1 OF 3

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Systems Impacts of Spent Fuel Disassembly Alternatives

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prepared for

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ABSTRACT

Three studies were completed to evaluate four alternatives to the disposal of intact spent fuel assemblies in a geologic repository. A preferred spent fuel waste form for disposal was recommended on consideration of (1) package design and fuel/package interaction, (2) long-term, in-repository performance of the waste form, and (3) overall process performance and costs for packaging, handling, and emplacement. The four basic alternative waste forms considered were (1) end fitting removal, (2) fission gas venting, (3) disassembly and close packing, and (4) shearing/immobilization. None of the findings ruled out any alternative on the basis of waste package considerations or long-term performance of the waste form. The third alternative offers flexibility in loading that may prove attractive in the various geologic media under consideration, greatly reduces the number of packages, and has the lowest unit cost.

These studies were completed in October, 1981. Since then Westinghouse Electric Corporation and the Office of Nuclear Waste Isolation have completed studies in related fields. This report is now being published to provide publicly the background material that is contained within.

TABLE OF CONTENTS

LIST OF TABLES

LIST OF TABLES (Continued)

LIST OF FIGURES

LIST OF FIGURES (Continued)

1.0 INTRODUCTION

National policies, announced in 1977, indefinitely deferred reprocessing of spent fuel from commercial light water reactors (LWRs). Since that time, the Department of Energy (DOE) has redirected its program for the development of a disposal capability for high-level radioactive reprocessing wastes to a disposal capability for either spent fuel or the high-level wastes resulting from reprocessing. Disposal of spent fuel or reprocessing waste in mined geologic repositories is being considered as a means of providing long-term isolation of radionuclides from the biosphere.

The Office of Nuclear Waste Isolation (ONWI) serves as the lead salt repository contractor in the Civilian Radioactive Waste Management Program (CRWM) geologic repository development effort and has responsibilities for program coordination and for developing the criteria and technology for permanent disposal of high-level nuclear waste, including spent fuel elements from commercial power reactors. Of particular importance to the program is the waste form, due to the impacts it will have on all repository operations.

The basic designs of facilities for the geologic disposal of spent fuel have considered handling and disposal of unmodified spent fuel assemblies. The possibility, however, that spent fuel may be stored in a modified form prior to its emplacement in a geologic repository has been considered. Some prospective advantages of various possible modifications are:

(1) minimization of the actual volume of the spent fuel package for disposal,

(2) venting fission gases from the spent fuel prior to disposal,

(3) optimization of thermal loading in the container, and

(4) immobilization of the spent fuel prior to disposal.

An assessment of the possible modifications to the spent fuel assembly has been made in order to determine the relative advantages of each and to select a reference waste form.

The issues related to modifying spent fuel fall into three areas:

(1) the relationship between the spent fuel waste form and the repository waste package that contains it,

(2) the impact of modifying spent fuel on the packaging and emplacement operations, including technical, operating, safety, and economic considerations, and

(3) the long term, in-repository performance of the waste form.

Four alternative waste forms have been identified by the Department of Energy (DOE) as candidates for use in geologic disposal (1). Each of these alternatives has been assessed and compared to the unmodified spent fuel assembly in each of the three issue areas in order to determine the relative merits of the modification. This report presents the results of these assessments and a recommendation for the preferred waste form. The specific alternatives which have been studied are: (1) end fitting removal, (2) fission gas venting and resealing, (3) fuel disassembly and close packing of fuel pins, and (4) fuel shearing and immobilization in a solid matrix.

1.1 DEVELOPMENT OF PREFERRED WASTE FORM

ONWI initiated parallel studies in each of the three issue areas identified above to assess each alternative spent fuel form and recommend the preferred waste form. Each issue area was addressed by a different contractor. This section discusses the objectives and methods of each study.

1.1.1 Effect of Spent Fuel Waste Form Alternatives on Waste Package Design

A variety of package concepts employing the multiple engineered barrier approach are being considered in a study by the Westinghouse Advanced Energy Systems Division (AESD), which has as its goal arriving at the best package concept for waste form and repository rock type (2). Although this study is not complete and waste package concepts are still evolving, a package design concept developed by AESD was selected as the standard, or reference, package. Effects of the alternative processes on this package were studied by

AESD; factors considered were those relative to containment capability, radiation safety, handling safety, retrievability, which constitute subjects of regulatory design criteria; the study also included non-regulatory issues such as standardization of packages and package costs.

1.1.2 Assessment of the Impacts of Spent Fuel Disassembly Alternatives on the Nuclear Waste Isolation System

E. R. Johnson Associates, Inc. (JAI) conducted a study on the impacts of the four alternatives on the nuclear waste isolation system (3). Assessments were made of the impacts of each disassembly alternative on the technical, operating, safety and economic aspects of all packaging and repository operations.

Each alternative was compared to disposal of unmodified spent fuel - the Reference Process. The technical assessment of each alternative included detailed review of the packaging facility, process and equipment; the repository facility and equipment for transfer, emplacement, and retrieval; the spent fuel storage facilities and equipment; the volume and characteristics of secondary wastes produced; and the safeguards considerations.

Operational factors pertinent to each of the disassembly alternatives which were considered were the complexity of the process involved; the operational reliability of the process and auxiliary equipment; the complexity of required operating controls and their probable reliability; the extent of in-process handling of radioactive materials; and the methods for establishing suitability of product.

A three part assessment of the radiological impacts of each of the alternative disassembly methods included consideration of handling, packaging, and disposal, and evaluated near-term radiological impacts for occupational exposure, near-term radiological impacts for non-occupational exposure, and long-term radiological impacts.

An estimate was made of the cost of each disassembly alternative in terms of the incremental capital and operating costs (including packaging and

emplacement) associated with the individual disassembly alternative relative to the Reference Process; estimated costs of reference disposal facilities and the cost of operation of the Reference Process served as a base for this incremental cost determination.

A Figure of Merit (FOM) was determined for each disassembly alternative which constituted a weighted evaluation of the technical, operational, safety and economic assessments thereof, and a comparison was made of the FOM's for the disassembly alternatives to obtain the preferred alternative.

1.1.3 Assessment of Spent Fuel Form/Stabilizer Alternatives for Geologic Disposal (In-Repository Performance)

Hanford Engineering Development Laboratory (HEDL) was selected to assess the expected relative in-repository performance of each waste form in combination with its associated generic stabilizer type (4). Four fuel forms and three generic stabilizers were considered in two plenum pressure states, for a total of 13 discrete assessment waste forms. The study focused on determining whether there was any expected interaction between the waste form/stabilizer and canister which might cause premature waste package breach or waste form degradation, and on assessing the relative ability of each of the proposed waste forms to resist radionuclide migration after water intruded into a breached waste package.

A set of criteria was developed and the performance of the waste forms was assessed with respect to these criteria for the intact waste package period (first 1,000 years) and the breached waste package period (post 1,000 years). A relative ranking was given to each waste form.

1.2 SCOPE OF THIS REPORT

Although each of the three studies described in Section 1.2 was independently accomplished by the three contractors, there was interaction among all three throughout the course of the studies, with coordination provided by ONWI personnel. Each study reached certain conclusions indicated by the results of analyses and evaluations in the area of concern assigned to the respective contractor. In all cases, the conclusions led to positive

recommendations concerning preferred waste forms. The recommendations were in most cases qualified, either explicitly by the authors, or implicitly by reason of collateral information provided. This report is intended to summarize the three independent reports as each relates to the objective of recommending a spent fuel waste form for disposal; it presents the pertinent conclusions and recommendations from each independent study as well as the joint recommendation.

All of the issues which formed the bases for the three studies must be collectively assessed, as none individually provides all the input necessary for the selection of a preferred waste form. Neither are the three issues totally independent of each other. Whereas each study reached conclusions relative to the issue studied, it was desirable to put together a summary of each with the methods employed and areas studied, and combine those areas of overlap within the studies.

Section 2.0 of this report provides an Executive Summary of the three studies. Section 3.0 presents a description of the reference spent fuel assemblies and possible variants thereon, and provides basic information on the reference package employed and how it was developed. Section 4.0 describes the four alternative processes which were assessed, and provides a summary of the technical, operating, safety, and economic assessments of these alternatives in comparison to the Reference Process. Section 5.0 summarizes the evaluation of the near- and long-term performance in the repository of the alternative waste forms in the reference package. Section 6.0 provides an overall summary of the comparisons and presents a recommendation for the preferred waste form.

1.3 REFERENCES

1. U.S. Department of Energy, Statement of Position of the United States Department of Energy, p. II-139, DOE/NE-0007, Washington, D.C., April 1980.

2. Westinghouse Electric Corporation, Engineered Waste Package Conceptual Design: Defense High-Level Waste (Form 1), Commercial High-Level Waste (Form 1), and Spent Fuel (Form 2) Disposal in Salt, ONWI-438, prepared for Office of Nuclear Waste Isolation, Battelle Memorial Institute, Columbus, OH, 1983.

3. E. R. Johnson Associates, Inc., Assessment of the Impacts of Spent Fuel Disassembly Alternatives on the Nuclear Waste Isolation System, BMI/ONWI-533, prepared for Office of Nuclear Waste Isolation, Battelle Memorial Institute, Columbus, OH, 1984.

4. R. E. Einziger, D. A. Himes, and R. J. Cash, Assessment of Spent Fuel Waste Form/Stabilizer Alternatives for Geologic Disposal, Final Draft Report, TC-1913, Rev. 1, Hanford Engineering Development Laboratory, Richland, Washington, April 1981.

2.0 EXECUTIVE SUMMARY

Three studies evaluating alternatives to the disposal of intact spent fuel assemblies in a geologic repository have been completed and summarized in this report. These studies led to a joint recommendation of a preferred spent fuel waste form for disposal based on consideration of:

(1) package design and fuel/package interaction,

(2) long term, in-repository performance of the waste form, and

(3) overall process performance and costs for packaging, handling, and emplacement.

The four basic alternative waste forms which were addressed were:

1) End Fitting Removal

2) Fission Gas Venting

3) Disassembly and Close Packing

4) Shearing/Immobilization

Based on evaluation of a reference waste package employing the multibarrier design concept, it was concluded that each alternative process under consideration could be accommodated equally well by the reference package, or minor dimensional modifications thereof, and that there were no significant interactions between the waste form and package involving any of the candidate processes.

The long term performance study examined thirteen variations on the four basic waste form alternatives, taking into account such factors as support against lithostatic pressure, hazard to canister from fuel pin plenum gas, heat transfer, cladding degradation, radionuclide retention, migration barriers, and criticality. With regard to the waste form/particulate stabilizer combinations employed in the systems study, the ratings indicate that the Reference Process and Alternative Processes 1-3 are approximately equivalent on all factors except stabilizer heat transfer and resistance to lithostatic pressure; Alternative 4 is rated highest on resistance to

lithostatic pressure and between Alternative 3 and the others on heat transfer. In the Final Draft Report of the in-repository performance study, HEDL recommended the intact assemblies with a solid stabilizer as the most desirable form, with a suitable alternative being the close-packed, bundled rods with a solid stabilizer; the particulate stabilizer was down-rated primarily due to its deficiencies in resisting lithostatic pressure. No criteria were presented for acceptable performance in this function, hence no basis exists for rejecting any of the waste forms on this factor.

The process evaluation study assumed, for purposes of identifying facility requirements and operating costs, that the Reference Process and Alternatives 1-3 would employ a particulate (sand) stabilizer; this selection was made early in the program. Conceptual processes were developed for each of the alternatives; this was followed by an assessment of the impacts of the alternative processes on the packaging, handling, and emplacement system at the repository site. This assessment included a technical evaluation of the processes and equipment involved in each of the alternatives, an operational analysis, an assessment of safety/risk factors, and a comparison of the overall economics of the alternative processes with the Reference Process. Based on these assessments, Alternative 3 ranked highest in the operational analysis and in economics, which on balance, made it the first preference over the Reference Process and the other alternatives. Alternative 3 entails a more complicated packaging procedure than the Reference Process, but it permits a reduction of 65 percent (relative to the Reference Process) in the number of packages which must be handled and emplaced. The reduction in overall cost afforded by the reduction in the number of packages required and in the level of operations both above ground and in the repository resulted in the conclusion that disassembly and close packing of fuel pins is the optimum method of preparation of spent fuel for disposal. A similar reduction in package requirements and level of operation would result from pursuing Alternative 4; it was, however, ruled out by reason of its even greater complexity and major uncertainties as to the operational feasibility of several of its process steps.

3.0 PACKAGE DESIGN

The spent fuel package can be defined as everything that is placed in the repository borehole. The package design used here is based on the concept of multiple barriers to the release of radionuclides. There are a number of barriers to dispersion of the radionuclides present in spent fuel, starting with the ceramic fuel itself, and followed by the fuel cladding, 99 percent or more of which can be expected to be intact at the time the fuel leaves the reactor. The first sealed barrier beyond the spent fuel itself is the canister into which the fuel assemblies are loaded. The canister also contains a stabilizer in order to assist in providing internal support against lithostatic pressure. The second external barrier is the borehole liner; the liner is separated from the canister by sand filler and from the borehole by bentonite backfill. A shield plug above the fuel assembly canister provides radiation protection in the vertical direction, and the liner is sealed by a welded cover plate above the shield plug.

This section will describe the fuel to be packaged, the package design constraints, the details of the waste package, and the influence of waste form on package costs.

3.1 DESCRIPTION OF FUEL TO BE PACKAGED

Commercial light water reactor (LWR) spent fuel discharged from either boiling water reactors (BWR) or pressurized water reactors (PWR) is contained in fuel assemblies designed and fabricated by:

General Electric - BWR fuel,
Westinghouse - PWR fuel,
Combustion Engineering - PWR fuel,
Babcock and Wilcox - PWR fuel, or
Exxon Nuclear - BWR and PWR fuel.

The spent fuel will typically have been out of reactor 10 years or more, although minor quantities may be only 3 years out of reactor. Irradiation exposures (burnup) of 27,500 and 33,000 MWD/MTU for BWR and PWR fuel, respectively, are anticipated.

Spent fuel assemblies consist of end fittings, spacers, and support hardware in addition to fuel pins containing the actual spent fuel. Ancillary components represent a significant percentage of the assembly volume, and the end fittings significantly extend the nonfueled length of the assemblies. The primary differences in fuel assembly configuration are those inherent in the BWR and PWR fuel. Minor configuration variations within each fuel type are attributable to the design philosophy of the various fabricators, and to the use of shorter assemblies in the earlier, lower-power reactors. The current 1,000 to 1,150 MWe LWR fuel designs have essentially become standardized concepts, and are unique to each fabricator.

For the purposes of this study, the Westinghouse 17x17 and the General Electric BWR-6 fuel assemblies have been selected as reference designs of PWR and BWR fuel. The minor variations from the designs of earlier fuel assemblies or from the designs of other fabricators are not regarded as significant to this study. The reference designs, along with these variations, are described in Sections 3.1.1 and 3.1.2. The physical characteristics of the reference fuel designs are given in Table 3-1.

3.1.1 PWR Fuel Assemblies

The Westinghouse standard 17x17 fuel assembly design contains 264 fuel pins, 24 guide tubes, and one instrumentation tube in a 17x17 array. Guide tubes provide axial structural support between the upper and lower end fittings and fix the spacers at their vertical positions. The guide tubes are fastened to the lower fitting by means of cap screws which pass through the end fitting web and through sleeves (which are brazed to the lower spacer and serve to fix it in position), engaging a threaded insert in the guide tube. The cap screws are secured to the fitting by welded pins. The upper end fitting is welded to sleeves which are brazed to the top spacer and are fixed on the guide tubes. Fuel pins are not in direct contact with either end fitting. Lateral fuel pin spacing is maintained by the spacer assemblies and axial fuel pin positioning is achieved by spring pressure applied to the pins

TABLE 3-1

PHYSICAL CHARACTERISTICS OF REFERENCE LWR FUEL ASSEMBLIES

Reactor Type	BWR	PWR
Fuel Designer	GE	Westinghouse
Fuel Pin Array	8x8	17x17
Fuel Pin Bundle Assembly		
Overall Length, m	4.470	4.058
Maximum Envelope, cm^2	$(13.9)^2$	$(21.4)^2$
Fuel Pin Pitch, cm	1.63 (b)	1.26 (b)
Number of Fueled Pins	62 (c)	264 (f)
Integral Poison Pins	yes (c)	no (e)
Weight - Fuel Bundle, kg	275.7	665
Weight - UO_2+Gd_2O_3, kg	208.0	N/A
Weight - UO_2, kg	207.5 (d)	523
Weight, U, kg	183.3	461
Weight, Zirconium Alloy, kg	57.9	127
Weight, Other Alloys, kg	9.77	15
Weight, Total Metals, kg	67.7	142
Material - Fuel Clad	Zirc-2 (b)	Zirc-4 (b)
Material - Poison Clad	N/A	304L (f)
Material - Guide Tube	N/A	Zircaloy
Material - Water Rod	Zirc-2 (e)	N/A
Material - End Fittings	SS	304 (f)
Material - Spacers	Zircaloy	Inconel
Material - Burnable Poison	Gd_2O_3 in UO_2 (c)	Borosilicate (f)
Material - Control Elements	B_4C (c)	Ag-In-Cd (f)
Fuel Pin Assembly		
Overall Length, m	4.064	3.84
Active Fuel Length, m	3.759	3.658
Pin Diameter, mm	12.27 (c)	9.5
Clad Wall Thickness, mm	0.81 (c)	0.57 (b)
Pellet Diameter, mm	10.41 (c)	8.19 (b)
Radial Gap, mm	0.12 (g)	0.084 (b)

N/A = Not Applicable

(a) Bechtel National, Inc., An Assessment of LWR Spent Fuel Disposal Options, ONWI-39, Vol. 3, p A-14, July 1979

(b) American Nuclear Society, American National Standard (Proposed) - Design Criteria for an Independent Spent Fuel Storage Installation (Water Pool Type), ANSI/ANS-57.7, Appendix G, p G-2, November 1978

(c) F. D. Judge, J. Jacobsen, D. R. Wilkins, J. B. Carr and S. R. More, Latest BWR Designed for Improved Operation, Nuclear Engineering International, Vol. 25, pp 37-38 and 41-42, September 1980

(d) Included 0.5 kg allowance for Gd_2O_3 as in (c)

(e) W. B. Weihermiller and G. S. Allison, LWR Nuclear Fuel Bundle Data for Use in Fuel Bundle Handling - Topical Report, PNL-2575, Battelle Pacific Northwest Laboratory, pp 25-29, September 1979

(f) Portland General Electric Company, Final Safety Analysis Report, Amendment 13, Table 4.1-1 and pp 4.2-43-44, May 1974

(g) Calculated from data of (c)

by spacer assembly springs. The guide tubes are capable of accepting either burnable poison clusters or control rod clusters. However, the fuel assembly does not contain either poison or control rods. The physical characteristics of the standard 17x17 fuel assembly were given in Table 3-1. The fuel assembly configuration is shown in Figure 3-1 and the details of removable and nonremovable fuel pins are shown in Figure 3-2. Removable fuel pins replace some of the standard pins in experimental fuel assemblies, where it is desired to remove fuel pins for detailed examination during the burnup cycle. There are certain other differences in structural detail of the top end fitting on these assemblies, but neither these nor the removable pin configuration affect the handling of these assemblies in the packaging operations.

In addition to the Westinghouse 17x17 fuel assembly, PWR fuel assemblies have been fabricated in a variety of fuel pin arrays as shown below:

Westinghouse - 14x14, 15x15, and 17x17 XL
Combustion Engineering - 14x14 and 16x16, and
Babcock and Wilcox - 14x14, 15x15, and 17x17.

Exxon Nuclear fabricates PWR fuel assemblies which essentially conform to the original designer's parameters. Tables 3-2 and 3-3 provide the available dimensional and physical characteristics of the various fuel assembly designs.

Westinghouse and Babcock & Wilcox utilize the integral guide tube method (described in the reference design) of accommodating burnable poison and control rods. Combustion Engineering also utilizes guide tubes as structural elements of the assembly, but the guide tubes are designed to accommodate only control element assemblies. The Combustion Engineering design incorporates poison pins, as required, directly into the fuel pin array. A typical Combustion Engineering 14x14 design contains, in addition to 5 guide tubes, an inventory of 176 pins of which 168, 172, or 176 will contain fuel, the balance being burnable poison pins.

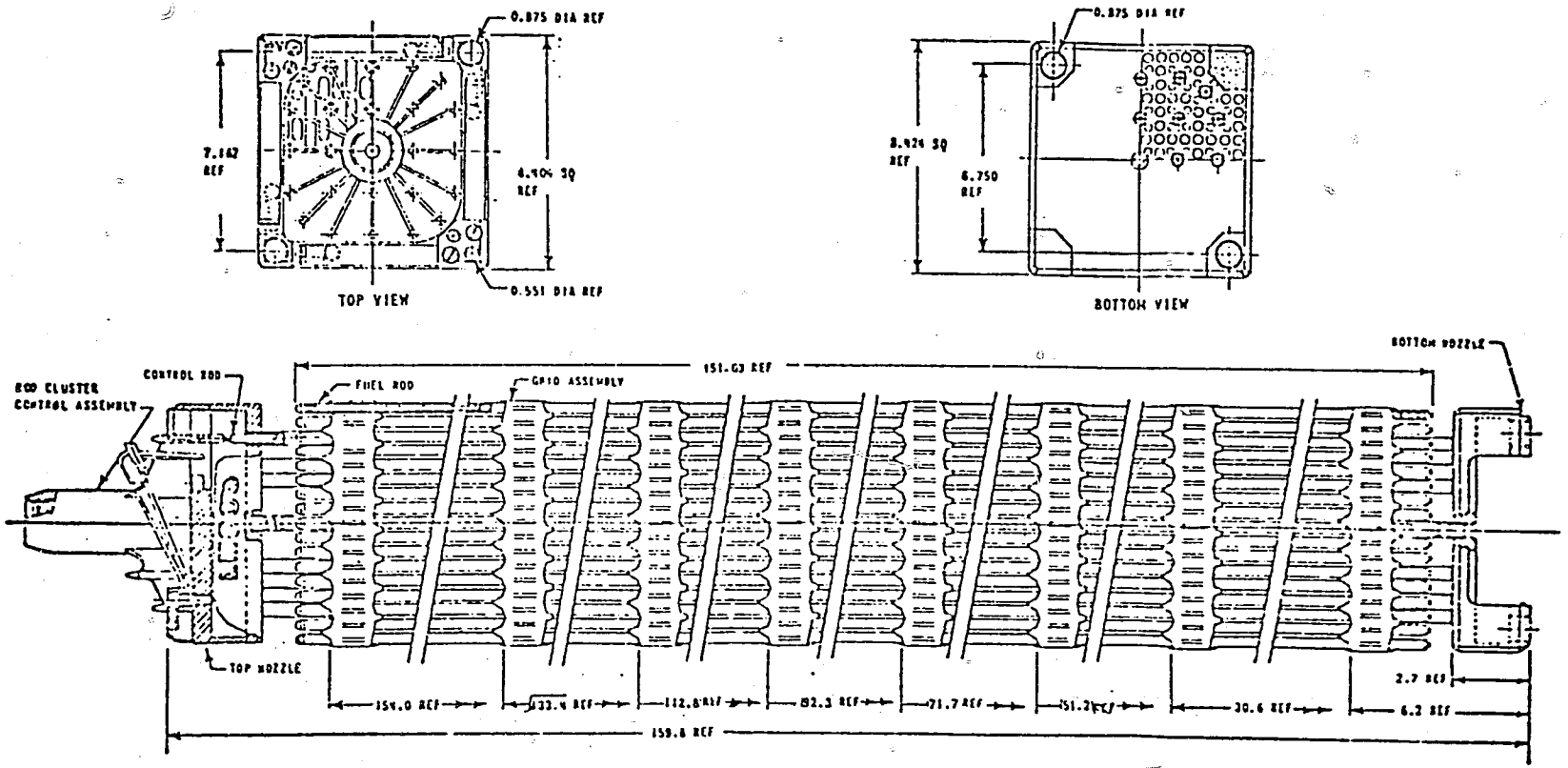

FIGURE 3-1

TYPICAL PRESSURIZED WATER REACTOR (PWR)

FUEL ASSEMBLY

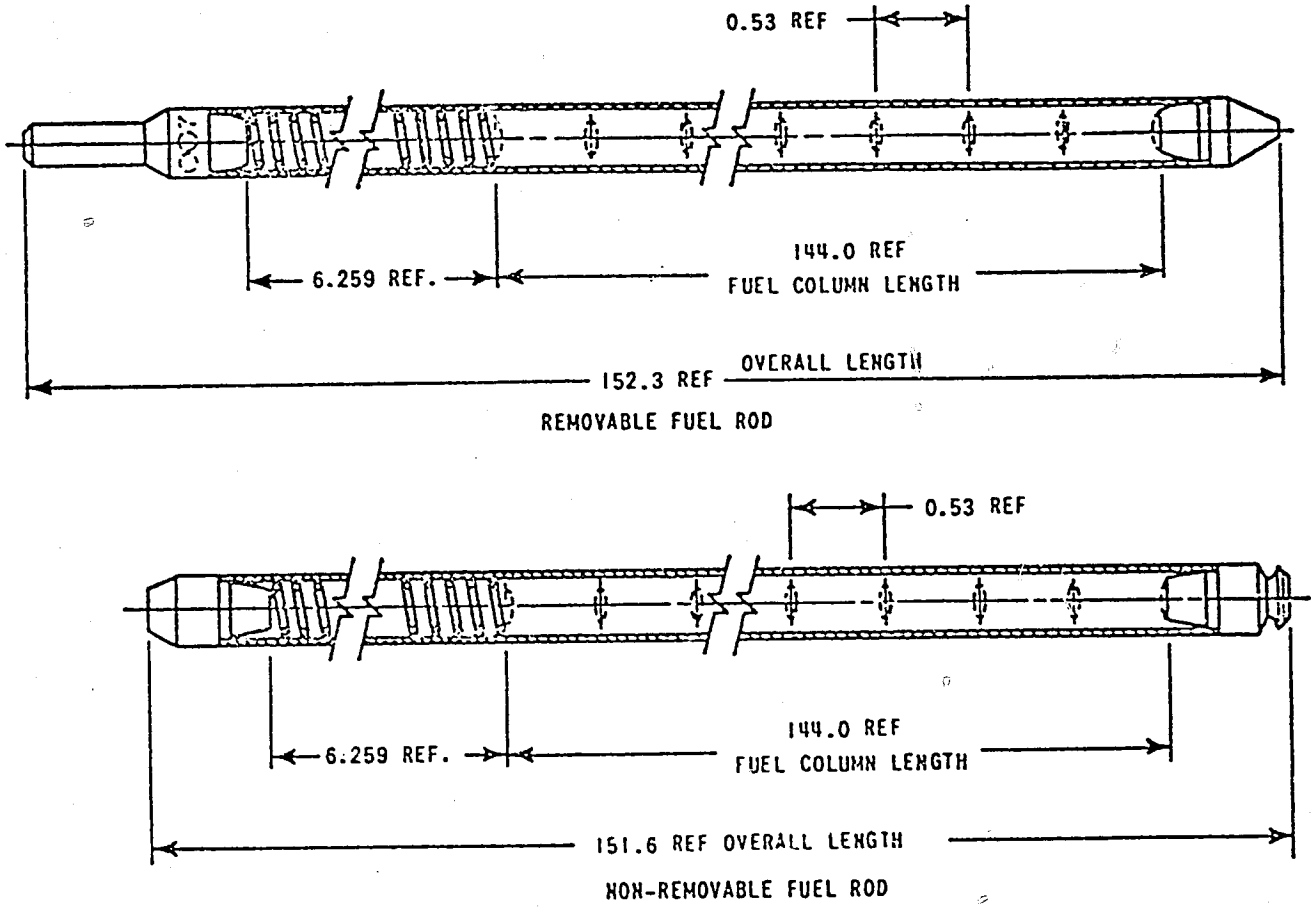

FIGURE 3-2

DETAIL OF PWR REMOVABLE AND NONREMOVABLE FUEL PINS

TABLE 3-2

PHYSICAL CHARACTERISTICS OF TYPICAL WESTINGHOUSE FUEL ASSEMBLIES

(PWR)

Fuel Designer	W	W	W	W	W	W	W	W	W
Fuel Pin Array	17x17	17x17	17x17	17x17	17x17	17x17	14x14	14x14	14x14
Designation	nr	nr	nr	nr	nr	nr	nr	nr	nr
Specific Reactor	nr	nr	nr	nr	Trojan	nr	nr	nr	nr
Source of Information	(a)	(b)	(a)	(c)	(d)	(c)	(e)	(c)	(c)
Fuel Bundle Assembly									
Overall Length, m	4.058	4.1	4.1	4.064	4.059	3.055	nr	4.064	3.48
Maximum Envelope, cm^2	$(21.4)^2$	$(21.4)^2$	$(21.4)^2$	$(21.4)^2$	$(21.4)^2$	$(21.4)^2$	$(15.7)^2$	$(19.72)^2$	$(19.72)^2$
Fuel Pin Pitch, cm	nr	nr	1.26	1.26	1.26	1.26	1.41	1.44	1.41
Number of Fueled Pins	nr	nr	264	nr	264	nr	179	nr	nr
Integral Poison Pins	nr	nr	none	nr	none	nr	nr	nr	nr
Weight-Fuel Bundle, kg	665	670	665	nr	nr	nr	nr	nr	nr
Weight-$UO_2Gd_2O_3$, kg	--	--	--	--	--	--	--	--	--
Weight-UO_2, kg	523	525	nr	nr	523.5	nr	nr	nr	nr
Weight-U, kg	461	460	nr	nr	461.5	nr	nr	nr	nr
Weight-Zirconium Alloy, kg	127	130	nr	nr	119.7	nr	nr	nr	nr
Weight-Other Alloys, kg	15	16	nr	nr	nr	nr	nr	nr	nr
Weight-Total Metals, kg	142	145	nr	nr	nr	nr	nr	nr	nr
Material-Fuel Clad	nr	SS	Zirc-4	Zirc-4	Zirc-4	Zirc-4	Zirc-4	Zirc-4	SS
Material-Poison Clad	nr	nr	nr	nr	304L	nr	nr	nr	nr
Material-Guide Tube	Zirc	Zirc	Zirc-4	nr	Zirc-4	nr	nr	nr	nr
Material-End Fittings	SS	SS	304SS	nr	304	nr	nr	nr	nr
Material-Spacers	Inconel	Inconel 718	Inconel 718	nr	Inconel	nr	nr	nr	nr
Material-Burnable Poison	nr	nr	nr	nr	Borosilicate	nr	nr	nr	nr
Material-Control Elements	nr	nr	nr	nr	Ag-In-Cd	nr	nr	nr	nr
Fuel Pin Assembly									
Overall Length, m	3.84	3.8	3.85-3.87	nr	3.851	nr	3.87	nr	nr
Active Fuel Length, m	3.658	3.7	3.658	3.658	3.658	4.166	3.658	3.658	3.048
Pin Diameter, mm	9.5	9.5	9.5	9.5	9.5	9.5	10.72	10.72	10.72
Clad Wall Thickness, mm	nr	nr	0.57	0.57	0.57	0.57	0.617	0.617	0.419
Pellet Diameter, mm	nr	nr	nr	8.19	8.19	8.19	nr	9.29	9.74
Radial Gap, mm	nr	nr	nr	0.084	0.084	0.084	nr	0.097	0.071
Pellet Densi , % theo.	nr	nr	nr	nr	95	nr	nr	nr	nr

nr = not reported N/A = not applicable SS = Stainless steel Zirc = zirconium alloy not specified

(a) Bechtel National, Inc., An Assessment of LWR Spent Fuel Disposal Options, ONWI-39, Vol. 3, p A-14, July 1979

(b) Department of Energy, Statement of Position of the United States Department of Energy, DOE/NE-007, Table IV-9 and Table IV-10, April 1980

(c) American Nuclear Society, American National Standard (Proposed) - Design Criteria for an Independent Spent Fuel Storage Installation (Water Pool Type), ANSI/ANS-../.., Appendix C, p C-2, November 1978

(d) F. D. Judge, J. Jacobsen, D. R. Wilkins, J. B. Carr and S. R. Mord, Latest BWR Designed for Improved Operation, Nuclear Engineering International, Vol. 25, pp 37-38 and 41-42, September 1980

(e) W. B. Wetherwiller and G. S. Allison, LWR Nuclear Fuel Bundle Data for Use in Fuel Bundle Handling - Topical Report, PNL-2575, Battelle Pacific Northwest Laboratory, pp 25-29, September 1979

TABLE 3-3

PHYSICAL CHARACTERISTICS OF TYPICAL COMBUSTION ENGINEERING AND BABCOCK AND WILCOX FUEL ASSEMBLIES

(PWR)

Fuel Designer	CE	CE	CE	CE	CE	CE	B&W	B&W	B&W	B&W
Fuel Pin Array	16x16	16x16	16x16	14x14	14x14	14x14	17x17	17x17	15x15	15x15
Designation	nr	nr	nr	nr	nr	nr	nr	nr	nr	nr
Specific Reactor	nr	nr	San Onofre 2	nr	nr	nr	nr	nr	nr	nr
Source of Information	(b)	(a)	(c)	(b)	(a)	(c)	(a)	(b)	(a)	(b)
Fuel Bundle Assembly										
Overall Length, m	4.49	4.456	4.406	3.716-3.994	4.166	3.994	4.216	4.208	4.216	4.206
Maximum Envelope, cm^2	$(20.68)^2$	$(20.78)^2$	$(20.90)^2$	$(20.68)^2$	$(20.78)^2$	$(20.90)^2$	$(21.68)^2$	$(21.68)^2$	$(21.68)^2$	$(21.68)^2$
Fuel Pin Pitch, cm	1.29	1.29	1.29	1.47	1.47	1.47	1.27	1.28	1.44	1.44
Number of Fueled Pins	236	nr	220-236	176	nr	168-176	nr	264	nr	208
Integral Poison Pins	nr	nr	yes (e)	nr	nr	yes (e)	nr	nr	nr	nr
Weight-Fuel Bundle, kg	650	nr	681	581	nr	545.7	nr	683	[illegible]	689
Weight-$UO_2Gd_2O_3$, kg	--	--	--	--	--	--	--	--	--	--
Weight-UO_2, kg	nr	nr	451.0-483.8	nr	nr	411.6-431.2	nr	nr	nr	nr
Weight-U, kg	nr	nr	397.6-426.5	nr	nr	362.8-380.1	nr	nr	nr	nr
Weight-Zirconium Alloy, kg	nr	nr	nr	nr	nr	nr	nr	nr	nr	nr
Weight-Other Alloys, kg	nr	nr	nr	nr	nr	nr	nr	nr	nr	nr
Weight-Total Metals, kg	nr	nr	nr	nr	nr	nr	nr	nr	nr	nr
Material-Fuel Clad	Zirc-4	Zirc-4	Zirc-4	Zirc-4	Zirc-4	Zirc-4	Zirc-4	Zirc-4	Zirc-4	Zirc-4
Material-Poison Clad	nr	nr	Zirc-4	nr	nr	Zirc-4	nr	nr	nr	nr
Material-Guide Tube	Zirc-4	nr	Zirc-4	nr	nr	Zirc-4	nr	Zirc-4	nr	Zirc-4
Material-End Fittings	SS	nr	308SS	nr	nr	308SS	nr	304SS	nr	304SS
Material-Spacers	Zirc-4 (d)	nr	Zirc-4 (d)	Zirc-4 (d)	nr	Zirc-4 (d)	nr	Inconel	nr	Inconel
Material-Burnable Poison	nr	B_4C/Al_2O_3	B_4C/Al_2O_3	nr	B_4C/Al_2O_3	B_4C/Al_2O_3	nr	nr	nr	nr
Material-Control Elements	nr	nr	B_4C	nr	nr	B_4C	nr	nr	nr	nr
Fuel Pin Assembly										
Overall Length, m	4.097	nr	4.099	3.48-3.73	nr	3.73	nr	nr	nr	nr
Active Fuel Length, m	3.810	3.810	3.810	3.25-3.48	3.48	3.47	3.63	3.63	3.66	3.66
Pin Diameter, mm	9.70	9.70	9.70	11.18	11.18	11.18	9.63	9.63	10.92	10.93
Clad Wall Thickness, mm	0.635	0.635	0.635	0.6660.71	0.66	0.66	0.597	0.609	0.673	0.671
Pellet Diameter, mm	nr	8.26	8.26	nr	9.64	9.65	8.21	nr	9.40	nr
Radial Gap, mm	nr	0.089	0.089	nr	0.11	0.11	0.112	nr	0.089	nr
Pellet Density, % theo.	nr	nr	95	nr	nr	95	nr	nr	nr	nr

nr = not reported N/A = not applicable SS = Stainless steel Zirc = zirconium alloy not specified

(a) American Nuclear Society, American National Standard (Proposed) - Design Criteria for an Independent Spent Fuel Storage Installation (Water Pool Type), ANSI/ANS-57.7, Appendix C, p C-2, November 1978

(b) W. B. Weihermiller and G. S. Allison, LWR Nuclear Fuel Bundle Data for Use in Fuel Bundle Handling - Topical Report, PNL-2575, Battelle Pacific Northwest Laboratory, pp 25-29, September 1979

(c) P. Ferwarda, Supervisor - QC Engineering, CE Nuclear Products Manufacturing, Windsor, Connecticut, October 24, 1980

(d) Bottom spacer Inconel

(e) Variable quantity dependent upon fuel assembly requirements

The Westinghouse 17x17XL fuel assembly is approximately one meter longer than the reference design and the Combustion Engineering 16x16 design is approximately 43 centimeters longer than the reference. Fuel pins of the 17x17XL and the CE 16x16 design are approximately 64 and 26 centimeters, respectively, longer than the reference design. The length variation may not impact upon the actual disassembly techniques, but must be considered in the establishment of storage canister length and repository emplacement cavity depth requirements, as well as in the flexibility of the encapsulation processing equipment and handling equipment. The length of the assemblies must also be considered in the vertical clearance provisions in the cells.

The variations of PWR fuel assembly design do not appear to extend to the actual fuel assembly structural elements except for the length of the guide tubes and fuel pins. The fuel pins are axially supported by spring pressure and do not contact the upper or lower end fittings.

3.1.2 BWR Fuel Assemblies

The General Electric BWR-6 and the BWR 2/5 retrofit fuel assemblies contain 62 fuel pins and 2 water rods in an 8x8 array. Eight of the 62 fueled pins double as tie rods to mechanically join the upper and lower end fittings. Tie rods are threaded into the lower fitting, passed through the upper fitting and fixed in place with lock washers and nuts. Lateral pin spacing is maintained by 7 spacer assemblies mechanically fixed in their vertical positions by locking tabs on one water rod. The fuel pins are axially supported by the lower end fitting and the upper end fitting exerts spring pressure upon fuel pins to assure proper seating. Supplementary reactivity control is provided by special pins which are included in the fueled pin inventory. These poison pins contain gadolina (Gd_2O_3) in solid solution in urania (UO_2), as sintered pellets. The physical characteristics of the BWR-6 fuel assembly were given in Table 3-1. A 8x8 fuel assembly and fuel pin are shown in Figure 3-3. The assembly shown contains one water rod whereas the BWR-6 and the BWR 2/5 retrofit assemblies contain two water rods. It should be noted that this assembly drawing and the upper tie plate drawing show the channel which would not accompany the assembly to the packaging facility.

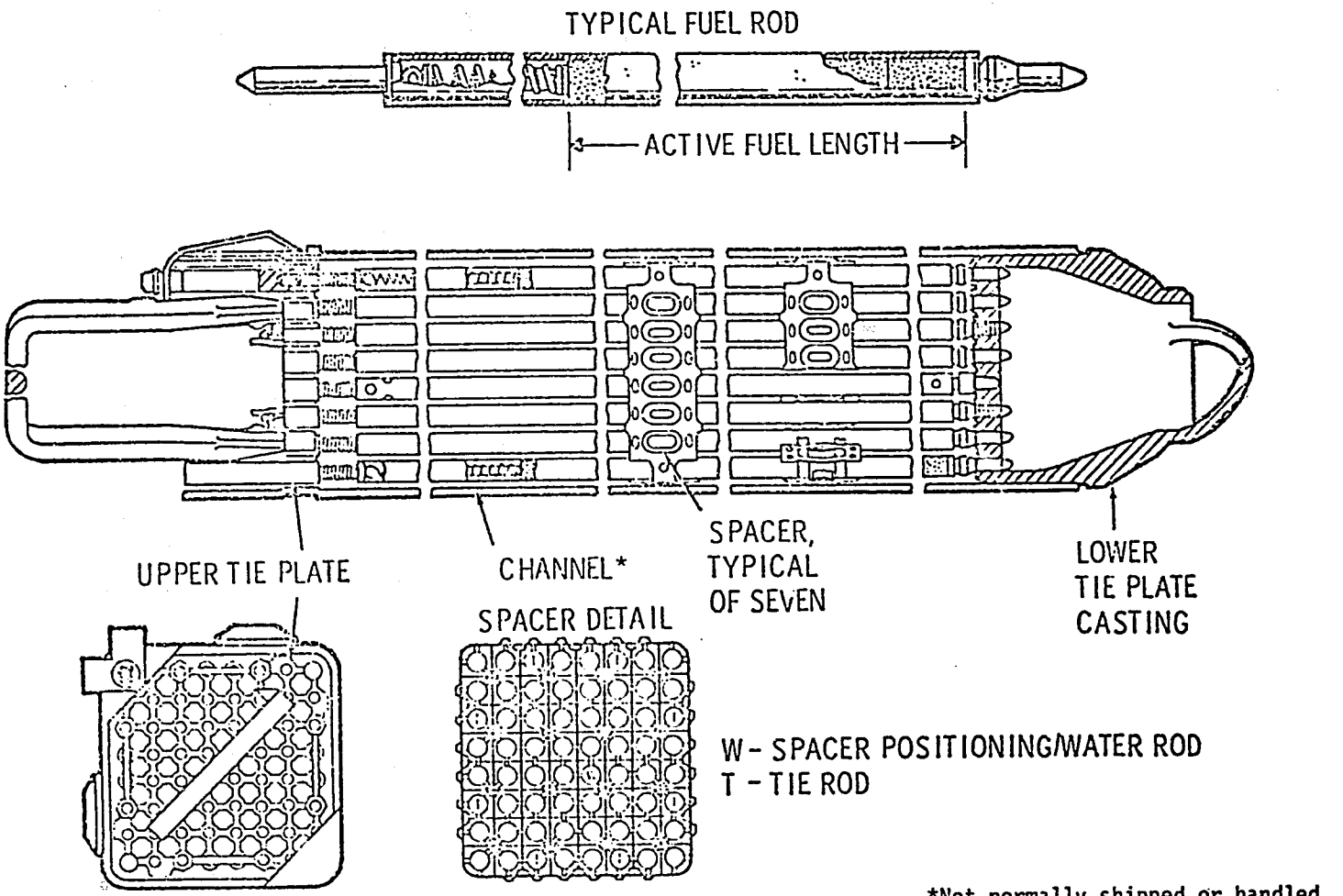

FIGURE 3-3

TYPICAL BOILING WATER REACTOR (BWR) FUEL ASSEMBLY AND FUEL PINS

Earlier General Electric fuel assemblies employed 6x6, 7x7, and a series of 8x8 pin arrays. The 6x6 design was the earliest commercial BWR fuel and was not extensively used. No dimensional data were obtained for the 6x6 design. The 7x7 pin array contains 49 fuel pins, one of which is a segmented pin which provides axial capture of the spacer assemblies. GE designed and fabricated a series of 8x8 fuel designs (designated BWR-2 through -5, BWR 2/5 Retrofit, and the BWR-6 Reference Case). Early 8x8 designs (BWR-2 through -5) contained 63 fuel pins and 1 water rod (used to axially fix the spacer assemblies). The BWR 2/5 Retrofit, as does the BWR-6, contains 62 fuel pins and 2 water rods, one of which is used to mechanically fix the spacer assemblies in the axial positions. The fuel pins in all instances are supported by the lower end fitting. Since the fuel assembly envelope for the 8x8 fuel designs are approximately equivalent to that of the 7x7 design, there are minor dimensional variations of fuel pin and pellet diameter from the 7x7 to the 8x8 design. End fitting dimensions, in all cases, appear to be approximately equivalent. Available dimensional and physical characteristics for the GE BWR fuel assemblies are shown in Table 3-4. Exxon Nuclear fabricates BWR fuel assemblies which essentially conform to the GE design parameters.

3.1.3 Impact of Process Alternatives on Package Dimensions

The Reference Process is based on the disposal of intact fuel assemblies; it is further based on canister loading of either one PWR or two BWR assemblies. The canister dimensions for the Reference Process are, therefore, determined by the fuel assembly dimensions.

The consideration of process alternatives is largely motivated by the reduction in waste package volume and the ability to vary the thermal loading which is made possible. Removal of end fittings from the fuel assemblies has no effect on the package diameter but does result in a reduction in package length. The disassembly of the fuel assembly and close packing of the individual fuel pins permit both the use of a shorter package and the loading of an increased quantity of fuel in a canister of a given diameter or the use of a smaller diameter canister. Certain geologic media

TABLE 3-4

PHYSICAL CHARACTERISTICS OF TYPICAL BWR FUEL ASSEMBLIES

Fuel Designer	GE	nr	GE	GE	GE	GE	GE	GE	GE
Fuel Pin Array	8x8	8x8	8x8	8x8	8x8	8x8	8x8	7x7	7x7
Designation	nr	nr	nr	nr	BWR 2/5R	BWR 2/5R	BWR/6	nr	nr
Specific Reactor	nr	nr	nr	nr	DAEC	NMP-1	Grand Gulf	nr	nr
Source of Information	(a)	(b)	(f)	(c)			(d)	(f)	(c)
Fuel Bundle Assembly									
Overall Length, m	4.470	4.5	4.354	4.470	4.474	4.35	nr	4.35	4.470
Maximum Envelope, cm^2	$(13.9)^2$	$(13.9)^2$	$(13.9)^2$	$(13.9)^2$	$(13.9)^2$	$(13.9)^2$	nr	$(13.37)^2$	$(13.81)^2$
Fuel Pin Pitch, cm	nr	nr	1.63	1.63	1.63	1.63	nr	1.87	1.87
Number of Fueled Pins	nr	nr	63	nr	62	62	62	49	nr
Integral Poison Pins	nr	nr	nr	nr	yes (e)	yes (e)	yes (e)	nr	nr
Weight-Fuel Bundle, kg	275.7	279	278	nr	272.7	265.9	nr	308	nr
Weight-$UO_2Gd_2O_3$, kg	nr	nr	nr	nr	207.4	200.8	nr	nr	nr
Weight-UO_2, kg	208.0	215	nr	nr	206.9	200.4	207.5	nr	nr
Weight-U, kg	183.3	190	nr	nr	182.4	176.7	nr	nr	nr
Weight-Zirconium Alloy, kg	57.9	57	nr	nr	nr	nr	nr	nr	nr
Weight-Other Alloys, kg	9.77	8	nr	nr	nr	nr	nr	nr	nr
Weight-Total Metals, kg	67.7	65	nr	nr	nr	nr	nr	nr	nr
Material-Fuel Clad	nr	Zirc 2+4	Zr-2	Zr-2	Zirc-2	Zirc-2	Zirc-2	Zirc-2	Zirc-2
Material-Poison Clad	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Material-Guide Tube	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Material-Water Rod	nr	nr	Zirc-2	nr	Zirc-2	Zirc-2	nr	N/A	nr
Material-End Fittings	SS	SS	304SS	nr	304SS	304SS	nr	304SS	nr
Material-Spacers	Zirc	Zirc	Zr-4	nr	Zirc	Zirc	nr	Zirc	nr
Material-Burnable Poison	nr	nr	nr	Gd_2O_3	Gd_2O_3 in UO_2	Gd_2O_3 in UO_2	Gd_2O_3 in UO_2	nr	Gd_2O_3
Material-Control Elements	nr	nr	nr	nr	B_4C	B_4C	B_4C	nr	nr
Fuel Pin Assembly									
Overall Length, m	4.064	4.1	nr	nr	4.10	3.95	4.064	nr	nr
Active Fuel Length, m	3.759	3.8	3.71-3.76	3.658	3.81	3.69	3.81	3.66-3.71	3.66
Pin Diameter, mm	12.52	12	12.52	12.52	12.26	12.26	12.268	14.30	14.48
Clad Wall Thickness, mm	nr	nr	0.86	0.86	0.81	0.81	0.81	0.81-0.94	0.99
Pellet Diameter, mm	nr	nr	nr	10.57	10.41	10.41	10.41	nr	12.40
Radial Gap, mm	nr	nr	nr	0.11	0.12	0.12	nr	nr	0.14
Pellet Density, % theo.	nr	nr	nr	nr	95	95	nr	nr	nr

nr = not reported N/A = not applicable SS = Stainless steel Zirc = zirconium alloy not specified

(a) Bechtel National, Inc., An Assessment of LWR Spent Fuel Disposal Options, ONWI-39, Vol. 3, p A-14, July 1979.

(b) Department of Energy, Statement of Position of the United States Department of Energy, DOE/NE-0007, Table IV-9 and Table IV-10, April 1980.

(c) American Nuclear Society, American National Standard (Proposed) - Design Criteria for an Independent Spent Fuel Storage Installation (Water Pool Type), ANSI/ANS-57.7, Appendix G, p G-2, November 1978.

(d) F. B. Judge, J. Jacobsen, D. R. Wilkins, J. B. Carr and S. R. More, Latest BWR Designed for Improved Operation, Nuclear Engineering International, Vol. 25, pp. 37-38 and 41-42, September 1980.

(e) Gd_2O_3 in UO_2 rods in array.

(f) W. B. Wetherwiller and C. S. Allison, LWR Nuclear Fuel Bundle Data for Use in Fuel Bundle Handling - Topical Report, PNL-2575, Battelle Pacific Northwest Laboratory, pp 25-29, September 1979.

may require thermal loadings which do not correspond to integral multiples of a full assembly, i.e., 2.5 assemblies. Disassembly of the fuel would allow a predetermined thermal load to be accommodated. Not only may the canister diameter be smaller, but the choice of a diameter is, with close packed pins, not constrained by fuel assembly dimensions. For the final process alternative in which fuel pins are sheared and immobilized in a solid matrix within the canister, there are no constraints on either canister diameter, length, or thermal load.

3.1.4 Effects of Irradiation on Fuel

Examinations of irradiated fuel have indicated that changes occur which could affect disassembly and packaging of consolidated pins. The effects of radiation on the fuel pins include changes in length, diameter, and ovality; the fuel pins may also become bowed; buildup of fission products within the fuel pellets tends to produce fuel irradiation swelling which, however, may be counteracted by fuel densification at service temperature; creepdown of the cladding and interaction of the cladding with the fuel pellets can cause ridging to occur; radiation also increases cladding tensile strength and reduces ductility. In addition to changes of the fuel pins, changes of the grid spring and holddown spring characteristics have been reported. Crud deposition on the fuel assemblies may also affect disassembly operations.

Information concerning the condition of irradiated fuel pins from Westinghouse 15X15 fuel assemblies and from second cycle Babcock and Wilcox (B&W) assemblies has been obtained, and provides an indication of the magnitude of changes caused by the reactor environment. Further information has been obtained from organizations which have disassembled spent fuel and from studies of spent fuel disposal.

B&W reports elongation of fuel pins by approximately 0.3 percent (slightly more than 10 mm). Their data indicates a trend of increasing fuel pin growth with increasing neutron fluence. Pin diameter typically decreased from the nominal pre-irradiation value of 10.9 mm by a creepdown of 0.07 mm.

An axial variation in creepdown was noted, with a minimum average creepdown of 0.05 mm observed near the top of the fuel column and a maximum average creepdown of 0.08 mm occurring near the bottom of the fuel column. Average maximum ovality values ranged from 0.031 mm to 0.36 mm. Of the 35 pins examined by profilometer, 12 exhibited evidence of ridging. Typical peak height of the ridges ranged from 0.019 mm to 0.025 mm. Examination of fuel assembly holddown spring force after irradiation showed minimal changes compared to the pre-irradiation values (1); though not directly relatable to the spring forces exerted by spacer springs, it provides some confidence that irradiation will not markedly change the restraining force of the spacer springs.

Examination of irradiated fuel pins of various burnup from Westinghouse 15x15 assemblies showed average length increases of 1.4 to 1.5 cm, from a pre-irradiation length of 386.0 cm or approximately 0.4 percent. Average overall diameter reductions of 0.08 mm from the end caps to the center of the pins were reported. Ovalities generally ranged from 0.025 to 0.051 mm, with extremes of up to 0.178 mm. The pins exhibited a degree of ridging, with ridge heights increasing in the center of the pins, generally averaging 0.025 mm, with a maximum ridge height of 0.05 mm (2). These data are in substantial agreement with the B&W data reviewed above.

A feasibility study concerning close packing of PWR fuel pins suggests that bowing and deposits built up on the surface of fuel pins may limit the degree of practical close packing which can be achieved. Examination of spent fuel pins was reported to indicate an average bowing of 0.588 mm, with a maximum of 1.27 mm (3).

The results to date of disassembly of irradiated spent fuel indicate that the irradiation will have little or no effect on the fuel pins that would cause interference with the pin pulling process, and little or no indication of bowing which would interfere significantly with close packing of pins (3).

3.2 PACKAGE DESIGN CONSTRAINTS

The package design concept for this study was selected from concepts being evaluated in a package design project being conducted by the Westinghouse Advanced Energy Systems Division (AESD), and is described in Reference (4). The package is intended for use in waste disposal in a repository borehole and employs the multiple engineered barrier approach. This section and the following sections summarize the study by Westinghouse AESD and the results which pertain to the waste form evaluation program.

Waste package design constraints applicable to the AESD package design are primarily those set forth in the federal regulations (5), although cost considerations and the extent of use of strategic materials must also be taken into account. Both the regulatory and non-regulatory constraints are discussed in this section.

3.2.1 Regulatory Requirements

3.2.1.1 Containment

The waste package design must provide reasonable assurance that the package will contain the radionuclides for a period of 1000 years in the salt environment of the Reference Repository. This is the time period during which radioactive decay is dominated by the relatively short-lived fission and activation products and there is significant thermal energy production. The ability of the waste package to provide such containment is primarily a function of the corrosion resistance of its components. Appendix A presents a review of the current knowledge of the corrosion of alloys of interest in oxic and anoxic salt waters and brine.

3.2.1.2 Retrievability

The need for retrievability occurs due to uncertainties associated with the deep geologic disposal of radioactive waste. As a result, it is conceivable that retrieval of a portion of the package containing the waste form may be desired early in the post-emplacement period to correct a package defect, to inspect packages, etc. Therefore, the package must be designed to facilitate that retrieval during the retrievability period which, at the present time, is assumed to be 50 years (5).

3.2.1.3 Safety/Risk Considerations

Both radiological and non-radiological safety considerations are specified by the regulations. The waste package has a primary function in radiological safety. The components of the waste package must prevent dispersion of radioactive material during handling and emplacement operations. Moreover, after emplacement, the total package must provide attenuation of the radiation from the spent fuel to levels which permit normal operations. Design values for radiation levels are based on meeting the requirements of 10CFR20.1(c) which requires that radiation exposures be "as low as reasonably achievable".

Criticality safety prior to emplacement is assured by careful control of potential moderator and safe geometry of the spent fuel. In the long term after emplacement, the waste package serves to prevent water intrusion and the redistribution of fissionable materials in the spent fuel.

Handling and transportation safety are to be achieved by waste package design. Acceleration and impact loads, specified in the federal regulations defining sealed sources must be considered in the design of the retrievable portion of the waste package.

3.2.1.4 Nuclear Material Control and Accountability

This requirement is intended to provide a permanent means of identifying and tracing each waste package and its contents during the repository operations period. The canister must have unique markings to assure traceability for the package components and contents.

3.2.2 Non-Regulatory Considerations

3.2.2.1 Cost

While minimum cost is a design objective, the cost of the waste package is severely constrained by the 1000 year containment requirement.

3.2.2.2 Materials and Processes

To expedite package implementation, the waste package design seeks to employ materials, parts and fabrication/assembly processes that are known and proven and for which extensive development efforts are not required. In addition, the design should avoid using critical materials and materials that could have substantial and attractive salvage values at some later date. Needless to say, this requirement is not entirely consistent with the requirement of 1000 year package life.

3.2.2.3 Standardization

Cost considerations will indicate the necessity for the most space-efficient package design. This consideration, in connection with the desirability of minimizing fabrication costs, leads to the design of a package which can be used with minimal dimensional change for the spectrum of spent fuel types described in Section 3.1.

3.3 DESCRIPTION OF REFERENCE PACKAGE

The reference package is shown in Figure 3-4; the dimensions shown are for the disposal of one PWR or two BWR intact fuel assemblies. The waste package components include the intact fuel assemblies, a sand stabilizer within the canister, a 0.64 cm thick titanium or titanium alloy canister with welded closure, a sand fill between canister and liner, a 2.54 cm thick sealed Inconel liner, and 30.5 cm of bentonite backfill surrounding the liner. The function, design, and basis of selection of each of these components are discussed in the following sections. Each disassembly alternative will employ this reference package concept, modified dimensionally to the requirements of that alternative.

3.3.1 Canister

Details of the Reference Process canisters and inner guide cages for PWR and BWR fuel assemblies are shown in Figures 3-5 and 3-6. The canister is fabricated of 0.64 cm titanium as shown.

The canister is the first sealed containment barrier. It performs the primary function of containing the radioactive material for the 1000 year design life of the waste package. In addition, when combined with the waste form and any other internal component, it becomes the retrievable package. Further, the canister may have to function as a surface storage container for the waste form during periods of lag storage both before emplacement and/or following retrieval. It must survive specified dropping accidents without loss of containment or handling ability and it must maintain its corrosion barrier performance under geology-induced crushing loads. However, the thickness of the canister is chosen to provide the requisite life against corrosion and to provide adequate handling strength. The resistance to lithostatic crushing pressures is to be provided by the presence of the spent fuel and to some extent by the particulate stabilizer; external resistance to lithostatic pressure is provided by the liner. The corrosion rates of titanium and titanium alloys in brine are discussed in Appendix A.

The fabrication of the canister and its internal guide cage present no unusual problems. Each will have to be welded so as to preserve straightness but with proper fixtures and weld sequencing this is not expected to be difficult.

3.3.2 Liner

The liner, composed of a nickel alloy such as Inconel 600 or 625, would be preplaced in the borehole to serve as a hole liner. In that capacity the liner must resist crushing loads during the retrievability period to enable waste form retrieval.

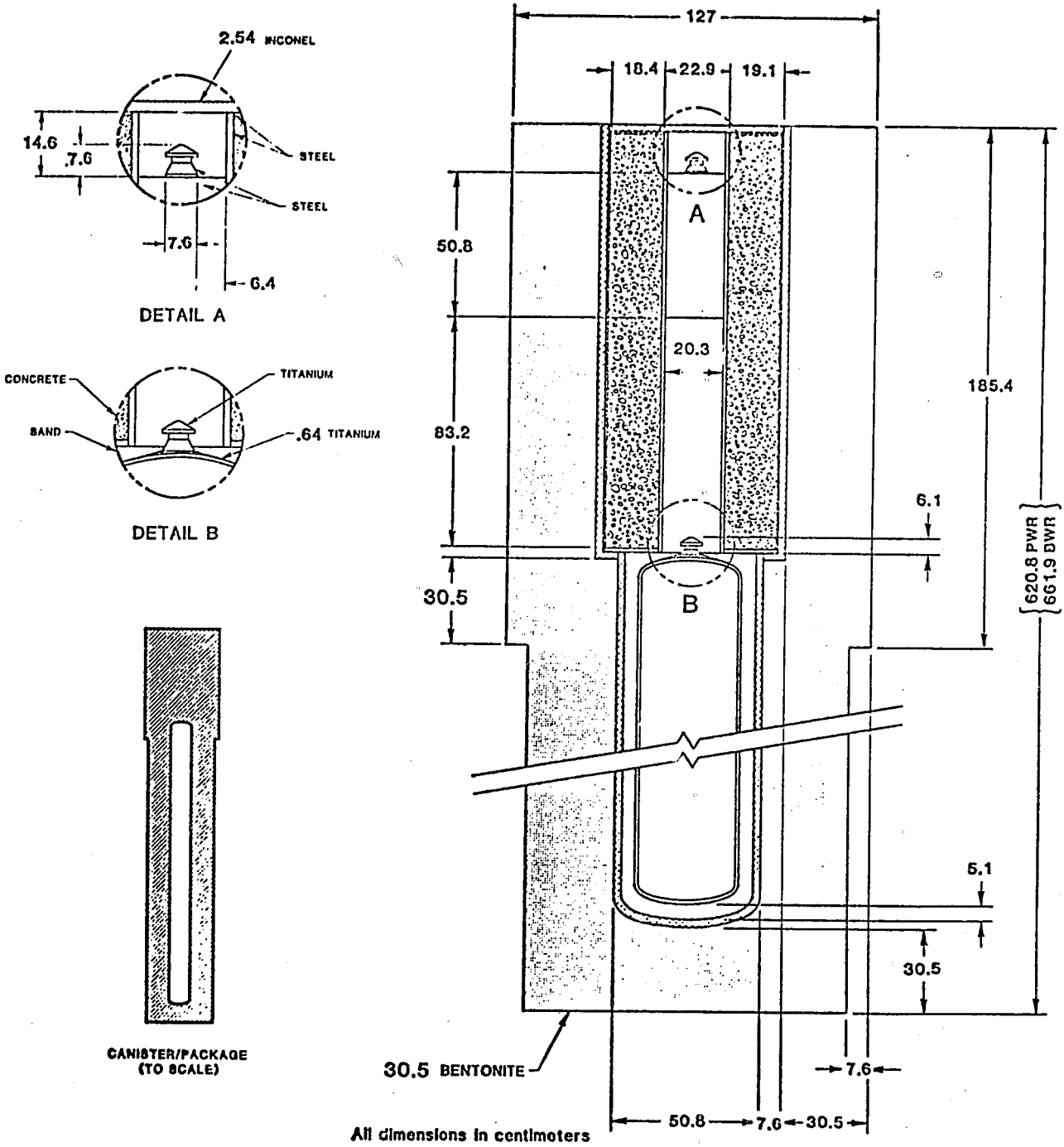

FIGURE 3-4

DETAIL OF REFERENCE PROCESS SPENT FUEL PACKAGE

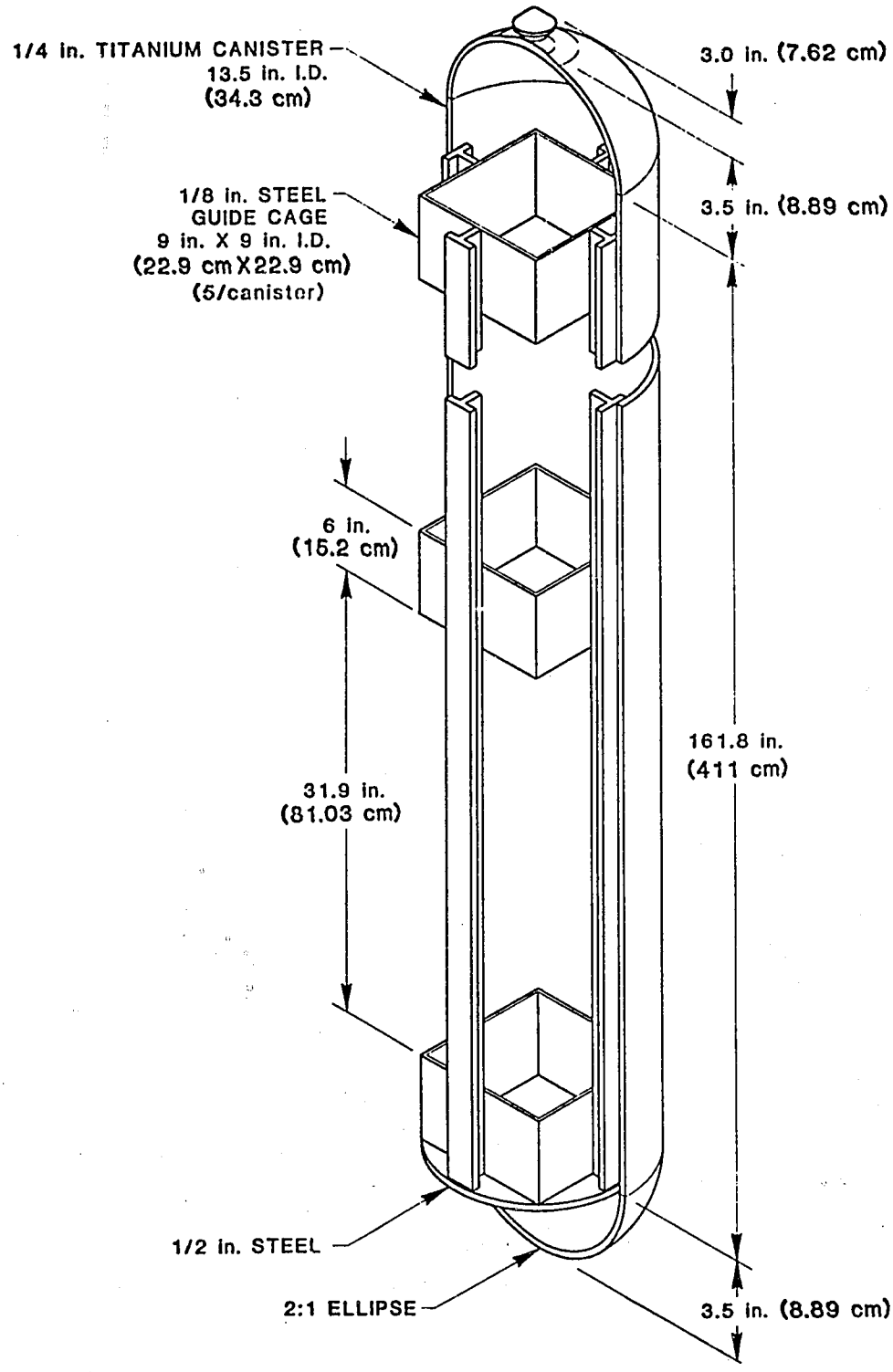

FIGURE 3-5

DETAIL OF REFERENCE PROCESS PWR SPENT FUEL CANISTER ASSEMBLY

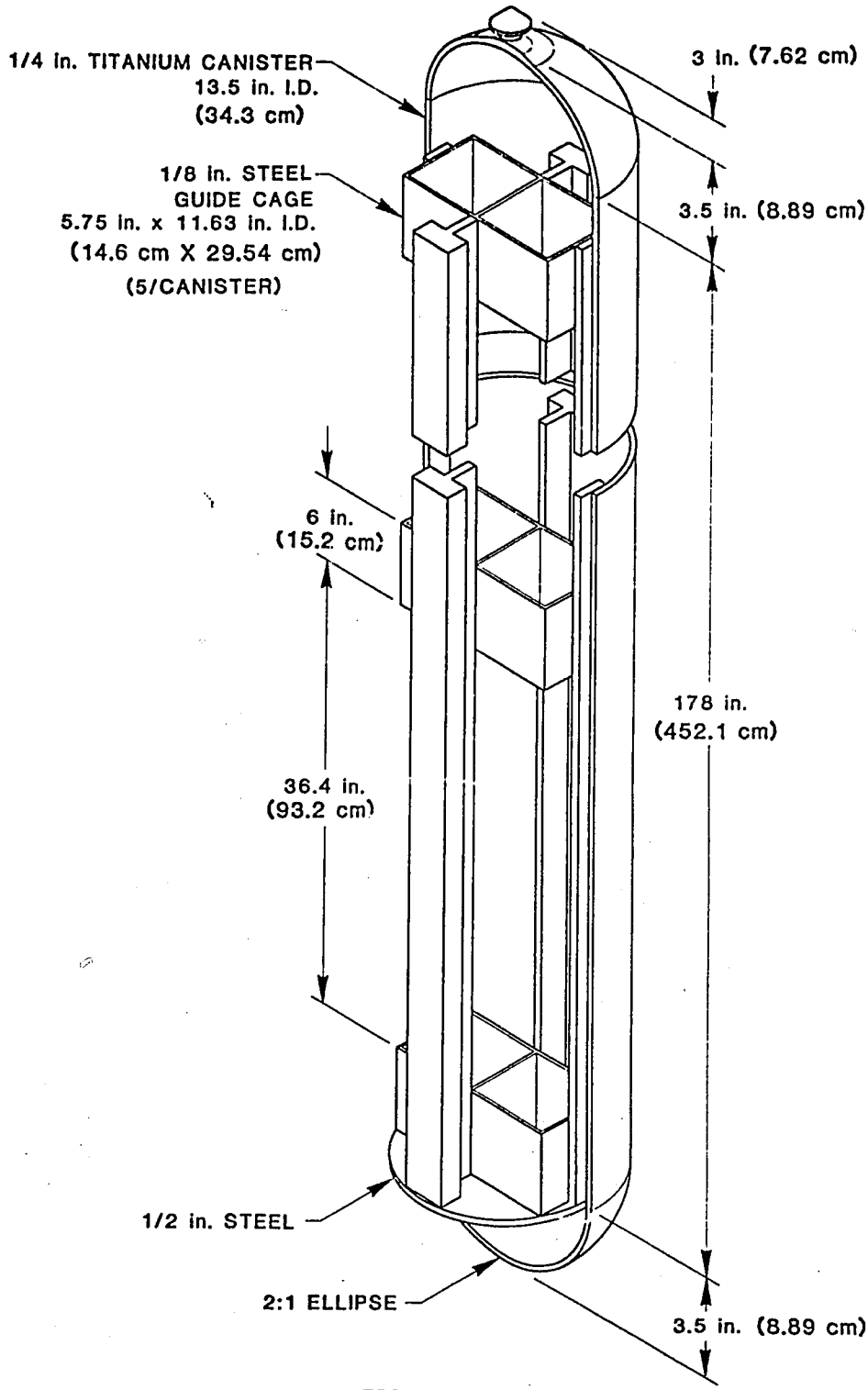

FIGURE 3-6

DETAIL OF REFERENCE PROCESS BWR SPENT FUEL CANISTER ASSEMBLY

Two liner conditions have been analyzed. With a "supported liner", the filler would be present between the canister and liner to provide crush resistance. In that case, the liner thickness could be independent of liner diameter and would be sized to provide satisfactory corrosion resistance and handling stability. An "unsupported liner" must have adequate thickness to resist crushing loads on its own since the filler gap between canister and liner is air-filled. Now the liner thickness must increase with liner diameter and possibilities such as roll-bonding a corrosion resistant nickel alloy to a thicker and less expensive backing liner must be considered.

In either the supported or unsupported liner, the thickness of the Inconel was chosen for corrosion resistance and was constant at 2.54 cm. In the unsupported case, the Inconel was backed with carbon steel and the thickness of the steel was chosen to resist the lithostatic pressure. The use of the unsupported liner was found to increase the cost of the reference package by approximately one-third. Subsequent analyses and cost estimates were, therefore, based on the use of a supported liner. Crushing resistance in this concept is enhanced by use of a stabilizer in the canister and of a filler in the radial gap between the canister and liner.

3.3.3 Stabilizer and Filler

3.3.3.1 Function of Stabilizer and Filler

Both stabilizer and filler have the primary function of providing resistance to lithostatic pressure; the filler provides support to the liner, while the stabilizer supports the canister. Secondary functions of the stabilizer include the provision of support to the fuel pins against handling stresses or stresses created by seismic events in the repository, provision of an additional barrier against radionuclide transport, prevention of redistribution of fissionable nuclides into a critical mass, and modification of the thermal environment in the canister. These functions could be accomplished by either a solid or a specially designed particulate stabilizer. Where the spent fuel cladding remains intact, that is, in every case except Alternative 4, the functions should be accomplished with minimal degradation of the spent fuel cladding.

3.3.3.2 Criteria for Selection of Stabilizer

There are two principal criteria for the selection of the stabilizer:

1. How well does it perform the required functions?
2. What complexities does it introduce into processing of the waste form?

With respect to the first of these criteria, it is patently obvious that a solid stabilizer is to be preferred over a particulate stabilizer. The solid would provide greater support, better heat transfer, and, if its corrosion resistance were adequate, another barrier against radionuclide migration. A particulate stabilizer could, with lesser effectiveness, perform the same functions. Einziger, et al, have recommended the solid stabilizer based on studies which are discussed in Section 5.0 (6); their recommendation appears to be based almost entirely on the performance of the solid in resisting lithostatic pressures. Although they have attempted to quantify the defects of a particular stabilizer in resisting lithostatic pressures, they essentially conclude that, lacking any criterion of acceptability/unacceptability of performance, the particulate stabilizer cannot be adjudged unacceptable (see Section 5.0).

The particulate stabilizer is much to be preferred, of course, in respect to the second criterion noted above. Filling the canister with stabilizer may be accomplished at ambient temperatures and control of the filling may be expected to be a straightforward operation. When in place, the stabilizer (and filler) must have the highest achievable density, in order to maximize the support it provides. In order to achieve this in a particulate stabilizer, a bi- or tri-modal particle size distribution in the particulate, coupled with vibration during loading, can be used; a particulate filler might have to be tamped during placement if further analysis indicates the necessity of a higher density than could be obtained by pouring a multi-modal particulate. Remotely-operated tamping equipment would have to be developed for this purpose.

For purposes of the process study, a stabilizer was selected prior to the time any recommendations were available from the other studies. The choice of a particulate stabilizer was based on the simplicity in processing techniques it afforded and was subsequently affirmed by the realization that a low-to-moderate melting point metallic alloy could be substituted in the Reference Process and Alternatives 1-3 without significant impact on the process analyses which had been developed, or the conclusions therefrom. The processes described in Section 4.0 therefore assume the use of sand as both filler and stabilizer, with the full recognition that other candidate materials, either particulate or solid, would have to be evaluated prior to final selection of the stabilizer.

3.3.4 Backfill

The backfill material fills the annulus between the liner and the host rock and provides the liner with lateral support. The material can also aid the transfer of heat from the liner as well as absorbing moisture, chemically conditioning that moisture, and acting as an oxygen scavenger and radionuclide sorber. The most convenient backfill material for the purpose would, of course, be crushed host rock. Use of crushed salt in the Reference Repository would not confer any of the possible advantages cited; for that reason, bentonite, a montmorillonite clay of high water absorption properties and excellent ion-exchange capacity, has been employed as the backfill for purposes of this study.

Use of bentonite for this purpose in a salt repository precludes access to the liner by migrating brine inclusions, thus providing additional delay in breaching the package. If all barriers have been penetrated and the fuel is in contact with a transporting fluid, the bentonite will provide a strong retarding effect on both migration of soluble ions (7) and transport of the fluid itself.

3.3.5 Shield Plug

The shield plug is placed above the retrievable package to attenuate radiation to hands-on working levels at the repository floor. The shield plug would be sealed inside the liner assembly with the retrievable package and must exhibit the same thermal, radiation, and chemical stability

as other internal package components. Remaining structurally sound over the 1000 year containment period, the shield plug would also provide support for the package cover plate.

When placed over the spent fuel canister, the 152 cm concrete shield plug is highly effective in attenuating the gamma rays from the fuel assembly. The limiting factor in shielding the assembly is the possibility of radiation streaming through the bentonite backfill. Since the surfaces of the various materials composing the waste package are not, in general, perpendicular to the direction of propogation of the radiation, it can be expected that the scattering of the radiation may cause it to take longer paths through the poorer attenuating medium. A conservative estimate can be calculated for the dose rate above the bentonite assuming a density of 1.75 g/cm^3 and the minimum path length through the bentonite. If the bentonite backfill is uniformly distributed and compacted about the shield plug, the dose rate will be limited to less than 0.1 mrem/hr. It is essential that all of the backfill is emplaced before the encapsulated spent fuel is placed into the storage hole.

3.3.6 Thermal Considerations

Tentative maximum temperatures have been set for the various possible host repository geologies: 284F (140C) salt, 329F (165C) basalt, 365F (185C) tuff, 302F (150C) granite, and 266F (130C) for shale. The waste form which generates heat will be hotter than the rock. The temperature differential between the host rock and the inside of the canister is a function of the heat generation rate within the waste form and the heat transfer properties of the package. This functional dependence is illustrated in Figure 3-7; the region which is applicable to the present problem, that is, canister radius of 17.15 to 24.1 cm and linear heat generation between 30 and 100 watts/foot is shown as the shaded area. Table 3-5 gives the package temperature differentials for the cases considered.

It is clear that the highest cladding temperature will obtain in the case of close packed PWR fuel pins. At the limiting borehole temperature of 140C, the inside canister temperature in this case is 227C. Then, assuming

homogeneous waste form with a thermal conductivity of 0.005 watts/cm^{o}C, the temperature rise in the waste form is found to be 67C. Using the thermal conductivity of sand gives a conservative value for the temperature differential. The compacted pins occupy 71 and 76 percent of the canister cross section for PWR and BWR spent fuel respectively. The thermal conductivity of the fuel pins will be much higher than sand (e.g., zirconium, k=0.242 watts/cm^{o}C), and the resulting temperature differential will be considerably less. Thus, the maximum temperature of the waste form is estimated to be 294C, or well within the limiting value of 375C set for the clad temperature*.

TABLE 3-5

WASTE PACKAGE TEMPERATURE DIFFERENTIAL

	Unmodified and End Fitting Removal		Disassembly and Pin Storage		Shearing and Immobilization	
	PWR	BWR	PWR	BWR	PWR	BWR
Canister Radius (cm)	17.15	17.15	17.78	17.78	24.13	24.13
Linear Power (watts/foot)	44.2	31.7	127.8	107.0	89.5	82.0
Temperature Differential (C)	30	22	87	73	54	47

*In Table 20 of Reference (8), the cladding temperature for close packed fuel pins in helium (gas stabilizer) is identical to that of close packed pins in a particulate stabilizer. In actuality, the mixture of a particulate stabilizer with helium backfill has a thermal conductivity greater than that of gas alone. On that basis, the maximum temperature was recalculated using the thermal conductivity of sand given in Table 19 of Reference (8), i.e., 0.005 watts/cm^{o}C.

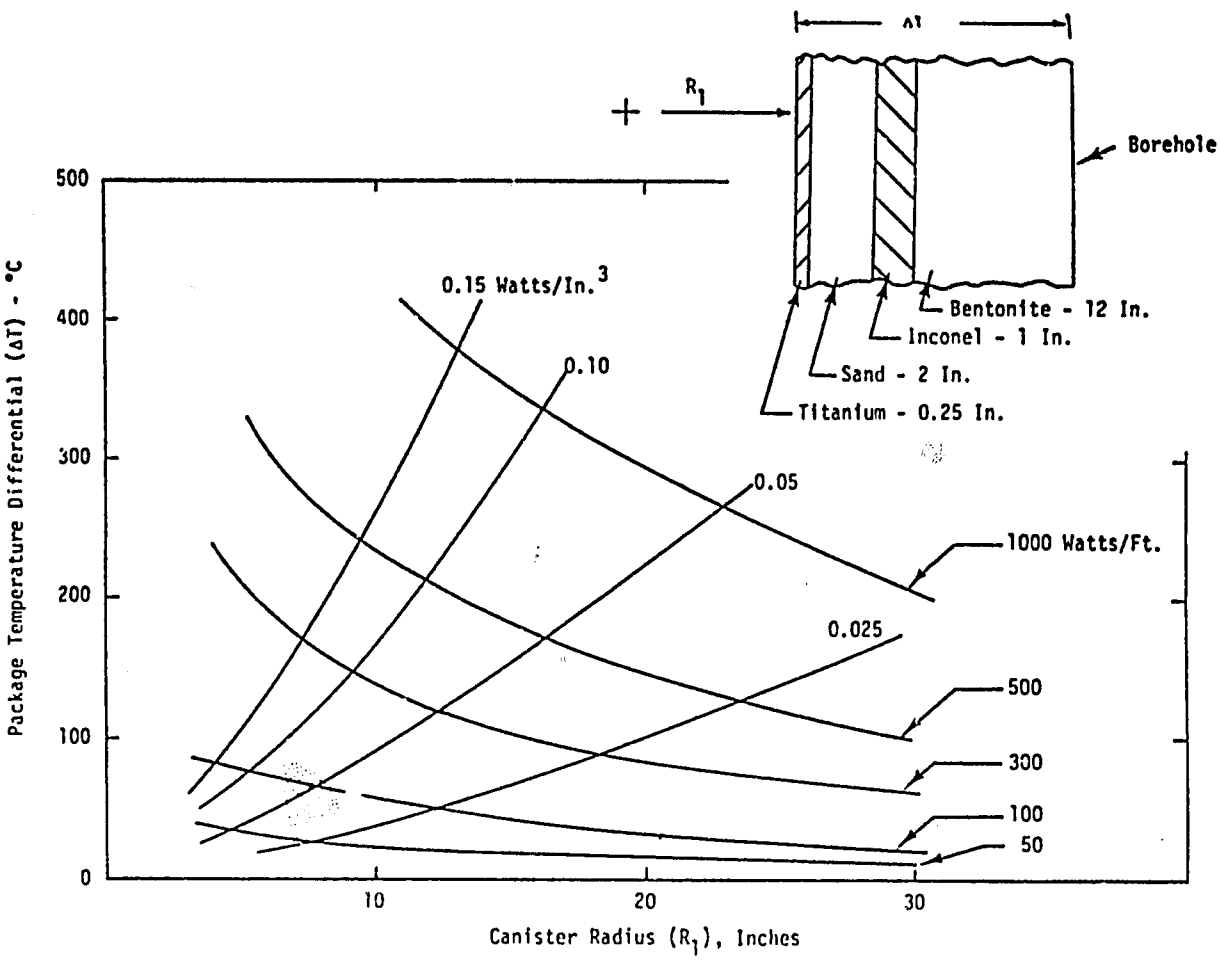

FIGURE 3-7 HEAT DROP ACROSS WASTE PACKAGE-CROSS HATCHING REPRESENTS REGION OF EXPECTED REPOSITORY OPERATION

3.4 INFLUENCE OF SPENT FUEL FORM ON PACKAGE COST

Intuitively, it would be expected that unit package costs ($/kg fuel) would be inversely proportional to the unit volumetric loading of fuel in a package. This has been shown to be correct, with package diameter the dominant parameter in overall package cost. Volumetric loading at a given package diameter is a function of pin-to-pin spacing in the canister, and it is obvious that decreasing the pin-to-pin spacing will result in lower unit package costs, all other factors remaining constant. The limiting factor on canister volumetric loading is imposed by thermal constraints, as discussed in Section 3.3.6. At three PWR or eight BWR assemblies per canister, the proposed canister loading for Alternative 3 gives the highest unit volumetric loading of the cases considered, and would thus be expected to result in the least package costs among the alternatives studied. (See Appendix B for details of the development of canister and liner costs.)

3.5 CONCLUSIONS

The reference waste package described here can accommodate each of the spent fuel waste form alternatives addressed in this study. The alternative fuel forms would not cause negative effects on package design that would render the reference package unworkable; in the case of the sheared/immobilized form with a hot-pour immobilizer, the package cani could not be used in the dual capacity of a process vessel and the containment barrier; therefore, the containment barrier is an overpack over a sealed inner container. This could be construed as a negative effect of waste form in the sense that it forces the extra packaging step. However, the result still largely resembles the reference package and would function identically.

The cost analyses show a clear cost advantage for the close-packed alternative. To arrive at a recommendation for a preferred spent fuel waste form, however, results from the in-repository performance and waste form process analyses must be evaluated to weigh trade-offs among all three issue areas. This is done in following sections of this report.

In summary, there were no findings which would rule out any alternative on the basis of waste package considerations or long-term performance of the waste form. Alternative 3 offers flexibility in loading which may prove attractive in the various geologic media under consideration, greatly reduces the number of packages, and has the lowest unit cost. ONWI and CRWM should proceed with the devlopment of the disassembly process technology, with the goal of incorporating disassembly of spent fuel in future test facilities as they move toward their goal of an operational disposal facility.

3.6 REFERENCES

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4.0 DESCRIPTION AND ASSESSMENT OF ALTERNATIVE WASTE FORM PROCESSES

A basic design of facilities for packaging and geologic disposal of unmodified spent fuel was established as a basis for evaluation of alternative fuel disassembly techniques in comparison to the disposal of unmodified spent fuel. The basic design, which is hereinafter referred to as the Reference Process, was presented in a study conducted by Bechtel Group, Inc., which incorporated three previously prepared conceptual design reports (1). The Bechtel Conceptual Reference Repository Description (CRRD) combines the key features of the following reports:

- National Waste Terminal Storage in a Bedded Salt Formation for Spent Unreprocessed Fuel (NWTS-R2), prepared by Kaiser Engineers (2)
- National Waste Terminal Storage Repository Number 1 (NWTS-R1), prepared by Stearns-Roger (3)
- Spent Fuel Receiving and Packaging Facility Conceptual Design, prepared by Rockwell Hanford Operations and Kaiser Engineers (4).

Operational modifications were required in the CRRD process and facility because of a change in the canister and emplacement package design from that described in the CRRD to that as described in Section 3.

The four alternative processes which were evaluated are (1) end fitting removal, (2) fission gas venting and resealing, (3) fuel disassembly and close packing of fuel pins, and (4) fuel shearing and immobilization in a solid matrix.

In this section the spent fuel disposal processes are described, and comparisons are made between the Reference Process and the alternative processes. Differences in the processing of the spent fuel, in facility and equipment modifications, and in the status of the technology for each of the fuel disassembly alternatives under consideration are examined relative to the Reference Process. The merits and disadvantages of each alternative process are then assessed.

Figure 4-1 shows the layout of the Reference Process Packaging Facility. Figure 4-2 shows the primary process steps for the Reference

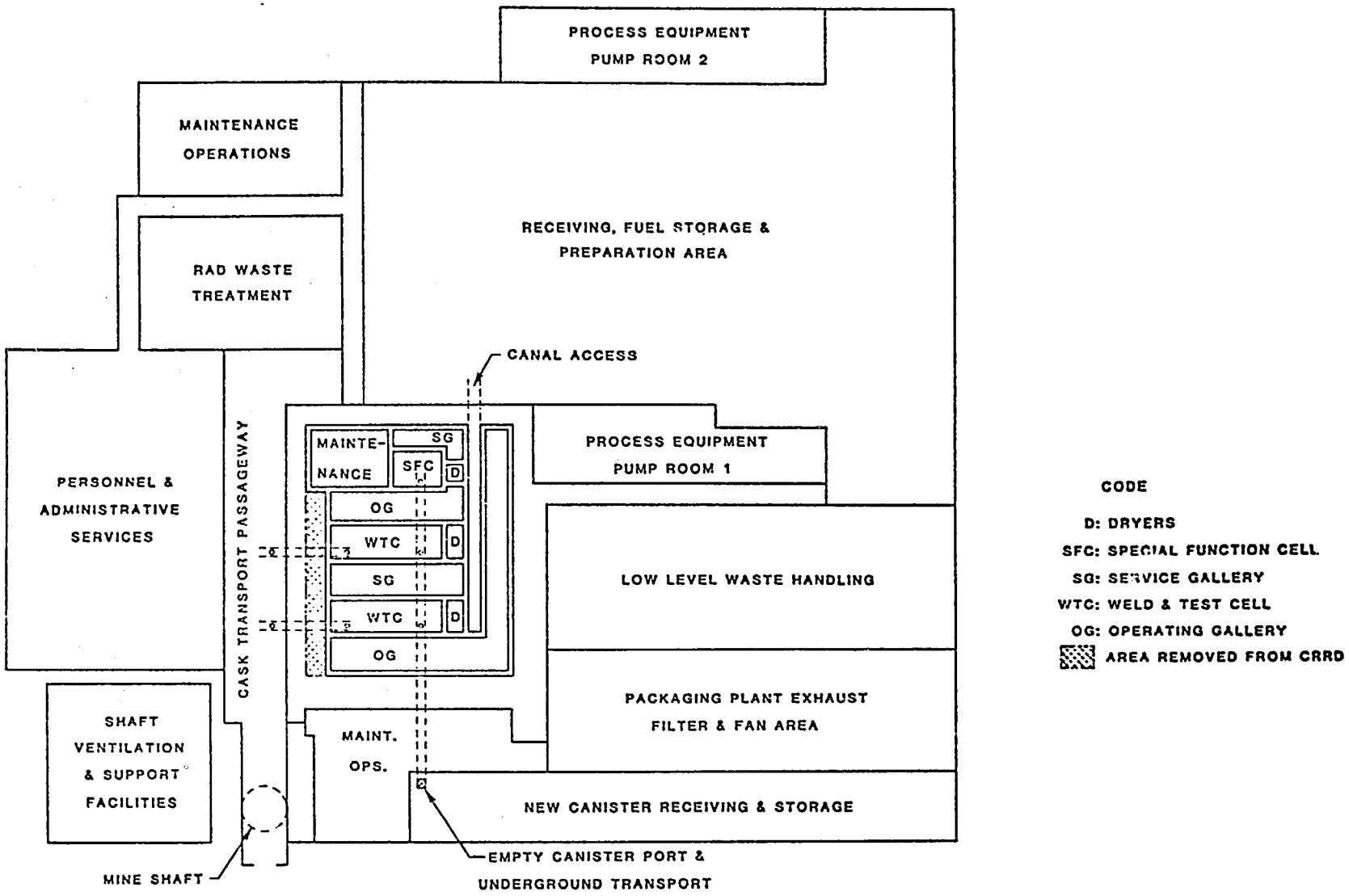

FIGURE 4-1

REFERENCE PROCESS PACKAGING FACILITY LAYOUT

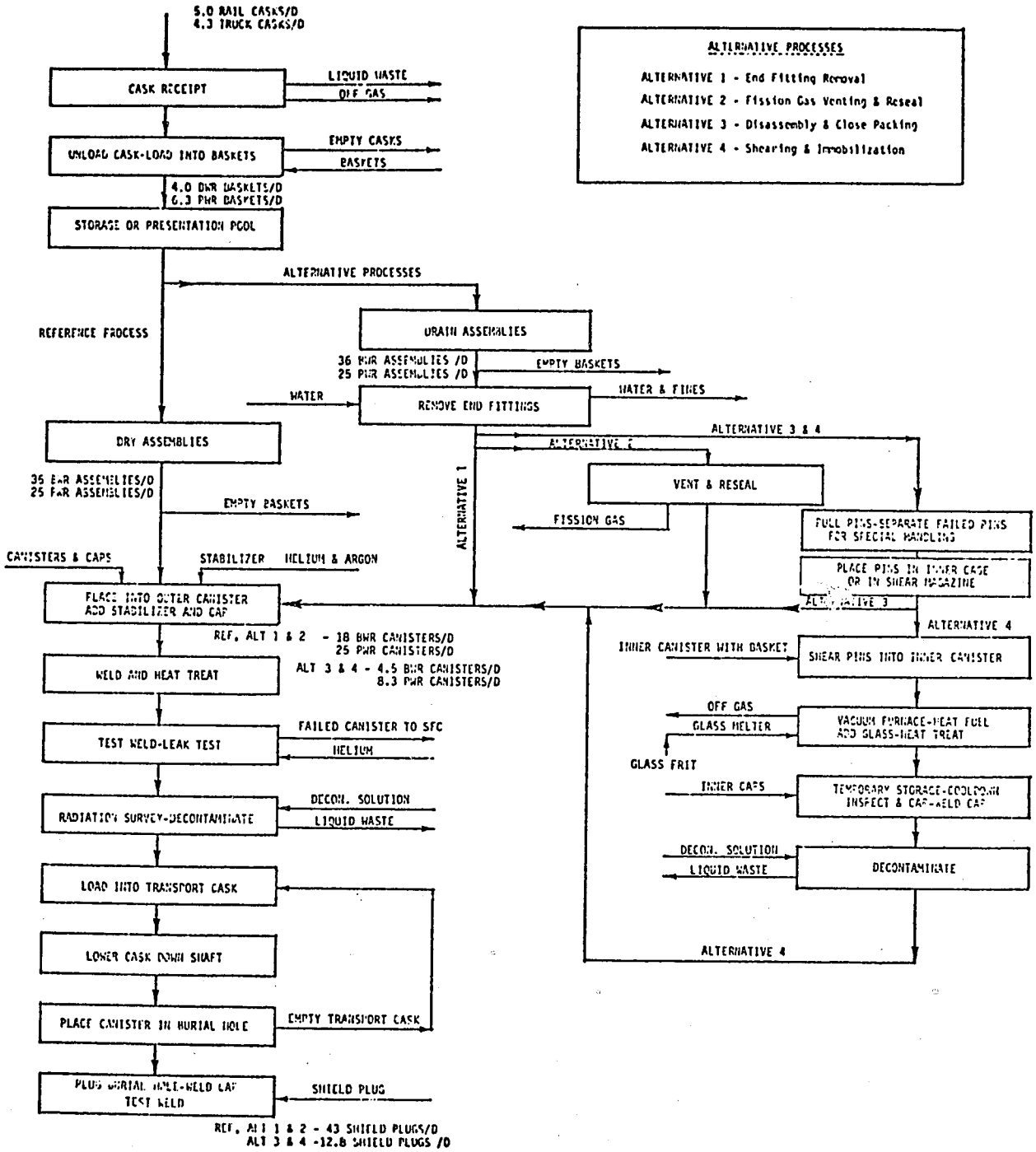

FIGURE 4-2

PROCESSES MATERIAL FLOW

Process and the additional or different process steps of the alternative processes. The processes are identical to the point where the spent fuel is removed from pool storage; after the spent fuel is canistered, the processes are again identical.

4.1 DESCRIPTION OF PROCESS AND FACILITIES

The Reference Process and the alternative processes have certain elements of commonality. These are discussed in Section 4.1.1. The elements of the alternative processes which differ from the Reference Process and among each other are detailed in Section 4.1.2 for each alternative.

4.1.1 Reference Process and Common Process Elements

The spent fuel is received by rail cask (90%) and truck cask (10%).* The carriers with shipping casks are inspected for sabotage and radiation contamination at outdoor inspection stations. Uncontaminated carriers with casks are washed to remove road dirt and moved into a preparation area through an airlock. Contaminated carriers are not washed at the outdoor inspection station; they are moved directly into the preparation area, where road dirt is removed. Cranes unload the casks from the carrier at the preparation station. The cask is moved into the wash and cooldown pit, where vent and cooldown hoses are manually connected. The vent gas and water coolant are monitored to detect radiation which would result from damaged fuel assemblies. The water effluent is treated and recycled; off-gas from the venting system is piped into the off-gas treatment system, then released into the stack system.

When the temperature is lowered to 115F (46C) the shipping cask is transferred to the stainless-steel-lined unloading pool by a 125-ton crane. The cask cover is removed, and the fuel assemblies are lifted and moved to an inspection station within the pool by a 5-ton gantry crane. There each fuel assembly is identified for accountability purposes and monitored for activity level. Damaged assemblies are isolated in special containers for transfer to

*Reference (1) assumes the rail cask to be the IF-300, manufactured by General Electric and the truck cask to be the NLI-1/2, manufactured by National Lead Industries.

a Special Function Cell for further processing. Leaking or damaged assemblies which require isolation and special containment are defined as those which evidence clad damage to the extent that particulate radioactive material may escape. Those from which the fission gas has been released due to damage to the clad, but which show no obvious clad damage, will be processed conventionally.

After inspection, the assemblies are loaded into baskets by the gantry crane. Three types of baskets enter the system as needed:

- PWR assembly baskets which have a capacity of 4 PWR assemblies
- BWR assembly baskets which have a capacity of 9 BWR assemblies
- Baskets for damaged fuel containers

Three of these baskets are handled on each canal buggy, which moves on corrosion-resistant rails located below the water surface. The gantry crane pushes the canal buggy from the spent fuel unloading pool under the storage separation wall to within reach of the storage gantry.

The spent fuel assembly baskets move directly from the unloading pool to the presentation pool or to the Weld and Test Cells. The presentation pool is equipped with 52 fuel basket racks; if it is full, the baskets are stored in the lag storage pool, which consists of two pools each equipped with 240 fuel basket racks. The storage pools and the presentation pool are connected by canals with the unloading pools and the Weld and Test Cells. All pools and canals have stainless steel liners, cooling, and decontamination systems. Double-gate locking systems isolate the canals and each pool. The gates permit draining of an isolated pool or section of a canal. The gates are sealed by double inflatable seals. Two 5-ton gantries travel on common rails over the lag storage pools, the presentation pool, the transfer canals and the canal buggy unloading area. These gantries are designed to receive the canal buggies from the unloading pool, remove the baskets from the canal buggy, transport the baskets to the storage or presentation pools and place the baskets in the storage rack, retrieve the baskets from the racks and place them in the transfer buggy for delivery to the Weld and Test Cells. The crane

can also take the basket directly through the canal to the transfer buggy loading area from the canal buggy unloading area outside the cask unload station.

The transfer buggy is loaded with a fuel basket at the front of the presentation pool and is propelled by electric-motored traction drive located above the transfer canal, moving the transfer buggy to the head of either of two Weld and Test Cells.

The subsequent processing of the spent fuel assemblies varies with each alternative process until the fuel is canistered, after which the processing of the canistered fuel is identical for each alternative process. The preparation of the spent fuel for the Reference Process is straightforward. The unmodified fuel assemblies are lifted from the transfer buggy into one of three fuel assembly driers, where heated air is blower-circulated to dry the fuel assemblies and basket. Accumulated moisture is exhausted from the drying chamber. At this point the unmodified fuel is ready for insertion into the canister as shown in Figures 3-5 and 3-6. The modified fuel of the alternative processes is not dried in this manner, since the additional time required to process the assemblies will allow the fuel to dry sufficiently. The perforated fuel baskets allow the majority of the water to drain as the basket is lifted from the transfer canal.

After the spent fuel has been inserted into the canister, the packaging operations are identical for all processes. The canister is fitted with a fixture which permits evacuation of the interior and introduction of the stabilizer. The canister cap is placed into position by a manipulator; the canister and cap are then transferred to the welding station.

At the welding station, helium is introduced through a valve built into the canister cap, displacing the air inside the cap. The welding positioner clamps the cap to the canister. A plasma-arc welder is positioned on the cap, indexed to the pintle, and the cap joint is welded while the welder head is rotated at welding speed. After the cap joint is welded, the helium pressure is increased to 5 psig. A probe inserts a mechanical sealing plug in the cap valve opening and a welding head automatically seal-welds the plug. On completion of all the welding operations the canister is transferred by the overhead 3-ton crane to the weld heat treat station.

The cap weld joint is examined by ultrasonic transducer equipment which is rotated in an orbital path around the canister cap joint to search for discontinuities in the weld. If no defects are found, a transfer jib hoist moves the canister into position for leak testing. Defective welds noted at this position will cause the canister to be routed to the Special Function Cell. In the Special Function Cell, the defective canister is overpacked in a titanium canister, filled with stabilizer, capped and welded.

The jib hoist moves the canister into position for the final leak test. Each Weld and Test Cell has two leak test stations equipped with tank type enclosures. Canisters acceptably passing the leak test are moved to the radiation survey station; those failing the leak test are moved to the Special Function Cell for rework as described above. The assumed failure rate is 2 percent of the total canistered assemblies.

At the radiation survey station the canister is monitored for external radiation and a master-slave manipulator obtains swipe samples which are sent to the laboratory for measurement of the canister's external radioactive contamination.

Canisters that pass the survey and swipe tests are transferred by the overhead crane through the cell floor port into the transporter below or to the temporary storage area in the cell. If excessive surface contamination is found, the canister is routed to the decontamination area. After a bath (agitated by ultrasonic transducers for cleansing action), the canister is dried and returned to the radiation survey station for retesting.

The sealed and decontaminated spent fuel canister is placed on a transporter and moved to a loading port, where it is transferred into a cask. The cask containing the spent fuel canister is moved to the burial shaft and placed by crane into a cage which transports the cask down the shaft to the underground repository. At the bottom of the shaft a crane picks up the cask and places in on a cask transporter, which then moves to the storage area.

A burial hole is prepared to receive the spent fuel canister in the following manner: A hole is drilled in the salt, and a 30 cm layer of bentonite is placed and packed into the bottom of the hole. The lower section of the Inconel liner is partially lowered into the hole, then held in

place while the upper section of the liner is positioned. The liner sections are welded, heat treated, and examined by non-destructive testing. The liner is lowered onto the bentonite, and the annulus between the outside of the liner and the inside of the borehole is filled with bentonite. The bentonite in the upper portion of the hole is packed to provide shielding protection from the emplaced canister. A layer of sand is placed in the bottom of the Inconel liner to serve as a separating support for the canister. A collar with shielding shutter is placed over the prepared burial hole. The collar rests on the salt floor and provides support for the cask as the canister is removed. The burial crane positions the transfer cask over the hole; the shutter in the collar and the cask bottom are remotely opened and the canister is lowered into the burial hole by the cask cable mechanism. The shutter is closed on the hole collar, and after closing the bottom opening of the cask it is removed by the burial crane. The space between the liner and canister is remotely filled with sand and a prefabricated shield plug in a shielded container is placed over the collar. The shield plug is placed in the burial hole in the same manner as the canister. The shielded container is lifted from the emplacement and the collar removed. An Inconel cap is placed on the shield plug and welded to the Inconel liner. Testing this weld completes the emplacement procedure. The waste package is now complete and the final configuration is as shown in Figure 3-4. The configuration for the alternative processes differ only in dimensional detail.

4.1.2 Process Variations in Alternatives 1-4

All of the alternative processes require the removal of the end fittings from the assembly; Alternatives 3 and 4 require separation of the pins from the rest of the structural components. The following paragraphs describe the processing of the assembly for each alternative through the insertion of the modified spent fuel into the canister.

4.1.2.1 Alternative 1

The assembly is lowered by a three-ton crane through the receiving port into the end fitting removal area, where it is received by a specially designed carousel. The carousel will secure the assembly, rotate from

vertical to horizontal, and position the assembly on the table for sawing. Two process lines are provided in each Weld and Test Cell (WTC). Two hacksaws are placed in position, one near the top end fitting and the other near the bottom end fitting. Positioning is monitored by an operator through a viewing window and remote television monitors. Adjustments are made for precise positioning of the saw at a location between the end fitting and the fuel pins so that the cut will not enter the fuel cladding. A shroud fits over the saw and end fitting to minimize spread of contamination and water to the cell. The sawed off end fittings are stacked on a table in the cell for further processing. Saw fines will be recovered by settling and filtration of the water and the water will be recirculated. The hacksaw also contains a small sawing unit to cut the bail off the top end fitting of a BWR assembly at the same time the end fitting is being sawed off.

Each process line is provided with a rail-mounted canister strongback. Each strongback is mounted on pivots and can be erected to a vertical position to receive the empty canister, which is lifted by the cell crane through a floor port located at the end of the process line near the strongback. Clamps on the strongback grasp the canister, and the strongback is lowered to the horizontal position and moved on its rails to a point contiguous to the saw table, and in line with the assembly. The saw table is equipped with a powered roller bed which rolls the fuel assembly into the canister. As in the Reference Process, one PWR assembly is placed in a canister and two BWR assemblies are placed in a single canister. The filled canister is transferred to the canister carousel after the canister strongback has been raised to vertical.

The end fittings are packaged in a canister similar to the assembly canister and proceed through the Weld and Test Cell in the same manner as the assembly canister. The Special Function Cell is modified so that the end fittings can be removed from the damaged BWR assemblies and the end fittings are packaged as in the Weld and Test Cell. The damaged PWR assemblies are placed in the longer BWR canister without removal of the end fittings. The canisters from the Special Function Cell enter the Weld and Test Cell at the weld heat-treat station. The above-ground packaging facility layout for Alternative 1 is shown in Figure 4-3. The Weld and Test Cell has been

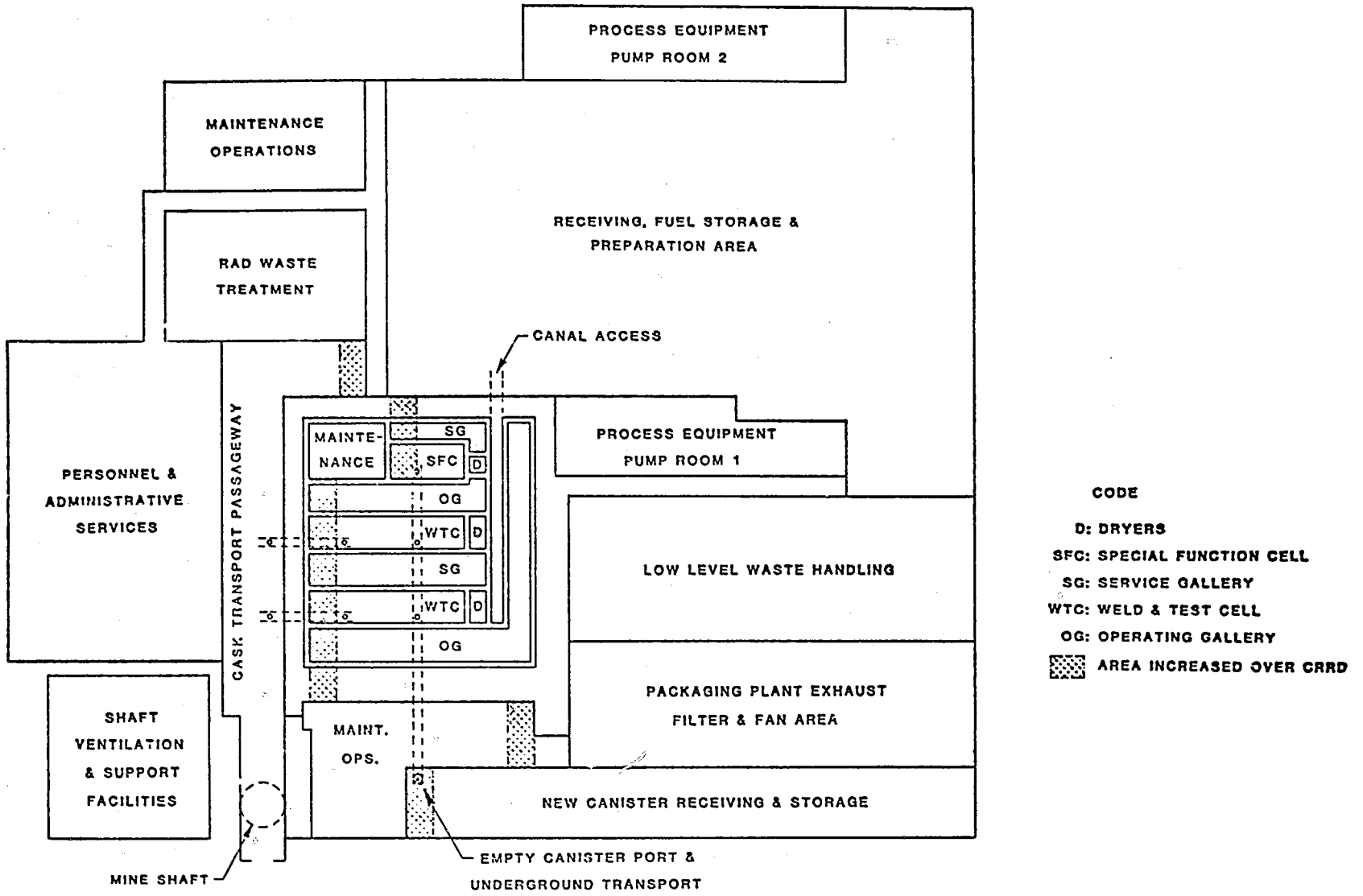

FIGURE 4-3

ALTERNATIVE 1 PACKAGING FACILITY LAYOUT

extended by 10 meters over the Reference Process WTC to accommodate the additional operations and equipment necessary. The canisters for this alternative are identical to the Reference Process except in length; shorter canisters may be used due to the removal of the end fittings from the assemblies.

4.1.2.2 Alternative 2

The end fitting removal in the Weld and Test Cell (WTC) follows the same process described in Section 4.1.2.1. Because of the additional process time required for the venting and resealing operation, it will be necessary to saw two BWR assemblies at the same time in order to provide an acceptable capacity reserve margin. The saw table of Alternative 1 will be modified to accept two BWR assemblies. As in Alternative 1, two process lines are provided in each WTC.

The empty canister, which has the same capacity and dimensions as the Alternative 1 canister, is brought into the WTC as described for Alternative 1. After the sawing operation is completed, the assemblies are inserted partially into the empty canister. The canister is then moved part way into the outer compartment of the vent/reseal chamber. An inflatable seal is secured around the open end of the canister and the assembly is positioned and secured with a clamping device which grasps the top spacer(s). The system is sealed and evacuated. Each fuel pin is then pierced by the laser device, whose optical system is mounted on an XYZ translator system. After venting, each pin is resealed by a defocused laser beam. The outer compartment is sealed off from the inner compartment, and is backfilled with argon. The gas which was removed from the chamber is treated if necessary before release. The seal on the canister is deflated, and the canister is removed from the chamber. A manipulator pad pushes the assembly end fully into the canister. The canister is returned to a vertical position and transferred to the canister carousel and from there to the stabilizer fill and cap welding stations. End Fittings are handled as described for Alternative 1. The Special Function Cell also operates the same as Alternative 1. The above-ground packaging facility layout is shown in Figure 4-4. The WTC has been extended over the Reference Process WTC by 14 meters to accommodate the additional operations and equipment necessary.

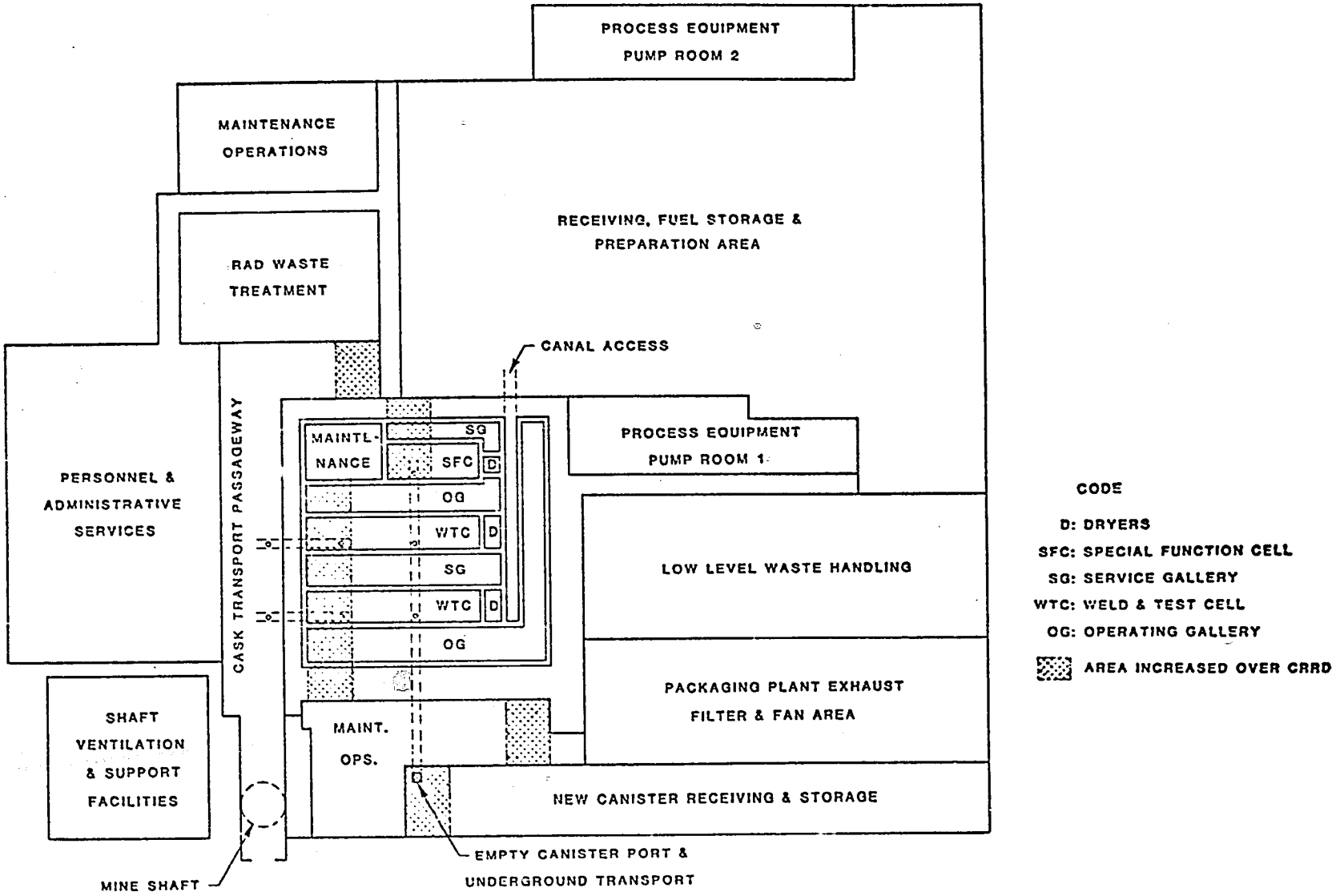

FIGURE 4-4

ALTERNATIVE 2 PACKAGING FACILITY LAYOUT

4.1.2.3 Alternative 3

The carousel receives the PWR assembly from the drain station and secures the assembly in the vertical position (Case 1). The carousel then rotates the assembly to horizontal and deposits the assembly on the saw table and the end fittings are removed in the same manner as in the Alternative 1 process. Two BWR assemblies are received and sawed together as in the Alternative 2 process. However, because of the rate of throughput of this operation, two entry ports with two receiving strongbacks are necessary for the BWR operations to provide an adequate margin of reserve capacity. Therefore, the second Weld and Test Cell has been further modified to provide these two entry ports from the drain station (Case 2). The carousel is not used in this cell and the saw table is designed to receive the two BWR assemblies vertically and then rotate to horizontal for the sawing operation. Both end fittings are removed.

The pins are pulled one row at a time; the pin pulling mechanism is the same design concept as that of the Barnwell Nuclear Fuel Plant (BNFP), which will be described in Section 4.2 in some detail. The receiving canister is designed to hold three PWR assemblies (792 pins) or eight BWR assemblies (512 pins). The PWR pin canister has three sections of 120 degrees, each holding pins from one assembly. The BWR pin canister has four sections of 90 degrees, each holding pins from two BWR assemblies. The canister and inner cage for PWR pins are shown in Figure 4-5. The collected pins are pushed from the collection trough into one section of the canister by a telescoping hydraulic pusher. The canister is then rotated 90 degrees or 120 degrees to position the next section for receiving fuel pins. These operations are repeated until all sections of the canister are filled.

End fittings will be packaged as in the Alternative 1 process. The additional hardware from the assembly skeleton are the grid spacers of the BWR assembly and the spacers and guide tubes from the PWR assembly. This hardware will be sheared and compacted in an assembly-type canister and proceed through the Weld and Test Cell operations when filled. The Special Function Cell will operate in the same manner as the Alternative 1 process, except that fuel pins which are damaged would be transferred from the WTC to the Special Function

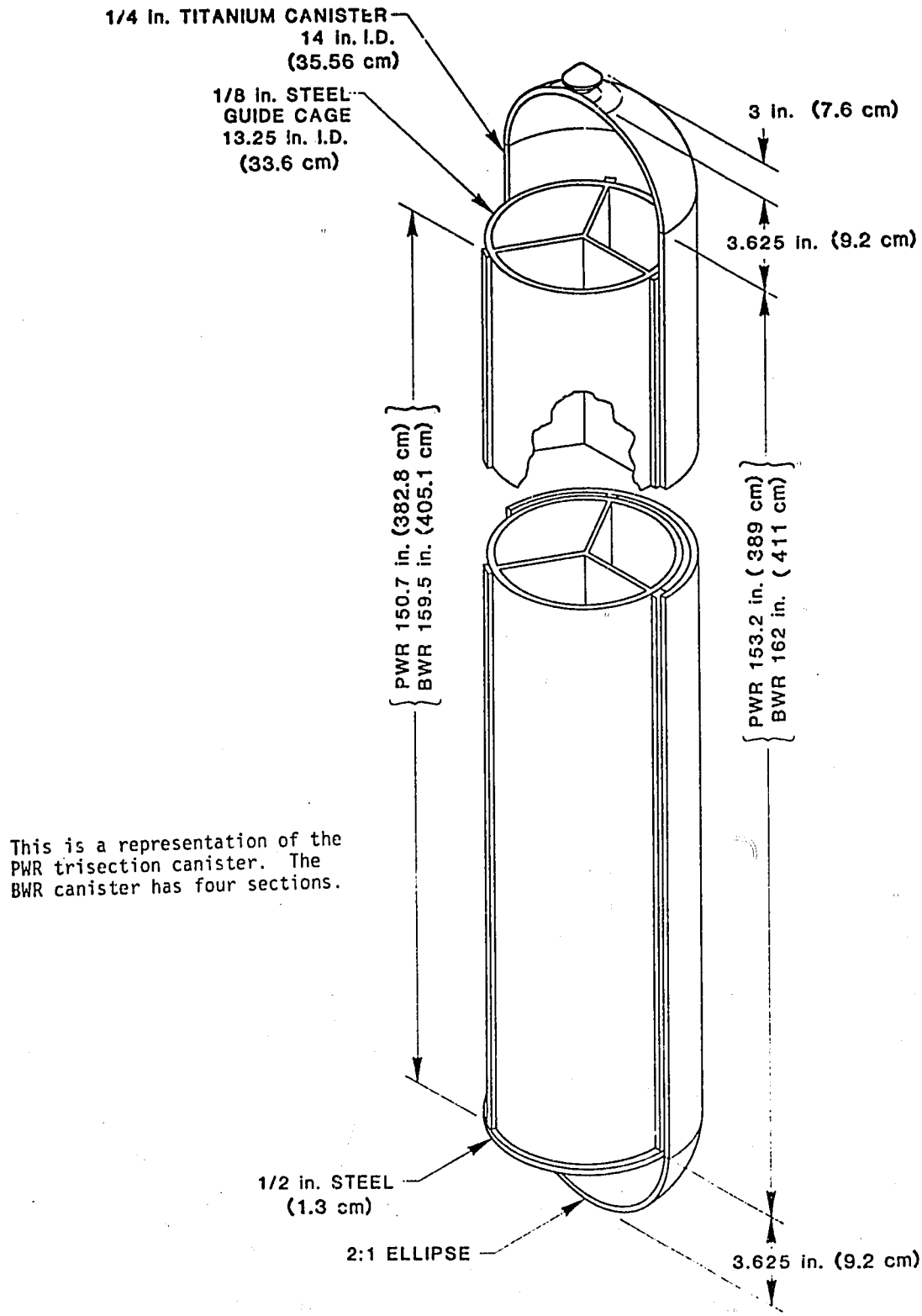

FIGURE 4-5

DETAIL OF ALTERNATIVE 3 SPENT FUEL CANISTER ASSEMBLY

Cell for canistering. The packaging facility layout for this alternative is shown in Figure 4-6. The Weld and Test Cell has been extended over the Reference Process WTC by 17.4 meters to accommodate the additional operations and equipment necessary. Several corridors have also been relocated.

4.1.2.4 Alternative 4

The spent fuel assemblies are received as described for Alternative 3, the end fittings removed and the fuel pins pulled.

The separated fuel pins are either collected in or subsequently pushed into a square container which serves as the feed magazine for the fuel pin shear. The loaded magazines are transferred through an airlock (for contamination control) into the Shear/Encapsulation Cell, in which are located the fuel shear, the vacuum casting stations, the heat treat stations, and the capping and welding stations for the inner canister. The fuel is sheared into 51 mm lengths which are dropped from the shear plenum directly into the perforated basket within the inner canister; the inner canister is a stainless steel can, fabricated from eighteen-inch, Schedule 10 pipe. The inner perforated basket is fabricated of 3.18 mm stainless steel sheet; the outer canister, inner canister and inner basket are shown in Figure 4-7. The purpose of the perforated basket is to provide an annulus just inside the inner canister which will not contain sheared fuel; this will allow for a monolithic shell around the sheared fuel.

The fuel pins from three PWR assemblies or eight BWR assemblies are sheared into the inner canister. Because of the greater bulk of the BWR fuel pins, the canister for use with this fuel is 46 cm longer than the canister used with PWR fuel. When the specified quantity of fuel has been sheared into the inner canister, the canister is moved to one of three vacuum furnaces where it is first heated to 2192-2282F (1200-1250C), evacuated, and filled with molten glass at 2282F (1250C). The molten glass is introduced through the central tube which extends to the bottom of the canister; thus the canister is filled from the bottom. The temperature will be maintained at a high enough level to assure fluidity of the glass. Volatile material released during the evacuation stage is collected for treatment in the off-gas system.

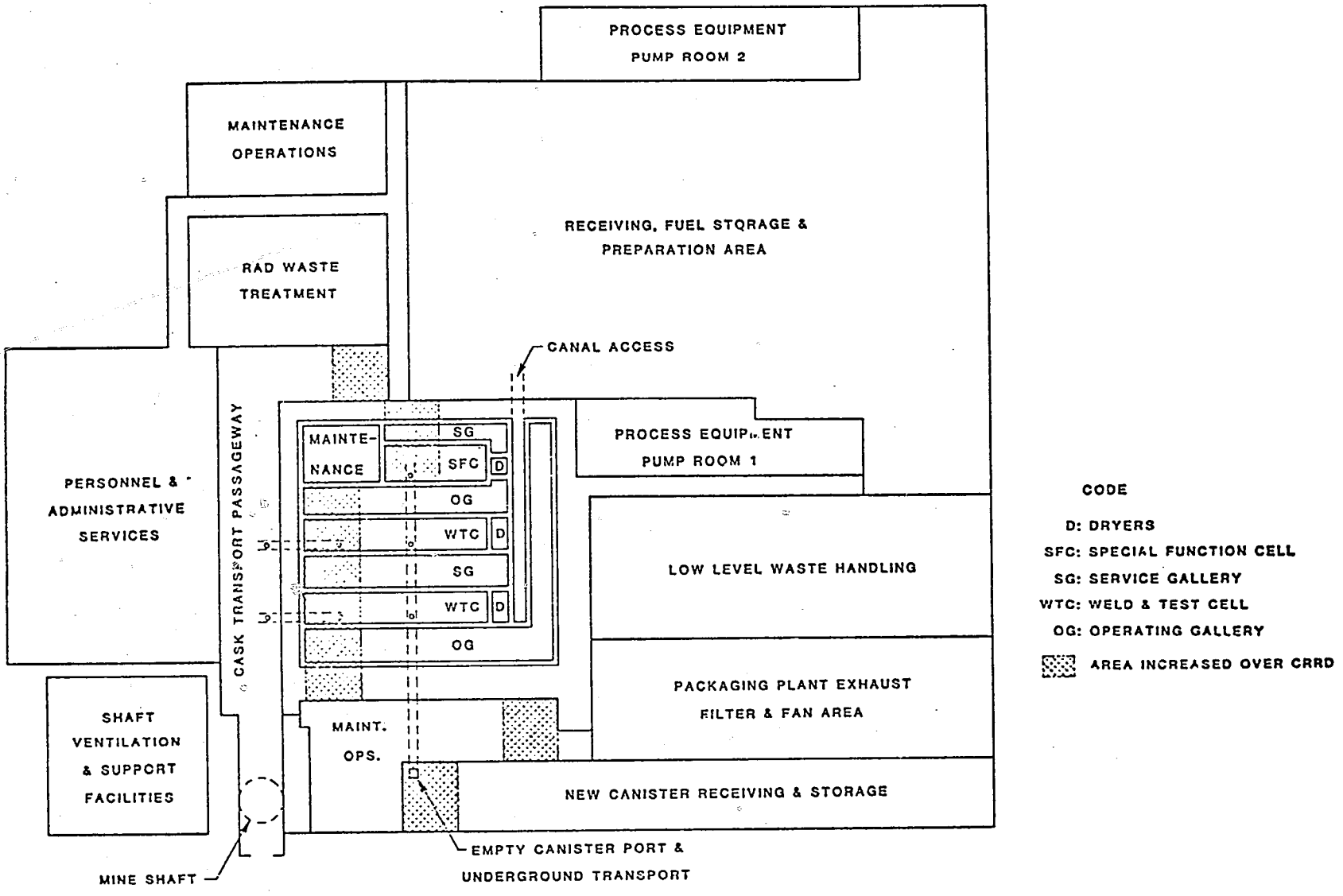

FIGURE 4-6

ALTERNATIVE 3 PACKAGING FACILITY LAYOUT

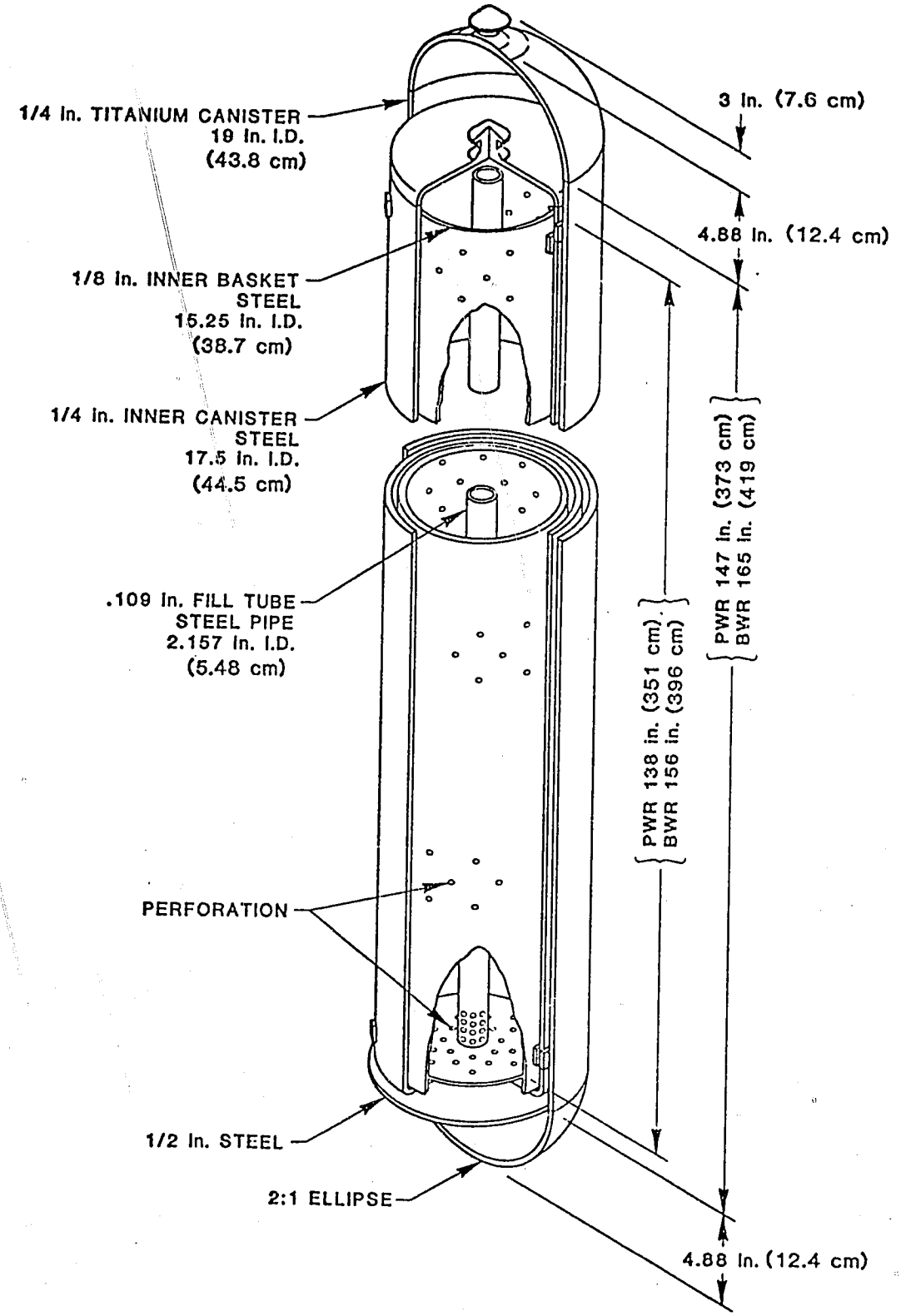

FIGURE 4-7

DETAIL OF ALTERNATIVE 4 SPENT FUEL CANISTER ASSEMBLY

After all glass has been transferred to the canister the vacuum is released, and the canister is cooled over a two-hour period to a temperature of 1200F (650C). It is then removed from the encapsulation furnace and transferred to one of six heat-treat furnaces where it is soaked at 1200F (650C) for three and one-half hours, following which it is heated to 1560F (850C) and annealed for six hours. At the end of the annealing, it is force-cooled to a temperature between 392F and 572F (200 and 300C). It is then transferred to lag storage, where it is held until ready for capping and sealing.

Fuel assemblies which are received in a damaged condition are transferred directly to the Special Function Cell as in the other processes. Here the end fittings are removed and the skeleton with the fuel pins in place, together with any broken pieces, are placed in a special thin-walled square aluminum container, which is seal welded for contamination control, and placed in a shear magazine for transport to one of the Shear/Encapsulation Cells. One standard PWR assembly is packaged per transfer box, or two standard BWR assemblies. Failed pins from the Weld and Test Cell do not go to the Special Function Cell, but are placed in the shear magazine with the other fuel.

Off-gas collected during the shearing operation and during the evacuation/heating cycle in the encapsulation operation is routed to an off-gas treatment system.

The packaging facility layout for Alternative 4 is shown in Figure 4-8. The increase in the length of the process module required to accommodate the added operations is approximately 45.7 meters. The orientation of the process cells has been rotated 90 degrees, necessitating some changes in the routing of the transfer tunnels. Some expansion of the Special Function Cell is also required.

4.2 ASSESSMENT OF ALTERNATIVES

Each alternative process was assessed and ranked relative to the Reference Process. The relevant concerns for the assessment were separated into four categories: the level of the technololgy required to effectively perform the processing of the spent fuel, the operational procedures necessary to carry out the process, consideration of the safety and risk aspects of the process, and the economics of the process

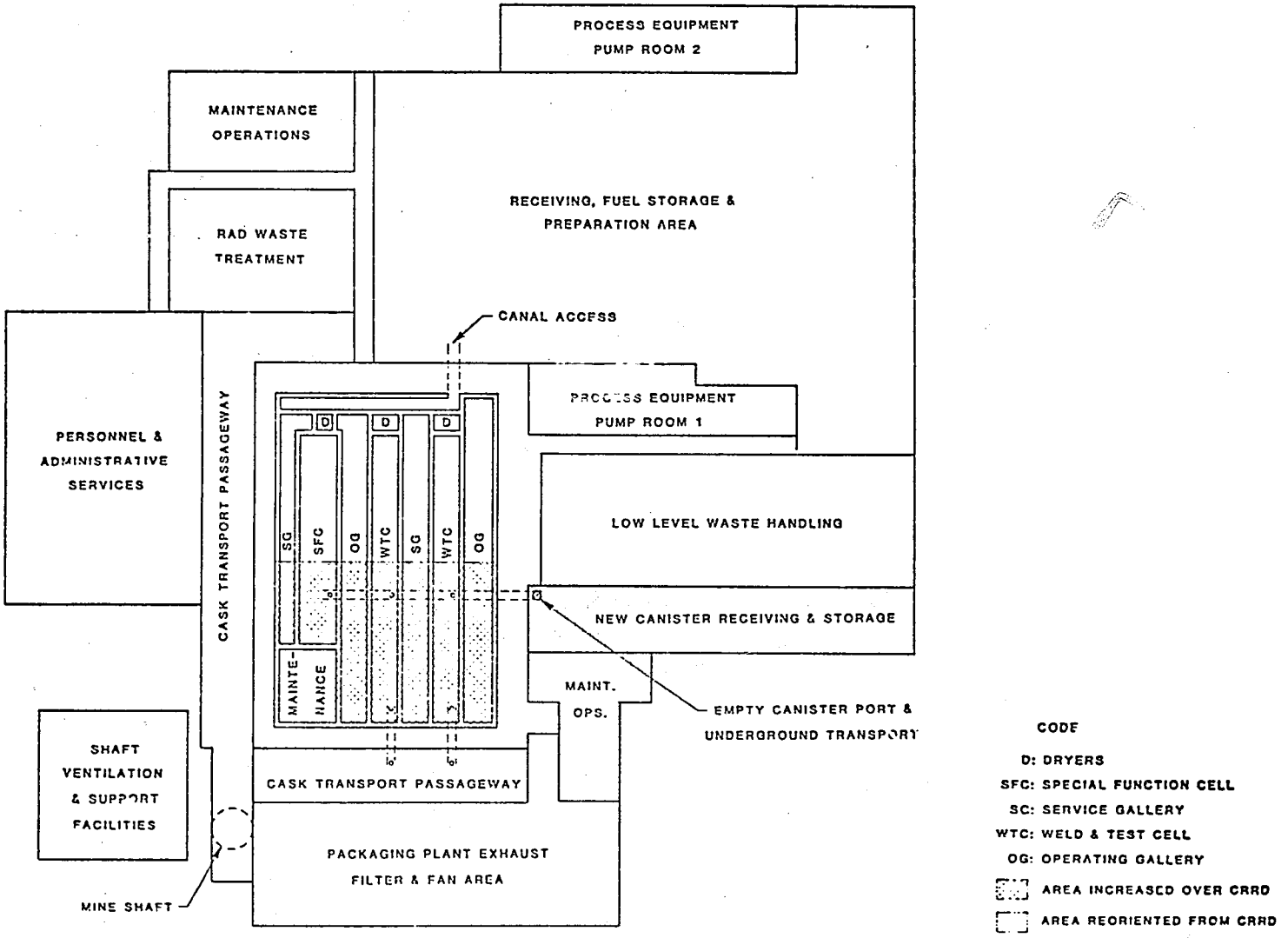

FIGURE 4-8

ALTERNATIVE 4 PACKAGING FACILITY LAYOUT

4.2.1 Technical Assessment

The technical assessment considered the overall technical and engineering practicability of the additional operations and procedures required over those in the Reference Process, including the status and development of the process technology and equipment, the amount of prior experience which could be drawn upon, an assessment of the development effort required, the effectiveness of safeguards, and the acceptability of the waste form.

4.2.1.1 Basis of Assessments

The technical assessment was based on first identifying the additional technology and equipment necessary to effectively implement each alternative over that of the Reference Process, then considering the status of the additional process technology and equipment in terms of the current state of development and the time required to complete development for application to the spent fuel disposal process on a large scale. An analysis of prior experience with similar technology and equipment was made, taking into consideration the unique need for remote handling and maintenance required due to the radiation hazards involved. The effectiveness of safeguards during processing and in respect to the final waste form were also considered. The additional equipment required for each alternative over that required for the Reference Process is shown in Figure 4-9.

4.2.1.2 Comparative Assessments

All of the alternative processes require the removal of the end fittings from the spent fuel assemblies. Allied-General Nuclear Services (AGNS) is currently conducting studies and research concerning spent fuel disassembly at the Barnwell Nuclear Fuel Plant (BNFP). Experimental demonstration efforts at BNFP were focused on the friction sawing and laser cutting of simulated PWR assembly end fittings (5,6,7). AGNS concluded that the friction sawing was a promising option but that further study would be necessary to arrive at a suitable facility design. The current tests of friction sawing have demonstrated the amount and size of fines generated, the

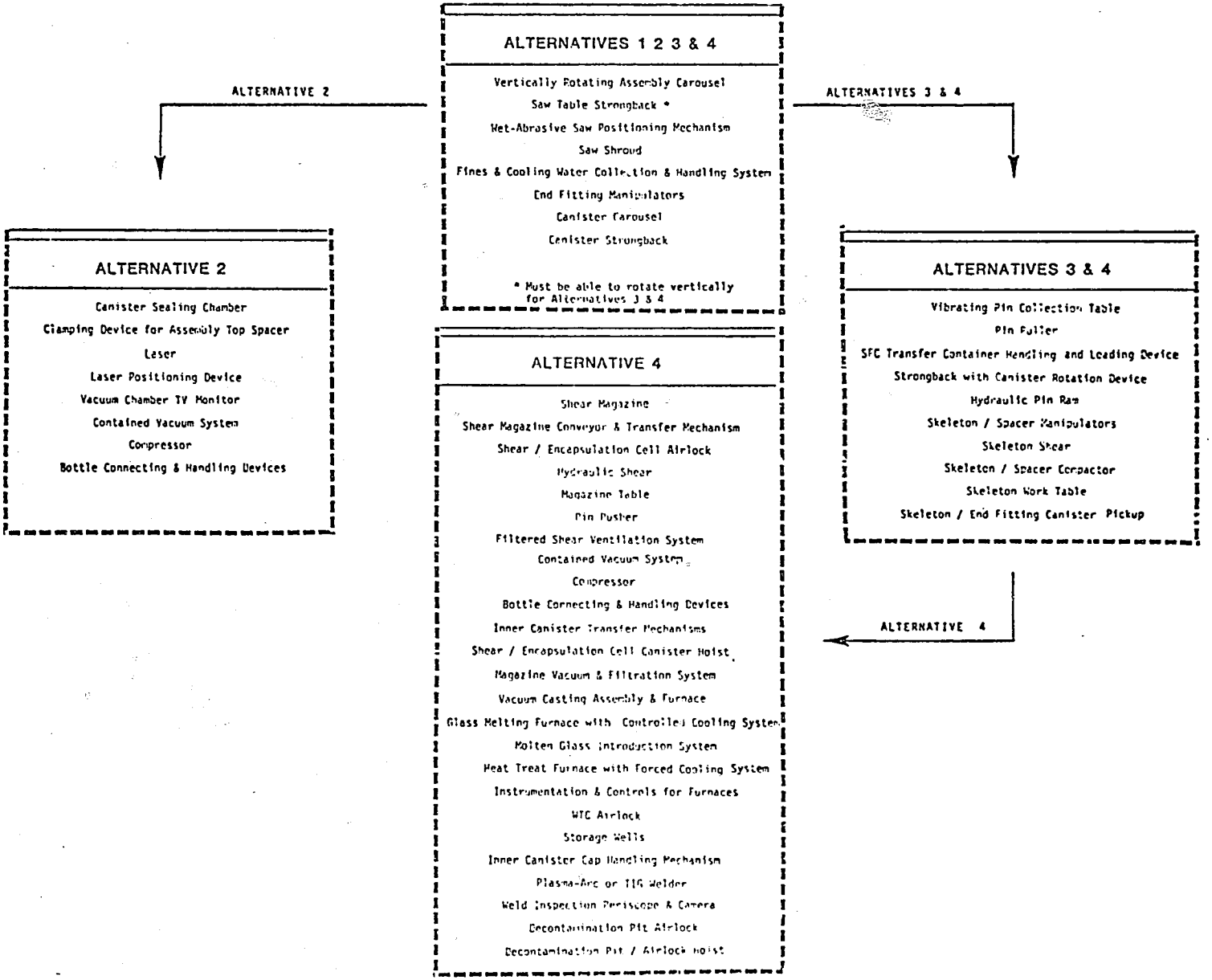

FIGURE 4-9

MAJOR EQUIPMENT REQUIREMENTS FOR ALL ALTERNATIVES

length of time for the operation, and assurance that the cutting operation can be accomplished without breaching the fuel pin cladding. AGNS further concludes that laser cutting, although feasible, requires further refinements in the observed cut quality prior to inclusion in the disassembly operation. Nuclear Assurance Corporation (NAC) describes in a conceptual report an air-powered cut-off wheel or arc-saw for removal of end fittings in a pool storage area. This sawing operation is accomplished with the assembly in a vertical position and only the top end fitting is removed (8).

Examining an alternative to the CRRD, Bechtel considers the removal of end fittings in a modified Weld and Test Cell using a cut-off saw, positioner and hydraulic ram, horizontal canister loader, remote manipulators for handling cut off end fittings, and a canister carousel (9). Considerable experience with end fitting removal was obtained at the NSF - West Valley reprocessing plant. After several modifications to the apparatus, a dry abrasive disc blade was used to cut off the end fittings from irradiated fuel assemblies. The saw was mounted on rails and the fuel assembly was lined up for cutting with a pushout ram (10).

Component development on hacksawing and bandsawing has indicated a decided advantage for hacksawing. While bandsawing was quicker, hacksawing minimized or eliminated burrs on the cut thimble tubes (7). The adaptability of the hacksaw to the remote operating and maintenance requirements for sawing the end fittings of a dry fuel assembly seemed to be essentially equal to or better than that of the bandsaw. Furthermore, a hacksaw is a rugged piece of equipment and involves less complicated blade changing for remote operations.

Technology and equipment are available for the design, fabrication and construction of a wet sawing table with provision for a stream of water circulating over the saw blades while in operation; the suspended saws required for cutting do not appear to present any significant design problems. Background experience pertaining to such an operation as envisaged for the alternatives is, however, rather limited (NFS and AGNS). Previous experience and the development efforts have primarily involved dry sawing methods. Considerable development needs to be done on the wet sawing process, as envisaged, specifically the components of the saw table including proper

mechanisms for alignment of the fuel assemblies, the sawing equipment and shrouds. This also would include developing a system which is operable in a remotely maintained cell for removing and collecting the saw cuttings (fines) from the circulating water. Estimated development time is 18 months.

Settling and filtration equipment suitable for removing the saw fines from the water used with the sawing operation are standard items. A series of filter cartridges in the water circulating system should be sufficient. The filtration equipment should be located within the cell and accessible to a manipulator for occasionally changing the container that collects the fines. The container with the fines then can be processed with simple equipment and transferred to the low-level waste treatment facilities.

Technology is available for the design, construction and operation of the tilting strongback table for receiving, holding, and rotating the empty canister to receive the fuel assemblies. However, a prototype strongback table will need to be constructed and operated in conjunction with a prototype sawing table for experience with the performance of the system and allow for any necessary adjustments to the design and operating controls.

The vertically rotating carousel with a horizontally rotating strongback is unique but there is no reason to suspect that such a carousel cannot be designed and successfully operated. A prototype could be constructed and tested within a year's time.

There are no safeguards areas of concern for end fitting removal operations. The fuel assemblies can be inspected and counted on completion of the operation. Canistering the assembly completes the differentiation between Alternative 1 and the Reference Process. The technical assessment of end fitting removal applies to all alternatives. There is no concern regarding the acceptability of the Alternative 1 waste form, as it is the same as for the Reference Process, minus the end fittings. The status of the equipment and process technology of end fitting removal for Alternative 1 do not indicate any serious drawbacks to its implementation.

The level of technology required by Alternatives 2, 3, and 4 is progressively higher than that of Alternative 1; those alternatives require

the same end fitting removal technology plus additional technology for the venting and resealing of the pins, pulling of pins, or shearing and immobilization of pulled pins.

In support of technology development for fuel pin venting, AGNS has developed a series of prototype laboratory tests on stainless steel and Zircaloy tubing in conjunction with a laser designer and vendor (Laser Incorporated, Sturbridge, Massachusetts) to demonstrate the laser venting and resealing process (6). The initial tests refined the parameters of laser power, pulsing rate, hole size, and beam optics. Later tests on both penetrating and resealing the simulated LWR fuel cladding were designed to demonstrate repeatability and consistency with a large number of samples. The following results were observed:

- optimal hole diameter appeared to be 0.25 mm,
- rewelding results in a new wall thickness of 0.46 to 0.51 mm (75 percent of original wall),
- inert cover gas prevents oxidation and results in a superior seal weld,
- cycle interval is 3 to 5 seconds (repositioning the fuel), recycle time of pulsed laser is one second,
- repeatability is excellent (99+ percent), and
- heat-affected zone is quite localized.

Further testing is planned on the drilling of pressurized tubes to develop optimal hole size under pressurized conditions to assure both adequate venting of gas and a hole suitable for rewelding.

Additional equipment required for Alternative 2 includes the following:

- wet sawing table modified to be capable of holding and sawing two BWR assemblies at a time,
- vacuum chamber with laser piercing and sealing device for venting the fuel pins, and
- treatment system for the off-gas vented from the pierced fuel pins.

The wet sawing table and its operations are covered in the assessment of Alternative 1. There are no foreseeable difficulties in modifying the table so that it will hold and saw two BWR assemblies at a time.

The vacuum chamber must be equipped with a device for individually piercing with a focused laser beam the exposed ends of fuel pins, evacuating the pin, and then sealing the hole by varying the laser beam strength and optical parameters. In addition to the AGNS work mentioned above, it is known that Exxon Nuclear uses a series of similar operations on unirradiated fuel to vent, pressurize with helium, and seal individual fuel pins; however, the rejection (or success) rate for the final product is not known. To perfect such an operation on irradiated fuel, piercing and resealing individual pins in an array with some of the pins possibly being out of line, will require additional development of appropriate technology and apparatus. Optimistically, it should take an additional two years of effort to develop the feasibility of the process and another year for demonstration with irradiated fuel.

There are no firm data upon which to estimate the composition or quantity of the gas from venting and evacuating the reservoir end of a spent irradiated fuel pin. The gaseous fission products, tritium and radioiodine, are reactive and presumably will be tied up in the solid structure of the uranium dioxide fuel material. They would not be released in significant quantities until the structure of the fuel is disturbed, e.g., by dissolution or oxidation. The same can be theorized for any carbon-14 that may be formed. However, it is known that some tritium does migrate through the uranium dioxide fuel and could be present in small quantities in the vented gas. Of the fission product noble gases, the radioxenons have half-lives of twelve days or less and will have decayed away by the time the fuel pins are vented, although there is a substantial quantity of stable xenons remaining. The other noble gas, krypton, contains significant quantities of the radioactive krypton-85; it is chemically unreactive and can migrate through the molecular structure of the fuel. Radiokrypton probably will be present to some extent in the gas reservoir at the end of a fuel pin.

Information from the Savannah River Laboratory (11), supplemented by information obtained in discussions with O. O. Yarbro of the Oak Ridge National Laboratory, indicates that even sheared irradiated fuel, several years out-of-reactor, will release very little tritium, radioiodine or carbon-14 if the molecular structure of the fuel is not disturbed. However, depending on the circumstances, a significant fraction of the radiokrypton may evolve when the fuel cladding is ruptured. Based on the foregoing and in the absence of firm contradictory information, it is concluded that the only significant quantity of radioactive material that can be expected to be included in the vented gas is the radiokrypton.

The krypton can be removed from the vacuum system off-gas that results from venting and evacuating the fuel pin by freezing it out with a cryogenic process, which will also recover the xenons. Equipment and technology for cryogenic processes are well known and routinely used in the commercial liquid air industry. A cryogenic system for the recovery of radioxenon and radiokrypton has been operated intermittently for over 15 years at the fuel reprocessing facility (Idaho Chemical Processing Plant) on the site of the Idaho National Engineering Laboratory (12).

The lack of firm data indicating the composition or quantities of gas released by evacuating the end of a pierced spent fuel pin suggests that further study is desirable to determine if a treatment process for the off-gas is necessary; it may be that the gas could be released through the cell ventilation exhaust treatment system with no significant risk entailed.

Venting of the fission gases by a laser is not required for Alternatives 3 and 4; there is the need for the equipment to accomplish the removal of the end fittings as described for Alternative 1. In addition to the specialized equipment required for Alternative 1, Alternative 3 requires the following:

- wet sawing table mechanically modified so it will tilt 90^{o} longitudinally to receive fuel assemblies; all sawing tables modified to be capable of holding and sawing two BWR assemblies at a time

- pin pulling machine with a pin collector trough
- pin pushing mechanism for pushing a group of pins into a sector of a canister or into a shear magazine
- device for rotation and seating of canister
- shear for fuel assembly skeletons
- compactor for fuel assembly hardware

Alternative 4 requires the above equipment as well as additional equipment discussed later in this section.

There are no foreseeable mechanical difficulties in modifying the wet-sawing table so that it will tilt forward 90^{o} longitudinally to receive and clamp in place two BWR fuel assemblies for wet sawing. Nor are there any foreseeable mechanical or operational difficulties in wet sawing two BWR fuel assemblies at a time compared with sawing them one at a time. It may be necessary to modify the clamping and holding devices for the PWR and BWR fuel assemblies on the sawing table so they are capable of firmly holding the fuel assemblies in the appropriate position for withdrawing the fuel pins (pin pulling), and it will be necessary to add longitudinal restraint to hold the PWR assembly against the pulling force.

The pin pulling mechanism required for Alternatives 3 and 4 requires a technology equivalent in complexity to the laser device required for Alternative 2. AGNS is conducting studies and research concerning spent fuel disassembly and fuel pin canistering at BNFP and have designed pin pulling equipment with the following specifications:

- processing rate of 12 to 15 assemblies/day,
- pin pulling rate of 2.54 cm/sec, and
- pulling one fuel pin row during a given pull.

The individual biter in the pin pulling equipment was designed to hold a minimum pull 200 lb and to release before an upper limit force of 250 lb was generated; it was also intended to minimize the biting force required to meet these previous criteria (50-100 lb), and dimensionally not to exceed the defined envelope. A prototype biter unit has been built and tested;

conclusion of the testing is that the device will function satisfactorily to remotely pull fuel pins from an assembly in a fuel disassembly process. Testing has not demonstrated any significant modifications required.

Based on the experimental work described, a multibiter head has been designed which will pull a row of pins at a time (7). A device to this design has been built and is undergoing testing; ultimately it must be tested with irradiated assemblies, to demonstrate the feasibility of the process and the extent that pin failure can occur. Further work is suggested by BNFP in scoping and investigating alternative handling, examination, and assay of the fuel pins once they are free of the fuel assembly. Results should impact both safeguarding and characterization of spent fuel. Also, verification of current results with simulated irradiated effects, such as dimensional changes generated by both nuclear and thermal processes, subsequent to an in-depth characterization study, should be undertaken (7).

After the fuel pins are removed from the PWR fuel assembly, an approximately 3.96 meter long skeleton consisting of the thimble tubes and spacers remains. Only the spacers remain after the fuel pins and water rods are removed from the BWR fuel assembly. A shearing mechanism is necessary to shear the PWR fuel skeleton into small lengths for placing in a canister. A suitable shearing mechanism that can be adapted for remote operation, with manipulator handling of the fuel skeletons should require minimal development work. However, a prototype unit should be employed to demonstrate the full range of operation. This same assessment also is applicable to the compacter that has to be adapted to compacting the pieces of fuel assembly hardware in the canister.

Alternative 4 requires much more equipment than for Alternative 3, since in addition to pulling the pins, the pins must be sheared into the inner canister and immobilized. The most extensive prior experience in shearing of irradiated fuel was that accumulated at the Nuclear Fuel Services, Inc., West Valley Irradiated Fuel Processing Plant during the period 1966-71. The shear employed at the West Valley Plant was fabricated by Birdsboro based on a mechanical design developed at the Oak Ridge National Laboratory (ORNL). A considerable amount of non-radioactive work with simulated fuel assemblies

had been conducted at ORNL, using a Birdsboro shear of essentially the same configuration as that subsequently installed at the West Valley Plant. Certain design features of this shear were less than optimum for the intended service and problems developed during the cold shakedown operations which persisted to some degree throughout the five and one-half years of operation. The difficulties generally were of relatively minor significance, and after a period of operation, the shear performed generally satisfactorily (13). A device for shearing fuel pins was also installed by General Electric at the Morris Plant (14).

The French firm Saint Gobain Techniques Nouvelles (SGN) has designed and fabricated five shears for processing irradiated fuel, one of which is installed at the Barnwell plant of AGNS. The shear proposed for this operation is that based on the SGN/AGNS machine, which employs several basically different design approaches to those components of the shear which gave problems in the NFS machine. Although the AGNS machine has not been operated on irradiated fuel, extensive cold shakedown operations, including checkout of the remote blade change and other maintenance procedures, does not appear to have indicated any fundamental problems with the shear. SGN shears of generally similar design have been installed in reprocessing plants in France, Japan, and India, and have been operated with irradiated fuel. Details of the operating experience with these shears, however, is not available.

The vacuum encapsulation and heat treat furnaces required for spent fuel immobilization are generally similar to equipment which has been used in similar types of commercial as well as radioactive operations, and should present no major equipment development problems.

A glass matrix has been chosen over cement or metallic alloys. There is an extensive background of work on the application of various glass formulations for the solidification of high level reprocessing waste, and a production facility employing a borosilicate glass formulation for solidification of reprocessing wastes has been in operation in France for more than three years (15). It does not appear, however, that any work has been done on the use of glass as an immobilization matrix for sheared fuel, thus the application proposed here will require a significant amount of

development work. Selection of the glass ceramic matrix as the glass formulation of choice for immobilizing the sheared fuel is based on the assumption that the thermal and mechanical properties of such a glass system would contribute better performance in the intended application than the standard borosilicate glass formulation.

The adaptation of a number of glass ceramic systems to the solidification of high level reprocessing wastes has been under investigation on a small scale by several investigators in recent years (16). In extending the small scale results to the application proposed here, a number of uncertainties are introduced. The ability to duplicate in a large scale container the temperature profile required for successful heat treating is questionable, and the influence of the thermal properties of the fuel materials on the behavior of the containers in the heat treat cycle is also an unknown. A substantial amount of development work will be required, therefore, and the final demonstration of a satisfactory process must be conducted on the same or nearly the same size scale as that proposed. Difficulties which might be expected would be the inability to control the crystal size adequately, with the development of relatively large crystals, rather than the micro-crystalline structure desired.

Some problems are foreseen in conducting the encapsulation operation according to the procedure set forth for Alternative 4, which will require considerable development work before the feasibility of the process as proposed can be demonstrated. Principal problems which are foreseen are:

a. The proposed glass formulation is expected to require a fusion and casting temperature in the range of 2200-2375F (1200-1300C); this can be expected to pose some problems in the melter, which would be the Pacific Northwest Laboratories (PNL) Joule-heated melter (17).

b. The operation would be conducted under vacuum, and would require that the canister of sheared fuel pieces be raised to approximately the casting temperature before the glass introduction is commenced. The vacuum on the system would be relieved as soon as the molten glass covered the sheared fuel body. The extent of out-gassing of the sheared fuel under these conditions cannot be predicted with accuracy, nor can the extent of migration of volatile fission products be quantified. An off-gas treatment system more complex than that described for Alternative 2 may be required.

c. The inner canister material proposed is 0.318 cm stainless steel, selected in order to provide adequate resistance to corrosion at the temperatures involved. Notwithstanding, it may prove necessary either to employ a more temperature resistant alloy, such as Inconel 600, for the inner canister material, or alternatively, it may be satisfactory to assure an inert atmosphere in the annulus between inner canister and the encapsulation furnace. Inasmuch as it is likely that the entire furnace interior would be evacuated during the vacuum casting operation, provision should be made in the design of the furnace to assure that any in-leakage during the high temperature operation is an inert gas, preferably argon.

d. The presence of a substantial fraction of the fuel in the form of fines can be expected to lead to some processing problems. Due to the density difference between the uranium oxide fuel material, and the glass (approximately 10.0 g/cm^3 versus 3.5 g/cm^3), it would be expected that the larger particles would remain toward the bottom of the canister; the particles of a sufficiently small size as to move upward with the rising glass are expected to react with the glass and become fixed as a part of the matrix. No information is available on the rate of reaction of sintered UO_2 with molten glass, and such would have to be obtained.

e. The lowest temperature at which a satisfactory rate of glass flow and encapsulation of the sheared fuel pieces can be achieved should be determined early in the development program, as lower temperatures during this phase of the operation would result in significant decrease in operating and material problems.

Safeguards for Alternative 4 require that a suitable system of traceability from identified assemblies to identified canisters be implemented, since individual accountability of spent fuel by assembly is lost as soon as the fuel bundles are disassembled. Any identification etched or stamped on the individual fuel pins is lost when the pins are sheared. The sheared fuel is not considered to be easily accessible because of the high level of radioactivity present, however in this form the fuel is in a more dispersible form than for the other alternatives.

The acceptability of the waste form of Alternative 4, from the standpoint of long term package integrity and radioactive releases, should be at least as good as that provided by the other alternatives. Although the barrier presented by the cladding has been breached by the shearing process,

the additional barriers presented by the glass matrix and inner stainless steel canister should provide the sheared fuel a high level of protection*.

4.2.1.3 Ranking of Alternatives

The relative simplicity and the great deal of prior experience in the removal of end fittings from assemblies causes Alternative 1 to be the most desirable method of spent fuel disposal of the four alternatives from a technical standpoint. Alternative 1 can be considered to be essentially technically equivalent to the Reference Process.

Following Alternative 1 in ranking by technical assessment are Alternatives 2 and 3. The complexity of the pin pulling mechanism of Alternative 3 and the lack of experience of pulling pins from irradiated fuel assemblies are its major technical drawbacks. The complexity and required precision of the laser needed to vent and reseal the fuel pins are even more severe drawbacks for Alternative 2, but to an extent, those factors are mitigated by the greater acceptability of the waste form because of the release of internal pressure within the cladding; therefore Alternatives 2 and 3 are considered to be technically equivalent. Alternative 4 is the least desirable method of fuel disposal from a technical standpoint. The process technology involving immobilization is not well developed and is likely to take considerable time to complete. Alternative 4 also shares with Alternative 3 the technical drawbacks associated with the pin pulling equipment. Furthermore, there is not a significant amount of prior experience to draw upon in the development of the immobilization process on the scale required, and the technical advantages of such a procedure are largely speculative.

*Einziger, Himes and Cash (18) state, "It is apparent that the release rates for all waste forms considered are comparable with none having any obvious superiority" (p 70). They down-rate the sheared/immobilized waste form, however, "because of uncertainties in fuel state due to loss of cladding integrity and additionally lack of time delay before onset of release in the case of a premature canister breach" (p 72). It should be noted that Einziger, et al, did not take into account the sealed stainless steel inner canister in their analysis; they also assume that the stabilizer is cracked and the sheared fuel is exposed to leaching immediately following breach of the canister. In fact there would be a delay equivalent to the time required to breach the inner canister.

4.2.2 Operating Assessment

This section considers the operational aspects of each alternative relative to the Reference Process, including an assessment of the additional material handling requirements and the product and operational quality assurance needs.

It has been assumed in this assessment that an appropriate development program has provided adequate information for good hardware and equipment design to perform the necessary operations, including a demonstration of the ability to perform the required operations repeatedly and safely. It is assumed, furthermore, that the operating life and maintenance requirements of the key components have been developed and demonstrated to a satisfactory degree of confidence.

4.2.2.1 Basis of Assessments

The operating assessment considers the aspects of the processes from the standpoint of problems likely to be encountered in operation, control, maintenance, operational and product quality assurance, and any other special considerations related to facility, equipment, and process materials. In developing an operating assessment of each process for purposes of comparing it to one or more alternative methods accomplishing the same objective, the complexity of the process in terms of the number and character of the operational steps involved and the problems likely to be encountered in adapting existing equipment and techniques are of considerable importance. The extent of mechanical operations which must be conducted in the remote radioactive environment is a significant consideration in evaluating one process against an alternative, as is also a requirement for equipment which must be serviced and maintained by hands-on methods. A matter of particular concern in assessing the feasibility of process equipment for remote operation in radiochemical environments is that of equipment reliability and ease of maintenance or replacement. Control of product quality to predetermined standards is an operational problem of considerable significance in processes for preparing radioactive material for long-term storage or disposal. The

need for special facilities or materials and the processing required for secondary wastes generated by each process were identified and evaluated.

The number of packages to be transferred from the above-ground facility to the repository is a consideration in assessing the operational complexity, as is also the underground handling involved in emplacing the packages. Material handling is, in fact, the dominant basis for assessing the mining activities, as the level of operating activity in the repository is determined by the number of packages received. The operational assessment thus considered both above-ground and below-ground factors.

The operational steps which differ for each alternative and the Reference Process were shown in Figure 4-10. These are the operational process steps which are to be evaluated. Those areas of commonality are not assessed except to recognize the reduction in material handling due to reduced number of canisters in Alternatives 3 and 4.

4.2.2.2 Comparison of Alternatives

The steps or operations which are required for removing the end fittings are common to each alternative and are straightforward and not complex. Additional handling of the assemblies occurs in placement of the assembly on the saw table and correct positioning for the cutting operation. Additional material handling over that of the Reference Process occurs in placing the end fittings (and bails) into a canister. There is an additional canister to be handled for approximately every 36 BWR assemblies processed and one for every 25 PWR assemblies. Changing the fines collection vessel and disposing of the fines should occur once a week.

The process control for the end fitting removal operation requires precision in adjusting the saws to proper cutting position on fuel assemblies and in aligning the canister for receiving a fuel assembly. Equipment will be designed for remote maintenance or replacement by remote manipulation. Worn or broken saw blades will need replacement periodically. The process equipment must be generally rugged yet capable of precise positioning; it should not be subject to a high rate of failure with the possible exception of

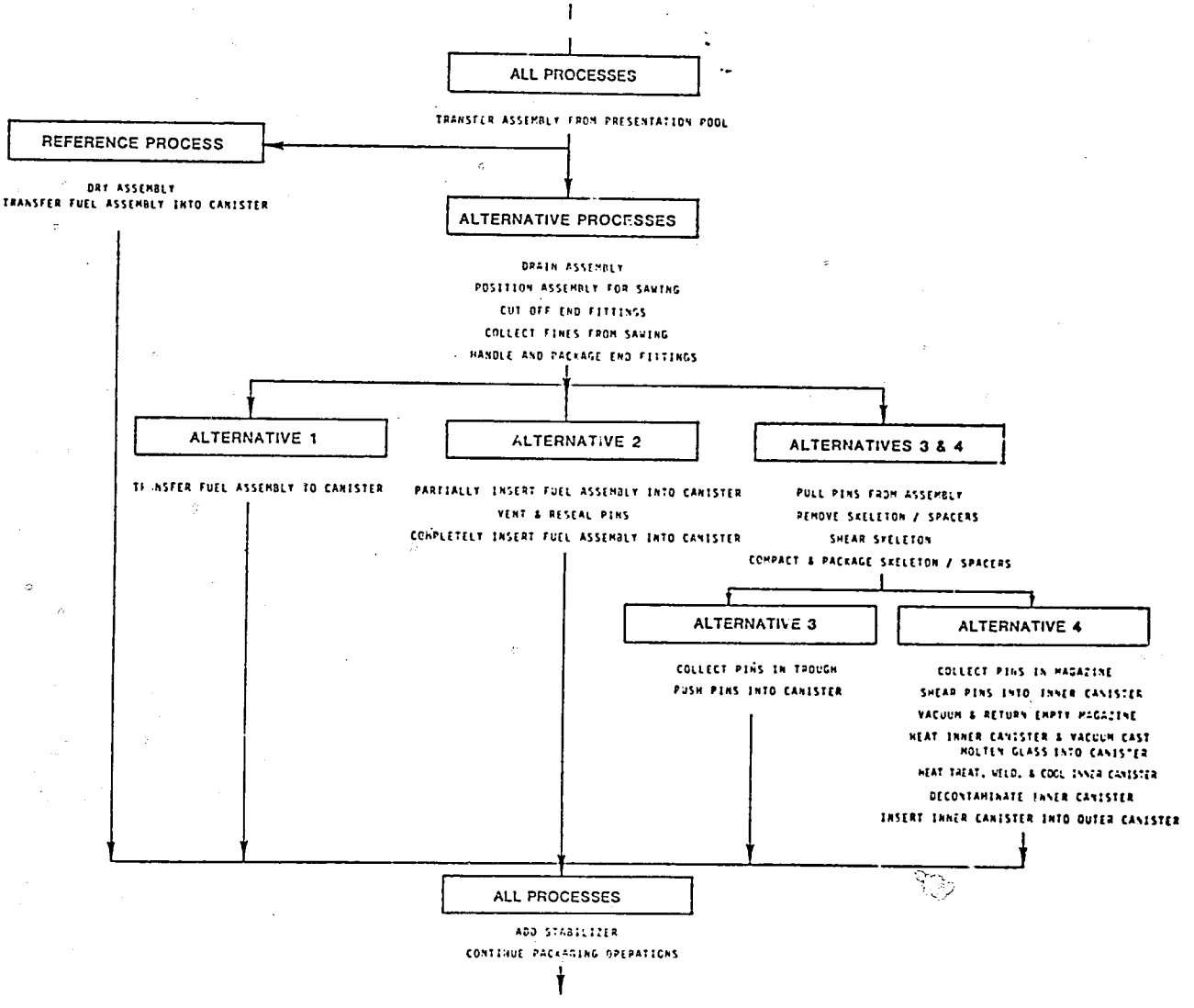

FIGURE 4-10

MAJOR OPERATIONAL PROCESS STEPS

FOR ALL ALTERNATIVES

the saw blades. The Weld and Test Cells of each alternative are designed so that there is reasonable plant downtime available to cope with both expected and unexpected equipment failures and still meet throughput capacity. Capability to handle variations in the design and/or type of fuel assemblies can be accommodated with adjustable positioners and clamps on the sawing table and adjustable sawing mechanisms or horizontally movable saws. To the extent required, variations in the lengths of assemblies will be accommodated by different lengths of canisters, and the equipment will be capable of accepting these variations as well.

Quality assurance measures for end fitting removal are not difficult and can be easily accomplished. Visual observation, together with interlocks between the saw table and saw mechanism will minimize the possibility of inadvertently cutting a fuel assembly at the wrong place. Records will be maintained identifying the fuel assemblies that are placed in each canister as well as any fuel assemblies having breached fuel pins and their disposition. Also, canisters containing the end fittings will be identified.

Operational and maintenance problems for all alternatives are therefore increased over the Reference Process by the end fitting operation and the above assessments are applicable to each alternative. For Alternative 1, insertion of the fuel assembly into the canister completes the discrimination between Alternative 1 process and the Reference Process as far as operations are concerned.

For Alternative 2, after end fitting removal, the process includes venting and resealing the top ends of the fuel pins. Piercing, venting and resealing the individual fuel pins is envisaged as being accomplished by a programmed sequence and movement of a laser that is correctly positioned to penetrate, vent and reseal each individual pin. This additional process step involves more material handling of the same material as the Reference Process and Alternative 1, with additional risk of accident.

The additional waste stream inherent in Alternative 2 results from the radioactive gases vented from the pierced fuel pins. Handling this vented off-gas stream is discussed in the preceding Section 4.2.1.

Process controls include an interlock between the laser operating mechanism and the positioning mechanism which precludes activitation of the laser if it is not focused on the end of a fuel pin, a scanning device, and controls for the vacuum apparatus and system which are standard industrial items.

It is difficult to conceive of remote maintenance being performed on the laser optical equipment in the hot cell; it will therefore have to be designed for ready removal for maintenance or repair; there is no experience to estimate a failure rate for the laser mechanism. There should be minimal maintenance required on the vacuum chamber once it is installed and tested; however, the apparatus that provides the vacuum tight seal around the girth of a canister should be capable of easy replacement. Both the vacuum pumping equipment and the cryogenic unit will have to be shielded because of the radiokrypton. Since krypton is a gas, they can be easily decontaminated for contact maintenance. The possibility of contamination of the vacuum inducing apparatus with radioiodine and tritium requires that means be provided to decontaminate this system.

The laser system should be capable of being programmed for various standardized fuel pin arrays, giving Alternative 2 the same process latitude as Alternative 1 or the Reference Process. Some level of quality assurance will be provided by appropriate programming of the laser cycle; visual monitoring would provide a coarse inspection of the venting and resealing operation. Incompletely resealed pins would not be reprocessed, nor would any intensive inspection procedures be adopted to identify such pins.

Following the reseal operation the partially inserted assembly is pushed all the way into the canister and proceeds through process operations as the Reference Process and Alternative 1. Number and size of canisters is the same as Alternative 1.

After the end fitting removal, the assemblies of Alternatives 3 and 4 go through an additional disassembly operation. The fuel pins are pulled from the assembly grid and collected for insertion in a canister (Alternative 3) or a shear magazine (Alternative 4). The remaining skeleton or spacers are compacted and packaged as well as the end fittings.

Pulling the fuel pins from an assembly, one row at a time, in principle, is a comparatively easy operation; however, problems can be expected if the fuel pins stick in their spacers. The multiple-head biter device for pulling a row of pins can also be expected to be a source of operational and maintenance problems. The long and narrow fuel pins are quite flexible and considerable care must be exercised during their handling as individual pins or as a collected group. Handling the individual pins and pushing groups of them into the multi-section canister or the square shear magazine can be accomplished with little difficulty provided these operations are done carefully. Proper precautions must be taken to avoid dropping the pins from the collection table and trough and to avoid excessive cross-overs and/or bending. The efficiency of the operation is dependent on a low pin pulling failure and/or sticking rate during the pulling operation. Once a fuel pin is stuck at the point of cut-off pulling force, it will have to be dislodged by special handling, and for this purpose, the assembly would be transferred to the Special Function Cell. In Alternative 3 broken or ruptured fuel pins will be placed in containers and sent to the Special Function Cell for processing. There is a greater possibility of contaminating the cell and equipment during the operation than for the Reference Process or Alternatives 1 and 2; therefore, the cell must have the capability to handle and control a considerable amount of decontamination waste.

The biter jaws are the most susceptible items for replacement; quick in-cell replacement is desirable. The skeleton shearing mechanism jaws should be capable of replacement by the use of remote manipulation. The pushing pad for the fuel pins will have to be replaced as a unit. The remainder of the equipment associated with the pin pulling operation is mechanically or hydraulically operated and can be ruggedly constructed so that little if any maintenance should be necessary.

There is no operating experience for estimating the failure rate of the biter jaws and unit; however, the Weld and Test Cells are designed to accommodate a reasonable amount of downtime and still meet production capacity.

Rigid quality assurance measures are required during design and fabrication of the biter units and the transducer unit in the pushing pad of

the hydraulic ram to assure proper functioning and reliability of those items. The quality control measures also extend to the maintenance and repair of the biter units and other critical hardware at the packaging facility. Documentation and control programs together with a records system which assures reliable information on the identity of the nuclear material must be implemented.

After the pins have been placed in the canister sections, subsequent operations are the same as for the Reference Process and Alternatives 1 and 2. Because of the increased fuel loading in the canister, there are only 30 percent of the number of spent fuel canisters required for Alternative 3 than for the Reference Process and Alternatives 1 and 2. Although the remaining process operations are the same as the Reference Process, the reduction in canisters means considerably less material handling for Alternative 3, and a sharply reduced requirement for package materials and emplacement hole operations.

The process of Alternative 4 picks up at the point where the pins from 3 PWR assemblies or 8 BWR assemblies have been placed in a shear magazine. Subsequent operations involve shearing the fuel pins and high temperature glass pouring and heat treating steps. Considerable radioactive particulate is generated, and radioactive off-gases may be released in sufficient quantities to be troublesome. Considerable remote handling of heavy, hot radioactive containers is required, and the shear equipment will require moderately complex remote maintenance procedures. At several points in the proposed process, canisters are moved from one location to another in the vertical position on dollies. In addition to lateral movement, some of these dollies must also be capable of raising and lowering a loaded canister from a work position to the transfer position. The requirement for radioactive dust handling in the Shear/Encapsulation Cell requires sophisticated design of the ventilation system. Installation of extensive decontamination facilities in the cell will be necessary.

The quantity of off-gas released in the Alternative 4 process is expected to require an off-gas processing system. This is an additional waste stream over the Reference Process and Alternatives 1 and 3; significantly

greater quantities of off-gas are likely to be encountered in Alternative 4 as compared to Alternative 2. The additional processing operations for the off-gas treatment have not been defined, but are likely to have several complications. Plateout of iodine in process lines and equipment may occur; the process may have to be operated under a vacuum, which will create further radioactive material control problems. Another secondary waste stream will be the decontamination solutions and loaded particulate air filters.

Precise control of the temperature profile during the heat treat cycle is required, including control of both the rate of heating as well as the rate of cooling of the inner canisters following the encapsulation operation. This is a reasonably straightforward control problem and should not present unusual difficulties in reliability.

In the Shear/Encapsulation Cell, it can be expected that significant quantities of radioactive particulate will be present and equipment requiring maintenance will have to be decontaminated before removal from the cell. Provisions for control of radioactive contamination include airlocks at the access points of the Shear-Encapsulation Cell. The shear proposed for Alternative 4 has been designed for ease of remote replacement of the knife and replacement of other components which may give difficulty. Maintenance on the furnaces is likely to be a source of operating difficulty. A significant failure rate is inevitable with the amount of mechanical equipment required. A reasonable level of maintenance outages can be sustained without prejudicing the production goals.

There do not appear to be any inherent restrictions in the capability of the process to handle any fuel which is likely to be presented; thus, Alternative 4 is the same as the Reference Process and the other alternatives in respect to process latitudes.

The operability of the process will be strongly influenced by the quality of the design and the conformance of the equipment installations to the design intent. Verification of the quality of the product by direct, non-destructive means does not appear feasible; thus, control of the product properties must be based on careful adherence to a predetermined process procedure. The formulation of the glass matrix material must be carefully controlled, and the prescribed heat treatment cycle must be followed

precisely. Short of sectioning a canister for internal examination, there appears to be no way to verify the integrity of the encapsulation.

Because of the nature of the quality control situation, as described in the preceding paragraph, it will be necessary to assure that considerable process documentation is maintained, as final product certification will have to be based on reliable information concerning the formulation of the glass matrix, processing times and temperatures, heating rates, cooling rates, and at least during the initial hot operations, internal canister temperatures. While not a difficult task, this will require a well-organized system and meticulous attention to detail in its implementation.

A number of special demands on the facility and equipment design have been discussed in the preceding paragraphs. These include the need for addressing such matters as the design of process equipment for ease of remote operation, remote servicing and replacement of components, and decontamination for hands-on maintenance work. The facility ventilation system must be designed with full consideration of the need for controlling airflows to minimize the possibilities of dispersion of particulate radioactivity from the Shear/Encapsulation Cell; airlocks must be provided at the process accesses to this cell, and ventilation air from this cell must be filtered before departure from the cell. Ventilation capacities in the various ventilation zones must be adequate, and the inter-zone controls designed to minimize the possibility of flow reversals due to pressure perturbations in the system, which might result in transport of particulate contamination to occupied areas of the plant.

The number of spent fuel canisters resulting from this process is the same as that of Alternative 3 and 30 percent of those required by the Reference Process and Alternatives 1 and 2. Following welding of the closure on the inner canister, the remaining operations are the same as the Reference Process. As was the case for Alternative 3, there will be considerably less material handling, sharply reduced usage of package materials, and reduced repository operations.

4.2.2.3 Ranking of Alternatives

In comparison to the Reference Process, all the alternatives require added operational steps. All the processes require additional facilities and equipment, which increase operational and maintenance problems. Alternative 4 is clearly the most complex, and involves the greatest uncertainties in the feasibility of the required operations. In terms of handling and the level of effort required in the repository, Alternatives 3 and 4 are outstanding due to the reduction by 65 percent of the number of canisters, both spent fuel and scrap, to be emplaced. Some of this advantage is offset by the substantial additional in-process material handling required by Alternative 4.

Alternatives 1, 2 and 3 are not sufficiently more complex than the Reference Process to make them unattractive from an operational standpoint. Alternative 3, however, offers the possibility of reducing the number of packages to be processed to 35 percent of those required for the Reference Process. This, and the similar reduction in cask transfer operations above and below ground and the reduction in repository operations, makes Alternative 3 the preferred process from an operational standpoint.

4.2.3 Risk Assessment

Each spent fuel disposal alternative was studied to compare its relative risk with that of the Reference Process. Radiation exposures to personnel from the Lag Storage Pool, Weld and Test Cell and Special Function Cell areas, transfer cask, and waste package were calculated. Exposure to the public from the disposal operations and from transportation of the spent fuel to the disposal facility was considered, as well as the probability and consequence of naive intrusion into the spent fuel package by a drilling crew 100 years after repository closure. The potential for accidents due to criticality, fire, or explosion were also considered.

4.2.3.1 Basis of Assessment

Calculations of radiation exposure to disposal operations personnel were based on ten year old spent PWR fuel. Source strength was based on the

dose rates calculated by Oak Ridge National Laboratory shown in Figure 4-11 (19). The maximum dose rate (at the assembly mid plane) was used as representative of the source strength, producing a worst case estimate. The canistered spent fuel was treated as an isotropic line source. Because of the varying number and positions of assemblies in the hot cells, the spent fuel in the cells was treated as an infinite plane source to provide a conservative estimate. Calculations of the shielding provided by the cell walls were performed using the buildup factors and density of ordinary concrete. The calculations for radiation exposure from the repository waste package emplacement considered radiation streaming through the bentonite backfill as the limiting factor, and it was assumed that the bentonite was packed about the shield plug at a density less than the theoretical maximum value. The calculated exposure levels were then compared to the occupational exposure limits contained in 10CFR20.101(a).

Limitation of exposure to the public from transportation of the spent fuel is governed by the Department of Transportation (DOT) regulations for rail or truck shipments of radioactive material (49CFR171-179), and by Nuclear Regulatory Commission (NRC) requirements for shipment of large quantities of radioactive material (10CFR71). Exposure to the public from the disposal operations was based on the release of radioactive material from the packaging facility as a result of sabotage. Risk to the public due to accidental intrusion by a drilling crew into a waste package 100 years after repository closure was determined using a scenario assuming that a typical wildcat petroleum well drilling operation strikes a canister and brings the radioactive material to the surface mixed with drilling mud. It is assumed, for the purposes of this scenario, that all records of the existence of the repository are lost or ignored, as well as all knowledge of radioactivity. A constant level of drilling activity is assumed, based upon the number of wildcat wells drilled in the United States in 1979. The probability of accidently intruding into the spent fuel canister was calculated for each alternative as a function of the spent fuel canister cross sectional area and the number of canisters. The probability was calculated for the partial as well as maximum possible intrusion into the radioactive material. The concept of risk was used to determine the worst case intrusion.

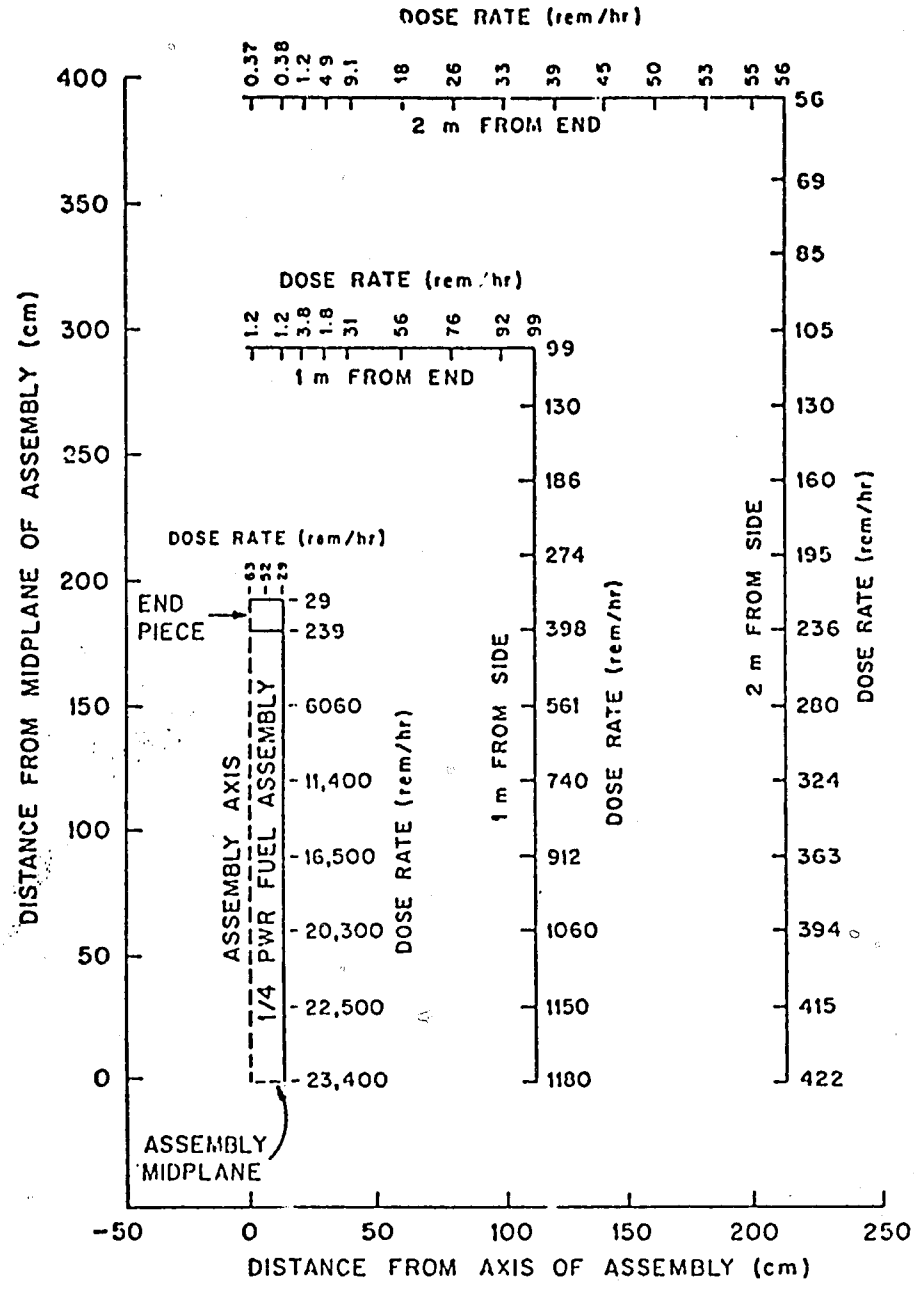

FIGURE 4-11

FISSION PRODUCT DOSE RATES FROM A 10-YEAR-OLD PWR FUEL ASSEMBLY

(From Reference 19)

4.2.3.2 Comparison of Alternatives

The radiation exposure to personnel from the Weld and Test or Special Function Cells, Lag Storage Pool, and transfer cask is acceptable for the Reference Process, and not significantly different for any of the alternatives. The possibility of radiation streaming through the bentonite backfill of the waste package is of greater concern for the Alternative 3 and 4 emplacements because of the greater amount of radioactive material contained. It is imperative that the bentonite be thoroughly packed about the shield plug to preclude any gaps or low density segments of material for those alternatives.

The most significant differences in radiation exposure to personnel between the Reference Process and alternative processes is likely to be among maintenance personnel. As discussed in Section 4.2.2, the alternatives all require more operations to package the spent fuel than does the Reference Process. The additional operations require more equipment, some of which is complex and may be expected to require more maintenance. Maintenance will require extensive decontamination of equipment and close monitoring of dose to maintenance workers. Each alternative requires a sawing operation for removal of the end fittings. Alternative 2 requires a complex laser mechanism to vent and reseal the fuel pins. Alternatives 3 and 4 require equipment to pull the fuel pins from an assembly; Alternative 4 additionally requires a shearing operation and subsequent immobilization of the sheared fuel. Any maintenance which is not possible by remote means will require equipment decontamination and subsequently more exposure to maintenance workers than expected for the Reference Process due to the additional equipment required by the alternatives.

Exposure to the public due to transportation of spent fuel is not likely to be significant, and is identical for each alternative and the Reference Process. Exposure to the public from the disposal facility during normal operations will be a fraction of that due to normal background radiation. In the event of an accident, however, the additional waste stream of the off-gas processing systems required for Alternatives 2 and 4 represent an increased potential for public exposure not present in the other processes.

Risk to the public in the long term by intrusion into the spent fuel by a drilling crew 100 years after repository closure has been determined to be insignificant for each disposal process.

The hazards of accidents due to criticality, fire, and explosion were also assessed for each alternative relative to the Reference Process. Criticality is not expected to be a problem for any of the processes. There has been considerable experience in water pool storage of fuel assemblies with no difficulty in maintaining subcriticality. Moderating materials introduced in the Weld and Test or Special Function Cells will be stringently controlled and geometrically safe positioning of the assemblies will be maintained. Once the fuel is placed into the canister, a stabilizer will be added minimizing the amount of moderator which can be introduced. The close packing of the fuel pins of Alternative 3 further precludes the possibility of significant amounts of moderator entering a canister. The potential of fire is somewhat greater for the alternative processes than for the Reference Process since the zirconium fines generated by the end fitting removal sawing process are pyrophoric. However, use of wet sawing should keep the additional fire hazard of the alternatives to a minimum. Alternative 4 presents the most significant risk of fire because of the elaborate heat treatment required during the immobilization process. The most credible potential explosion would be a result of sabotage. The consequences of an explosion would be most severe for Alternative 4 because of the higher dispersibility of the sheared fuel, but it is not likely that the guidelines of 10CFR100 would be exceeded at the site boundary if such an event did occur.

4.2.3.3 Ranking of Alternatives

The only significant difference from the Reference Process from the standpoint of risk is that of Alternative 4. The higher dispersibility of the sheared fuel would cause more severe consequences in the event of a disruptive event than would occur for the other alternatives; therefore in terms of risk evaluation Alternative 4 is the least attractive process.

The relative differences among the remaining three alternatives are minor. Of these, Alternative 2 is least advantageous because of the complexity of the laser mechanism for the venting and resealing of the fuel pins and the resulting fission gas waste stream, producing a somewhat greater risk of contamination in the event of an accident and during maintenance of the complex equipment. The equipment necessary to remove and close pack the fuel pins in Alternative 3 is also more complex, but this is partially offset by the lesser handling of material because of the reduction in number of spent fuel and scrap packages (65 percent). The remote sawing operation during end fitting removal is an additional potential hazard for all the alternatives, but the past experience of related operations in the industry indicate that it entails very low risk. Therefore, from the safety aspect, there is no significant element of risk in Alternatives 1, 2, or 3 which would override any advantages in respect to the Reference Process which any of these may possess.

4.2.4 Economic Comparison

The basis used for estimation of packaging and disposal costs for spent fuel was that set forth in the Bechtel CRRD report (1). However, these costs were modified to provide for the inclusion of a stabilizer between the spent fuel and the canister, to cover the added expense of the recommended spent fuel package design (see Section 3.3), to account for the different capital and operating costs of the alternative processes relative to those of the Reference Process, and to adjust for escalation.

The unit cost for packaging and disposal was calculated using the same basic methodology employed by DOE in developing its estimated charges for spent fuel storage and disposal (20), assuming that the facilities would be operated at about one-third capacity during the initial five years of operation and at full capacity thereafter.

4.2.4.1 Capital Costs

The capital costs for packaging and repository facilities for the disposal of spent fuel by the Reference Process and the alternative processes

are set forth in Table 4-1. The costs which were based on the CRRD were escalated from the second quarter of 1979 to June 1980 based on changes to U.S. Department of Commerce Composite Construction Cost Index (factor--1.1324). The Bechtel estimates associated with underground development of the repository were modified to cover the additional cost of welding and radiography equipment and additional liner emplacement equipment necessary to effect welding on-site, testing and emplacement of the reference package Inconel liner.

The lower costs shown for the repository facilities of the alternatives was due to the slightly smaller volume of salt removed during the mining of rooms for the first five years of operation (which were included in the initial capital costs). The lesser amount of mining for Alternatives 1 and 2 is due to the shorter spent fuel packages resulting from removal of the end fittings. The effect of a shorter spent fuel package for Alternative 3 is somewhat mitigated by the slightly larger diameter required, but the greatest reduction is the result of a lesser amount of emplacement equipment required inasmuch as the fuel pins from either 3 PWR fuel assemblies or 8 BWR fuel assemblies could be contained in a single emplacement canister, thus reducing the emplacement requirements of the spent fuel and the scrap by about 65 percent. The same reduction in emplacement requirements is possible for Alternative 4 but the larger spent fuel package diameter requires more excavation than for Alternative 3.

The higher costs shown for the waste handling facilities of the alternatives is due to the increase in equipment and space necessary to effect the additional operations required to modify the spent fuel assembly. Alternative 1 required more equipment and hot cell space to conduct the end fitting removal operation and associated auxiliary facilities, and this increase was common to the other alternatives as well. Alternative 2 required hot cell space and equipment in addition to that required by Alternative 1 to effect the venting of the fission gases, resealing of the fuel pins, and handling of the off-gas waste. The additional hot cell space and equipment required by Alternative 3 was related to the removal of the fuel pins from the assembly skeleton and their subsequent insertion into the canisters. Alternative 4 also required the removal of the fuel pins from the assembly

TABLE 4-1

ESTIMATED CAPITAL COSTS OF PACKAGING AND REPOSITORY FACILITIES FOR THE DISPOSAL OF SPENT FUEL

($-Millions, 1980)

	Reference Process	Alternative 1	Alternative 2	Alternative 3	Alternative 4
General Site Facilities	$ 32.8	$ 32.8	$ 32.8	$ 32.8	$ 32.8
Utilities and Site Distribution Systems	89.5	89.5	89.5	89.5	89.5
Repository Facilities	294.9	293.0	293.0	281.3	282.5
Waste Handling Facility	200.3	218.8	242.9	240.8	368.0
Other Buildings	43.0	43.0	43.0	43.0	43.0
Sub Total	$ 660.5	$ 677.1	$701.2	$687.4	$ 815.8
Engineering (15%)	99.1	101.6	105.2	103.1	122.4
Contingency (20%)	132.1	135.4	140.2	137.5	163.2
Total	$ 891.7	$ 914.1	$946.6	$928.0	$1,101.4

2 OF 3

skeleton, plus additional hot cell space and equipment to shear the fuel, immobilize the sheared fuel in glass, and package the gaseous and solid process wastes.

4.2.4.2 Operating Costs

The operating costs estimated by Bechtel for the CRRD included the cost of canisters as well as depreciation. Moreover, the Bechtel estimates were based on what appeared to be an average for each of three periods of operation of the facilities. In each period an increasing amount of fuel was scheduled to be emplaced in accordance with a specific scenario of fuel availability. However, for the purposes of this study, it was assumed that only two scenarios of disposal operations would be involved--during the first 5 years of operation spent fuel would be packaged and emplaced at a rate of 1500 MTU/year and thereafter packaging and emplacement would proceed at a rate of about 4500 MTU/year. Accordingly, the operating costs for these two scenarios of operation were determined as follows:

(1) Each element of operating costs was determined by interpolating (for the 1500 MTU/year rate), or extrapolating (for the 4500 MTU/year rate), the individual costs in the CRRD for the various average capacities of operation described therein. (The costs of canisters, liners, and depreciation was excluded from the operating costs and were handled separately).

(2) The operating costs obtained from (1), above were escalated to June 1980, from the second quarter of 1979 based on changes to the Wholesale Price Index--Industrial Commodities for Materials (factor--1.181), the Average Hourly Earnings of Workers on Mining Payrolls for Labor and Supervision (factor--1.004), and the Wholesale Price Index--Fuels for Utilities (factor--1.527) as appropriate.

(3) The estimated cost of labor and supervision associated with the welding of Inconel liners and the added effort of emplacement thereof was added to the labor and supervision cost obtained in (2), above.

Summaries of the annual operating costs thus obtained for the initial 5-year operation period and for the period of full operation are set forth in Tables 4-2 and 4-3, respectively.

Although there was a slight increase in the operating costs associated with the process and packaging operation of Alternative 1 over those of the Reference Process due to the additional processing steps involved, these were more than offset by lower costs in the repository operation due to a lesser quantity of salt that had to be mined (about a 6% reduction in emplacement room volume). Alternative 2 benefited from the same lower cost associated with repository operation since the same lesser quantity of salt had to be mined, but the higher costs of operations associated with end fitting removal, fuel pin venting and resealing, and waste gas handling operations combined to produce a higher operating cost than for the Reference Process. Alternative 3 also required a relative increase in operating costs associated with the disassembly operation compared to the Reference Process, but these were more than offset by lower costs in the repository operation due to the lesser quantity of salt that had to be mined and the need to effect fewer emplacements (only about 35 percent of those required by the Reference Process). Alternative 4 benefited from the same decrease in the number of emplacements required and in a similar lesser quantity of salt that had to be mined, but these savings were more than offset by the increased cost of operation associated with the disassembly, shearing, and immobilization process.

4.2.4.3 Package Costs

The costs of materials required for packaging were estimated based on the assumption that all fabrication of the canisters, liners, cages, and plugs would be performed by outside suppliers. Of course, the welding of the cap on the canister after the spent fuel had been inserted therein would have to be accomplished at the packaging and repository facility, as would inclusion of the stabilizer, the assembly of the liner (from two pieces), the liner emplacement and the final weld sealing of the liner. The costs for these activities have been included in the capital and operating costs as described in Sections 4.2.4.1 and 4.2.4.2.

TABLE 4-2

ESTIMATED ANNUAL OPERATING COSTS FOR PACKAGING AND REPOSITORY FACILITIES FOR THE INITIAL 5-YEAR OPERATION PERIOD

($ Millions, 1980)

	Reference Process	Alternative 1	Alternative 2	Alternative 3	Alternative 4
Labor and Supervision	$ 28.3	$ 27.9	$ 28.1	$ 24.8	$ 26.1
Supplies	2.2	2.2	2.3	2.3	2.6
Maintenance Materials	9.7	9.8	10.0	9.9	10.5
Equipment Replacement	5.9	5.8	5.8	5.7	5.7
Utilities	9.8	10.1	10.4	10.3	13.3
Administration & Overhead	10.4	10.2	10.3	10.1	10.4
Sub Total	$ 66.3	$ 66.0	$ 66.9	$ 63.1	$ 68.6
Contingency (15%)	9.9	9.9	10.0	9.5	10.3
Total	$ 76.2	$ 75.9	$ 76.9	$ 72.6	$ 78.9

TABLE 4-3

ESTIMATED ANNUAL OPERATING COSTS FOR PACKAGING AND REPOSITORY FACILITIES FOR THE PERIOD OF FULL OPERATION

($ Millions, 1980)

	Reference Process	Alternative 1	Alternative 2	Alternative 3	Alternative 4
Labor and Supervision	$ 29.9	$ 29.4	$ 29.7	$ 26.2	$ 27.6
Supplies	4.6	4.6	4.9	4.7	5.4
Maintenance Materials	9.7	9.8	10.0	9.9	10.5
Equipment Replacement	5.9	5.8	5.8	5.7	5.7
Utilities	14.7	15.1	15.6	15.4	18.4
Administration & Overhead	12.3	12.1	12.2	11.7	12.3
Sub Total	$ 77.1	$ 76.8	$ 78.2	$ 73.6	$ 79.9
Contingency (15%)	11.6	11.5	11.7	11.0	12.0
Total	$ 88.7	$ 88.3	$ 89.9	$ 84.6	$ 91.9

The estimated costs of packaging for both unmodified PWR and BWR assemblies for the Reference Process and each alternative are summarized in Tables 4-4 and 4-5 (See Appendix B for details of the development of canister and liner costs).

TABLE 4-4

ESTIMATED COSTS OF PACKAGING COMPONENTS FOR PWR SPENT FUEL

($/Package, 1980)

Component	Reference Process	Alternatives 1&2	Alternative 3	Alternative 4
Canister	$14,830	$14,240	$14,400	$16,600
Cage	800	790	890	2,400
Liner	59,220	57,660	58,760	72,410
Plug	770	770	790	920
Total	$75,620	$73,460	$74,840	$92,330

TABLE 4-5

ESTIMATED COSTS OF PACKAGING COMPONENTS FOR BWR SPENT FUEL

($/Package, 1980)

Component	Reference Process	Alternatives 1&2	Alternative 3	Alternative 4
Canister	$16,250	$14,970	$15,140	$18,390
Cage	1,110	1,080	900	2,610
Liner	62,860	59,610	60,840	77,610
Plug	770	770	790	920
Total	$80,990	$76,430	$77,670	$99,530

The total package costs (for both PWR and BWR fuel) of Alternatives 1 and 2 are lower then those of the Reference Process because a shorter package is required for the fuel with the end fittings removed than for the

unmodified fuel. The Alternative 3 packages share a similar advantage in length, but a slightly larger (1.3 cm additional) canister diameter is required to contain the fuel from 3 PWR or 8 BWR assemblies which offsets some of the gain achieved by the shorter length. The Alternative 4 packages cost more than those of the Reference Process as a result of the use of two canisters--an inner canister which serves as the immobilization process vessel in addition to the standard titanium canister, and the larger diameter (14 cm additional) canister required, which also caused the overall size of the emplacement package to be larger.

Direct comparison of the costs of the alternative packaging components with the Reference Process can be misleading because of the different capacities for spent fuel of the Alternative 3 and 4 canisters. Since those alternatives' packages can contain the fuel from 3 PWR assemblies or 8 BWR assemblies compared to 1 PWR assembly or 2 BWR assemblies for the packages of the Reference Process or Alternatives 1 and 2, comparatively fewer spent fuel packages (70 percent less) are required. The resulting savings in package costs for Alternatives 3 and 4 produce a distinct economic advantage relative to the Reference Process, Alternative 1, or Alternative 2. The package cost per fuel assembly provides for a clear economic comparison of the alternatives with the Reference Process and are shown in Table 4-6.

TABLE 4-6

PACKAGE COST/ASSEMBLY FOR PWR AND BWR SPENT FUEL

($/Assembly, 1980)

Reactor Type	Reference Process	Alternatives 1&2	Alternative 3	Alternative 4
PWR	$75,620	$73,460	$24,950	$30,780
BWR	40,500	38,220	9,710	12,440

4.2.4.4 Ranking of Alternatives (Unit Cost)

For the purpose of determining the unit cost ($/kgU) for packaging and disposal of spent fuel for the Reference Process and each alternative, the following assumptions were used:

(1) The packaging and repository facility would commence operation at the beginning of 1997.

(2) The construction of the facilities would be conducted during the period 1990 to 1996 inclusive; expenditures of total construction funds would be made as follows:

1990	5%
1991	10%
1992	20%
1993	20%
1994	25%
1995	15%
1996	5%

(3) Spent fuel would be received, packaged, and emplaced at a rate of 1512 MTU/year during the first 5 years of operation and thereafter at a rate of 4535 MTU/year until the capacity of the repository was reached.

(4) The annual package costs tabulated for the alternative processes assume that the end fittings removed from the spent fuel would be packaged in the same type and size of canister used for the spent fuel, but would not be emplaced in a liner assembly. It was estimated that there would be a total of 250 canisters each of end fittings from PWR and BWR fuel for disposal in each full operating year and 83 canisters of each of the two end fitting types in each year of the initial five year operational period. It was assumed that the only package cost associated with the end fittings would be that of the canisters required to contain them. Alternative 3 requires an additional 96 canisters of compacted PWR skeletons and 16 canisters of compacted BWR spacers for disposal in each full operating year, while Alternative 4 requires an additional 54 canisters of compacted PWR skeletons and 12

canisters of compacted BWR spacers. One-third of these amounts are required in each year of the initial five-year period.

The costs incurred each year were then discounted to 1980 using a 7.5% discount rate, which was that used by DOE in developing its latest estimate of charges for spent fuel storage and disposal. The unit cost was obtained by dividing the discounted costs by the discounted quantities of spent fuel received at the packaging and repository facility; this is the same as equating discounted costs to discounted revenues inasmuch as discounted revenues can be obtained by the product of discounted quantities and a fixed unit cost. Table 4-7 sets forth a schedule of discounted quantities of spent fuel and Tables 4-8, 4-9, 4-10, 4-11, and 4-12 set forth a schedule of the total costs for packaging and disposal of spent fuel and the unit cost calculation derived therefrom.

TABLE 4-7

DISCOUNTED AMOUNTS OF SPENT FUEL RECEIVED FOR PACKAGING AND DISPOSAL

(MTU)

Year	Actual Amounts Received	Amounts Discounted @ 7.5% to 1980
1997	1512	442
1998	1512	411
1999	1512	383
2000	1512	356
2001	1512	331
2002	4535	924
2003	4535	859
2004	4535	799
2005	4535	692
2006	4535	692
2007	4535	644
2008	4535	599
2009	4535	557
2010	4535	518
2011	4535	482
2012	4535	448
2013	4535	417
2014	4535	388
2015	4535	361
2016	4535	336
2017	1995	137
	77,580	10,828

TABLE 4-8

TOTAL AND UNIT COSTS FOR PACKAGING AND DISPOSAL OF UNMODIFIED SPENT FUEL (REFERENCE PROCESS)

($-Millions, 1980)

Year	Capital Costs	Operating Costs	Package Costs	Total Cost	Total Cost Discounted @ 7.5% to 1980
1990	$ 45			$ 45	$ 22
1991	89			89	40
1992	178			178	75
1993	178			178	70
1994	223			223	81
1995	134			134	45
1996	45			45	14
1997		$ 75	$ 279	354	104
1998		75	279	354	97
1999		75	279	354	90
2000		75	279	354	84
2001		75	279	354	78
2002		89	837	926	189
2003		89	837	926	175
2004		89	837	926	163
2005		89	837	926	152
2006		89	837	926	141
2007		89	837	926	131
2008		89	837	926	122
2009		89	837	926	114
2010		89	837	926	106
2011		89	837	926	98
2012		89	837	926	92
2013		89	837	926	85
2014		89	837	926	79
2015		89	837	926	74
2016		89	837	926	69
2017		39	368	407	28
	$ 892	$1,754	$14,318	$16,964	$2,618

$$\frac{\$2618\text{-million}}{10828\ \text{MTU}} = \$242/\text{kgU}$$

TABLE 4-9

TOTAL AND UNIT COSTS FOR PACKAGING AND DISPOSAL OF SPENT FUEL WITH END FITTINGS REMOVED (ALTERNATIVE 1)

($-Millions, 1980)

Year	Capital Costs	Operating Costs	Package Costs	Total Cost	Total Cost Discounted @ 7.5% to 1980
1990	$ 46			$ 46	$ 22
1991	91			91	41
1992	183			183	77
1993	183			183	71
1994	228			228	83
1995	137			137	46
1996	46			46	14
1997		$ 76	$ 270	346	101
1998		76	270	346	94
1999		76	270	346	88
2000		76	270	346	81
2001		76	270	346	76
2002		88	810	898	183
2003		88	810	898	170
2004		88	810	898	158
2005		88	810	898	147
2006		88	810	898	137
2007		88	810	898	127
2008		88	810	898	119
2009		88	810	898	110
2010		88	810	898	103
2011		88	810	898	95
2012		88	810	898	89
2013		88	810	898	83
2014		88	810	898	77
2015		88	810	898	71
2016		88	810	898	66
2017		39	356	395	27
	$ 914	$1,739	$13,856	$16,509	$2,556

$$\frac{\$2556\text{-million}}{10828\ \text{MTU}} = \$236/\text{kgU}$$

TABLE 4-10

TOTAL AND UNIT COSTS FOR PACKAGING AND DISPOSAL OF SPENT FUEL WITH FISSION GAS VENTED & PINS RESEALED (ALTERNATIVE 2)

($-Millions, 1980)

Year	Capital Costs	Operating Costs	Package Costs	Total Cost	Total Cost Discounted @ 7.5% to 1980
1990	$ 47			$ 47	$ 23
1991	95			95	43
1992	189			189	79
1993	189			189	74
1994	238			238	86
1995	142			142	48
1996	47			47	15
1997		$ 77	$ 270	347	101
1998		77	270	347	94
1999		77	270	347	88
2000		77	270	347	82
2001		77	270	347	76
2002		90	810	900	183
2003		90	810	900	171
2004		90	810	900	159
2005		90	810	900	148
2006		90	810	900	137
2007		90	810	900	128
2008		90	810	900	119
2009		90	810	900	111
2010		90	810	900	103
2011		90	810	900	96
2012		90	810	900	89
2013		90	810	900	83
2014		90	810	900	77
2015		90	810	900	72
2016		90	810	900	67
2017		40	356	396	27
	$ 947	$1,775	$13,856	$16,578	$2,579

$$\frac{\$2579\text{-million}}{10828\ \text{MTU}} = \$238/\text{kgU}$$

TABLE 4-11

TOTAL AND UNIT COSTS FOR PACKAGING AND DISPOSAL OF DISASSEMBLED SPENT FUEL (ALTERNATIVE 3)

($-Millions, 1980)

Year	Capital Costs	Operating Costs	Package Costs	Total Cost	Total Cost Discounted @ 7.5% To 1980
1990	$ 46			$ 46	$ 22
1991	93			93	42
1992	186			186	78
1993	186			180	73
1994	232			232	84
1995	139			139	47
1996	46			46	14
1997		$ 73	$ 84	157	46
1998		73	84	157	43
1999		73	84	157	40
2000		73	84	157	37
2001		73	84	157	34
2002		85	253	338	69
2003		85	253	338	64
2004		85	253	338	60
2005		85	253	338	55
2006		85	253	338	52
2007		85	253	338	48
2008		85	253	338	45
2009		85	253	338	42
2010		85	253	338	39
2011		85	253	338	36
2012		85	253	338	33
2013		85	253	338	31
2014		85	253	338	29
2015		85	253	338	27
2016		85	253	338	25
2017		37	111	148	10
Total	$ 928	$1677	$4326	$6931	$1225

$$\frac{\$1225\text{-Million}}{10828\ \text{MTU}} = \$113/\text{kgU}$$

TABLE 4-12

TOTAL AND UNIT COSTS FOR PACKAGING AND DISPOSAL OF IMMOBILIZED SPENT FUEL (ALTERNATIVE 4)

($-Millions, 1980)

Year	Capital Costs	Operating Costs	Package Costs	Total Cost	Total Cost Discounted @ 7.5% To 1980
1990	$ 55			$ 55	$ 27
1991	110			110	50
1992	220			220	92
1993	220			220	86
1994	276			276	100
1995	165			165	56
1996	55			55	17
1997		$ 79	$ 105	184	54
1998		79	105	184	50
1999		79	105	184	47
2000		79	105	184	43
2001		79	105	184	40
2002		92	314	406	83
2003		92	314	406	77
2004		92	314	406	72
2005		92	314	406	67
2006		92	314	406	62
2007		92	314	406	58
2008		92	314	406	54
2009		92	314	406	50
2010		92	314	406	46
2011		92	314	406	43
2012		92	314	406	40
2013		92	314	406	37
2014		92	314	406	35
2015		92	314	406	32
2016		92	314	406	30
2017		40	138	178	12
Total	$1101	$1815	$5373	$8289	$1460

$$\frac{\$1460\text{-Million}}{10828\ \text{MTU}} = \$135/\text{kgU}$$

The unit cost for the packaging and disposal of spent fuel are summarized in Table 4-13.

TABLE 4-13

UNIT COSTS FOR THE PACKAGING AND DISPOSAL OF SPENT FUEL

Process	Unit Cost ($/kgU,1980)
Reference	242
Alternative 1	236
Alternative 2	238
Alternative 3	113
Alternative 4	135

It should be pointed out that these costs are not directly comparable to the unit cost developed by DOE in DOE/SR-0006, since the latter document also included the cost of research and development, government overhead, decommissioning, and the operation of a repository system over a finite time period (1997-2010) in which the packaging and repository facilities were not used over their full lifetime nor was the capacity of any of the repository facilities reached. Rather the cost calculations contained in this analysis were based on the operation of a single packaging and repository facility over its lifetime, operating at about one-third of its capacity during the first five years of operation and thereafter at its full design capacity until the total disposal capacity of the repository was reached. It was believed that the unit costs obtained by this methodology would provide a better means of comparing the costs involved in the various disassembly alternatives. It should be emphasized that the dominant economic advantage shown for Alternatives 3 and 4 are results of the need for a lesser number of expensive packages; it should, however, be noted that the economic advantage of these alternatives is retained even at substantially lower package costs as is shown in Figure 4-12, which compares the unit cost of the Reference Process and Alternative 3 as a function of the relative package costs.

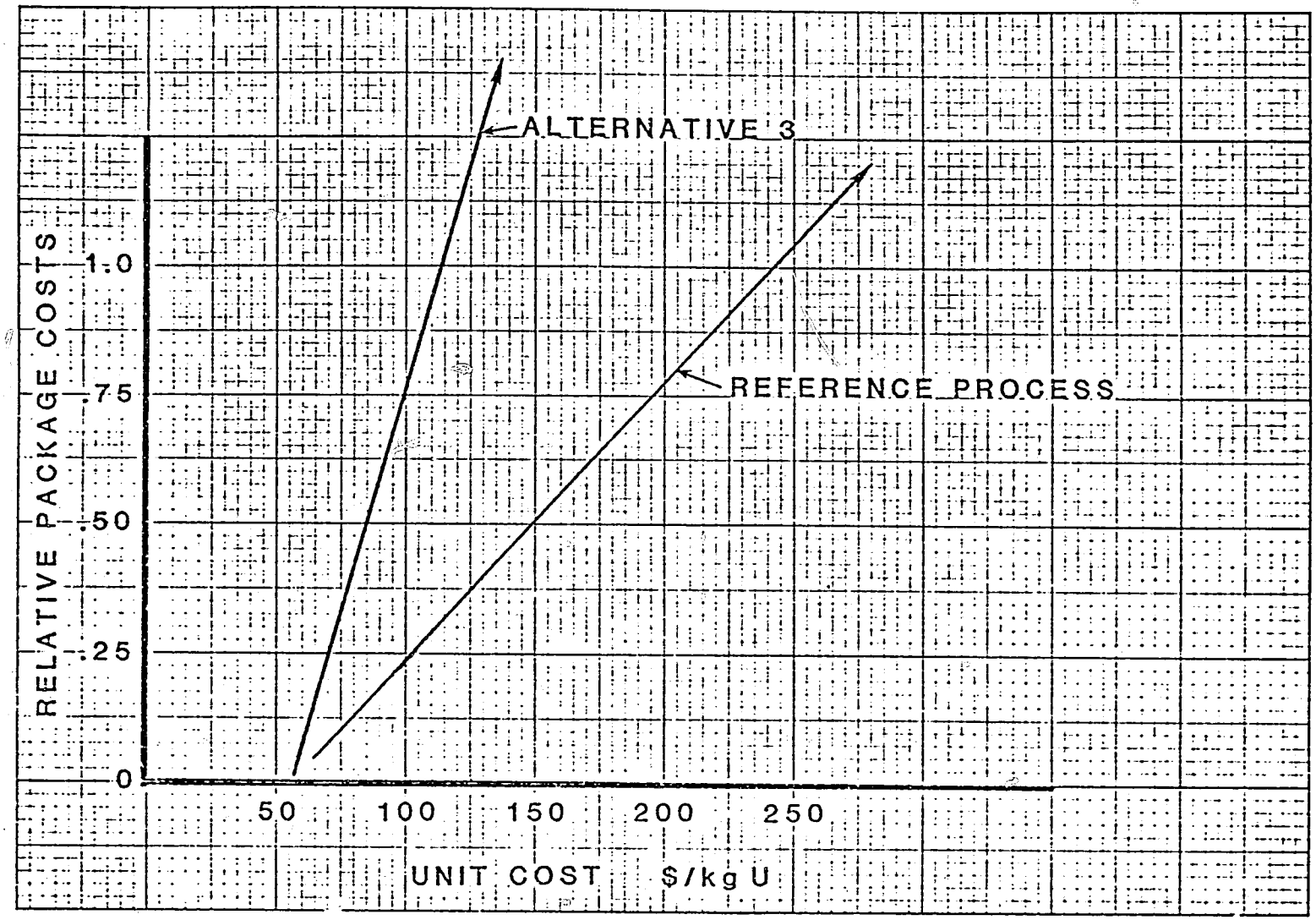

FIGURE 4-12

UNIT COST OF DISPOSAL AS A FUNCTION OF RELATIVE PACKAGE COSTS

4.3 CONCLUSIONS

A systems analysis approach was used to develop a basis of comparison of the alternatives. Assessments were made of the technical, operational, safety/risk, and economic considerations related to each of the alternatives, including both the surface packaging and underground repository operations.

The objective of the assessments was to provide an evaluation of the four alternative methods in reference to the disposal of unmodified spent fuel (Reference Process).

Sixteen principal attributes were identified among the factors assessed for each process, and each was assigned a weight representing its significance in the overall evaluation. With the exception of the economics assessment, which was expressed in terms of comparative unit costs in dollars/kilogram of fuel, each of these principal attributes was divided into two or more sub-attributes, to which the assigned weight was distributed.

A quantitative basis of comparison was then developed by assigning to each sub-attribute a number expressing its merit relative to that of the Reference Process. Summing the products of the individual figures of merit and the weight assigned to the sub-attribute permitted a ranking of each of the processes in each of the four assessment areas. The resulting rankings are given in Table 4-14.

TABLE 4-14

RELATIVE RANKINGS OF ALTERNATIVE PROCESSES AS DETERMINED BY FIGURE-OF-MERIT EVALUATIONS

Assessment Area	Reference Process	Alternative Processes 1	2	3	4
Technical	1	2	3	3	4
Operating	2	3	4	1	5
Safety/Risk	1	1	3	2	4
Economics	4	3	3	1	2
Overall	4	3	5	1	2

It will be apparent that the operating and economic assessments dominate the overall ranking of the alternatives. This is largely a result of the fact that the package design employed is a very costly item, and Alternatives 3 and 4 required only 30 percent of the number of spent fuel packages required for the Reference Process and Alternatives 1 and 2. Thus, not only are the overall packaging costs substantially lower for these processes, but the level of operational activity required to handle and emplace the waste packages is greatly reduced. It should be noted also that the overall cost advantage of Alternative 3 would be retained - although the magnitude of the difference would be less - even if a much less costly package were used, as is clearly shown by Figure 4-12.

The weighting factors were developed through extensive consultation with the JAI staff. These factors are described in detail in Appendix C and a sensitivity to the ranking by weighting is presented.

As a result of the assessments and comparative evaluations made during the course of this study, it is concluded that the preferred method of disposal of spent fuel is that of Alternative 3 - fuel bundle disassembly and close packing of fuel pins. Although Alternative 4 is ranked second in the overall rating, it is unattractive because of the increased complexity and major uncertainties in the operational feasibility of several of the processes involved. The overall advantage of Alternative 3 would be further improved if the package-to-package spacing in the repository could be reduced by increasing the allowable thermal loading.

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5.0 WASTE FORM PERFORMANCE IN REPOSITORY

An important consideration in the comparative assessment of the different disposal alternatives is the long term performance of the waste form in the repository. Of concern in this connection are factors which affect the degradation of the canister, the fuel cladding, or the fuel itself, and factors which affect the rate of transport of radionuclides from the package after breach of both the canister and the cladding. The waste forms cannot be assessed alone, but must be considered as waste forms/stabilizer/canister combinations.

The evaluation of waste form performance was performed by HEDL (1). Thirteen combinations of fuel form, stabilizer type, and fuel pin plenum condition were examined. Stabilizer types considered were gas, particulate, and solid, and the fuel forms corresponded to the Reference Process and the four Alternative Processes 1 through 4 previously identified. Table 5-1 identifies the relationships among the fuel forms and the "assessment waste forms". In conducting the evaluations, a number of possible functional requirements were assessed. These included:

1) Stabilizer Support Against Lithostatic Pressure
2) Long-Term Stability for Radionuclide Retention
3) Minimization of Cladding Degradation
4) Prevention of Canister/Repository Breach Due to Pressurization
5) Stabilizer Heat Transfer
6) The Stabilizer as an Independent Barrier to Radionuclide Migration
7) Prevention of Criticality

HEDL's overall ranking of the five waste forms in the various configurations is presented in Table 5-2 for the temperature regime considered most likely to prevail in the repository. Conclusions of the HEDL study were not available to JAI until after the process study (which was based on the use of sand as a particulate stabilizer) had been completed. Thus, although the HEDL study resulted in recommendations of other than a particulate-stabilized waste form as the preferred configuration for disposal, the following conclusions are pertinent to consideration of the sand-stabilized system which formed the basis of the process evaluations:

TABLE 5-1

RELATIONSHIP BETWEEN DOE WASTE FORMS AND THE ASSESSMENT WASTE FORMS

(from Reference 1)

DOE Waste Forms		Assessment Waste Forms								
		Stabilizer			Plenum State		Config-uration		Form	
		Gas	Particulate	Solid	Vented	Unvented	In Grid	Close Packed	Intact	Chopped
Unaltered Assembly	1.	X				X	X		X	
End Bells Removed From Assembly	2.		X			X	X		X	
	3.			X		X	X		X	
Rods Vented to Remove Gas and Resealed	4.	X			X		X		X	
	5.		X		X		X		X	
	6.			X	X		X		X	
Close-Packed Fuel Rods	7.	X				X		X	X	
	8.		X			X		X	X	
	9.			X		X		X	X	
	10.	X			X			X	X	
	11.		X		X			X	X	
	12.			X	X			X	X	
Rods Chopped and Immobilized in Matrix Material	13.			X	X			X		X

TABLE 5-2

WASTE FORM RANKING BY CONCERN -<425° C

(from Reference 1)

DOE Waste Forms		Assessment Waste Forms																
		Stabilizer			Plenum State		Config-uration		Form									
		Gas	Particulate	Solid	Vented	Unvented	In-Grid	Close-Packed	Intact	Chopped	Support Against Lithostatic Pressure	Breach of Canister due to Pressuri-zation	Stability Heat Transfer	Minimizing Cladding Degradation	Long Term Stability for Radionuclide Retention	Independent Barrier to Migration	Criticality
Unaltered Assembly	1.	X				X	X		X		4	1	2	1	1	1	2
	2.		X			X	X		X		2	1	2	1	1	1	1
End Bells Removed From Assembly	3.			X		X	X		X		1	1	1	1	1	1	1
Rods Vented to Remove Gas and Reseal	4.	X			X		X		X		4	1	2	1	1	1	2
	5.		X		X		X		X		2	1	2	1	1	1	1
	6.			X	X		X		X		1	1	1	1	1	1	1
Close Packed Fuel Rods	7.	X				X		X	X		2	2	4	1	1	1	1
	8.		X			X		X	X		2	2	4	1	1	1	1
	9.			X		X		X	X		1	2	2	1	1	1	1
	10.	X			X			X	X		2	1	4	1	1	1	1
	11.		X		X			X	X		2	1	4	1	1	1	1
	12.			X	X			X	X		1	1	2	1	1	1	1
Rods Chopped and Immobilized in Matrix Material	13.			X	X			X		X	1	2	3	4	2	2	1

"The stabilizer plays a minor role as an independent barrier unless its corrosion resistance is nearly the same or better than Zircaloy.

"As long as the cladding stays intact during the thermal period, all waste forms perform equally well in limiting radionuclide migration. If the cladding is compromised during the thermal period, fuel-stabilizer-canister interactions may occur which degrade the fuel significantly. This interaction must be studied if temperatures are set above 425°C since cladding integrity may be compromised.

"Waste forms with particulate stabilizers may have trouble helping the canister resist lithostatic pressure." (1)

Based on these and other conclusions presented. The authors note that:

"There is not enough available information to make a definitive division between acceptable and non-acceptable waste forms.

"The best waste form/stabilizer combination is the intact assembly, with or without end bells, vented or unvented, with solid stabilizer.

"A suitable alternative is the combination of bundled close-packed rods with a solid stabilizer around the outside of the bundle.

"The other possible waste forms are of lower ranking with the worst waste form/stabilizer combination being the intact assembly with a gas stabilizer or the chopped fuel." (1)

Although sand was selected as the particulate stabilizer in the JAI process study, it is recognized that several candidate materials would have to be evaluated prior to final selection, if a particulate stabilizer is to be used. It is also recognized, as pointed out in Section 3.3.3.2, that a metal alloy of low-to-moderate melting point could be substituted for sand without materially affecting the evaluations presented. The following summary of the waste form in-repository performance study is based on Reference (1) but attention is focused on that material relating to particulate stabilizers and the performance of waste forms incorporating them. Table 5-3 summarizes the waste form/stabilizer/canister combinations which have been studied in the process evaluation.

TABLE 5-3

WASTE FORM/STABILIZER/CANISTER COMBINATIONS

	Waste Form	Stabilizer	Canister
Reference Process	Intact Fuel Assemblies	Sand	Titanium Alloy
Alternative 1	End Fittings Removed	Sand	Titanium Alloy
Alternative 2	Vented Fuel Pins	Sand	Titanium Alloy
Alternative 3	Close Packed Pins	Sand	Titanium Alloy
Alternative 4	Sheared/Immobilized Fuel	Glass and Sand	Stainless Steel and Titanium Alloys

There are three questions which are addressed in the analyses underlying the assessment presented in this section. These are:

1. can the waste form/stabilizer and canister "interact" to cause premature waste package breach or waste form degradation, and its corollary

2. can the stabilizer act to aid the canister in the resisting influences which might result in premature degradation, and

3. what is the relative ability of each of the proposed waste forms to resist radionuclide migration after water intrudes into a breached waste package.

5.1 STABILIZER FUNCTION

As has been noted in Section 3, the primary function of a stabilizer, when used in conjunction with intact or partially disassembled fuel bundles, is to support the canister against lithostatic pressure. A secondary mechanical function is to provide some measure of support to the fuel pins against stresses sustained in handling the loaded canisters and in any seismic events after emplacement. A third function, of significantly less importance than the first two, is to modify the heat transfer environment in the canister. An additional benefit of having a stabilizer is the prevention of redistribution of fissionable nuclides into a critical mass at a point in time when the structural integrity of both the cladding and the fuel material is gone.

The stabilizer employed in Alternative 4 - a glass matrix to encapsulate the sheared fuel - serves an additional special function; it provides a means of sequestering the fuel fines which are released from the cladding when the fuel pins are sheared. In terms of its ability to fulfill requirements of the first function, either a solid stabilizer, such as is used in Alternative 4, or the particulate stabilizer employed in the Reference Process and the first three alternatives are acceptable, although it is obvious that more support would be provided by a solid, rigid stabilizer which completely filled all space in the canister not occupied by fuel and hardware. The presence of voids in the particulate stabilizer, and the propensity of the particulate to decrease in bulk volume when subjected to pressure, raise some questions as to the extent of support the particulate stabilizer would provide. This aspect is considered in the evaluation presented herein. In the absence of a detailed description of the force vectors acting in an asymmetric lithostatic pressure field (i.e., non-hydraulic), the analysis presented herein assumes resultant forces acting equally but in opposite directions along a diameter of the canister, and evaluates the extent to which the canister/stabilizer would react to such forces.

5.2 TECHNICAL ASSESSMENT OF WASTE FORM/STABILIZER COMBINATIONS

There are two time periods of concern. The first is prior to canister breach or a period of approximately 1,000 years* (during the early part of which fission products are the primary hazard concern) and the second is after canister breach. For the first 1,000 years, concern is with any interactions which could possibly compromise the waste package integrity, and any degradation of the fuel form which would eventually lead to an increased radioactivity release after waste canister breach. After the canister is compromised, the prime areas of concern are the prevention of radionuclide transport and an assurance that a critical mass will not be formed.

---

*Note that the analyses presented herein take no credit for the protection provided by the package liner either in respect to corrosive attack by formation fluids or in respect to resistance to lithostatic pressure. They assume breach of the canister 1,000 years after emplacement; they also assume direct application of the lithostatic pressure forces to the canister.

5.2.1 Assessment of Unbreached Package

5.2.1.1 Support for Canister Against Lithostatic Pressure

While in the repository, the canister will have to resist the lithostatic pressure in order to prevent rupture or even gross deformation of the retrievable package. The canister wall thickness of 0.64 cm titanium was chosen to provide corrosion allowance and handling stability but it is insufficient to resist deformation under lithostatic pressure which may be as high as 4,400 psi. Two situations exist in respect to the Reference Process and Alternatives 1-3; in all but Alternative 3, the particulate stabilizer must resist this pressure unaided, while in Alternative 3, the possibility exists that the compacted pins themselves would carry part of the load.

Two effects might occur when lithostatic pressure is transmitted to the compact particulate:

1) Rearrangement of the particulate to a higher packing density.

2) Fracturing of the particulate followed by rearrangement to a higher packing density.

Due to pin bow, the interpin spacing in a close packed waste form is not uniform. One cannot custom size a particulate which would just fit into the interpin gaps and act as a framework which holds the pins in place. Therefore, the best particulate stabilizer would have to have a particulate size distribution which minimized stabilizer compaction under lithostatic pressure.

In practice for a single particle size, 60% dense packing is the maximum achievable density without pressure compaction (2,3). Theoretically close packing should achieve 76% density. If one uses a binary particulate distribution sequentially packed with a 10 to 1 coarse-to-fine particle size ratio, 70% of the material being coarse, an 84% packing density can be achieved (2,3). Initially, the coarse fraction fills 60% of the void volume and the fine fraction fills 60% of the remaining 40% void or a 24% fill of the fine fraction in the void volume.

Due to the lack of complete close packing, particulates in the stabilizer can move under lithostatic pressure until a close-packed density is achieved. If the total void volume available to particulate was V_o, then this total volume would originally be filled with 60% coarse material and 24% fine material. Upon compaction, the coarse material will fill 76% of the new void volume V' and the fine particles will fill 76% of the remaining void for a fill of 18% of the new void, V'. The packing density is then 94% of the available void for a decrease of 10% in the void volume over the original stabilizer/pin mixture.

If all of this volume reduction were due to compression of the canister in one direction, an elliptical cylinder is formed. Since lithostatic deformation occurs in the transverse cross-section of the canister, the three dimensional problem can be reduced to two dimensions in the cross-sectional plane of the canister. The initial area occupied by fuel and stabilizer is:

$$A_{init} = A_{fuel} + A_{stab} \qquad [1]$$

The final area occupied by fuel and stabilizer is:

$$A_{final} = A_{fuel} + fA_{stab} \qquad [2]$$

where f is the reduced percentage of the stabilizer area due to compaction. A_{final} would be the area of the deformed elliptical canister which has semi-major and semi-minor axes a and b so that:

$$\pi ab = (A_{fuel} + fA_{stab}) = A_{final} \qquad [3]$$

The initially circular canister had a circumference

$$C = 2\pi\left(\frac{A_{fuel} + A_{stab}}{\pi}\right)^{1/2}$$

which must be conserved during the deformation process so that

$$\frac{a^2 + b^2}{2} = \frac{A_{fuel} + A_{stab}}{\pi} \qquad [4]$$

a and b are given by the two roots of the solution of the system of equations [3] and [4] .

$$a,b = \left[\frac{A_{fuel} + A_{stab} \pm \left[(A_{fuel} + A_{stab})^2 - (A_{fuel} + fA_{stab})^2\right]^{1/2}}{\pi}\right]^{1/2}$$

since the deformation forces are perpendicular to the axis of the canister, the area in the plane of deformation will only be reduced to 0.946% of its original value or f = 0.946 (i.e., 0.92 raised to the 2/3 power). With this value of f, the axes of the ellipse can be calculated from

$$a,b = \left[\frac{A_{fuel} + A_{stab} \pm \sqrt{0.108\ A_{fuel}A_{stab} + 0.105\ A^2_{stab}}}{\pi}\right]^{1/2}$$

Results are shown in Table 5-4. The eccentricity (ϵ) of the resulting deformed canister is given by:

$$\epsilon = \left(1 - \left(\frac{b}{a}\right)\right)^{1/2}$$

The values for eccentricity for the canisters of the Reference Process, Alternatives 1, 2, and 3 are also shown in Table 5-4. These are the maximum eccentricities which could occur from unfracturing compaction if the lithostatic pressure is great enough to cause unidirectional deformation of the canister.

TABLE 5-4

EFFECT OF TOTAL COMPACTION OF STABILIZER*

		Reference Process Alternatives 1 &2	Alternative 3	
Canister I.D. (in.)		13.5	12.5	14
No. of Assemblies PWR		1	3	3
BWR		2	8	8
Total Internal Canister Cross Section in.^{2}		132	123	153.9
Total Pin Cross Section, in.^{2}	PWR 264 pins/assy	29	87	94.8 +
	BWR 64 pins/assy**	23.5	93.8	101.6 +
Cross Sectional Area for Stabilizer, in.^{2}:				
	PWR	114	36	59.1
	BWR	119.5	29.2	52.3
Deformed Ellipse:				
Major semiaxis a/minor semiaxis b				
	PWR	7.66/5.68	6.79/5.68	7.67/6.25
	BWR	7.68/5.66	6.74/5.74	7.64/6.30
Eccentricity:	PWR	0.67	0.55	0.58
	BWR	0.68	0.52	0.57

* Adapted from Table B1 in Reference (1). Several changes to the original entries have been made in order to conform the parameters to those for the reference PWR and BWR fuel assemblies and the corresponding canisters, and to correct several errors in the original.

** Includes two water rods

\+ Includes the cross sectional area of the inner guide cage.

The two columns under Alternative 3 in Table 5-4 represent two different size canisters. The first canister (12.5 in. ID) assumes the pins are banded together and inserted with no internal guide cage. The second canister (14 in. ID) assumes an inner steel cage of the type shown in Figure 4-6 which requires a larger diameter canister for the same capacity of spent fuel pins.

In addition, fracturing of the particulate could occur which would change both the particulate size and shape distribution and hence the packing density. The propensity for this to happen is dependent on the compacting pressure and the compressive strength of the particulate material. Typical compressive strengths for particulates under consideration are given in Table 5-5. If the compressive strength of the particulate is below the lithostatic pressure, which could be the case for graphite and granite for instance, then crushing of the particulate would occur. Even if the compressive strength is above the lithostatic pressures, there could be considerable fracturing of the particulate. This has been demonstrated for quartz where compacting pressures of 2.2×10^3 psi (1/3 of the compressive strength) resulted in a significant change in the particulate size distribution (4).

Unfortunately, the relative ratio of compaction to fracturing in any given situation is unknown. Compaction studies, although conducted as low as 7000 psi, are usually conducted on loosely packed powder, i.e., 70% void fraction (5). On the other hand, fracture studies are usually done at 22,000 psi which is much higher than lithostatic pressures (3,5).

An accurate assessment of a particulate as a stabilizer which resists lithostatic pressure on the canister is difficult to make. First, while there is the possibility of compaction, it is not known if the lithostatic pressure is high enough to cause this compaction. Curves of final density versus density of the particulate at lithostatic pressures are necessary to answer this question. Second, while uniform compaction has been assumed in this analysis, a density gradient is established in pressed powders (4). Third, it is not known whether the canister can withstand the eccentricity at maximum compression. Furthermore, recognizing that no criteria for permissible eccentricity have been established, ratings based on the usefulness of resistence to lithostatic pressure can only be judged if the critical eccentricity of the canister is established (1).

TABLE 5-5

COMPRESSIVE STRENGTHS OF PARTICULATES

Material	Strength (psi)
Alumina	$1.4 - 4 \times 10^5$ [6]
Basalt	$2.8 - 5 \times 10^4$ [6]
Silica (amorphous/quartz)	6.4×10^4 @ 240C [3]
Granite	2.9×10^3 [6] 1.9×10^4 @ 240C (464F) [3]
Graphite	1.8×10^3 [6]
UO_2 (irradiated)	1.2×10^5 [7]

5.2.1.2 Long-Term Stability for Radionuclide Retention

The ability of the waste form to resist leaching after the waste package breaches will be dependent on the characteristics of the spent fuel pellet. These characteristics include surface area, grain boundary conditions, location of fission products in fuel structure, phases and surface structure. During the thermal period, prior to intrusion of water into the waste package, these characteristics may change with time. In order to reduce the leaching propensity, it would be desirable to choose a waste form and specify an upper repository temperature so that the fuel pellet does not deteriorate from its initial condition during the thermal period.

Five mechanisms have been identified which might alter the fuel pellet condition:

1. Chemical changes.
2. Volatile fission product migration.
3. Gas diffusion to grain boundaries.
4. Radiation damage due to decay of fission products with change in crystalline structure.
5. Fragmentation due to thermal and mechanical effects.

These mechanisms are analyzed in detail in Appendix D for an isolated intact pin. Based on these analyses, it can be concluded that there is no reason to expect significant changes in fuel pellet structure under the repository conditions expected to prevail. There are some changes in the chemical microstructure of the fuel, but these are unlikely to result in enhanced availability of radionuclides for transport.

5.2.1.3 Preservation of Cladding as a Barrier

Physical support of the cladding available from a particulate stabilizer is difficult to assess. Under repository conditions, it would not be expected that cladding creep due to the internal pin pressure would be sufficient to cause compaction of a particulate stabilizer; conversely, it is clear that the stabilizer would not provide support to the cladding. It must still be determined if in fact support for the cladding is necessary. This will be relevant only if the pin remains pressurized during disposal. Blackburn (8) identified stress-rupture and stress-corrosion cracking (SCC) as the two most likely stress-related cladding degradation mechanisms. A conservative estimate of 377C (711F) as the maximum clad temperature in the repository, based on stress rupture, was established (8). Whole pin anneals have demonstrated that this temperature limit for stress-rupture is probably conservative by at least 50C (90F) (9) (see Figure 5-1). This new temperature limit (425C or 797F) may still be conservatively low and could possibly be increased based on further analysis of presently available data.

The Stress Corrosion Cracking Initiation and Growth (SCCIG) Model (10) developed for the Electric Power Research Institute (EPRI) was used by Miller to evaluate the propensity of fuel pins for stress corrosion cracking (SCC). The model was used to estimate the maximum radial crack that could be present in an unfailed spent fuel pin after normal LWR exposure and to develop a 1,000-year fuel pin fail/no-fail boundary based on crack size, temperature, iodine concentration and internal pin stress.

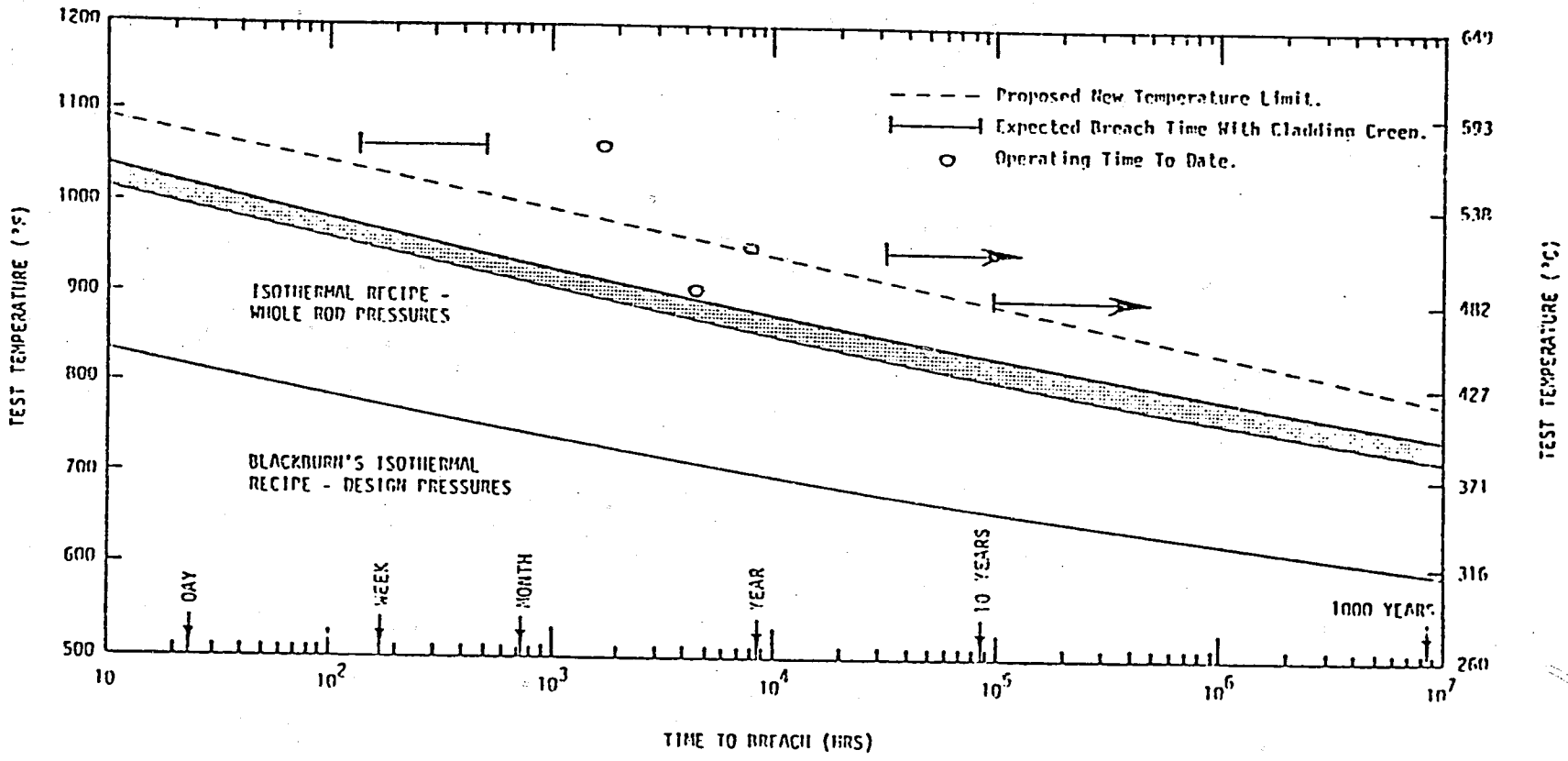

FIGURE 5-1

WHOLE PIN TEST RESULTS INDICATING THAT BREACH TIME PREDICTED BY BLACKBURN'S FORMULATION (8) IS VERY CONSERVATIVE

(from Reference 1)

As long as the probability of in-reactor failure is less than 1 percent, (the case for the vast majority of LWR fuel pins) then 99 percent of the remaining unfailed pins will enter the repository with incipient SCC cracks, if present, of less than 20 percent of the wall thickness. When the model is applied for repository/storage conditions, the 1,000-year fail/no-fail lines in Figure 5-2 are obtained. With realistic iodine concentrations representative of the majority of the pins ($5 \times 10^{-5} \rightarrow 5 \times 10^{-4}$ kg/m^2), and a 20 percent radial wall crack, Zircaloy cladding temperatures up to at least 600C (1112F) can be tolerated without SCC causing cladding breach.

Below 425C (797F), the cladding should survive the 1,000-year thermal period without failure. Should the temperature be increased above 425C (797F), the lifetime would be shorter in the absence of physical support of the cladding.

5.2.1.4 Stabilizer Heat Transfer

The maximum cladding temperature for in-grid and close-packed pins in a salt medium with a particulate stabilizer has been estimated to be 215C (419F) and 294C (561F), respectively. These values are less than the lower limit of the range in which cladding survival is considered questionable. Section 3.3.6 discusses the temperature differentials of the various waste forms.

5.2.2 Assessment of Breached Waste Package

Once the package is breached, the alternatives essentially reduce to only two; one is the fuel pins (whether in-grid or close-packed in the sand stabilizer) and the other is the sheared fuel pins immobilized in glass. These two alternatives are compared on the basis of radionuclide release rate and potential for criticality in this section.

5.2.2.1 Radionuclide Release Rates

The long term leach rate of radionuclides from the spent fuel material was assumed to be controlled by the dissolution rate of UO_2 and taken

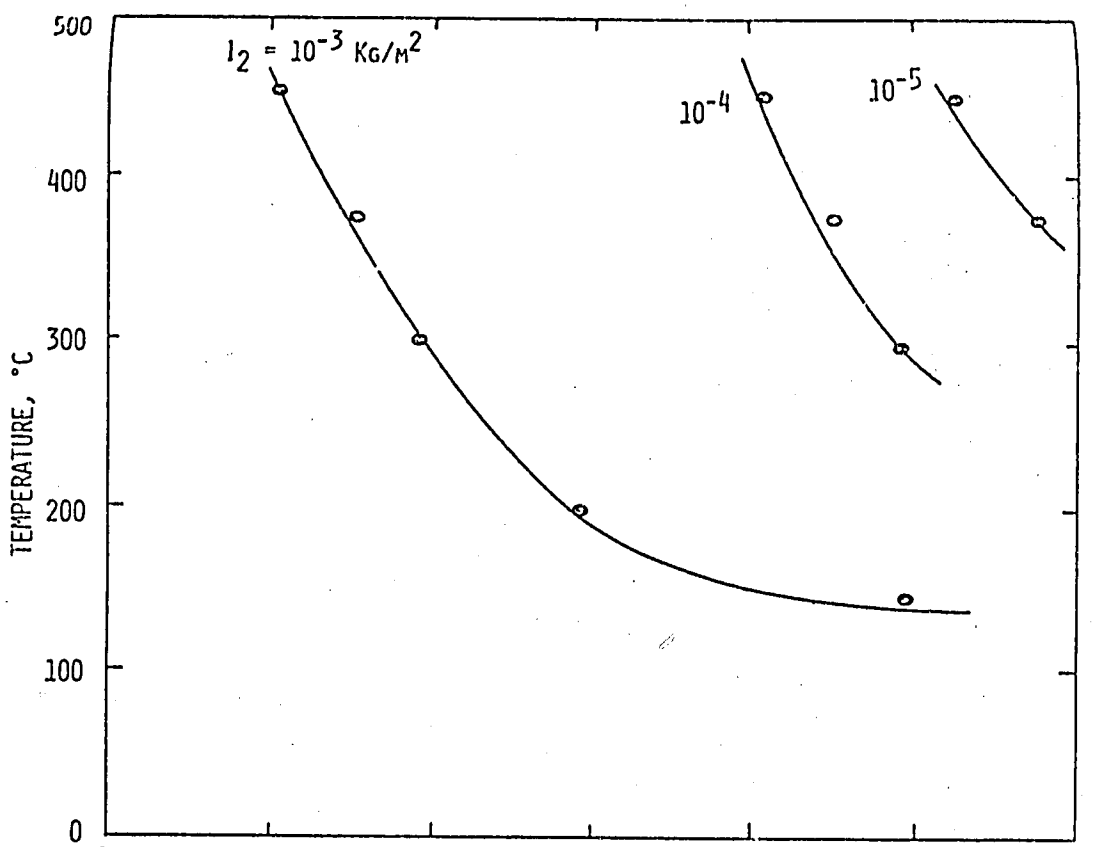

FIGURE 5-2

1,000-YEAR ISOCHRONS FOR IRRADIATED ZIRCALOY CLADDING WITH VARIOUS INTERNAL IODINE CONCENTRATIONS

σ_H(MPa) = 0.11 T (^{o}K) WAS USED TO CALCULATE THE PIN HOOP STRESS

(From Reference 1)

to be 10^{-5} g/cm^{2}day (11,12) (a volumetric release rate of approximately 10^{-11}/cm^3/cm^{2}s). This is based on the exposed geometric surface area of cracked irradiated fuel in cladding. The source formulation for the fission products in the gap assumed zero internal resistance to transport; i.e., there is no resistance to transport of nuclides from the surface of the pellet. The fraction of solid fission products residing in the gap would be highly variable; however, for purposes of this analysis, 10 percent of the solid fission products were assumed to be in the gap.

Zircaloy-2 cladding corrosion in brine at 250C (482F) has been measured at approximately 10^{-3} mm/yr (13) based on a maximum credible temperature in a salt repository. This temperature is much higher than is expected when the canister has breached, however. Data for corrosion of Zircaloy-2 in water as a function of temperature (14) indicate that the corrosion rate would probably be two to three orders of magnitude less for temperatures expected at the time of cladding attack. Therefore, the cladding corrosion rate in the model was assumed to be 10^{-4} mm/yr.

The internal pressure in unvented PWR fuel pins is assumed to be approximately 600 psi (8) and would be expected to cause cladding failure before the tube wall is completely corroded. Although a wall thickness of about 7 percent of the initial value would be sufficient to contain the pressure, based on an ultimate strength of 70 ksi (8), it is assumed here that the wall fails at 20 percent of its original thickness to allow for nonuniform corrosion and possible cladding flaws. Unvented BWR pins are assumed to fail at 10 percent of original cladding wall thickness due to the lower pressure (8). The geometric area available for transport through the failed cladding for the Reference Process, Alternatives 1 and 3 is based on the assumption that the initial breach size is approximately 0.1 mm in diameter.

A reasonable value for radionuclide transport speed was assumed to be 4 x 10^{-6} cm/s based on thermally driven ground water motion through typical repository basalt in the volume immediately surrounding the canister (15). In all cases examined for purposes of this assessment, the following assumptions were made:

- radionuclides are transported at the same rate as the groundwater

- transport rates and radionuclide releases are taken to be those at the inside of the canister boundary

- no credit is taken for retardation of radionuclide transport by the remains of the corroded barriers

- no credit is taken for retardation of radionuclide transport by the particulate stabilizer

- radioactivity levels of the fuel are taken to be those at 1,000 years from emplacement, the time of assumed package (canister) breach; no credit is taken for radioactive decay beyond this point

- no credit is taken for the delay in canister breach afforded by the Inconel liner provided in the overall waste package.

The barrier resistance of the cladding was assumed to decrease linearly from the initial value at time of cladding breach to the minimum value corresponding to the above transport speed at the time when the cladding is completely consumed by uniform corrosion. Vented and resealed fuel pin cladding is assumed to fail when uniform corrosion completely penetrates the cladding wall thickness. The cladding transport resistance characteristics are given in Table 5-6. Since the water must penetrate and exit the cladding, the barrier areas for transport are one half the circumferential area of the cladding tubes. The cladding transport resistance is assumed to vary linearly between time t_1 and time t_2 for the unvented pin alternatives, t_0 being the time of initial canister breach.

The resulting individual source responses for the PWR pins are given as an example in Table 5-7. The total fractional release rates as functions of time are plotted in Figure 5-3 for both the PWR and the BWR pins with unvented (pressurized) pins. The peak at time of initial cladding breach is due to the rapid release of the highly mobile fission products in the gap which are quickly depleted. The following build-up and peak at time t_2 corresponds to the matrix fuel leaching and dissolution as the cladding completely fails. The exponentially decaying tail after time t_2 indicates depletion of the fission products in the fuel matrix. It should be carefully

TABLE 5-6

CLADDING TRANSPORT RESISTANCE CHARACTERISTICS

(From Reference 1)

Item	PWR	BWR
Transport speed	4×10^{-6} cm/s	4×10^{-6} cm/s
Transport area	546 cm^2	725 cm^2
Time to initial breach of cladding (t_1)	1.44×10^{11} s (4560 yr)	2.30×10^{11} s (7290 yr)
Initial geometric parameter (x) at $t=t_0$*	10^6	10^6
Time to cladding destruction (t_2)	1.80×10^{11} s (5700 yr)	2.56×10^{11} s (8110 yr)
Transport resistance at $t=t_1$ (initial breach)	4.58×10^8 s/cm^3	3.45×10^8 s/cm^3
Transport resistance at $t=t_2$ (cladding completely gone)	4.58×10^2 s/cm^3	3.45×10^2 s/cm^3

* Reciprocal of the fraction of the area of cladding in initial breach

TABLE 5-7

EXAMPLE OF INDIVIDUAL SOURCE CONTRIBUTIONS (PWR, PRESSURIZED PINS)

(from Reference 1)

Time (10^{11}s)	Fuel Matrix Source		Gap Source		Total	
	Fractional Release Rate (1/s)	Integrated Release	Fractional Release Rate (1/s)	Integrated Release	Fractional Release Rate (1/s)	Integrated Release
1.44	8.26 (-12)	0	2.77 (-11)	0	3.60 (-11)	0
1.50	9.01 (-12)	5.17 (-2)	5.42 (-12)	8.40 (-2)	1.44 (-11)	1.36 (-1)
1.60	1.08 (-11)	1.50 (-1)	1.44 (-13)	1.00 (-1)	1.09 (-11)	2.50 (-1)
1.70	1.39 (-11)	2.72 (-1)	2.91 (-16)	1.00 (-1)	1.39 (-11)	3.72 (-1)
1.80	2.23 (-11)	4.44 (-1)	5.56 (-65)	1.00 (-1)	2.23 (-11)	5.44 (-1)
1.90	1.37 (-11)	6.20 (-1)	0	1.00 (-1)	1.37 (-11)	7.20 (-1)
2.00	8.37 (-12)	7.28 (-1)	0	1.00 (-1)	8.37 (-11)	8.28 (-1)
3.00	6.31 (-14)	8.98 (-1)	0	1.00 (-1)	6.31 (-14)	9.98 (-1)
4.00	4.76 (-16)	8.99 (-1)	0	1.00 (-1)	4.76 (-16)	9.99 (-1)

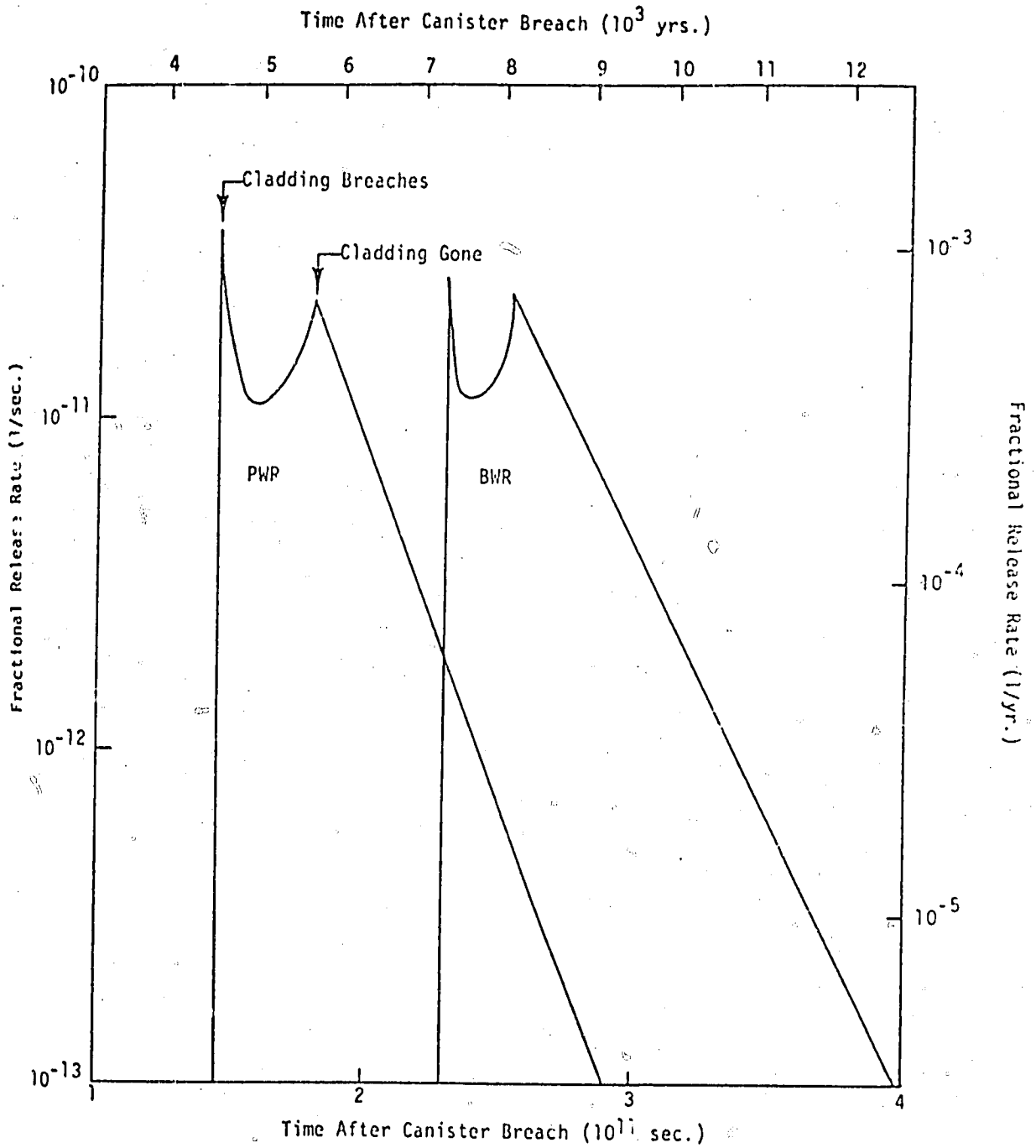

FIGURE 5-3

INDIVIDUAL PIN FRACTIONAL RELEASE RATES FOR UNVENTED (PRESSURIZED) PINS

(From Reference 1)

noted that the release rates are not corrected for radioactive decay since this would require more specific information than is contained in the general model.

In the case of vented and resealed fuel pins, the cladding is assumed to have an infinite transport resistance up to time t_2 when the cladding is completely corroded. The cladding barrier characteristics after time t_2 are given in Table 5-6. The resulting release rates are shown in Figure 5-4. Notice that very high peak release rates occur as the gap material is released over a relatively short time span.

For purposes of the release analysis, a standard PWR and BWR individual fuel pin was characterized in terms of its fuel and fuel/clad configuration. The source concentrations were normalized to be fractions of the total solid fission product inventory in the pin. The release rates are, therefore, fractional release rates of the total inventory. Since all pins in the canister are assumed to be equivalent, the fractional release rates just derived can be applied to the contents of the canister as a whole, and are independent of the number of fuel pins in the package.

In the case of sheared fuel in a glass stabilizer, it was assumed that the homogenized fuel and stabilizer corrode at the same rate as UO_2, i.e., 1×10^{-11} cm^3/cm^2-sec. Furthermore, it is assumed that shearing the fuel reduces 25 percent of the UO_2 to a granulated material which reacts with and becomes uniformly distributed in the glass. Two subcases will be considered here: Case 1 assumes that the radionuclides in the fuel-cladding gap remain in the gap during the stabilizer fill and cooling process and Case 2 assumes that the fuel-cladding gap material exits the gap and mixes uniformly with the stabilizer material during fabrication.

The resulting source concentrations (normalized to total canister inventory) are given in Table 5-8. These concentrations are based on the currently proposed canister dimensions for the sheared fuel alternative. The resulting release rates are shown in Figure 5-5. It should be noted that the low release rates indicated depend on the optimistic assumption that the glass stabilizer is completely intact (i.e., uncracked).

To determine the possible effect of an extensively cracked stabilizer glass, Case 1 was recalculated assuming the stabilizer was permeable with 1 percent of its cross-section available for transport and that

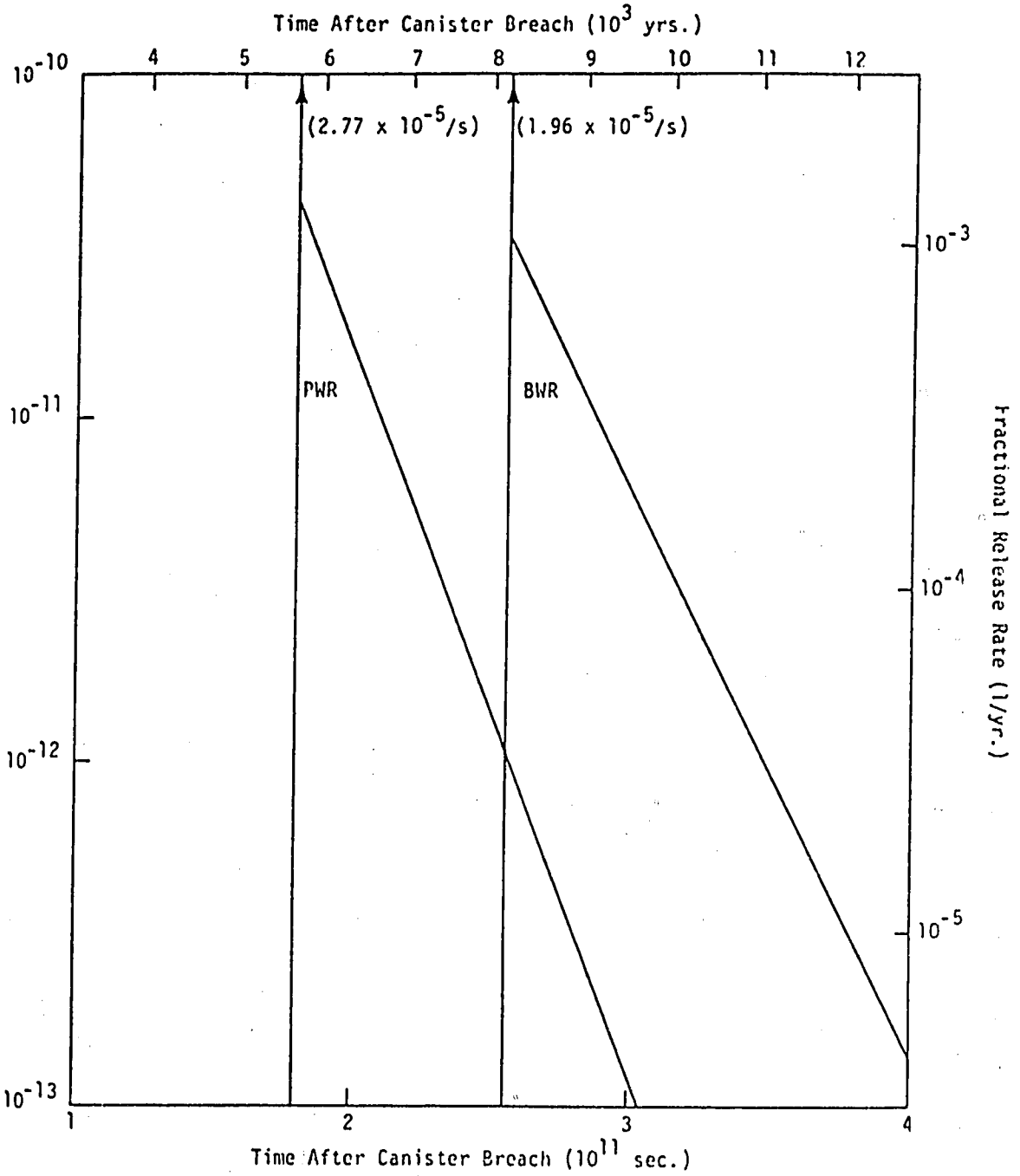

FIGURE 5-4

INDIVIDUAL PIN FRACTIONAL RELEASE RATES FOR VENTED AND RESEALED PINS

(From Reference 1)

TABLE 5-8

SOURCE CONCENTRATIONS FOR THE SHEARED FUEL CASE

(From Reference 1)

Item	Case 1	Case 2
Fraction of inventory in stabilizer	0.25	0.325
Concentration in stabilizer, cm^{-3}	4.36×10^{-7}	5.67×10^{-7}
Fraction of inventory in fuel chunks	0.75	0.675
Homogenized concentration in fuel chunks, cm^{-3}	1.44×10^{-6}	1.30×10^{-7}
Total concentration inside cage, cm^{-3}	1.88×10^{-6}	1.87×10^{-6}

Note to Table 5-8: The calculations presented here are based on canister inside radius of 21.6 cm and a fill height of 391 cm. These values differ from those presented in Figure 4-8 for the Alternative 4 package. The conclusions drawn from Figures 5-5 and 5-6 are not significantly altered if these differences are taken into account.

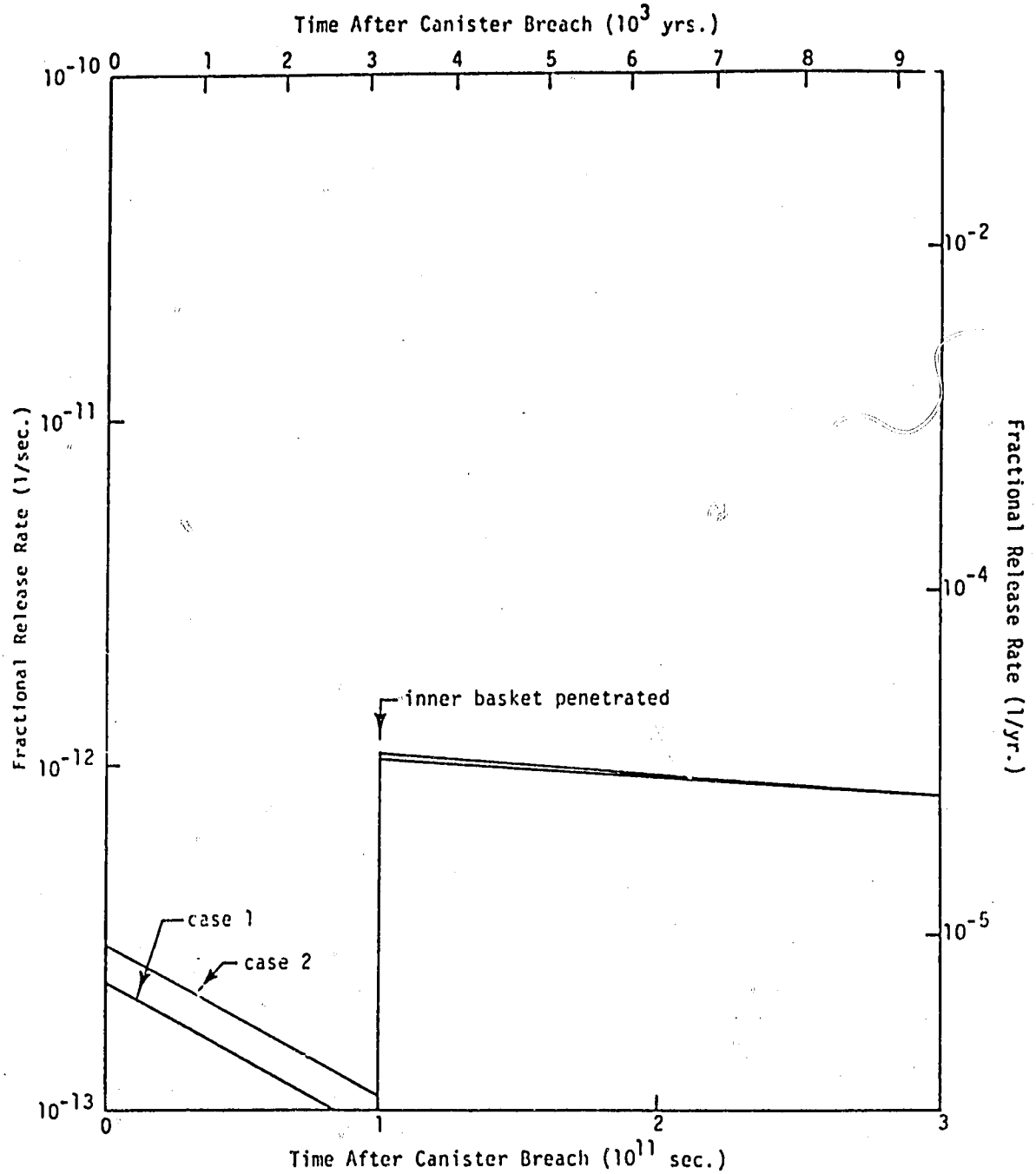

FIGURE 5-5

RELEASE RATES FOR SHEARED FUEL IN UNCRACKED STABILIZER

(From Reference 1)

all of the sheared fuel pin pieces are exposed at the time of canister breach.* The resulting release rate is shown in Figure 5-6. The initial spike at time t=0 is due to the sudden release of the material in the fuel cladding gap. If all the material from the gap were dissolved in the matrix (Subcase (b)), the response would be very similar except the initial spike would be absent.

Initial release rate spikes associated with the sudden release of gap fission products were observed in the Alternative 2 waste form and in the sheared fuel alternative with cracked stabilizer. These very high instantaneous release rates shown in Figures 5-4 and 5-6 are artifacts of the model assumptions and would probably be reduced by several orders of magnitude by probabilistic and internal transport delay mechanisms not included in this model. For example, the release rates indicated in Figure 5-3 imply that all the gap material is released within a few hours of cladding breakdown. In reality, all the fuel pins would not fail at one time and the cladding on any given pin would not corrode uniformly. This could easily extend the release period for the gap material over several years thereby reducing the peak release rate by about 4 orders of magnitude. In addition, the radionuclide retention in the stabilizer remains due to adsorption has not been included. This would, again, possibly reduce the apparent release rate from the waste form. The presence of these release rate spikes should not, therefore, be used as a basis of rating the general release rates of the various waste forms.

It is apparent that the release rates for the waste forms considered are comparable, with none having any obvious superiority. The only important difference is the delay time between canister breach and onset of release. It should be noted that:

*This implicitly assumes that the mass of sheared fuel pieces has not been penetrated by the matrix - an unlikely situation. It would be appropriate to assume that some fraction of the sheared fuel in excess of that exposed directly to the cracks in the stabilizer is available for transport, due to the existence of interconnected voids resulting from incomplete penetration of the stabilizer. Thus, Figure 5-6 will overestimate the release in comparison to the other cases examined. It should also be noted that neither Case 1 nor Case 2 take any credit for the sealed stainless steel inner canister as a transport barrier.

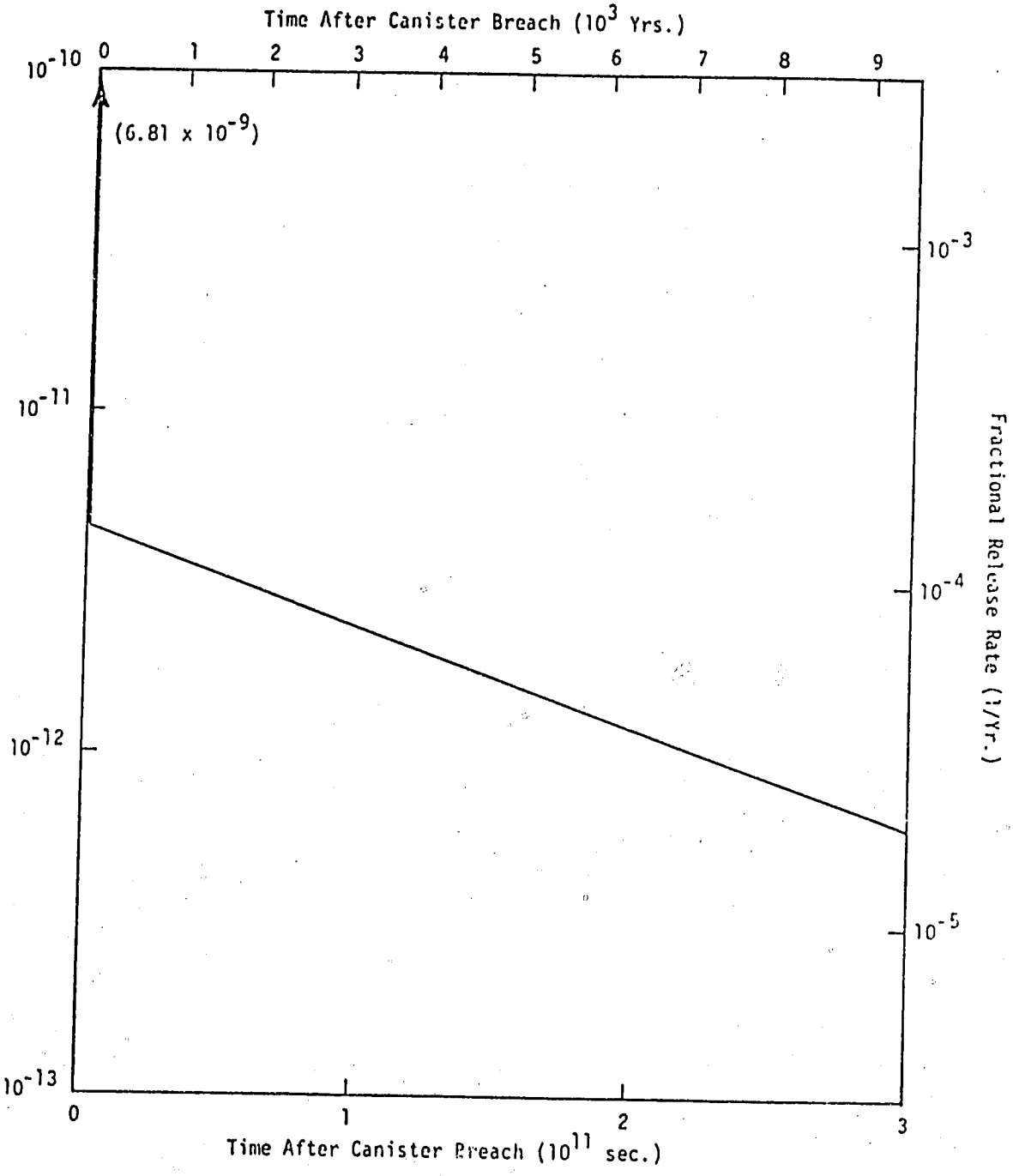

FIGURE 5-6

RELEASE RATES FOR SHEARED FUEL IN CRACKED STABILIZER

(From Reference 1)

1. The 4,000 to 8,000 year time delay before onset of release from intact fuel pins is due mainly to the presence of the intact cladding.

2. The sheared fuel in a cracked stabilizer produces zero time delay due to the absence of an intact cladding barrier*.

Since the fission product activity at times after the thermal period (~1,000 years) is due to long-lived isotopes, the time delays produced by the intact cladding do not produce a large reduction in activity released as shown in Figure 5-7 (16). If the canister were to breach prematurely during the time when short-lived radionuclides dominate, however, a built-in time delay of a few thousand years to onset of release would become a critical safety factor. The presence of the liner in the emplacement package would provide such a safety factor, as it would extend the time at which formation fluids first come in contact with the canister by 500-1,000 years.

The release characteristics of the various waste forms, once the cladding is breached, is dominated by the internal resistance of the fuel matrix source, i.e., the leaching characteristics of the fuel. Any degradation of the fuel prior to canister breach could drastically affect the absolute leaching characteristics and, hence, the overall performance of the waste form. It is important, therefore, to maintain cladding integrity prior to canister breach since breach of the cladding (e.g., shearing the fuel) would lead to large uncertainties in the state of the fuel after the waste package breaches.

5.2.2.2 Criticality

Until the burnup of LWR fuel exceeds approximately 20,000 MWD/MTU (17-20), there is enough fissile material in a single assembly to go critical if the correct ratio of fuel to moderation is allowed to occur. Inasmuch as (1) some assemblies received may have burnups less than this value, and (2) multiple packaging of fuel assemblies in a canister is

*The authors also note that a metallic stabilizer in the intact assembly case (i.e., the Reference Process) "may add 1000 to 2000 years to the delay time" (p 70).

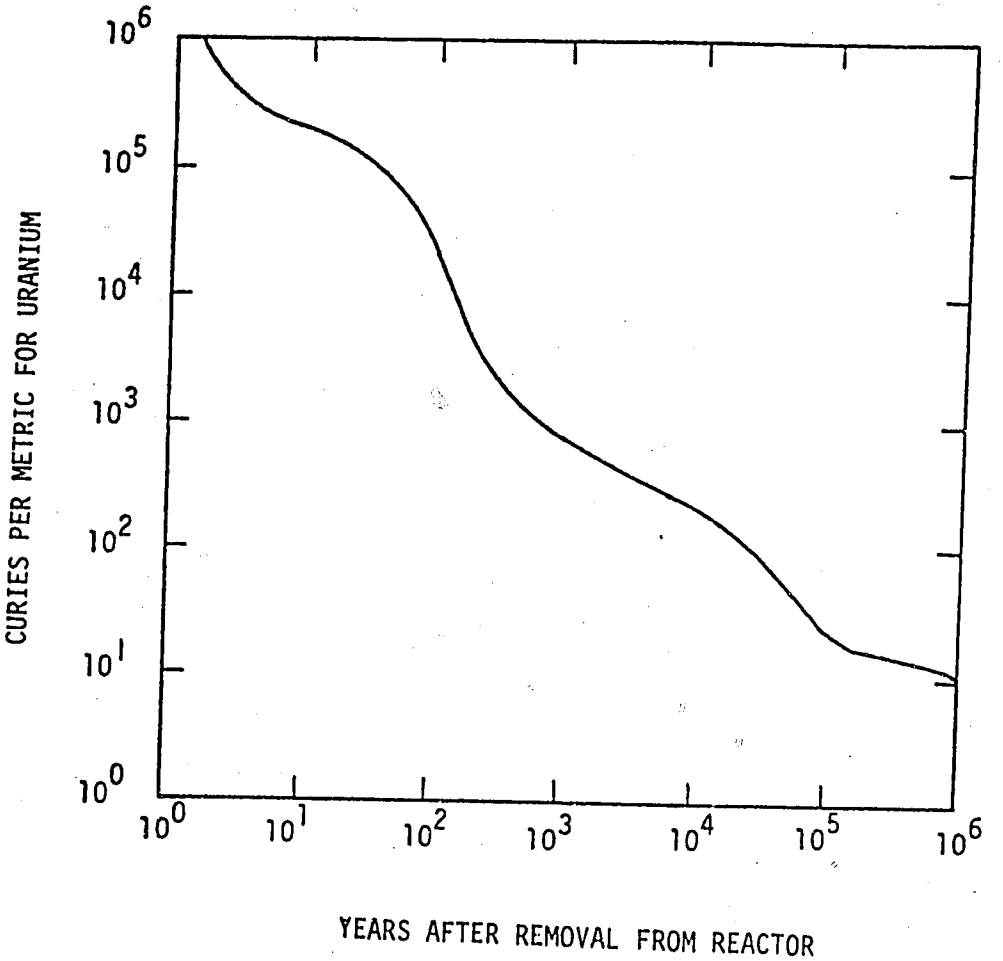

FIGURE 5-7
RADIOACTIVE DECAY OF PWR SPENT FUEL
(From Reference 16)

contemplated (up to 3 PWR or 8 BWR assemblies/canister), steps should be taken to prevent criticality from occurring in the repository. If criticality occurred, the heat load of the repository is increased at least an order of magnitude (at shut down, decay heat produces 7 percent of a reactor thermal load) and the radiation field is also significantly larger. In addition to considering the generation of gas due to radiolysis and radiation damage to the package components, it is necessary to reassess the integrity of the package and repository at the higher temperature. Furthermore, arguments relating long term safety of the repository to eventual decay of the spent fuel to the activity levels of natural uranium are no longer valid since new short-lived isotopes will be generated.

The Pacific Northwest Laboratory (PNL) has studied the question and concluded that for certain configuration of fuel (17-20), there is a possibility of criticality occurring. These results were questioned since they represented highly improbable and ideal situations. An independent assessment of the PNL criticality calculations (17-20) was made under more plausible scenarios to evaluate the criticality possibility, with the result that it is concluded that there is a very small, but real, possibility of criticality in the canistered fuel.

The calculations were made with the KENO-IV Monte Carlo code (20) and compared to the PNL results to provide verification of the present calculations. Although small differences are seen, the PNL studies are expected to be more accurate since their detailed pin cell calculations provide better effective cross sections; general trends should remain unchanged.

Based on both the Battelle calculations and those conducted at HEDL, a number of simplifications can be made to an otherwise complex multivariable problem. First, for a reasonably long cylinder ($\frac{\text{length}}{\text{diameter}} > 10$), the length of the cylinder is relatively unimportant so that geometry effects can be discussed in the context of only the cylinder diameter. For the same reason, the absolute amount of fissile material is unimportant, only its homogenized density. Secondly, hydrogen moderation is

very important so that the hydrogen to uranium ratio is a key variable. These considerations lead to the representation of criticality parameters shown in Figure 5-8. Figure 5-8 displays k_{eff} as a function of the ratio, volume of water to volume of UO_2:

$$R = V_{H_2O}/V_{UO_2}$$

Separate curves for $V_{H_2O} + V_{UO_2}$ = constant are shown for different cylinder radii (10", 13", 16"). The base case results in Figure 5-8 were computed for the following conditions: UO_2 and H_2O are assumed to completely fill the cylinder and are homogenized over the entire cylinder, the outside of the canister cylinder is water reflected, and the fuel is fresh (no burnup) at 3.5 percent enrichment.

Battelle studies (17-19) indicate that a water to uranium-oxide volume ratio of about R = 3.3 is optimally moderated for these conditions. The HEDL work is in agreement with this conclusion. Battelle's absolute values of k_{eff} are slightly higher than HEDL results but the densities are also slightly higher for the same postulated conditions.

From Figure 5-8, it is clear that criticality is a definite possibility in the Reference Process at a diameter, d, of 13 inches, if only fuel and water and some cladding are in the canister. For a single spent fuel assembly the H_2O to UO_2 ratio is close to R = 5 for dispersed fuel. Clearly a settling of fuel as the fuel pins disintegrate could drive this ratio towards the optimum value near R = 3.3. Moreover, if the canister disintegrated and, along with a shifting of the surrounding medium, allowed the effective homogenized radius to increase, then the system would become more reactive.

Since there is much more than a critical mass available, the surest way to avoid criticality is to limit the amount of moderation due to water intrusion. One way is to simply store more fuel in the canister which leaves less room for water. But this approach of course increases the amount of fissile material so that if gross distortions of the geometry at very long disposal times are considered, it becomes more difficult to rule out criticality scenarios. A more obvious approach is to ensure the displacement

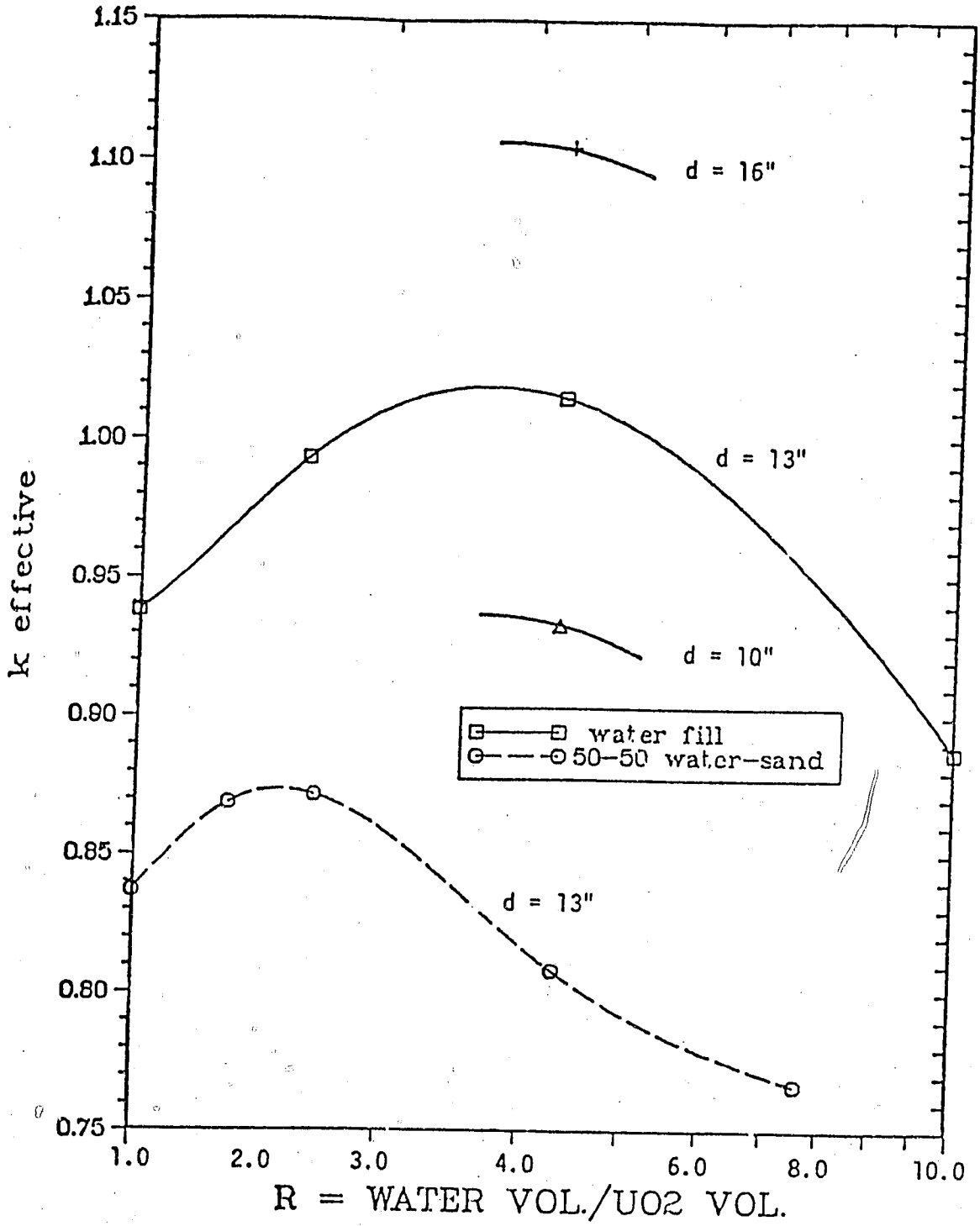

FIGURE 5-8

k_{eff} AS A FUNCTION OF THE H_2O VOLUME RATIO AND FOR DIFFERENT CYLINDER DIAMETERS, d

(Note: For each curve, both the water and uranium oxide volumes are varied to homogeneously fill the entire canister volume.)

(From Reference 1)

of significant amounts of water by an appropriate non-fissile filler. This is apparent in the "SAND" curve in Figure 5-8 where sand and water were assumed to occupy equal volumes. The value R retains its meaning as the ratio V_{H_2O}/V_{UO_2}. For a fixed volume, the sand replaces both water and U for a fixed ratio R, thus significantly lowering the reactivity.

5.3 CONCLUSIONS

Analyses presented in this section indicate that the reference fuel (waste) form should survive the projected 1,000 year life of the package (canister) barrier without any degradation which would be likely to result in an enhanced release of radioactive material at the time of package breach. Calculations show that, once a normal package (i.e., containing intact fuel pins) is breached, delays of the order of 4 to 8 thousand years occur before there is a release of radioactivity. In the sheared fuel alternative, no delay in release is apparent if it is assumed that the glass stabilizer is cracked and if no credit is taken for the double containment in the Alternative 4 Process; an uncracked stabilizer is estimated to provide more than three thousand years delay.

There is no essential difference in the performance of the various intact fuel pin waste forms with sand stabilizer, although the use of close-packed pins reduces the fraction of volume occupied by stabilizer and thus may reduce the amount of deformation of the canister under lithostatic pressure; Alternative 4, with its solid stabilizer, shows minimum deformation. Lack of a specific criterion for acceptable canister deformation in this circumstance precludes a determination of whether the sand is acceptable or unacceptable in respect to this attribute. Table 5-9 presents the relative rankings of the Reference and Alternative Waste Forms in respect to this and other attributes related to long-term waste form performance.

TABLE 5-9

WASTE FORM RANKING AT < 425C CLAD TEMPERATURE

(Extracted from Table 30 of Reference 1)

Functional Criteria	Reference Process and Alternative 1	Alternative 2	Alternative 3	Alternative 4
Support Against Lithostatic Pressure	2	2	2	1
Breach of Canister Due to Pressurization	1	1	2	2
Stability of Heat Transfer[a]	2	2	4[b]	3
Minimizing Cladding Degradation	1	1	1	4
Long Term Stability for Radionuclide Retention	1	1	1	2
Independent Barrier to Migration	1	1	1	2
Criticality	1	1	1	1

[(1 (best) → 4 (worst)]

(a) Reference to "Stability of Heat Transfer" should probably read "Stabilizer Heat Transfer"

(b) It appears that these rankings were derived from the ΔT's reported in Section 3.3.6. As was pointed out in Section 3.3.6, using the thermal conductivity of sand and giving no credit to the compacted fuel rods which occupy approximately 70 percent of the cross section of the canister results in a higher calculated temperature differential, and thus a lower ranking on this attribute than is justified.

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6.0 OVERALL SUMMARY AND RECOMMENDATION

Three essentially parallel studies were conducted to arrive at a conclusion regarding a preferred waste form/package combination for the disposal of spent nuclear fuel. One study was concerned with the elements of the disposal package and their interaction with the waste forms, one considered the impacts of alternative waste forms on the overall packaging/disposal system, and the third addressed the long-term performance in the repository of the waste form/canister combination. Based on the first study, a package configuration was selected and the same configuration was used in all of the comparative processes; similarly, a Reference Process was designated, based on the packaging and disposal of unmodified spent BWR and PWR fuel assemblies as described in a Bechtel Group report titled National Waste Terminal Storage (NWTS) Conceptual Reference Repository Description (CRRD). The alternative waste form processes were compared to the Reference Process and ranked in order of their overall attractiveness. Finally, the results of the waste form performance evaluation were combined with the process evaluations to arrive at a recommendation for a preferred waste form.

6.1 SELECTION OF PACKAGE

The waste package was defined as all elements of the fuel confinement system which are placed in the cavity bored in the repository host rock.

Elements of the standard, or reference package were:

- a canister of 0.64 cm titanium alloy (TiCode-12)
- a stabilizer inside the canister
- a borehole liner of 2.54 cm Inconel, with seal welded cover
- sand filler in the annulus between the canister and the liner

- a shield plug between the canister and the liner cover
- bentonite backfill around the liner, 30.5 cm thick.

Only dimensional differences distinguished the packages in the various alternatives, and no incompatibilities between the waste forms and the package were identified in the study of the package/waste configurations. In order to provide a practical basis for process comparison, sand was selected as the stabilizer, recognizing that a particulate stabilizer is not optimum for the function of resisting lithostatic pressure. A low melting metallic alloy could be substituted for the sand stabilizer without significantly affecting the process evaluations.

The Reference Process and the alternatives assumed that the spent fuel for processing is the standard Westinghouse 17x17 assembly (PWR) and the General Electric 8x8 BWR-6 assembly (BWR). Variations on these basic fuel assemblies can be handled without materially affecting the processes.

6.2 SYSTEM EVALUATION

A common basis of comparison of the four alternative processes against the Reference Process was developed by first identifying for each the process steps necessary to reach the objective of placing the fuel in a repository in the prescribed package configuration. Following identification of the process steps required, a process method which could reasonably be expected to be feasible under conditions of remote operation was devised; a conceptual process equipment layout was then made to estimate the added facilities required over those necessary for the Reference Process. Material and equipment requirements for each alternative were identified, and the impacts of the differences among the four alternative processes and the Reference Process on the packaging facility and the repository were identified.

With this common basis established for the processes, a systematic comparison of the relative merits of the different alternatives was conducted in the manner discussed in the following paragraphs of this section.

6.2.1 Method of Comparison

The pertinent concerns in respect to the five processes were divided into four areas: technical, operational, safety/risk, and economic. Assessments were made of each of the processes in each of these four areas and these assessments formed the basis for the quantitative comparisons.

The technical assessment considered the number and nature of additional operations required over those in the Reference Process and the nature of the equipment and operational procedures required to accomplish the desired objective. The state of development of the technology and the equipment, and the level of experience which has been acquired in the same or similar operations was evaluated. Other factors considered in the technical assessment included the waste form effectiveness, and the safeguards effectiveness of the process.

The operations assessment considered: the complexity of the process, as reflected by the number of operational steps required; the nature of the equipment, its reliability and maintenance requirements; the nature of and processing required for secondary wastes generated in the process; and the material handling in the packaging facility and repository. Radioactive material control problems inherent in the process were identified and their operational impact assessed.

The safety/risk assessment considered the probable sources of exposure both to plant workers and to the public which were considered to be inherent in the process, and an assessment was made of the accident potential including the probability and consequences of criticality, fire, explosion, or effects of natural phenomena events (earthquake, tornado). Occupational safety aspects were considered, and the alternatives were compared in respect to the probable effects of an intrusion into the repository by a drilling crew one hundred years after closure of the repository.

The economic comparison of the various alternatives addressed the capital costs of the respective facilities required, the operating costs, and the packaging costs. These were expressed finally as the total unit cost over the life of the facility in dollars per kilogram of uranium emplaced in the repository.

6.2.2 Results

In comparison to the Reference Process, all the alternatives require additional operational steps, and employ processes which are less well demonstrated than the Reference Process. All of the processes require additional packaging facilities and equipment, which increase operational and maintenance problems in the packaging facility. Alternative 4 is clearly the most complex of the alternatives, and involves the greatest uncertainties in the feasibility of the required operations. In terms of material handling and the level of effort required in the repository, Alternatives 3 and 4 are outstanding -- nearly sixty-five percent fewer emplacements are required for these alternatives. Some of this advantage in the case of Alternative 4 is offset by the substantial additional in-process material handling required. A significant increase in the advantage of these two alternatives would result if the thermal loading of the repository could be increased, permitting a closer spacing between the larger packages than is required by the 60 kW/acre loading assumed in this study; a decrease in the spacing for the larger packages would substantially decrease the amount of mining required to establish the storage corridors.

In terms of safety/risk considerations, all alternatives are less advantageous than the Reference Process, although the differences become significant only in connection with Alternative 4, where there is a substantial increase in the risk of facility contamination from the shearing operation, and where a difficult off-gas treatment is required. Potential hazards from criticality, fire, explosion or natural phenomena events do not appear to be significantly different for any of the alternatives, although the sheared fuel of Alternative 4 would increase whatever potential risk there is from any of these disruptive events.

The economic comparison was affected by three principal factors: operational complexity, facility requirements, and the costs of packaging materials. Of these, the material costs dominated the comparison; high material costs for the Reference Process and Alternatives 1 and 2 were a result of the much larger number of packages required in comparison with Alternatives 3 and 4, and established these latter alternatives as clearly the most advantageous economically.

Although the factors considered in the technical and operational assessments described earlier have a profound impact on the economics of the respective processes, an effort was made to separate the technical and operational assessments from any influence by the economic consequences of technical and operational factors. The economic evaluation was likewise made independent of considerations related to the technical uncertainties or to the magnitude of probable operational and maintenance problems likely to be encountered with the more complex processes. It might also be noted that the economic analysis did not consider the probable costs of development programs which would be required to demonstrate the respective alternative processes at a level which would provide reasonable assurance of success in their application in a production facility. The principal impact of taking this factor into account would be to reduce somewhat the economic advantage of Alternative 4 over the Reference Process and Alternatives 1 and 2.

6.3 WASTE FORM PERFORMANCE IN REPOSITORY

The assessment of waste form performance considered two time periods: the period from emplacement to breach of the canister, which is assumed in these analyses to be approximately 1000 years, and the period of time following canister breach.

6.3.1 Method of Evaluation

Thirteen combinations of spent fuel configurations and stabilizers were evaluated in respect to stabilizer functions and to any factor or combination of factors which might influence the integrity of the canister/waste form/stabilizer combinations (and hence the time to canister breach), or might alter the transport of radioactive material from the package after the canister barrier has been breached. Of these, only five combinations are pertinent to this study--the Reference Process Waste form and those of Alternatives 1-4, all with particulate stabilizer.

The ability of the particulate stabilizer to resist effects of lithostatic (i.e., non-hydraulic) pressure forces on the canister, and the

probable extent of distortion of the cylindrical canister under such forces were examined. Stability of the fuel form and probable effects of possible fuel/pellet degradation mechanisms on radioactivity release rates were examined, as also was the expected longevity of the cladding as a barrier in the alternatives involving canistering of intact fuel pins.

After breach of the canister, the evaluation was primarily concerned with the probable rates of release of radioactive material from the fuel as a function of time after canister breach, and the factors which affect these rates. The potential for assembling a critical mass in the repository after loss of cladding and fuel structural integrity was also examined.

6.3.2 Results

In general, all intact pin forms were found to be about equivalent in long term performance, with the vented pin having a slight advantage in delaying onset of release of radioactivity in the distant future. No degradation mechanisms were found which would impair the fuel/canister/stabilizer integrity during the first 1000 years, and no significant differences were observed in release rates after breach of the barriers. The only important difference found was the time delay between canister breach and onset of release, which ranged from 4000 to 8000 years for the alternatives involving packaging of intact rods, and zero delay for the sheared fuel with a cracked (glass) stabilizer, when it is assumed that the inner canister of Alternative 4 loses its integrity at the same time as the titanium alloy canister does.

Criticality in fuel material, due to rearrangement of fuel geometry following loss of integrity of the clad and fuel structure, was found to be a credible possibility. Although no evaluation of the probability of a long term criticality problem was made, it was concluded that the only case in which there appeared to be a possible criticality problem was the intact assembly with a gas stabilizer. Absent specific criteria as to allowable distortion of the canister under lithostatic pressure, no conclusion could be drawn as to the acceptability or unacceptability of the particulate stabilizer to resist that pressure.

6.4 SUMMARY OF EVALUATIONS

In order to arrive at an overall rating of the waste form/package alternatives, the relative merits of each in respect to each of the issues identified in Section 1.1 must be considered. The issue of compatibility of waste form with the overall repository package does not provide a basis for discriminating among the alternatives, as all packages were essentially identical and no incompatibilities were found.

A qualitative comparison of the four alternative processes with the Reference Process in respect to their impacts on packaging and repository operations reveals some obvious advantages and disadvantages to each, as summarized in Table 6-1. A more quantitative evaluation, taking into detailed account considerations relating to the technical status of the process and equipment, the nature and severity of potential operating problems, the relative safety/risk inherent in the respective processes, and the overall economics of the packaging, handling, and disposal operation, make possible a ranking of the alternatives in respect to these assessment areas. This ranking is shown in Table 6-2. Not immediately apparent from Table 6-2 is the extent of the influence of the economic comparator in the determination of the overall ranking; Alternative Processes 3 and 4 require 65 percent fewer packages for the same process throughput, and both the decreased cost of the packages and the decreased cost of the handling and emplacement operations offset the impact of the increased complexity of these two processes.*

Assessment of the long-term performance in the repository of the alternative waste form/stabilizer combinations showed significant difference among the candidate processes only in respect to the ability of the stabilizer to resist lithostatic pressure and to the influence of cladding degradation. The alternatives with the solid stabilizer were clearly more able to resist lithostatic pressure. With the particulate stabilizer the close-packed pins

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*The advantage of reduction in number of packages could be attained by emplacing more assemblies in each canister. Appendix E presents the results of an analysis of package costs and package material requirements using the Reference Process and an increased diameter canister which would hold 3 PWR assemblies or 8 BWR assemblies.

TABLE 6-1

TRADE-OFFS: ALTERNATIVE PROCESSES VERSUS REFERENCE PROCESS

Principal Operational Steps	Positive Factors	Negative Factors
ALTERNATIVE 1		
Remove End Fittings Package End Fittings	Shortens Canisters Reduces Weight to Handle	Creates Additional Waste Presents Small Additional Risk Requires Additional Facilities
ALTERNATIVE 2		
Remove End Fittings Vent and Reseal Pins Package End Fittings	Shortens Canisters Reduces Weight to Handle Relieves Internal Pressure Increases Temperature Margin	Creates Additional Waste Presents Small Additional Risk Requires Equipment Development and Demonstration Requires Additional Facilities
ALTERNATIVE 3		
Remove End Fittings Pull Fuel Pins Compact and Package Hardware	Shortens Canisters Increases Density of Packing Increases Flexibility in Canistering Greatly Reduces Canister Requirements Increases Repository Loading Options Reduces Handling of Material	Creates Additional Waste Requires Significant Additional Facilities Slightly Increases Safety Risk Requires Considerable Equipment Design Development Requires Equipment Demonstration
ALTERNATIVE 4		
Remove End Fittings Pull Fuel Pins Shear Fuel Pins Immobilize in Glass-Ceramic	Increases Flexibility in Canistering Greatly Reduces Canister Requirements Reduces Handling of Material Removes Some Fission Gases Provides Matrix of Low Solubility Provides Means of Sequestering Fuel Fines Glass-Ceramic Matrix Has Good Thermal/Mechanical Properties	Destroys Barrier Provided by Fuel Cladding Requires Double Canistering Substantially Increases Safety Risk Creates a Troublesome Waste Significantly Increases Facility Requirements Presents Difficult Maintenance Problems Requires Large Scale Development Program Involves High Temperature Operation Quality Control Difficult Requires Large Amount of Additional Facilities, Both Radioactive and Non-Radioactive

TABLE 6-2

SUMMARY OF RANKING OF ALTERNATIVE PROCESSES

Assessment Area	Reference Process	End Fitting Removal	Fission Gas Venting	Disassembly and Pin Storage	Shearing And Immobilization
Technical	1	2	3	3	4
Operating	2	3	4	1	5
Risk	1	1	3	2	4
Economic	4	3	3	1	2
Figure Of Merit	4	3	5	1	2

are more effective against lithostatic pressure than the intact assemblies, which have considerably more void area for stabilizer fill and compaction. The assessment for impacts of cladding degradation resulted in downgrading the sheared/immobilized waste form significantly. Ranking of the alternatives in respect to the in-repository criteria is set forth in Table 6-3.

It should be noted that the overall ratings in the performance study favored the solid stabilizer waste forms over the particulate by a substantial margin, which was almost entirely due to the influence of the rating on resistance to lithostatic pressure. In view of the lack of any criteria for acceptable distortion of the canister, this cannot be used as a basis for rejecting the particulate stabilizer. It is of course obvious that a solid stabilizer would be preferable for resisting canister distortion; as noted elsewhere in this report, a low-melting metallic alloy could be substituted for the particulate, by making relatively minor changes in process equipment, without altering the process evaluation presented in Section 4.0.

TABLE 6-3

WASTE FORM RANKING AT <425C CLAD TEMPERATURE

(Extracted from Table 30 of Reference 1)

Functional Criteria	Reference Process and Alternative 1	Alternative 2	Alternative 3	Alternative 4
Support Against Lithostatic Pressure	2	2	2	1
Breach of Canister Due to Pressurization	1	1	2	2
Stability of Heat Transfer	2	2	4	3
Minimizing Cladding Degradation	1	1	1	4
Long Term Stability for Radionuclide Retention	1	1	1	2
Independent Barrier to Migration	1	1	1	2
Criticality	1	1	1	1
[1 (best) → 4 (worst)]				

(a) Reference to "Stability of Heat Transfer" should probably read "Stabilizer Heat Transfer"

(b) It appears that these rankings were derived from the ΔT's reported in Section 3.3.6. As was pointed out in Section 3.3.6, using the thermal conductivity of sand and giving no credit to the compacted fuel rods which occupy approximately 70 percent of the cross section of the canister results in a higher calculated temperature differential, and thus a lower ranking on this attribute than is justified.

6.5 RECOMMENDATION

As a result of the assessments made during the course of these studies and comparative analyses of the relative merits of each disassembly alternative, as summarized in Sections 3.0, 4.0, and 5.0, it was concluded that Alternative 3, involving the disassembly of spent fuel and packaging and disposal of the resulting fuel pins, is the preferred method of disposing of spent fuel. Alternative 3 does not rate as the first preference in all areas of evaluation. It is concluded that, on balance, the disassembly and close-packing of fuel pins is the optimum approach; a principal reason for this is the reduction of nearly 65 percent (relative to the Reference Process) in the number of packages which must be handled and emplaced which is made possible by this process.

The recommended process entails a more complicated packaging procedure than the Reference Process or Alternative 1, and is about equal (overall) in process complexity with Alternative 2. The economic evaluation places it as the most attractive in overall cost, followed by Alternative 4; the latter, even though superior in ability to resist lithostatic pressure, is ruled out due to its increased complexity and the major uncertainties as to the operational feasibility of several of the processes involved.

From the standpoint of the suitability of the waste form, Alternative 3 is about even with the Reference Process and Alternatives 1 and 2, although Alternative 2 might receive a slightly higher rating than Alternative 3 on the basis that relief of the internal pressure in the fuel pins delays from 4000 to 8000 years the time after canister breach in the repository at which release of radioactivity commences. It would be a relatively simple matter, however, to combine the benefits of both alternatives by venting and resealing the pins either before pulling as in Alternative 2 or after pulling. In comparison with Alternative 4, Alternative 3 may be equal or better, depending on how one assesses the value of the Zircaloy fuel cladding barrier versus the double encapsulation in Alternative 4 in a glass matrix and a stainless steel container. As has been observed, this was not taken into account in the in-repository performance study.

It is clear that economic advantage of Alternatives 3 and 4 over the Reference Process and Alternatives 1 and 2 is dominated by the effects of reducing the number of packages to be emplaced. The total costs for the five processes under consideration were all dominated by the package costs, and are quite expensive because of the complexity of the package design which was employed, and by the high cost of the materials required. Notwithstanding, the relative advantage of Alternative 3 over the Reference Process and Alternatives 1 and 2 would be preserved regardless of the package costs; thus, if even the simplest of the packages which have been previously considered for disposal of spent fuel (or high level solidified waste) were used as the packaging basis for this study, Alternative 3 will show an economic advantage over the Reference Process and Alternatives 1 and 2. The unit costs of the Reference Process and Alternative 3 were calculated at 75, 50, and 25 percent of the Reference Waste Package cost. The results are plotted in Figure 6-1, and clearly illustrate that the magnitude of the difference is sensitive to package costs, but that the sign of the difference is unaffected by package costs down to essentially zero package cost. The Alternative 3 process is also attractive as a means of tailoring the thermal loading per package to the specific requirements of a repository; as the per package loading decreases from three PWR or eight BWR assemblies per canister, the economic advantage of Alternative 3 will diminish, but will not disappear.

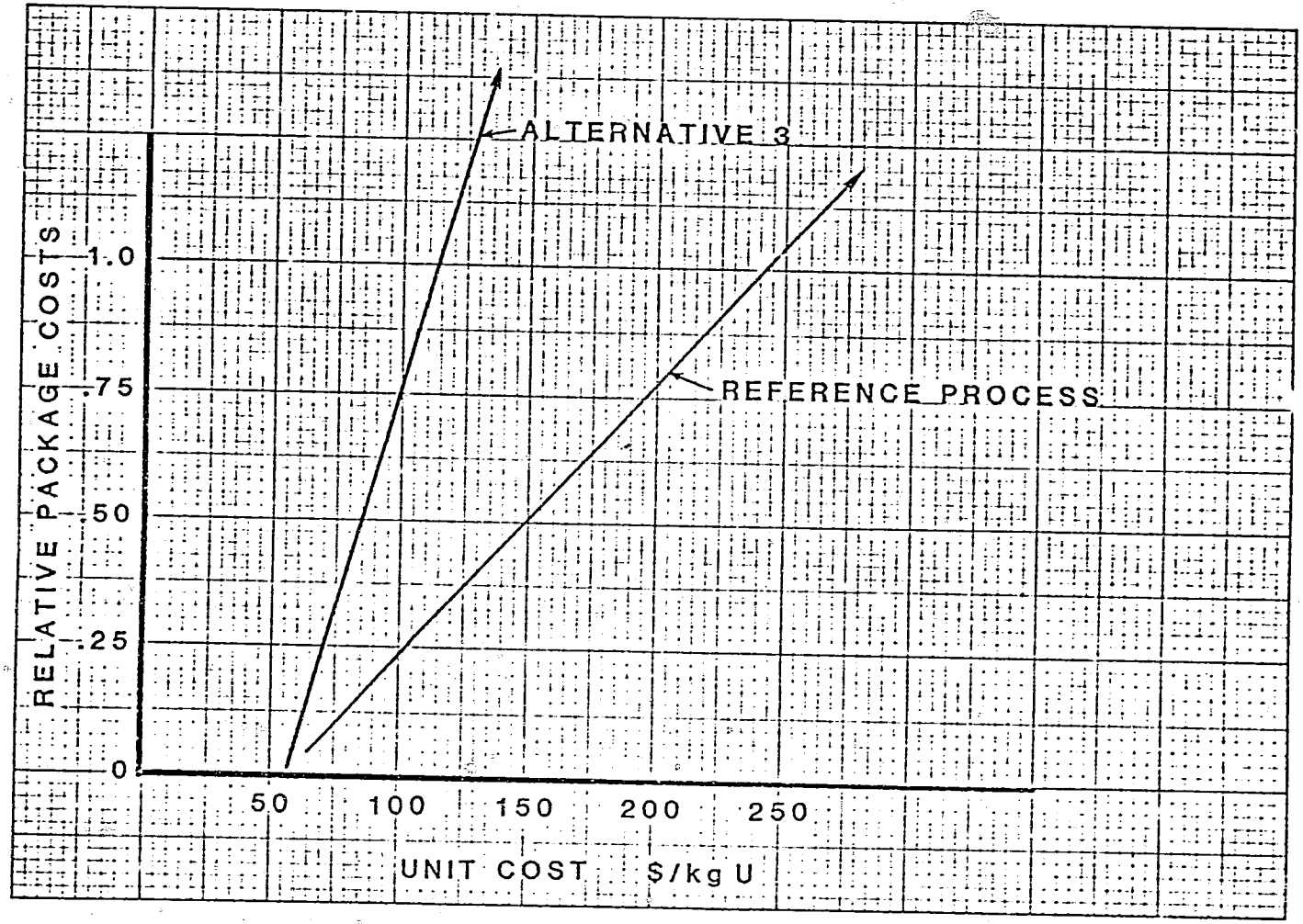

FIGURE 6-1

UNIT COST OF DISPOSAL AS A FUNCTION OF RELATIVE PACKAGE COSTS

APPENDIX A

THE INFLUENCES OF TEMPERATURE AND RADIATION ON THE GENERAL CORROSION OF TITANIUM AND NICKEL ALLOYS IN BRINE

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January 1981

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INTRODUCTION

A 1000-year containment life has been specified for nuclear waste packages designed for geologic isolation (A1). Consequently, waste packages must be constructed from materials that will resist failure under all possible thermal, radiation, pressure and water chemistry conditions which may develop over this extended period of time.

Corrosion resistance in aqueous saline solutions is a primary property considered when selecting waste package materials for salt isolation; available information indicates that titanium and nickel alloys are the best metallic candidates (A2). A review and analysis of the available saline corrosion data is presented in this appendix to estimate how temperature and radiation will affect the general corrosion* of these alloys. It should be possible to combine the analytical results with predictions of thermal conditions developed during salt isolation to estimate minimum general corrosion allowances for waste package barriers fabricated from titanium and nickel alloys.

ASSUMPTIONS

1) The dependency of corrosion rate (R) on absolute temperature (T) is given by the Arrhenius cquation, i.e.,

   $R = ae^{-b/T}$ where a and b are constant

2) Tha Arrhenius relaionship governing the temperature dependency of nickel alloy corrosion in brine and sea water exhibit similar values of the b constant (plots of ℓ nR vs 1/T exhibit the same slope for corrosion in brine and seawater)

3) The reported differences in the brine and seawater corrosion behavior of nickel alloys** are not statistically significant.

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*Uniform surface attack

**The nickel alloys referred to are those in which the base metal is alloyed principally with Cr and frequently also with additions of Fe, Mo, W, Co, Nb, Ti or Al.

4) Radiation influences corrosion behavior in brine in a manner similar to oxygenation.

DISCUSSION

Corrosion Data

A summary of the available corrosion data for titanium and nickel alloys in brine and seawater is given in Table A-1 (A3, A4, A5, and A6). The saturated solution of salts commonly referred to as brine is the corrosive fluid which can form in a salt repository. Seawater is a corrosive fluid obviously related to brine in that it contains similar dissolved salt constituents but is less concentrated by about a factor of ten (A3). The general brine and seawater corrosion data summarized in Table A-1 include information on commercially pure titanium (C.P.Ti), two titanium alloys (TiCode-12 and Ti-Pd), and four nickel alloys (In600, In625, In825 and Hastelloy C-276). These materials were tested at temperatures ranging from about room temperature to 250C under anoxic and oxic conditions. Also included in Table A-1 is a material referred to as a "Typical Nickel Alloy" which represents the corrosion behavior of the four tested compositions with the data averaged where possible.

Titanium Corrosion

One observation to note with regard to titanium is the difference in corrosion behavior of the chemically pure material compared to the alloys. The chemically pure material exhibited a pronounced increase in corrosion rate and transition from general to crevice attach at 250C in brine when the solution chemistry was changed from the anoxic to the oxic condition. Under the identical conditions, general corrosion was maintained for the titanium alloys with the corrosion rates being substantially lower in the oxic case. These observations were jenerally confirmed by the behavior exhibited in seawater at 250C where change in fluid conditions from anoxic to oxic caused an increase in the corrosion rate of commercially pure titanium but decreased that of the TiCode-12 alloy.

TABLE A-1

THE CORROSION BEHAVIOR OF TITANIUM AND NICKEL ALLOYS, REFERENCES A3,A4,A5, AND A6

MATERIAL	TEST CONDITIONS AND CORROSION RATES $\frac{in}{y}$ ($\frac{mm}{y}$)[h]							
	250°C Anoxic[a] Brine	150°C Anoxic[a] Brine	70°C Anoxic[a] Brine	250°C Oxic Brine	250°C Anoxic[a] Seawater	250°C Oxic Seawater	90°C Anoxic Seawater	25°C Anoxic Seawater
C.P. Ti	.00055 (.014)	.0001 (.0026)	.0000024 (.00006)	.13[b,c] (3.2)	.00046 (.0117)	.0[illegible] (.[illegible])	.000003 (.0001)	.0000005 (.000013)
TiCode-12	.00013 (.0032)	.000035 (.0009)	.0000028 (.00007)	.000071[c] .000016[d]	.000043 (.0110)	.000024[e] (.00060)		
Ti-Pd	.000095 (.0024)	.000012 (.0003)	.0000036 (.00009)	.000016[c] (.0004)	.000045 (.00114)	.000024 (.00062)		
In600	.00035 (.009)				.0002 (.005)	.0039[d] (.1)		
In625	.00020 (.005)				.00047[f] (.012)			
InS25	.00024 (.006)				.00016 (.004)			
Hast. C-276	.00028 (.007)			.0024[d,f] (.06)	.000059 (.0015)	.0079[d,f] (.2)	.0000025 (.000065)	
Typ. Ni Alloy	.00027[g] (.0068)			.0024 (.06)	.00022[g] (.0057)	.0059[g] (.15)	.0000025 (.000065)	

[a] 0.03 ppm O_2 [b] crevice attack [c] 450 ppm O_2 [d] 500 ppm O_2 [e] 500 ppm, O_2

[f] material pitted [g] average [h] original data reported in $\frac{mm}{y}$ to the number of significant figures given

The aforementioned titanium and titanium alloy corrosion results have three implications in regard to the selection of these materials for salt applications and their anticipated corrosion behavior.

1. The susceptibility toward crevice corrosion exhibited by the commercially pure titanium implies that use of this material would have to be justified by careful analytical work and additional corrosion testing. This is necessary to quantitatively determine how all solution conditions (T, P, pH, $[O_2]$, etc.) interact to govern whether general or crevice attack occurs, and establish that operating conditions favoring crevice attack cannot develop during the design life of the waste package. Consequently, in the absence of this data, the selection of one of the two titanium alloys represents the more conservative choice of material for salt repository waste packages.

2. The decreased rate of general corrosion caused by increasing the oxygen level of the solution (anoxic to oxic change) implies that a similar effect might result from radiation exposure, since it is generally conceded that the oxidation potential of any brine in the vicinity of a geologically isolated nuclear waste package will be increased by its reaction with radiation (A3, A5, and A6). On the other hand, it would be more conservative to simply conclude that the general corrosion resistance of titanium alloys is not expected to be significantly changed in the presence of radiation as opposed to being improved. This approach is substantiated to a degree by result of Braithwaite et al (A3), who reported that the general corrosion rate of TiCode-12 under a radiation exposure of 10^7 rads (T)/h at 90C remained unchanged in seawater, and was only doubled* in brine by comparison to the levels measured without presence of radiation.

3. General corrosion allowance calculations based on the behavior under anoxic conditions should be conservative in regard to the use of the titanium alloys for salt repository waste packages. (The general corrosion rates measured for the titanium alloys were higher under anoxic compared to oxic conditions, and radiation effects favor development of oxic conditions).

*Owing to the extremely low corrosion rates measured for this titanium alloy at 90C ($\sim 10^{-5}$ in./yr), a factor of two increase may be well within the accuracy of experimental measurements.

The corrosion data reported for TiCode-12 and Ti-Pd tested in anoxic brine are plotted in Figure A-1, assuming that the corrosion rates follow an Arrhenius equation temperature dependency, i.e., $R=ae^{-b/T}$**. An accurate description of the influence that temperature has on the rate of a chemical reaction is typically given by this expression over small temperature ranges when reaction mechanisms remain unchanged (A7).

The equations given in Figure A-1 to define the dependency of corrosion rates on temperature are least square curve fits. They could be used to calculate general corrosion allowances for TiCode-12 or Ti-Pd alloy waste package barriers, provided that the dependency of barrier temperature on residence time in the repository has been analytically determined.

Temperature appears to have approximately the same influence on the general corrosion behavior of both titanium alloys, but the rate of attack is slightly higher for the TiCode-12 composition. Regardless, the TiCode-12 should be less expensive than Ti-Pd (A5); therefore, it is the best currently available titanium alloy for salt repository waste package applications.

Nickel Alloy Corrosion

Most of the corrosion rate data available for nickel alloys represents anoxic tests at 250C. Depending upon the particular alloy, these rates range from ~ 0.0002 to 0.0004 in./yr in brine, and ~ 0.0001 to 0.0005 in./yr in seawater. However, these data typically represent one or two short duration tests on each material, consequently, differences in reported corrosion rates ae probably not significant. This conclusion was applied to the "Typical Nickel Alloy" summary given in Table A-1 where the 250C anoxic brine and seawater corrosion rates are averages of all the reported data.

**R = General Corrosion Rate
a and b = Constants
T = Absolute Temperature

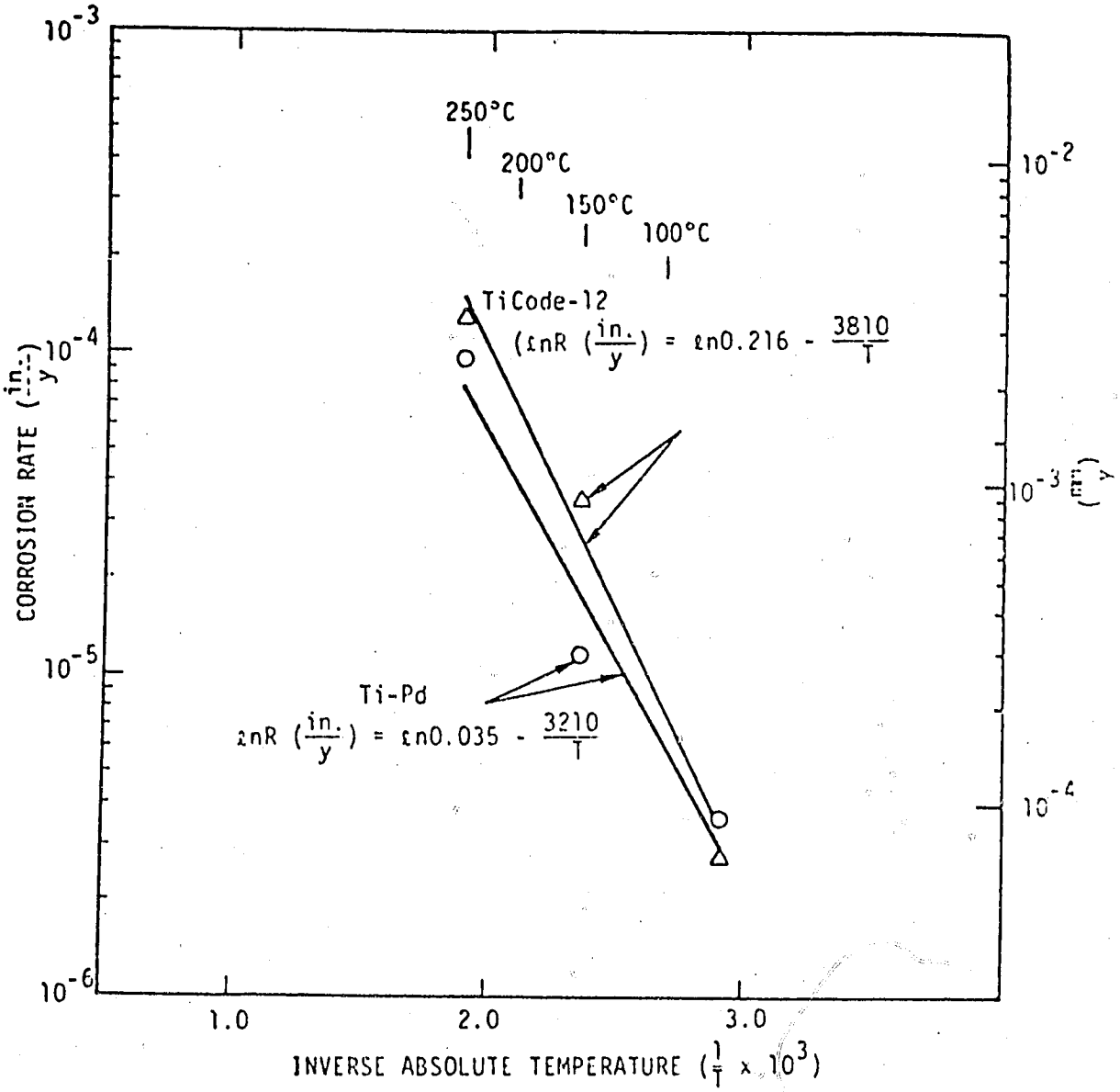

FIGURE A-1

TEMPERATURE DEPENDENCY OF TICODE-12 AND

Ti-Pd CORROSION IN ANOXIC BRINE

Only three tests have been run on nickel alloys under oxic conditions, but the results consistently reveal a corrosion behavior distinctly different from that previously described for the titanium alloys. This difference is graphically illustrated in Figure A-2 where the corrosion rate of nickel and titanium alloys are plotted as a function of the oxygen content of the test solutions. The nickel alloys, In600 and Hastelloy C-276, displayed roughly a factor of ten increase of corrosion rate at 250C when test conditions were changed from anoxic to oxic which is opposite to the observed titanium alloy behavior. The Hastelloy test material was also pitted by oxic brine and seawater which proved to be a particularly surprising result since the test autoclaves were fabricated from this alloy and none underwent similar attack (A2). This result illustrates how subtle conditions are frequently involved in determining whether or not localized attack occurs, and serves to emphasize that corrosion behavior is best established by careful statistical experiments.

The implication of the observed increase in nickel alloy corrosion behavior caused by an anoxic to oxic change of solution condition is that radiation (increased oxidation potential) will have a similar effect. This negative influence of radiation could be minimized in waste package design by placing any nickel alloy barrier furthest from the waste form or shielding it.

To be conservative in determining waste package corrosion allowances for nickel alloys, additional data are required on their behavior in oxic brine. Further testing is necessary to statistically determine whether the many candidate nickel alloys differ significantly in corrosion behavior, which compositions maintain general corrosion behavior as opposed to undergoing localized attack such as crevice, pitting or stress corrosion, and also how corrosion rates are influenced by temperature.

The corrosion behavior of nickel alloys under worst-case repository conditions, approximated by oxic brine conditions, and how corrosion rates

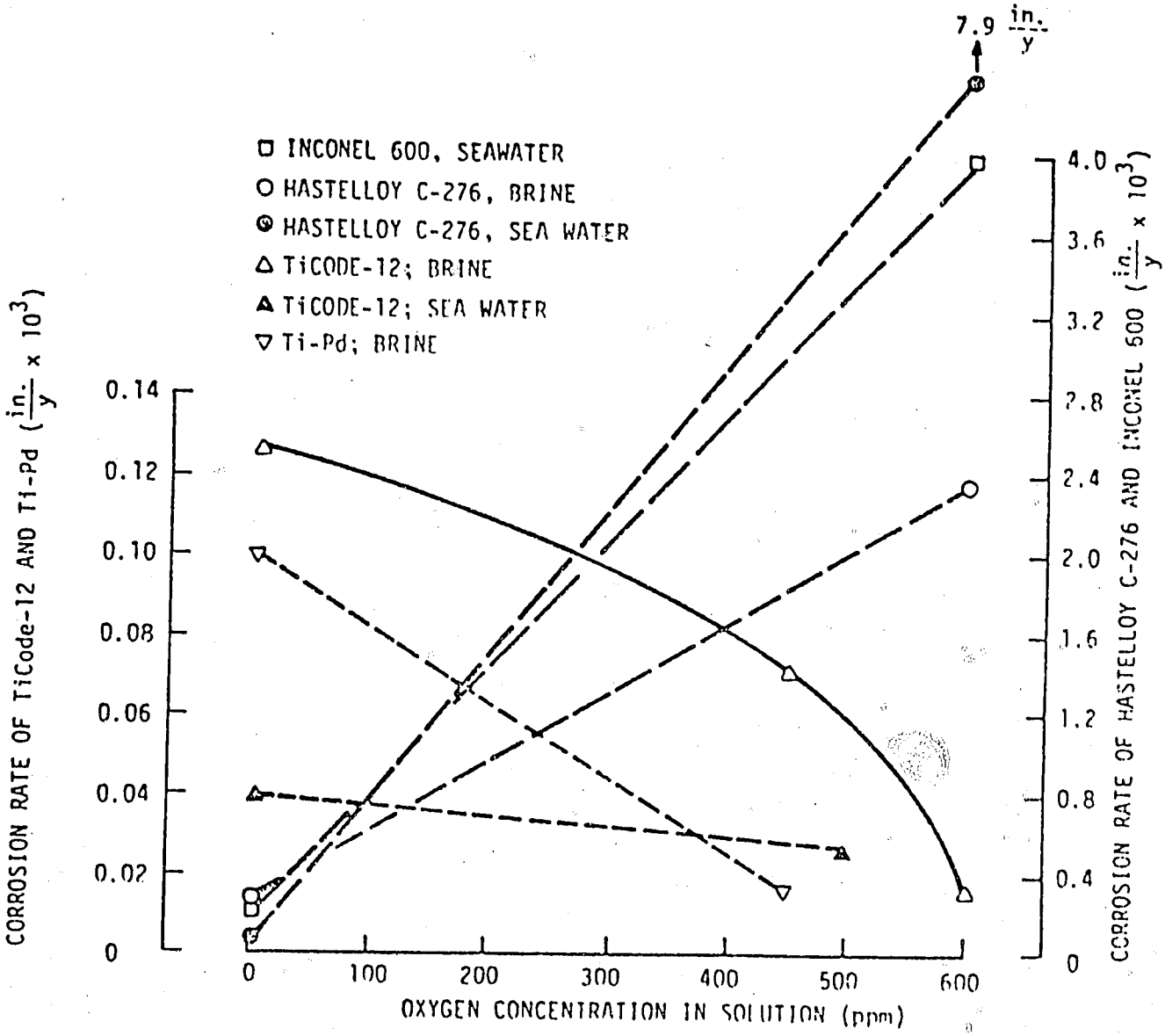

FIGURE A-2

THE INFLUENCE OF OXYGEN CONTENT ON THE CORROSION OF NICKEL AND TITANIUM ALLOYS AT 250° C IN BRINE & SEAWATER

are influenced by temperature, can only be roughly estimated from the data currently available. Such an estimate is presented in Figure A-3 using information summarized in Table A-1 for a "Typical Nickel Alloy." The procedure used to estimate the temperature dependency of brine corrosion simply involved assuming that it would be similar to that observed for corrosion in seawater; i.e., the n vs 1/T slopes for brine and seawater corrosion were assumed to be equal. The corrosion behavior observed for commercially pure titanium in brine and seawater (Figure A-4) provided some justification for making this assumption.

CONCLUSIONS

1) The presence of radiation should have little influence on the brine corrosion behavior of the principal titanium alloy waste package barrier candidate, TiCode-12.

2) A reasonable estimate of the general corrosion allowance for TiCode-12 used in a waste package designed for isolation in salt can be obtained by reference to behavior in anoxic brine. The temperature dependency of TiCode-12 corrosion in anoxic brine in the 343 to 523K range is given by:

$$\ell n R = \ell n\ 0.216 - \frac{3810^*}{T}$$

3) The superiority of any one nickel alloy for waste package barrier applications involving isolation in salt cannot be clearly established on the basis of the available brine corrosion information.

4) Radiation will increase the rate of nickel alloy corrosion and may cause localized attack in some cases.

5) A reasonable estimate of the general corrosion allowance for a nickel alloy used in a waste package designed for isolation in salt can be obtained by reference to behavior in oxic brine. A rough estimate of the temperature dependency of nickel alloy corrosion in oxic brine in the 363 to 523K range is given by:

$$\ell n R \sim \ell n 60 - \frac{5300}{T}$$

*R = General Corrosion Rate (in./yr)
T = Absolute Temperature (Kelvin)

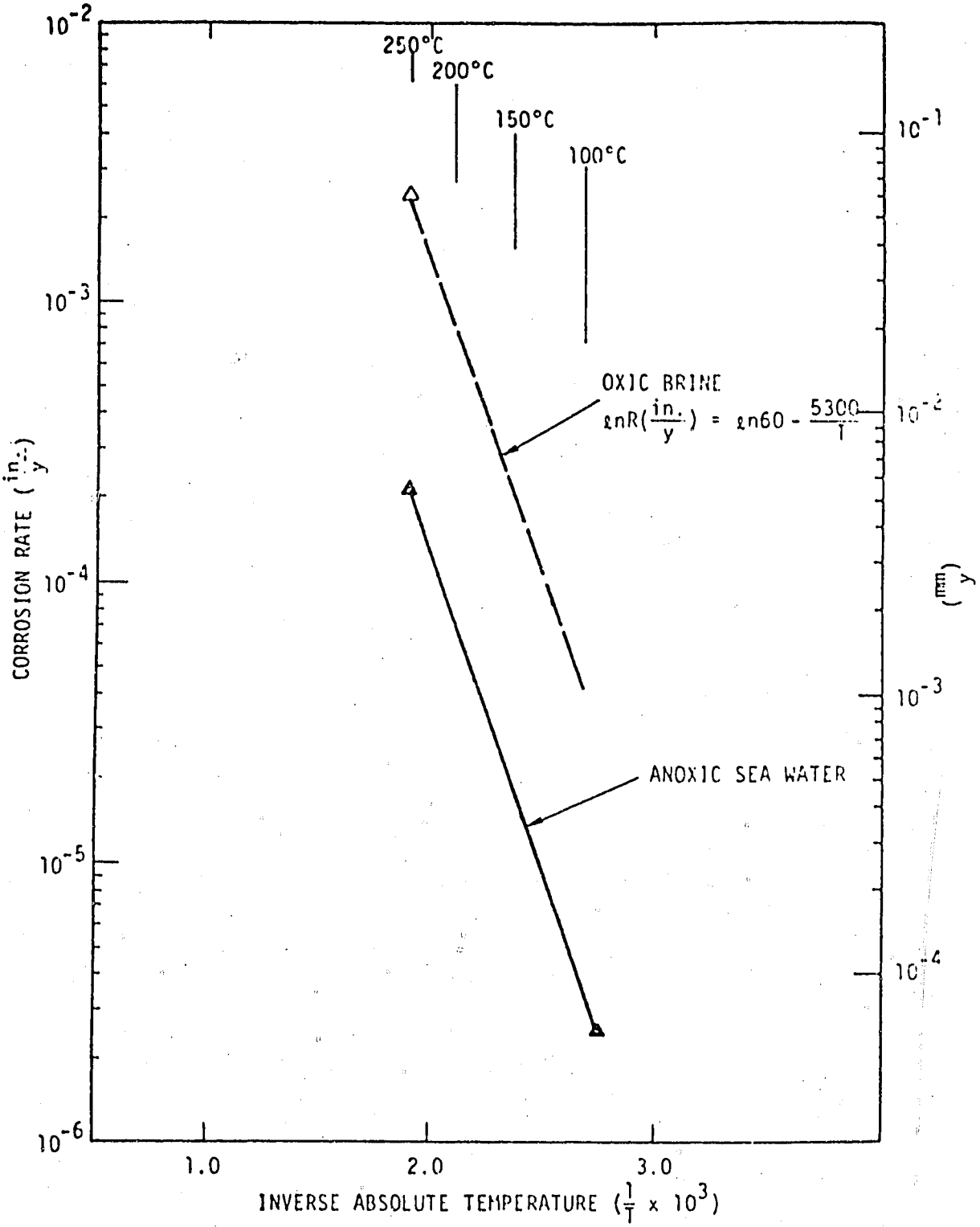

FIGURE A-3

THE ESTIMATED TEMPERATURE DEPENDENCY OF NICKEL ALLOY CORROSION IN OXIC BRINE AND ANOXIC SEAWATER

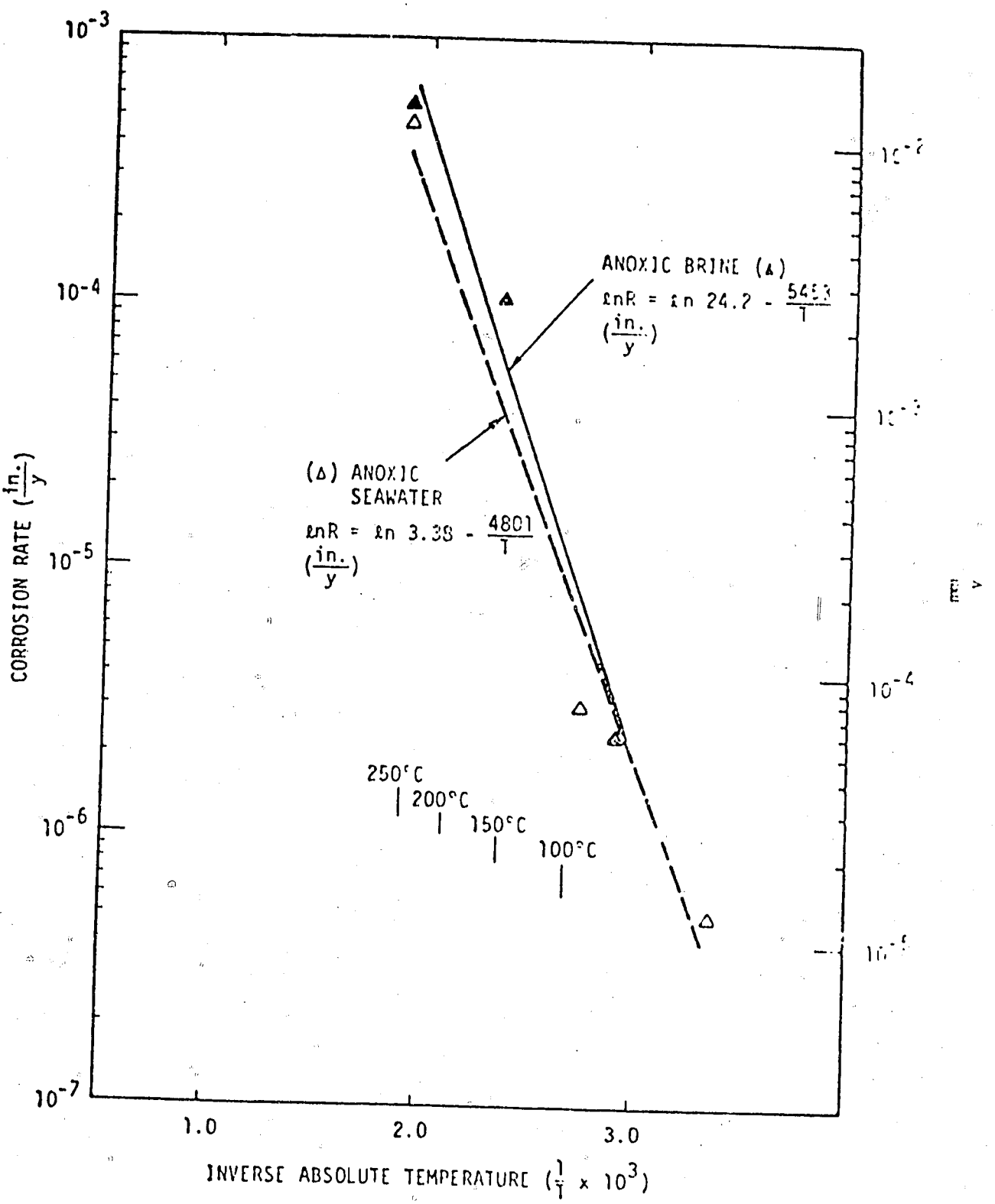

FIGURE A-4

TEMPERATURE DEPENDENCY OF TITANIUM CORROSION IN ANOXIC BRINE AND SEAWATER

REFERENCES

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APPENDIX B

ANALYSIS OF SPENT FUEL PACKAGE PARAMETERS AND COST FOR THE VARIOUS DISASSEMBLY ALTERNATIVES CONSIDERED IN THIS STUDY

In view of the fact that the cost of the titanium canisters and Inconel liners were expected to represent a major cost consideration in the comparison of the spent fuel disassembly alternatives considered in this study, a careful analysis of the package size was made for each alternative. The smallest possible packages which could be utilized to adequately contain the spent fuel and stabilizer were established, thereby minimizing the amount and cost of titanium and Inconel required. A preliminary analysis of the cost of each type of material composing the package indicated that titanium and Inconel were by far the most expensive constituents of the package, and therefore the titanium canister and Inconel liner were the principal items of consideration. A summary of the spent fuel packaging requirements for each disassembly alternative which resulted from this analysis is set forth in Table B-1.

Producers and fabricators of these materials were contacted to obtain cost estimates of plate, pipe, and the fabricated canisters and liners. The volumes of titanium and Inconel required by the packages were calculated. Because of the large size of the package and the number of packages required, it was necessary to analyze the total amount of raw materials consumed for the repository and examine the impact of such consumption on the available natural resources.

The prices obtained from producers and fabricators of titanium and Inconel are described by the following sections and are presented in summary form in Table B-2.

TABLE B-1

SPENT FUEL PACKAGE PARAMETERS

Canister Description	Reference Unmodified Spent Fuel	Alt. 1 End Fitting Removal	Alt. 2 Fission Gas Venting	Alt. 3 Disassembly & Pin Storage	Alt. 4 Shearing and Immobilization
Outside Diameter (in.)	14.0	14.0	14.0	14.5	19.5
Wall Thickness (in.)	0.25	0.25	0.25	0.25	0.25
Length (in.)					
PWR	171.8	164.7	164.7	163.5	159.8
BWR	188.0	173.5	173.5	172.3	177.8
Assemblies Contained					
PWR	1	1	1	3	3
BWR	2	2	2	8	8
Heat Load/Canister (kW)(1)					
PWR	0.53	0.53	0.53	1.59	1.59
BWR	0.38	0.38	0.38	1.52	1.52
Number Canisters/yr.					
PWR	6,300	6,300	6,300	2,100	2,100
BWR	4,450	4,450	4,450	1,113	1,113
Liner Description					
Liner Outside Diameter (in.)	20.0	20.0	20.0	20.5	25.5
Wall Thickness (in.)	1	1	1	1	1
Liner Length (in.)					
PWR	232.4	225.3	225.3	224.1	220.4
BWR	248.6	234.1	234.1	232.9	238.4
Excavation Required ($ft.^3$)					
PWR	233.8	227.5	227.5	231.4	280.8
BWR	248.0	235.3	235.3	239.4	300.9
Excavation Reduction from Reference Process Package (ft^3)					
PWR	-	6.3	6.3	2.4	-47.0
BWR	-	12.7	12.7	8.6	-52.9

(1) Based on 0.53 kW/assembly PWR and 0.19 kW/assembly BWR.

TABLE B-2

SUMMARY OF COST ESTIMATES FOR TITANIUM AND INCONEL

Source	Alloy	Thickness(in)	Material Costs(Plate) $/pound	Material Costs(Plate) $/ft³	Fabrication Costs $/pound	Fabrication Costs $/ft³	Combined Material and Fabrication Costs $/pound	Combined Material and Fabrication Costs $/ft³
Titanium								
Astro Metallurgical[a]	Ti-50A	1/4	-	-	-	-	18.60	5,230
Titanium Industries[b]	TiCode-2	1/4	-	-	-	-	39.10	11,000
TIMET	TiCode-12	1/4	12.22	3,434	-	-	-	-
Futura Titanium[c]	TiCode-12	1/4	19.60	5,500	20.50	5,750	40.10	11,250
Inconel								
Huntington Alloys	Inconel 600	1	6.25	3,310	-	-	-	-
Williams & Co.	Inconel 600	1	6.52	3,460	-	-	-	-
Midland Steel[d]	Inconel 600	1	-	-	1.73	919	-	-
Youngstown Welding[e]	Inconel 600	7/8	-	-	-	-	8.02	4,250
J. M. Tull[f]	Inconel 600	1	-	-	-	-	9.40	5,000

(a) For 14" OD pipe.

(b) Based on production of 24" OD Canister, with 2 domed heads and a nozzle, 9' in length. Quote not available for bulk quantity.

(c) Includes welding on lower end cap with top cap left loose. Based on average costs of several canister sizes.

(d) Based on experience fabricating 1" thick stainless steel.

(e) Based on 20" OD, 8' long sections of pipe.

(f) Based on average unit costs of various sizes of pipe (20" OD X 170" long, 23" OD X 160" long, 25" OD X 60" long, 29" OD X 60" long). Unit cost was not strongly dependent on OD or length.

(1) Titanium

Although TiCode-12 was the titanium alloy under consideration, quotes for TI-50A and TiCode-2 (a commercial grade TiCode) were included for purposes of comparison.

(a) Astro Metallurgical estimated the cost per foot of 14-inch OD pipe of Ti-50A with a wall thickness of 0.25 inches to be $392.56/ft. A one foot length of this pipe was calculated to contain 0.075 ft^3 of titanium. The resulting cost per unit volume was calculated to be $5,230/$ft^3$. To arrive at the cost per pound, the value of $5,230/$ft^3$ was divided by the density of titanium (281 lb/ft^3), resulting in a cost of $18.60/pound.

(b) Titanium Industries estimated the cost of a 9-foot long TiCode-2 canister with a wall thickness of 0.25 inches, which had been produced previously at $13,000. This cost was divided by the estimated volume of the titanium contained in the canister (1.182 ft^3) to arrive at a unit cost of $11,000/$ft^3$. The cost per pound was calculated by dividing the cost per cubic foot by the density of titanium, resulting in a cost of $39.10/pound.

(c) TIMET supplied the cost of 0.25-inch TiCode-12 plate as $12.22/pound. Multiplying by the density of titanium results in a cost of $3,434/$ft^3$.

(d) Futura Titanium provided the following formulas for use in calculating the costs for material and fabrication for large quantities of various sizes of TiCode-12 canisters:

(i) For diameters greater than 12-inches and lengths between 120 and 180-inches:

Material cost in $ = $2d\ (1+1.25\ell)$

Where d = diameter in inches

ℓ = length in inches, including end caps

Labor Cost in $ = $25d + 34\ell + 300$

(ii) For diameters greater than 12-inches and lengths greater than 180-inches, the material cost formula is the same as in (i) and labor is estimated as follows:

Labor Cost in $ = $25d + 34\ell + 400$

(d) The cost per cubic foot was calculated by averaging the cost per cubic foot of several different canister sizes. The costs per cubic foot did not show a strong dependence on canister length or OD. The average canister cost per cubic foot was calculated to be $\$5,750/ft^3$ with a standard deviation of only $\$400/ft^3$. It is therefore felt that the average value indicated in Table B-2 is an accurate representation of the unit cost of a typical canister. The cost per pound was then calculated by dividing the cost per cubic foot by the density of titanium.

(2) Inconel

The alloy under consideration for the package liner is Inconel 600. The suppliers of Inconel who were contacted felt they could meet the projected annual demand for this project only with sufficient advance planning. Many fabricators were contacted but most were unable to fabricate large OD or 1 inch thick pipe wall. It appears likely that the present fabrication capacity of Inconel pipe would have to be expanded to meet the annual needs of the repository.

(a) Huntington Alloys quoted 1-inch Inconel 600 plate at $6.25/pound. Multiplying by the density of Inconel (530 pounds/ft^3) results in a cost of $3,310/$ft^3$.

(b) Williams and Company quoted 1-inch Inconel 600 plate as $6.52/pound. Multiplying by the density of Inconel results in a cost of $3,460/$ft^3$.

(c) Because of the paucity of Inconel fabricators able to produce pipe of the required OD and wall thickness, a steel fabricator (Midland Steel) was contacted and provided an estimate of the cost of fabricating a typical Inconel liner based on their experience with stainless steel. Midland Steel estimates typical liner fabrication to cost $8,580. Dividing by the average liner volume of 9.34 ft^3 results in a cost of $919/$ft^3$. Dividing by the density of Inconel yields a cost per pound of $1.73/pound.

(d) Youngstown Welding stated that they were unable to fabricate pipe with a 1-inch wall thickness, but were able to quote the cost of 7/8-inch wall Inconel 600 pipe in eight foot lengths with a 20-inch OD at $1,550/ft. The volume of a one-foot length of such pipe was calculated to be 0.365 ft^3. Dividing the cost of one foot of pipe by the volume thereof results in a cost per cubic foot of $4,250/$ft^3$. Dividing the cost per cubic foot by the density yields a cost per pound of $8.02/pound.

(e) J. M. Tull Metals contacted an Inconel fabricator who was able to produce pipe of the required OD and thickness. SWEPCO provided J. M. Tull with fabrication costs, and the cost of Inconel 600 plate was provided by Huntington Alloys. J. M. Tull then quoted the cost of four different pipe sizes for purposes of comparison. The cost of each pipe was divided by the calculated volume of each pipe; the four resulting costs per cubic foot were then averaged to obtain a mean cost per cubic foot of approximately $5,000/$ft^3$, with

a standard deviation of $\$120/ft^3$. The lowest individual cost per cubic foot was $\$4,830/ft^3$; the highest was $\$5,120/ft^3$. Since the variation in cost per cubic foot of each pipe was very small, $\$5,000/ft^3$ was used as a typical cost for fabricated Inconel 600 pipe. The cost per pound was calculated by dividing the cost per cubic foot by the density of Inconel, resulting in a cost per pound of $9.40/pound.

Based on comparison of the quotes provided by the previously discussed sources and the relevance of each quote to the specific spent fuel package application, cost estimates for fabricated titanium and Inconel were established for the purposes of this analysis.

The cost estimate provided by Futura Titanium for TiCode-12 canisters was chosen to be the most appropriate. Costs for material and labor for each canister size were calculated using the equations provided by them to give the total cost per canister. This result was increased by 20% to account for the quality assurance requirements.

The cost estimate provided by J. M. Tull Metals for Inconel 600 pipe was chosen as a basis for estimating the cost of the liner. The cost per cubic foot of $\$5,000/ft^3$ was increased by $\$400/ft^3$ to provide for the extra labor necessary to form an ellipsoid end cap for the liner. The resulting total of $\$5,400/ft^3$ was then increased by 20% for the quality assurance requirements to yield a final cost per cubic foot of $\$6,500/ft^3$.

The volume and weight of titanium and Inconel contained in the waste packages for the various spent fuel disassembly alternatives were determined, as were the total annual requirements for these materials (at capacity operation of the repository) and the total requirements for these materials over the lifetime of the repository operation. The unit costs for titanium and Inconel were applied to each of these requirements and the differences in costs for these materials between the various disassembly alternatives were calculated. The results of these calculations are set forth in Tables B-3, B-4 and B-5.

TABLE B-3

SPENT FUEL PACKAGE REQUIREMENTS (INDIVIDUAL)

		Reference Unmodified Spent Fuel	Alt. 1 End Fitting Removal	Alt. 2 Fission Gas Venting	Alt .3 Disassembly & Pin Storage	Alt. 4 Shearing and Immobilization
TiCode Canister						
Volume of TiCode-12: ($ft.^3$/Canister)	PWR	1.08	1.03	1.03	1.06	1.41
	BWR	1.18	1.09	1.09	1.12	1.56
Weight of TiCode-12 (lb.) (ρ=281 lb/$ft.^3$)	PWR	303	289	289	298	396
	BWR	332	306	306	315	438
Cost/Canister ($)	PWR	14,830	14,240	14,240	14,400	16,600
	BWR	16,250	14,970	14,970	15,140	18,390
Canister Cost/Assembly ($)	PWR	14,830	14,240	14,240	4,800	5,530
	BWR	8,130	7,490	7,490	1,890	2,300
Canister Cost Reduction From Reference Process Canister($/Canister)	PWR	-	590	590	430	- 1,770
	BWR	-	1,280	1,280	1,110	- 2,140
Canister Cost Reduction From Reference Process Canister($/Assembly)	PWR	-	590	590	10,030	9,300
	BWR	-	640	640	6,240	5,830
Inconel-Liner						
Volume of Inconel 600 ($ft.^3$/Canister)	PWR	9.11	8.87	8.87	9.04	11.14
	BWR	9.67	9.17	9.17	9.36	11.94
Weight of Inconel 600 (lb/Liner) (ρ=530 lb/$ft.^3$)	PWR	4,838	4,701	4,701	4,791	5,904
	BWR	5,125	4,860	4,860	4,961	6,328
Cost/Liner ($)	PWR	59,220	57,660	57,660	58,760	72,410
	BWR	62,860	59,610	59,610	60,840	77,610
Liner Cost/Assembly ($/Assembly)	PWR	59,220	57,660	57,660	19,590	24,140
	BWR	31,430	29,810	29,810	7,610	9,700
Liner Cost Reduction From Reference Process ($/Liner)	PWR	-	1,560	1,560	460	-13,190
	BWR	-	3,250	3,250	2,020	-14,750
Liner Cost Reduction From Reference Process ($/Assembly)	PWR	-	1,560	1,560	39,630	35,080
	BWR	-	1,620	1,620	23,820	21,730
TOTALS						
Canister and Liner Cost ($)	PWR	74,050	71,900	71,900	73,160	89,010
	BWR	79,110	74,580	74,580	75,980	96,000
Canister and Liner Cost/Assembly ($/Assembly)	PWR	74,050	71,900	71,900	24,390	29,670
	BWR	39,560	37,300	37,300	9,500	12,000
Cost Reduction per Canister and Liner from Reference Process ($/Package)	PWR	-	2,150	2,150	890	-14,960
	BWR	-	4,530	4,530	3,130	-16,890
Cost Reduction per Assembly from Reference Process ($/Assembly)	PWR	-	2,150	2,150	49,660	44,380
	BWR	-	2,260	2,260	30,060	27,560

TABLE B-4

SPENT FUEL PACKAGE REQUIREMENTS (ANNUAL)

Description	Reference Unmodified Spent Fuel	Alt. 1 End Fitting Removal	Alt. 2 Fission Gas Venting	Alt. 3 Disassembly & Pin Storage	Alt. 4 Shearing and Immobilization
TiCode-12 volume/yr. ($ft.^3$/yr.)	12,055	11,340	11,340	3,473	4,697
TiCode-12 weight/yr. (millions of lb/yr.)	3.39	3.19	3.19	0.98	1.32
Incogel 600 volume/yr. ($ft.^3$/yr.)	100,425	96,688	96,688	29,402	36,683
Inconel 600 weight/yr. (millions of lb/yr.)	53.2	51.2	51.2	15.6	19.4
TiCode and Inconel cost/yr. (millions of $/yr.)	819	785	785	238	294
Annual Cost Reduction From Reference Process (millions of $/yr.)	-	34	34	581	525

TABLE B-5

SPENT FUEL PACKAGE REQUIREMENTS (LIFE[1] OF REPOSITORY)

Description	Reference Unmodified Spent Fuel	Alt. 1 End Fitting Removal	Alt. 2 Fission Gas Venting	Alt. 3 Disassembly & Pin Storage	Alt. 4 Shearing and Immobilization
TiCode-12 Volume (thousands of ft.^{3})	204	192	192	58.6	79.3
TiCode-12 Weight (millions of lb.)	57.3	54.0	54.0	16.5	22.3
Inconel 600 Volume (thousands of ft.^{3})	1,702	1,638	1,638	496	619
Inconel 600 Weight (millions of lb.)	902	868	868	263	328
Total Excavation Required (millions of ft.^{3})	43.7	42.0	42.0	12.7	15.6
Excavation Reduction from Reference Process (millions of ft.^{3})	-	1.7	1.7	31.0	28.1
TiCode and Inconel Cost (billions of $)	13.87	13.29	13.29	4.02	4.96
Cost Reduction from Reference Process (billions of $)	-	0.58	0.58	9.85	8.91

(1) Based on 260,000 assemblies, 40% PWR (104,000) and 60% BWR (156,000)

APPENDIX B

ANALYSIS OF SPENT FUEL PACKAGE PARAMETERS AND COST FOR THE VARIOUS DISASSEMBLY ALTERNATIVES CONSIDERED IN THIS STUDY

APPENDIX C

METHODOLOGY FOR SYSTEMATIC COMPARISON

APPENDIX C
METHODOLOGY FOR SYSTEMATIC COMPARISON

Each alternative process has been evaluated with respect to the Reference Process on the basis of technical assessment, operating assessment, risk assessment, and economic assessment. In order to reach a clear relative evaluation, these assessment areas have been weighted with respect to each other. Further, where it was reasonable to do so, the assessment area was subdivided into sub-criteria and these sub-criteria were also weighted with respect to each other. Finally, the alternative processes were rated with respect to the Reference Process and a Figure of Merit for each process was determined by the sum of products of weighting factors and ratings.

It is the purpose of this appendix to define the subdivisions of the assessment areas and to present the weighting factors. The next section will present the process ratings and Figure of Merit determination.

C.1 ASSESSMENT CRITERIA

Table C-1 lists the four assessment areas and the subdivision of each into individual criteria. The choice of the criteria in each area is briefly discussed below.

C.1.1 Technical Assessment

Based on the process flowsheets and equipment layouts which have been presented for each alternative and upon a comparison of the known technology with that deemed to be necessary for reliable and safe operation, the technological desirability of each process was evaluated. The subjects of this evaluation included not only the state of knowledge of the process and equipment but also the effectiveness of safeguards during processing and in respect to the final waste form.

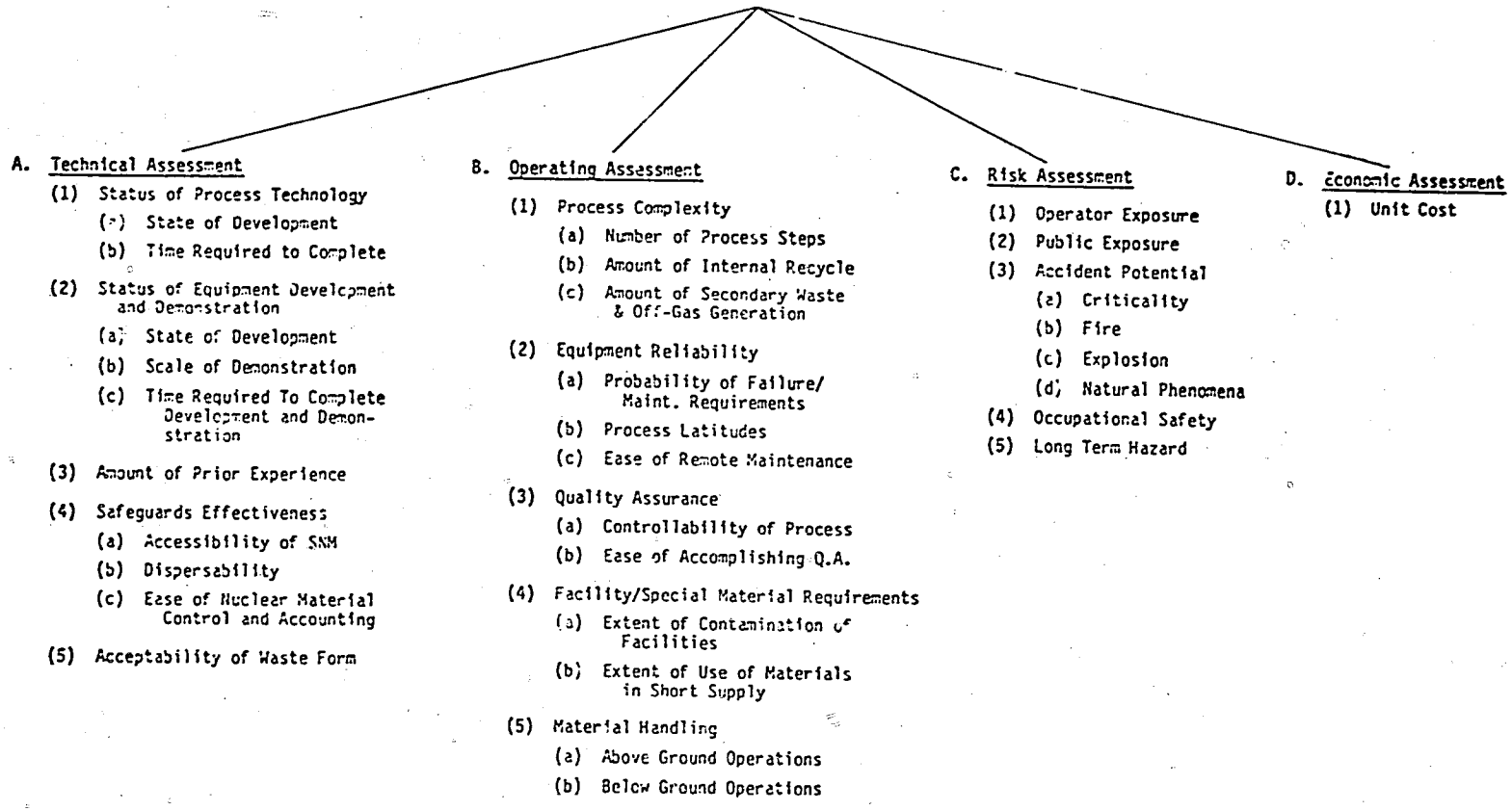
C.1.2 Operating Assessment

Difficulty of operation of each spent fuel packaging facility and of the repository for each alternative was determined based on process

TABLE C-1

CRITERIA FOR COMPARATIVE EVALUATION OF SPENT FUEL DISASSEMBLY ALTERNATIVES

FIGURE OF MERIT



complexity and equipment reliability. These attributes are particularly important in operations which must be conducted remotely. In addition to projected difficulty of operation, consideration was also given to the requirements for and ease of quality assurance. and to the need for special facilities or materials.

C.1.3 Risk Assessment

The risk assessment of each spent fuel disassembly alternative was evaluated primarily in connection with radiation exposure. Such exposure includes both occupational exposure under routine and accident conditions and non-occupational exposure due to possible off-site releases. Moreover, the possibility of radiation exposure of future populations as a result of naive intrusion into the repository was subjected to comparative evaluation.

C.1.4 Economic Assessment

The unit cost of each spent fuel disposal alternative was determined as a comparator of the alternatives.

C.2 WEIGHTING FACTORS

The hierarchy of weighting factors is illustrated in Tables C-2 through C-5. It will be noticed that the weighting factors for the four assessment areas sum to unity; that the weighting factors for criteria in any one assessment area sum to unity; and that, where sub-criteria are used, their weighting factors also sum to unity. The relative weights in each category are thus easily compared.

Obviously, the assignment of weighting factors reflects a particular point of view regarding the relative importance of the various criteria. Of the assessment areas, the economic assessment is weighted most heavily. This results from the recognition that the cost considerations are basic to any comparison of competing routes to a defined objective, and that higher costs will reflect higher overall levels of operational complexity or activity in each of the other assessment areas.

3 OF 3

TABLE C-2

WEIGHTING FACTORS FOR ASSESSMENT AREAS

Technical Assessment	0.25
Operating Assessment	0.20
Risk Assessment	0.15
Economic Assessment	0.40
	1.00

TABLE C-3

WEIGHTING FACTORS FOR TECHNICAL ASSESSMENT CRITERIA

1.	Status of Process Technology		0.20
	a) State of Development	0.50	
	b) Time Required to Complete Development	0.50	
2.	Status of Equipment Development and Demonstration		0.20
	a) State of Development	0.30	
	b) Scale of Demonstration	0.30	
	c) Time Required to Complete Development and Demonstration	0.40	
3.	Amount of Prior Experience		0.20
4.	Safeguards Effectiveness		0.20
	a) Accessibility of SNM	0.40	
	b) Dispersibility of SNM	0.30	
	c) Ease of Nuclear Material Control and Accounting	0.30	
5.	Acceptability of Waste Form		0.20
			1.00

TABLE C-4

WEIGHTING FACTORS FOR OPERATING ASSESSMENT CRITERIA

1.	Process Complexity		0.20
	a) Number of Process Steps	0.40	
	b) Amount of Internal Recycle	0.30	
	c) Amount of Secondary Waste and Off-Gas Generation	0.30	
2.	Equipment Reliability		0.20
	a) Probability of Failure/Maintenance Requirements	0.40	
	b) Process Latitudes	0.25	
	c) Ease of Remote Maintenance	0.35	
3.	Quality Assurance		0.15
	a) Controllability of Process	0.55	
	b) Ease of Accomplishing Quality Assurance	0.45	
4.	Facility/Special Material Requirements		0.20
	a) Extent of Contamination of Facilities	0.55	
	b) Extent of Use of Materials in Short Supply	0.45	
5.	Material Handling		0.25
	a) Above Ground Handling	0.40	
	b) Below Ground Handling	0.60	
			1.00

TABLE C-5

WEIGHTING FACTORS FOR RISK ASSESSMENT CRITERIA

1.	Operator Exposure		0.25
2.	Public Exposure		0.20
3.	Accident Potential		0.25
	a) Criticality	0.30	
	b) Fire	0.25	
	c) Sabotage	0.25	
	d) Natural Phenomena	0.20	
4.	Occupational Safety		0.15
5.	Long Term Hazard		0.15
			1.00

C.3 FIGURE OF MERIT COMPARISONS

The disassembly alternatives were rated with respect to the Reference Process by assigning a rating of 5 to the Reference Process and rating the alternatives on each criterion over a scale from 0 to 10 corresponding to unacceptable to very desirable. The assigned ratings as well as the weighted ratings for the individual criteria are shown in Table C-6. On virtually all criteria with the exception of the use of materials in short supply, material handling, and the unit cost of spent fuel disposal, the alternative processes are equal to the Reference Process or less desirable. The advantages accrue exclusively from the volume reduction of the spent fuel package or the reduction in number of packages and the consequent more efficient utilization of waste package materials and repository volume. Thus, not only are the overall package costs substantially lower, especially for Alternative 3 and 4, but the level of operational activity required to emplace the waste packages is reduced.

TABLE C-6

RATINGS OF DISASSEMBLY ALTERNATIVES

	End Fitting Removal		Fission Gas Venting		Disassembly and Pin Storage		Shearing and Immobilization	
	Rating	Weighted Rating	Rating	Weighted Rating	Rating	Weighted Rating	Rating	Weighted Rating
A. Technical Assessment								
(1) Status of Process Technology								
(a) State of Development	4.5	.11	3.0	.08	4.0	.10	2.0	.05
(b) Time Required to Complete	4.5	.11	4.0	.10	4.0	.10	2.0	.05
(2) Status of Equipment Development and Demonstration								
(a) State of Development	5.0	.08	4.0	.06	4.0	.06	2.0	.03
(b) Scale of Demonstration	5.0	.08	3.0	.04	4.0	.06	1.5	.02
(c) Time Required To Complete Development and Demonstration	5.0	.10	4.0	.08	4.0	.08	2.0	.04
(3) Amount of Prior Experience	5.0	.25	3.0	.15	3.5	.18	1.5	.08
(4) Safeguards Effectiveness								
(a) Accessibility of SNM	5.0	.10	5.0	.10	4.0	.08	4.0	.08
(b) Dispersability	5.0	.08	5.0	.08	5.0	.08	3.0	.04
(c) Ease of Nuclear Material Control and Accounting	5.0	.08	5.0	.08	4.0	.06	2.5	.04
(5) Acceptability of Waste Form	5.0	.25	5.5	.28	5.0	.25	4.5	.22
B. Operating Assessment								
(1) Process Complexity								
(a) Number of Process Steps	4.5	.07	4.0	.06	3.5	.06	2.0	.03
(b) Amount of Internal Recycle	5.0	.06	5.0	.06	5.0	.06	4.5	.05
(c) Amount of Secondary Waste & Off-Gas Generation	4.5	.05	4.0	.05	4.0	.05	2.5	.03
(2) Equipment Reliability								
(a) Probability of Failure/ Maint. Requirements	4.5	.07	3.5	.06	3.5	.06	2.0	.03
(b) Process Latitudes	5.0	.05	4.5	.04	5.0	.05	4.5	.04
(c) Ease of Remote Maintenance	5.0	.07	4.0	.06	4.5	.06	3.0	.04
(3) Quality Assurance								
(a) Controllability of Process	5.0	.08	4.5	.07	4.5	.07	3.0	.05
(b) Ease of Accomplishing Q.A.	5.0	.07	4.5	.06	4.5	.06	3.0	.04
(4) Facility/Special Material Requirements								
(a) Extent of Contamination of Facilities	4.5	.10	4.0	.09	4.5	.10	2.0	.04
(b) Extent of Use of Materials in Short Supply	5.5	.10	5.5	.10	9.0	.16	8.5	.15
(5) Material Handling								
(a) Above Ground Operations	5.0	.10	5.0	.10	7.0	.14	2.0	.04
(b) Below Ground Operations	5.0	.15	5.0	.15	9.0	.27	9.0	.27
C. Risk Assessment								
(1) Operator Exposure	5.0	.19	4.0	.15	4.0	.15	2.5	.10
(2) Public Exposure	5.0	.15	4.5	.14	5.0	.15	4.0	.12
(3) Accident Potential								
(a) Criticality	5.0	.06	5.0	.06	4.5	.05	4.0	.04
(b) Fire	5.0	.05	4.5	.04	4.5	.04	4.0	.04
(c) Explosion	5.0	.05	5.0	.05	5.0	.05	4.0	.04
(d) Natural Phenomena	5.0	.04	5.0	.04	5.0	.04	4.5	.03
(4) Occupational Safety	4.5	.10	4.0	.09	4.5	.10	3.5	.08
(5) Long Term Hazard	5.0	.11	5.0	.11	5.0	.11	5.0	.11
D. Economic Assessment								
(1) Unit Cost	5.5	2.20	5.5	2.20	10.0	4.00	9.0	3.60

A summary of the Figure of Merit components by assessment areas is given in Table C-7. On the basis of these summed weighted ratings, the disassembly of the spent fuel and close packing of the fuel pins is the preferred alternative.

TABLE C-7
FIGURE OF MERIT SUMMARY*

Assessment Area	Unmodified Spent Fuel	End Fitting Removal	Fission Gas Venting	Disassembly and Pin Storage	Shearing And Immobilization
Technical	1.25(1)	1.24(2)	1.05(3)	1.05(3)	0.65(4)
Operating	1.00(2)	0.97(3)	0.90(4)	1.14(1)	0.81(5)
Risk	0.75(1)	0.75(1)	0.68(3)	0.69(2)	0.56(4)
Economic	2.00(4)	2.20(3)	2.20(3)	4.00(1)	3.60(2)
Figure Of Merit	5.00(4)	5.16(3)	4.83(5)	6.88(1)	5.62(2)

*The number in parentheses is the comparative ranking

C.4 SENSITIVITY OF RATINGS

Because of the weight given to the economic assessment, it might be construed that the rating was slanted to give preference to the lowest cost alternative. To test the validity (or lack thereof) of this hypothesis, a series of calculations were completed which used three different weightings by assessment area, varying the relative weights assigned to economics and safety/risk. These weightings are shown in Table C-8, C-9, and C-10.

TABLE C-8

SENSITIVITY ANALYSIS NO. 1, WEIGHTING FACTORS FOR ASSESSMENT AREAS

Technical Assessment	0.25
Operating Assessment	0.25
Risk Assessment	0.25
Economic Assessment	0.25
	1.00

TABLE C-9

SENSITIVITY ANALYSIS NO. 2, WEIGHTING FACTORS FOR ASSESSMENT AREAS

Technical Assessment	0.25
Operating Assessment	0.20
Risk Assessment	0.35
Economic Assessment	0.20
	1.00

TABLE C-10

SENSITIVITY ANALYSIS NO. 3, WEIGHTING FACTORS FOR ASSESSMENT AREAS

Technical Assessment	0.25
Operating Assessment	0.20
Risk Assessment	0.45
Economic Assessment	0.10
	1.00

The rating values from Table C-6 were kept constant and the Figure of Merit derived from the weighed values in Tables C-8, C-9, and C-10 are shown in Tables C-11, C-12, and C-13. Alternative 3 remained the preferred alternative in each of these analyses.

TABLE C-11
SENSITIVITY ANALYSIS NO. 1
FIGURE OF MERIT SUMMARY*

Assessment Area	Unmodified Spent Fuel	End Fitting Removal	Fission Gas Venting	Disassembly and Pin Storage	Shearing and Immobilization
Technical	1.25(1)	1.24(2)	1.05(3)	1.05(3)	0.65(4)
Operating	1.25(2)	1.22(3)	1.12(4)	1.44(1)	1.04(5)
Risk	1.25(1)	1.24(2)	1.12(4)	1.16(3)	0.93(5)
Economic	1.25(4)	1.38(3)	1.38(3)	2.50(1)	2.25(2)
Figure Of Merit	5.00(3)	5.08(2)	4.67(5)	6.15(1)	4.87(4)

*The number in parentheses is the comparative ranking

TABLE C-12

SENSITIVITY ANALYSIS NO. 2

FIGURE OF MERIT SUMMARY*

Assessment Area	Unmodified Spent Fuel	End Fitting Removal	Fission Gas Venting	Disassembly and Pin Storage	Shearing and Immobilization
Technical	1.25(1)	1.24(2)	1.05(3)	1.05(3)	0.65(4)
Operating	1.00(2)	0.97(3)	0.90(4)	1.14(1)	0.81(5)
Risk	1.75(1)	1.72(2)	1.57(4)	1.61(3)	1.30(5)
Economic	1.00(4)	1.10(3)	1.10(3)	2.00(1)	1.80(2)
Figure Of Merit	5.00(3)	5.03(2)	4.62(4)	5.80(1)	4.56(5)

*The number in parentheses is the comparative ranking

TABLE C-13

SENSITIVITY ANALYSIS NO. 3

FIGURE OF MERIT SUMMARY*

Assessment Area	Unmodified Spent Fuel	End Fitting Removal	Fission Gas Venting	Disassembly and Pin Storage	Shearing and Immobilization
Technical	1.25(1)	1.24(2)	1.05(3)	1.05(3)	0.65(4)
Operating	1.00(2)	0.97(3)	0.90(4)	1.15(1)	0.81(5)
Risk	2.25(1)	2.22(2)	2.01(4)	2.08(3)	1.68(5)
Economic	0.50(4)	0.55(3)	0.55(3)	1.00(1)	0.90(2)
Figure Of Merit	5.00(2)	4.98(3)	4.51(4)	5.27(1)	4.04(5)

*The number in parentheses is the comparative ranking

APPENDIX D

EVALUATION OF FUEL PELLET DEGRADATION MECHANISMS

(Quoted in its entirety from Appendix A
to Hanford Engineering Development Laboratory
Report TC-1913,

ASSESSMENT OF SPENT FUEL WASTE FORM/STABILIZER
ALTERNATIVE FOR GEOLOGIC DISPOSAL, April 1981)

APPENDIX D
EVALUATION OF FUEL PELLET DEGRADATION MECHANISMS

A. CHEMICAL CHANGES IN THE FUEL

Previous studies (D1,D2) of possible mechanisms of fuel degradation have not been able to identify any chemical mechanisms which could cause significant degradation of the fuel material as long as the cladding remains intact. It is expected that chemical compounds formed during irradiation will remain essentially unchanged at the much lower temperatures in pool storage or repository.

The only effect which would be expected to alter the chemical state significantly over a period of time is the decay of fission products into other chemical species. It is evident from the rapid decrease in decay heat during the first few years (D3) that much of the degradation from this source would occur during the initial cooling period due to relatively short-lived radionuclides. Metallographic examinations of intact spent fuel after up to 11 years of pool storage have disclosed no noticeable degradation of the fuel (D4,D5). Damage in a repository (after pool storage) would have to be due to the daughters of long-lived radionuclides. The quantities of such fission products are generally small and they tend to decay into chemically similar elements (D6). Most fission products, being neutron rich, decay by β emission thereby increasing their atomic number by one. Thus, for the most part,

a) Oxide formers decay into oxide formers
b) Noble metals decay into other noble metals
c) Reactive metals decay into other reactive metals

Exceptions to this are decays involving the inert gases.

$$^{85}Kr \xrightarrow[10.76\ y]{\beta} {}^{85}Rb$$

and

$$^{129}I \xrightarrow[1.7 \times 10^{7}\ y]{\beta} {}^{129}Xe$$

Based on half-life considerations, only the decay of ^{85}Kr is of any concern.

Probably the most important exception to the above argument is ^{137}Cs which decays to stable ^{137}Ba with a half-life of 30 years. Cesium is monovalent while barium is divalent indicating that this process will have a chemical effect since there is a high in-reactor yield of the highly mobile Cs nuclide. For example, if Cs is present as CsI, the expected change will be $2\ CsI \rightarrow BaI_2 + Ba$. The free Ba will remain as a metal or act as a getter for oxygen or any other available atom. In ten year old fuel, ^{137}Cs is present at a concentration of about 0.15 atom percent of heavy metal in PWR fuel irradiated to 30,000 MWD/MTU and about 0.10 atom percent in BWR fuel irradiated to 20,000 MWD/MTU (D8). This may increase near the pellet outer surface (D9) since cesium tends to migrate down the temperature gradient. Essentially all of the ^{137}Cs initially present will decay to barium during the 1,000 year thermal period and the magnitude of the effect as a fuel degradation mechanism is unclear. It should be noted, however, that about 21 percent of the ^{137}Cs initially present decays during the first 10 years of pool storage, and, as previously mentioned, examinations have disclosed no evidence of change indicating that the effect is probably minor. These metallographic examinations, however, were directed mainly at investigation of cladding degradation. More detailed experimental studies of spent fuel are, therefore, needed in order to reliably assess chemical changes, if any, which are occurring in spent fuel.

B. VOLATILE FISSION PRODUCT MIGRATION

Significant migration of volatile fission products within the fuel could cause a decrease in resistance of the fuel to attack by a leachant if the volatiles accumulate near the outer edge of the fuel. Migration of volatile fission products as a function of temperature was extensively investigated by Cubicciotti (D9,D10) who found that migration rates were extremely sensitive to temperature. He deduced the existence of a threshold for migration of volatile solids at a fuel center temperature of 1000C, and reported a threshold for release from the fuel at a centerline temperature of approximately 1400C. Behavior of the volatile fission products as a function of burnup and fuel center temperature is shown schematically in Figure D1

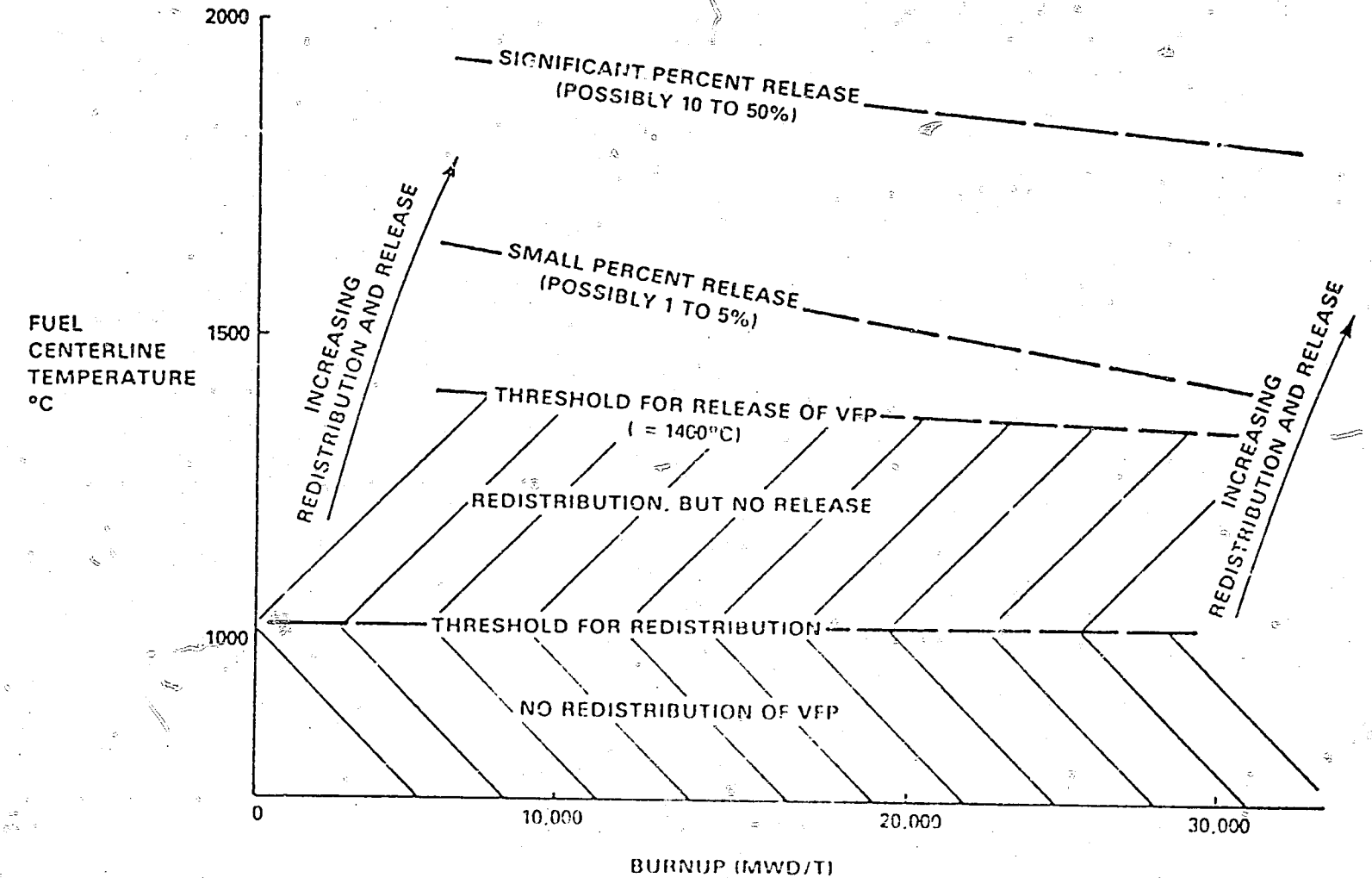


FIGURE B-1

SCHEMATIC REPRESENTATION OF REDISTRIBUTION AND RELEASE BEHAVIOR OF VOLATILE FISSION PRODUCTS

(from reference D9). The reason for the sensitivity of migration rate to temperature can be appreciated by examining Figure D2, which shows approximate vapor pressure as a function of temperature for the important group of volatile solid fission products including CsI, RbI, CsBr, RbBr, Cs_2Te, and Cs_2Se. As can be seen, the vapor pressure rapidly becomes vanishingly small at temperatures much below 1000C. The vapor pressure at 400C is about 6-1/2 orders of magnitude below that at the redistribution threshold which, along with the relatively insignificant temperature gradient across the fuel in repository (see Section E), renders vapor transport of these volatile fission products virtually inoperative.

C. GAS DIFFUSION

The principal fission gases produced in LWR fuel are Kr and Xe with Xe predominating at about 90 atom percent of gas produced (D8). These gases are produced in the fuel matrix during in-core irradiation and at temperatures above about half the absolute melting temperature begin to form gaseous precipitates (D12). These precipitates contain a few to several hundred vacancies per gas atom. They tend to grow into bubbles by collecting additional gas atoms and vacancies, but can also be destroyed by a thermal spike from a nearby fission event. An equilibrium condition is established between gas in-bubbles and in-solution within the fuel grain (D13,D14,D15). Baker (D16) calculated the fraction of total gas generated residing in the intragranual bubbles by measuring bubble diameters and densities in fuel between 950C and 1800C. Results indicated the fraction of gas in intragranular bubbles to be about 20 percent up to 1400C and then increasing to about 40 percent above 1600C. The gas in solution diffuses to form bubbles at the grain boundaries which gradually increase in size and density until interconnection occurs and gas begins venting to the free volume of the pin. This interconnection of intergranular porosity by means of grain-edge tunnels is usually well developed at less than 1 atom percent burnup (D9,D17,D18,D19).

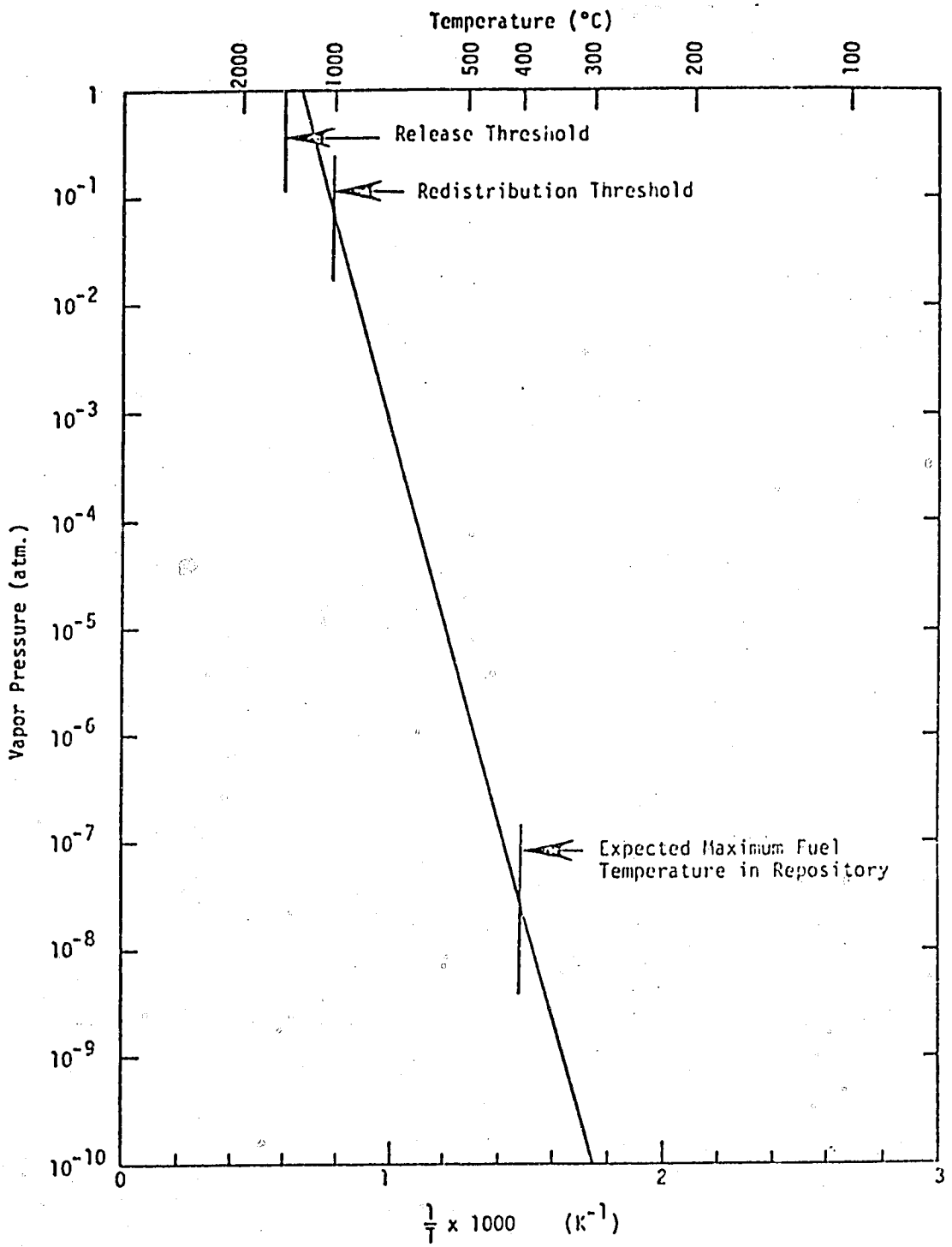


FIGURE D-2

APPROXIMATE VAPOR PRESSURE FOR IMPORTANT VOLATILE FISSION PRODUCTS

(C_sI, RbI, C_sBr, Cs_2Te, and Cs_2Se) (D9,D11)

Although gas diffusion will be a slow process at the relatively low temperatures expected in a repository, it is expected that some gas will diffuse out of the grains over a long period of time and could conceivably accumulate at any unvented intergranular porosity possibly leading to weakening of the grain boundaries.

Diffusion of xenon out of sintered UO_2 at low temperatures has been experimentally studied by Taketani and Ikawa between 400C and 800C (D20). They found the effective diffusion coefficient to undergo a transition at about 600C as shown in Figure D3. Below the transition temperature effective diffusion coefficient can be represented by:

$$D\left(\frac{m^2}{s}\right) = 2.54 \times 10^{-22} \exp\left[-\frac{4.003 \times 10^3}{T\,(K)}\right] \qquad [1]$$

This diffusion coefficient has been used in our assessment since it is a conservative upper bound on the data extrapolated from higher temperature measurements C9 (see Figure D3). Extrapolating to repository temperatures produces very low estimates for the diffusion coefficient as shown in Figure D4.

Total gas release fractions have been estimated for an equivalent spherical grain after 1,000 years at various constant temperatures. The calculations were based on simple diffusion of the gas in solution in the grain and took no credit for the presence of intragranular bubbles as trapping centers. A fundamental mode concentration profile within the grain was assumed at the beginning of the repository period, and the grain surface was conservatively assumed to be a perfect sink. The resulting release fraction estimates are shown in Figure D5 for the typical range of grain sizes in LWR Fuel (D13). As would be expected, the release fraction increases very rapidly above 500C, and after 1,000 years at 800C, nearly all the gas has either left solution, or reached equilibrium with the internal pressure of the pin. It should be noted that the temperatures shown in Figure D5 are assumed to be constant over the thermal period (first 1,000 years).

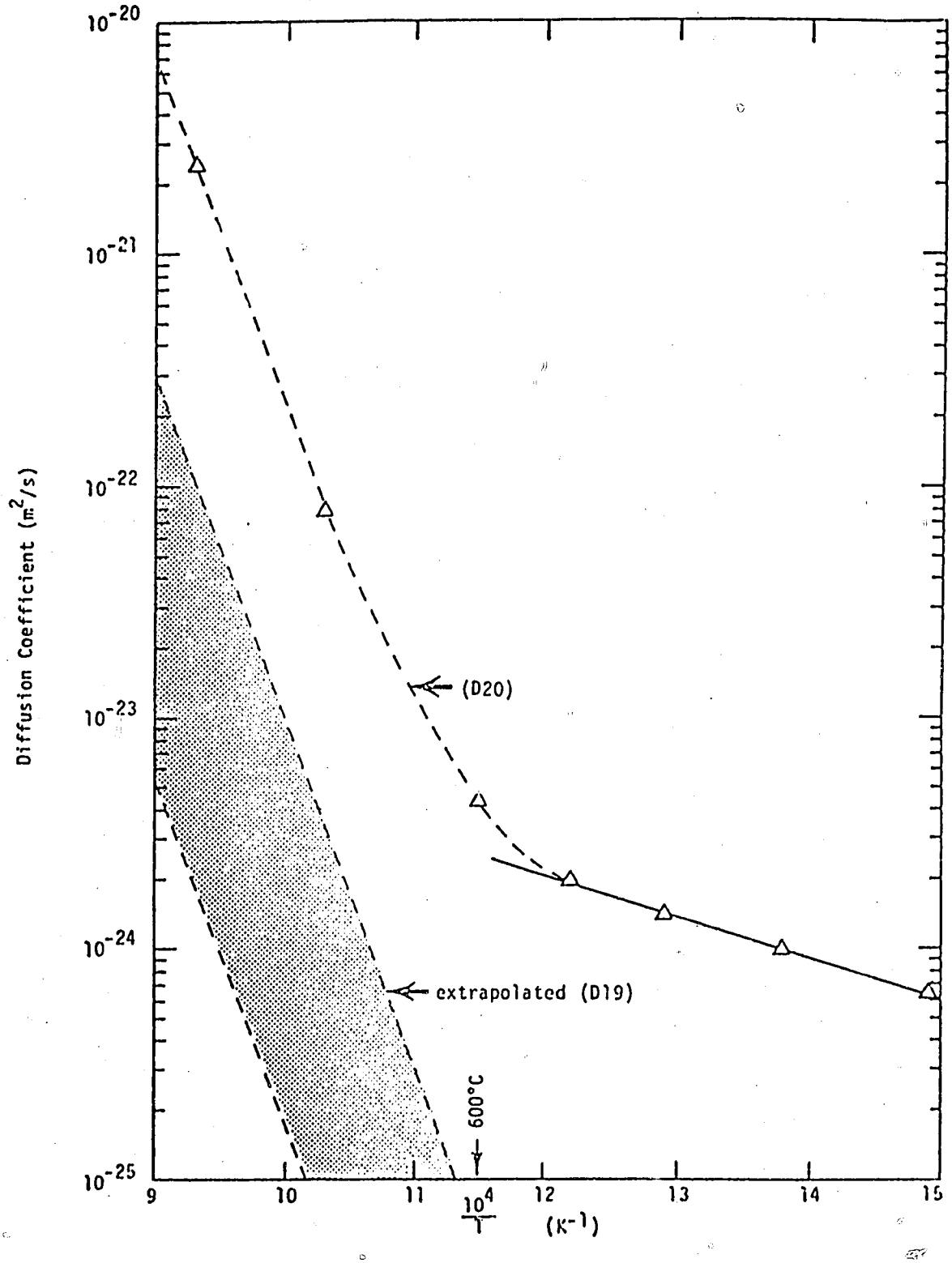


FIGURE 7-3

DIFFUSION COEFFICIENT OF Xe in SINTERED UO_2

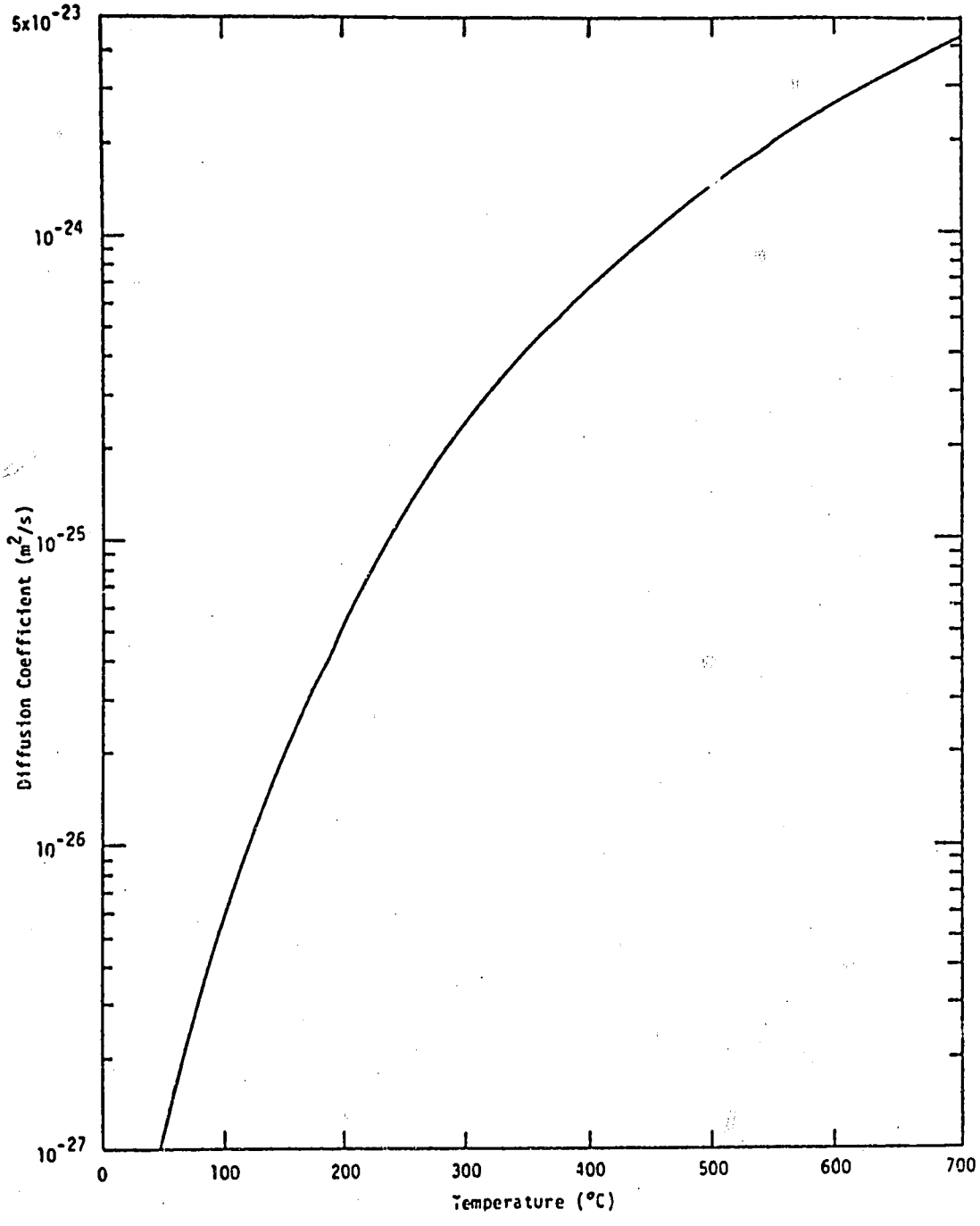


FIGURE D4

LOW TEMPERATURE THERMAL DIFFUSION COEFFICIENT FOR Xe IN UO_2

(Extrapolated From the Data of Reference D20)

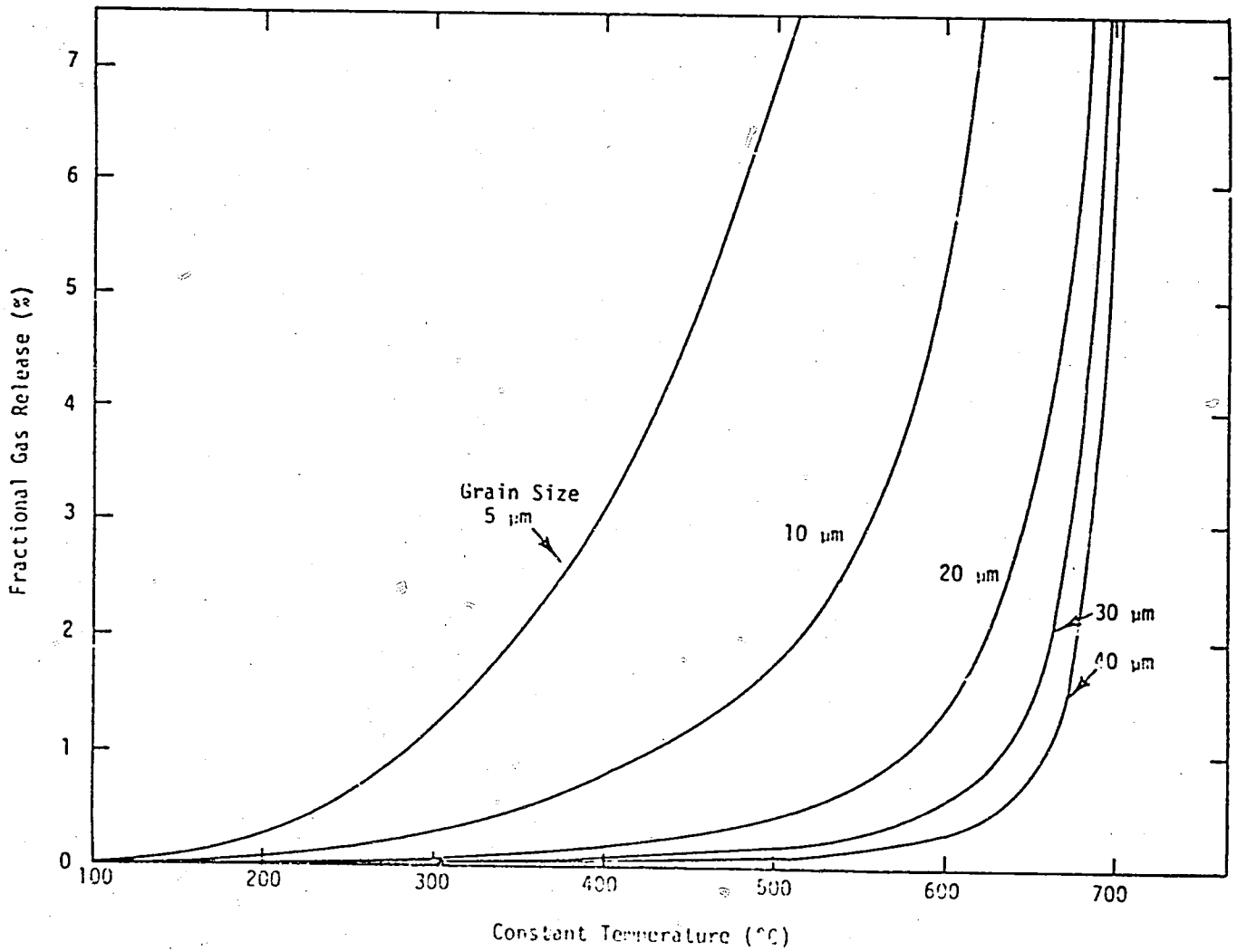


FIGURE D-5

ESTIMATED FISSION GAS RELEASE FROM THE FUEL IN 1000 YEARS AS A FUNCTION OF GRAIN SIZE AND TEMPERATURE

D. RADIATION DAMAGE

The effects of radiation damage in single crystal UO_2 (analogous to individual grains) at low temperatures (less than 400C) and relatively high exposures were investigated by Bates, et al (D21). They found UO_2 to be extremely resistant to radiation damage with a total exposure of 1.44×10^{20} f/cm^3 producing a lattice strain of about 0.024 percent and only "subtle and minute" changes in microstructure along with some hardening. This exposure is equivalent to a total energy deposition of about 4.6×10^9 J/cm^3 compared to a conservatively estimated deposition rate of about 3.2×10^6 J/cm^3 per year for typical PWR fuel only one year old (D3). According to lower exposure data reviewed by Lustman, (D22) the lattice strain appears to pass through a maximum of about 0.16 percent at an exposure of about 5×10^{16} f/cm^3 with the damage annealing out above this exposure (D21,D22). The maximum lattice strain of 0.16 percent is comparable to the effect of a solution of about 7 percent ThO_2 indicating that the changes in observed properties would be expected to be minor and, in addition, appear to anneal rapidly at temperatures above 200C (D22,D23). These data are based on irradiation of UO_2 in a reactor at low temperatures. Most of the damage is due to fission fragments which are much more disruptive than, for example, atom recoil or alpha emission processes. Damage associated with in reactor, low temperature irradiation can therefore be considered an upper limit on the damage due to the less disruptive processes associated with radioactive decay. The UO_2 structure is, therefore, not expected to change significantly due to self-radiation while in repository.

E. THERMAL AND MECHANICAL FRAGMENTATION

Fragmentation of the fuel beyond the extent existing at end of irradiation would have the effects of increasing the surface area available to a leachant and providing access to previously isolated fission product phases within the fuel.

The total fuel pellet centerline to surface temperature drop was calculated for typical LWR fuel as a function of fuel age and temperature using power levels and fuel thermal conductivities shown in Tables D1 and D2 assuming a pellet radius of 0.5 cm (D24). The resulting temperature drops are very small even for freshly irradiated fuel as shown in Figure D6. A thermal stress analysis by Bosi (D26) indicates the minimum temperature drop to fracture an intact pellet (worst case) to be approximately 66C. The thermal stresses due to shocks and gradients are far more severe during reactor transients and shutdown (when the fuel actually fragments) than during time in pool storage or repository. The fuel is, therefore, not expected to suffer any additional fragmentation due to thermal effects during geologic disposal.

The possibility of fuel degradation due to mechanical shock during handling and geologic disposal was assessed by Bosi (D26). He considered a peak transportation shock loading to consist of a half sine wave pulse with a duration of 0.059 msec peaking at 2.9 g acceleration (D27). Application of this loading to intact pellets (worst case) in cladding produced contact stresses of approximately 1317 psi and 1390 psi for PWR and BWR fuel, respectively, which are well below the fracture strength of approximately 11.5 ksi at zero degree Celsius. Since for closely fitting nested cylinders the contact stress varies with the applied force to the 1/2 power (D28), it would require about 68 times the postulated maximum loading to fracture a pellet.

In order to determine the seismic loading force, Bosi (D26) surveyed the data on siesmicity in deep geologic formations and adopted a peak acceleration of 0.25 g with a frequency of 3 Hz based on a Mercalli intensity of 7.5 and a mean distance from the causitive fault of 15 miles. Since expected seismic shock loadings are far less than those due to handling and transportation (barring physical disruption of the pin) further fracturing of the fuel due to seismic effects is not expected.

TABLE D1

DECAY POWER FOR PWR FUEL (3.3% ENRICHED, 37.3 MW/MTU, 33 MWD/kgU) (D3)

FUEL AGE	DECAY POWER (kW/MTU)	HEAT SOURCE @ 92% TD (w/cm^3)
10d	87	0.88
180d	19	0.19
1yr	11	0.11
5yr	2	0.02
10yr	1	0.01

TABLE D2

THERMAL CONDUCTIVITY OF UO_2 AT 92% T.D. (D25)

Temperature C	k (w/cm C)
50	0.070
100	0.064
200	0.055
300	0.048
400	0.043
500	0.039
600	0.035
700	0.032
800	0.030
900	0.028
1000	0.026

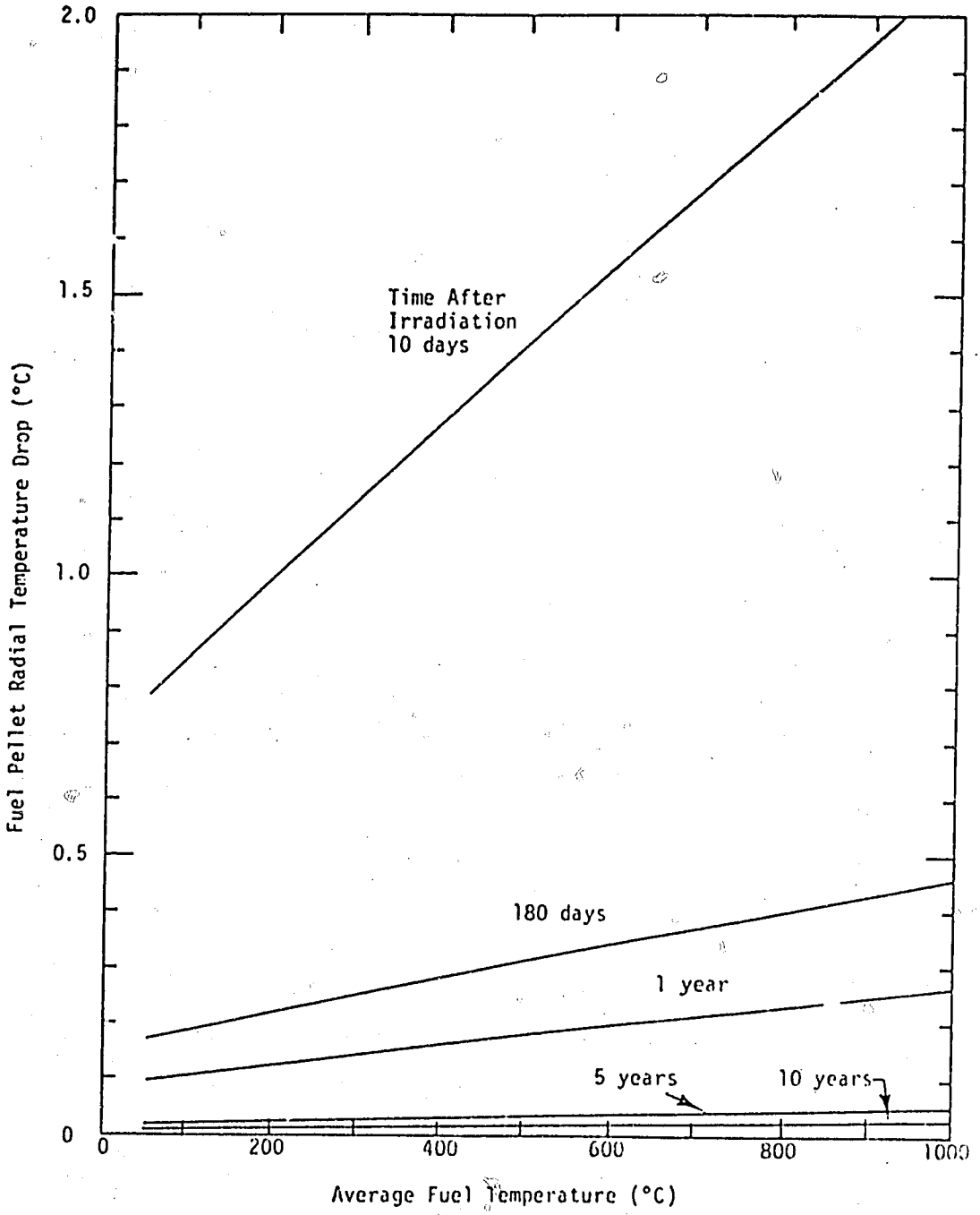


FIGURE D-6

FUEL PELLET RADIAL TEMPERATURE DROP AS A FUNCTION OF FUEL TEMPERATURE AND AGE

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APPENDIX E
ANALYSIS OF PACKAGE PARAMETERS AND COST OF REFERENCE PROCESS INCREASED CANISTER LOADING

APPENDIX E

ANALYSIS OF PACKAGE PARAMETERS AND COST OF REFERENCE PROCESS INCREASED CANISTER LOADING

The economic analyses of the Reference Process and the alternatives revealed that a high percentage of the cost was due to the package costs, specifically the titanium and Inconel requirement, giving Alternatives 3 and 4 significant advantage. Additionally, the operating assessment showed considerable advantage for Alternative 3 because of the reduced number of packages for handling and emplacement following canistering. An analysis was done of the incremental package costs considering the Reference Process (unmodified spent fuel) loaded at the same fuel volume per canister as Alternative 3.

In order to accommodate 3 PWR assemblies or 8 BWR assemblies in one canister, it was determined that a canister of 24.25 inches outside diameter would be required. This is outside the bounds of the maximum 18-inch diameter canister specified for this study. The operating impacts of the larger diameter canister were not assessed. However, if it proved feasible from an operational viewpoint, increased canister loading would have the advantage of decreasing the package costs per assembly by $36,610 and $24,530 for PWR and BWR loadings, respectively. Over the life of the repository, the reduction in packaging cost over the Reference Process would be $7.64 billion.

The following Tables E-1 through E-4 show the quantities and costs of package requirements for increased loading of the Reference Process canister.

TABLE E-1

SPENT FUEL PACKAGE PARAMETERS

Canister Description	PWR	BWR
Outside Diameter (in.)	24.25	24.25
Wall Thickness (in.)	0.25	0.25
Length (in.)	176.9	193.1
Assemblies Contained	3	8
Heat Load/Canister (kW)	1.59	1.52
Number Canisters/yr	2,100	1,113
Package Description		
Liner Outside Diameter (in.)	30.25	30.25
Wall Thickness (in.)	1	1
Liner Length (in.)	237.5	253.7
Excavation Required (ft^3)	356.6	378.3
Size Reduction from Reference Process Package (ft^3)	-122.8	-130.3

TABLE E-2

SPENT FUEL PACKAGE REQUIREMENTS (INDIVIDUAL)

Description	PWR	BWR
TiCode Canister		
Volume of TiCode-12 (ft^3/canister)	1.95	2.12
Weight of TiCode-12 (lb)	548	596
Cost/Canister ($)	20,930	22,850
Canister Cost/Assembly ($)	6,980	2,860
Canister Cost Reduction from Reference Process Canister ($/canister)	-6,100	-6,600
Canister Cost Reduction from Reference Process Canister ($/assembly)	7,850	5,270
Inconel Liner		
Volume of Inconel 600 (ft^3/canister)	14.06	14.92
Weight of Inconel 600 (lb/liner)	7,452	7,908
Cost/Liner ($)	91,390	96,980
Liner Cost/Assembly ($)	30,460	12,120
Liner Cost Reduction from Reference Process ($/liner)	-32,170	-34,120
Liner Cost Reduction from Reference Process ($/assembly)	28,760	19,310
Totals		
Canister and Liner Cost ($)	112,320	119,830
Canister and Liner Cost/Assembly ($/assembly)	37,440	14,980
Cost Reduction per Canister and Liner from Reference Process ($/package)	-38,270	-40,720
Cost Reduction per Assembly from Reference Process ($/assembly)	36,610	24,580

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