with less than 5 year half life	700	a	a
H-3	40	a	a
Co-60	700	a	a
Ni-63	3.5	70	700
Ni-63 in activated metal	35	700	7,000
Sr-90	0.04	150	70,000
Cs-137	1	44	46,000

[a] There are no limits established for these radionuclides in Class B or C wastes. Practical considerations such as the effects of external radiation and internal heat generation on transportation, handling, and disposal will limit the concentrations for these wastes. These wastes shall be Class B unless the concentrations of other nuclides in Table 2 determine the waste to the Class C independent of these nuclides.