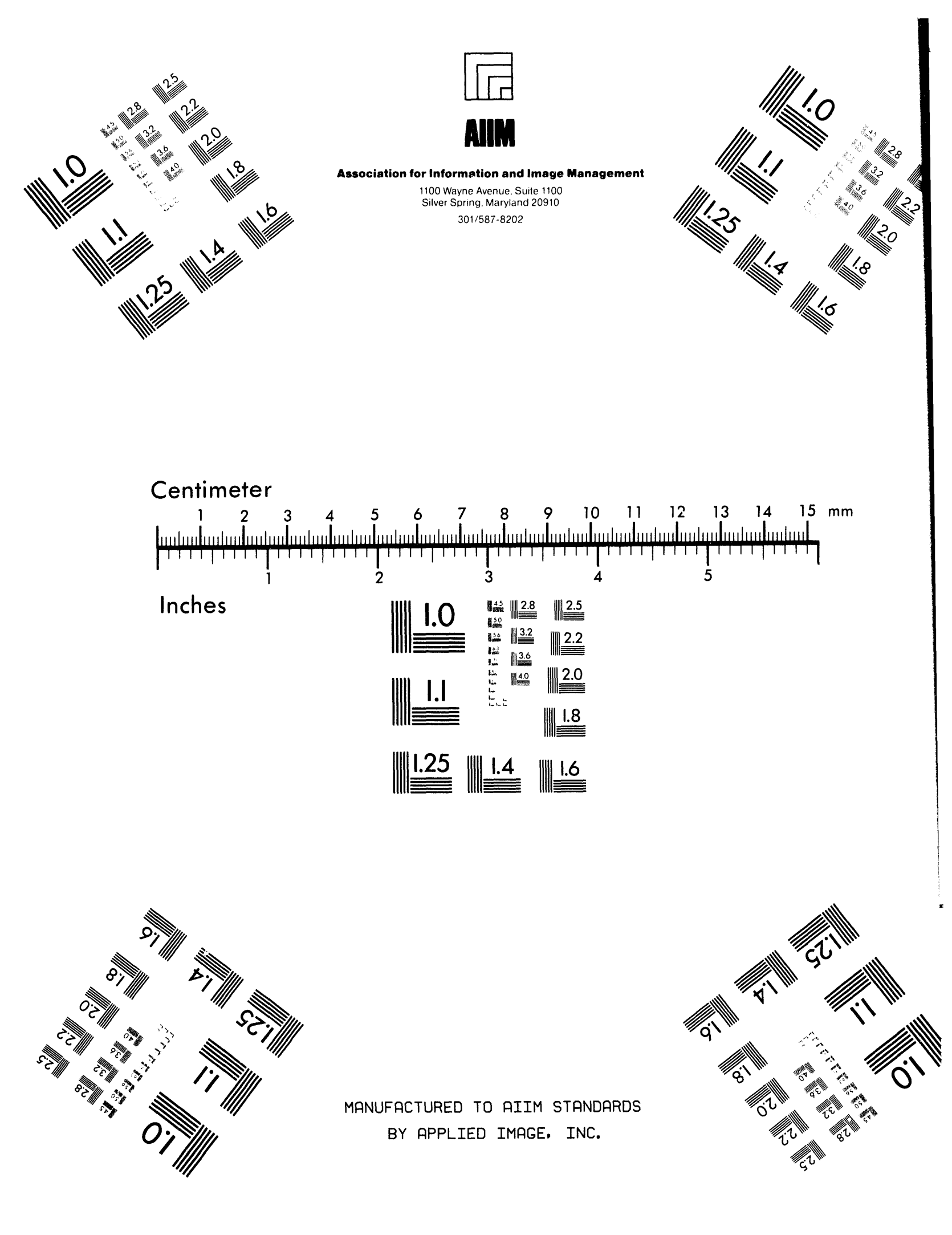
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Weapons-Grade Plutonium Dispositioning Volume 2

Comparison of Plutonium Disposition Options

D. A. Brownson
D. J. Hanson
H. S. Blackman
R. L. Bratton
C. H. Cooper
K. S. Dawson Jr.
L. B. Lundberg
C. S. Olsen
A. M. Ougouag
P. D. Wheatley
R. E. Felt[a]
R. N. Henry[a]

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a. Westinghouse Idaho Nuclear Company, Idaho Falls, Idaho 83415.

ABSTRACT

The Secretary of Energy requested the National Academy of Sciences (NAS) Committee on International Security and Arms Control to evaluate disposition options for weapons-grade plutonium. The Idaho National Engineering Laboratory (INEL) offered to assist the NAS in this evaluation by investigating the technical aspects of the disposition options and their capability for achieving plutonium annihilation levels greater than 90%. This report was prepared for the NAS to document the gathered information and results from the requested option evaluations.

Evaluations were performed for 12 plutonium disposition options involving five reactor and one accelerator-based systems. Each option was evaluated in four technical areas: (1) fuel status, (2) reactor or accelerator-based system status, (3) waste-processing status, and (4) waste disposal status. Based on these evaluations, each concept was rated on its operational capability and time to deployment. A third rating category of option costs could not be performed because of the unavailability of adequate information from the concept sponsors.

Based on these evaluations, the INEL believes that if plutonium annihilation levels greater than 90% are desired by policy, only those options that reprocess irradiated fuel can reasonably achieve this goal. Half of the disposition options evaluated included reprocessing. The four options achieving the highest rating, in alphabetical order, are the Advanced Light Water Reactor with plutonium-based ternary fuel, the Advanced Liquid Metal Reactor with plutonium-based fuel, the Advanced Liquid Metal Reactor with uranium-plutonium-based fuel, and the Modular High Temperature Gas-Cooled Reactor with plutonium-based fuel. Of these four options, the Advanced Light Water Reactor and the Modular High Temperature Gas-Cooled Reactor do not propose reprocessing of their irradiated fuel. Time constraints and lack of detailed information did not allow for any further ratings among these four options. The INEL recommends these four options be investigated further to determine the optimum reactor design for plutonium disposition.

SUMMARY

The Secretary of Energy requested the National Academy of Sciences (NAS) Committee on International Security and Arms Control (CISAC) to evaluate disposition options for weapons-grade plutonium. The Idaho National Engineering Laboratory (INEL) offered to assist the NAS in this evaluation. The NAS indicated that gathering information from the sponsors of reactor and accelerator-based options on the capability of their systems to annihilate plutonium (destruction of 90% to 99.9% of the plutonium) and providing an evaluation of this information would be very beneficial in selecting a preferred plutonium disposal method. This report was prepared for the NAS to document the gathered information and the results from the requested option evaluations.

The following four areas were identified where additional information could provide increased understanding of the capability of the reactor and accelerator-based options to annihilate plutonium: (1) fuel status, (2) reactor and accelerator-based system status, (3) waste-processing status, and (4) waste-disposal status. A set of questions was developed in each of these areas and transmitted to the following option sponsors for response:

- Brookhaven National Laboratory—Particle Bed Reactor (PBR)
- General Atomics—Modular High Temperature Gas-Cooled Reactor (MHTGR)
- General Electric Company/Argonne National Laboratory—Advanced Liquid Metal Reactor (ALMR)
- Los Alamos National Laboratory—Accelerator-Based Conversion System (ABC)
- Oak Ridge National Laboratory—Molten Salt Reactor (MSR)
- Westinghouse Savannah River Company—Advanced Light Water Reactor (ALWR).

Responses from sponsors, together with the original proposals, were reviewed and evaluations of option capabilities were made. A summary of results for each of the four areas and a recommended grouping of plants according to plutonium annihilation capabilities follows.

Fuel Status

Previous fuel development work for many of the options concentrated on uranium-based fuel. As a result, plutonium-based fuel development has been limited and demonstration of full-scale fuel fabrication has not been made. Based on the current fuel status, the development and fabrication of plutonium-bearing fuels will be on the critical path if annihilation of a high percentage of the plutonium (90% or greater) is desired. Fuel development will consume much of the time required to design and construct any of the reactor or accelerator-based options.

Two fuel forms have been proposed for both the ALMR and ALWR options. A uranium-plutonium-based metal fuel has been proposed for the ALMR reference fuel cycle (referred to as ALMR-R) and a plutonium-based metal fuel for a maximum burner or annihilation cycle (ALMR-MB). A mixed oxide (MOX) fuel has been proposed for the ALWR (ALWR-MOX) and also a ternary fuel (ALWR-T) to provide more rapid plutonium annihilation.

Fuel development of the ALWR-MOX has been completed. Irradiation testing of the ALMR-R fuel is under way at Argonne National Laboratory (ANL). Vital developmental work remains for the ALMR-MB, ALWR-T, and MHTGR fuels. Significant development work will be required for the fuel and the components associated with the core region (frits, etc.) for the PBR. Insufficient information on fuel development needs was provided by sponsors of the ABC and MSR, but reviews by the INEL indicate that additional fuel development is required.

The INEL believes that sponsor estimates for fuel fabrication facility costs and schedules may be optimistic because experience with uranium oxide fuel cannot be directly extrapolated to plutonium-based fuel fabrication. Plutonium fuel facilities will have more complex and difficult safety and

environmental issues to resolve. In addition, all fabrication and storage facilities will likely be funded by the Department of Energy (DOE) and reside on a highly secure DOE reservation, which could result in higher costs and longer schedule durations. Operating costs and durations are expected to be greater than currently encountered for uranium-based fuel. However, operating costs are a small part of the overall costs of the plant.

Reactor and Accelerator System Status

The INEL believes that annihilation of plutonium is preferred over denaturing. Furthermore, because it is possible to construct a nuclear explosive device from a wide range of plutonium isotopic concentrations, the INEL believes that the capability to annihilate all five plutonium isotopes (^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, and ^{242}Pu) is a better measure of a concept's effectiveness than the capability to annihilate just the ^{239}Pu and ^{241}Pu isotopes. A discussion of the capabilities of the reactor and accelerator-based options to annihilate plutonium follows.

The original ALWR option description proposed using mixed oxide (MOX) fuel for disposition of plutonium. The ALWR sponsor response states that ternary fuel (PuO_2-ZrO_2-CaO) is now the preferred fuel. The INEL believes that the change in ALWR fuel choice is possibly because use of MOX fuel to annihilate 50 MT of plutonium would require long periods of time or large numbers of reactors. Numerous fuel-reprocessing cycles would be necessary. Although the use of existing commercial reactors may be possible, reprocessing would likely become the critical path for mission completion. Large amounts of spent fuel would have to be shipped to centralized reprocessing facilities, increasing the potential for diversion. In addition, the time and cost to license a large number of commercial light water reactors (LWRs) for MOX fuel could be substantial. Because commercial LWRs are not standardized, it is likely that the Nuclear Regulatory Commission (NRC) would have to consider the licensing of MOX fuel on a plant-by-plant basis.

ALWR-T and MHTGR options are capable of annihilating high percentages of ^{239}Pu and ^{241}Pu without reprocessing. The ALWR sponsor proposes using a ternary fuel to achieve annihilation of a significant fraction of the total plutonium isotopes without reprocessing. ALWR-T high annihilation percentages can only be reached through in-reactor fuel assembly resident times significantly longer than current LWR fuel assemblies typically experience. Further investigation would be required to determine whether ALWR-T fuel assembly materials could survive long resident times without being refurbished. The MHTGR sponsor proposes employing a fuel management scheme that involves shuffling irradiated fuel blocks into the core reflector region to achieve annihilation of large fractions of all plutonium isotopes. Further investigation would be required to determine the practicality of such a fuel management scheme.

Calculations by option sponsors indicate that concepts with reprocessing can achieve near total plutonium annihilation in shorter durations than nonreprocessing options. A comparison of the plutonium depletion rates of all reactor and accelerator-based concepts is provided in Figure S-1. As seen from this figure, the ABC, ALMR-MB, MSR, and PBR reprocessing concepts can achieve near total annihilation in a 32- to 45-year period. The MHTGR fuel scheme does not allow for reprocessing of all irradiated fuel and annihilation levels achieved by other reprocessing options cannot be attained by the MHTGR. Total annihilation of the weapons-grade plutonium can be accomplished for the ALMR-R concept, but over an estimated 160-year period. The INEL believes that the best nonreprocessing option is the ALWR-T. The maximum achievable annihilation level for this concept is 65% over a 32-year period. Table S-1 shows the time required to annihilate various percentages of 50 MT for the options that use reprocessing. Results from this table indicate that the concepts that can annihilate total plutonium quickest are the ABC and ALMR-MB. These concepts are closely

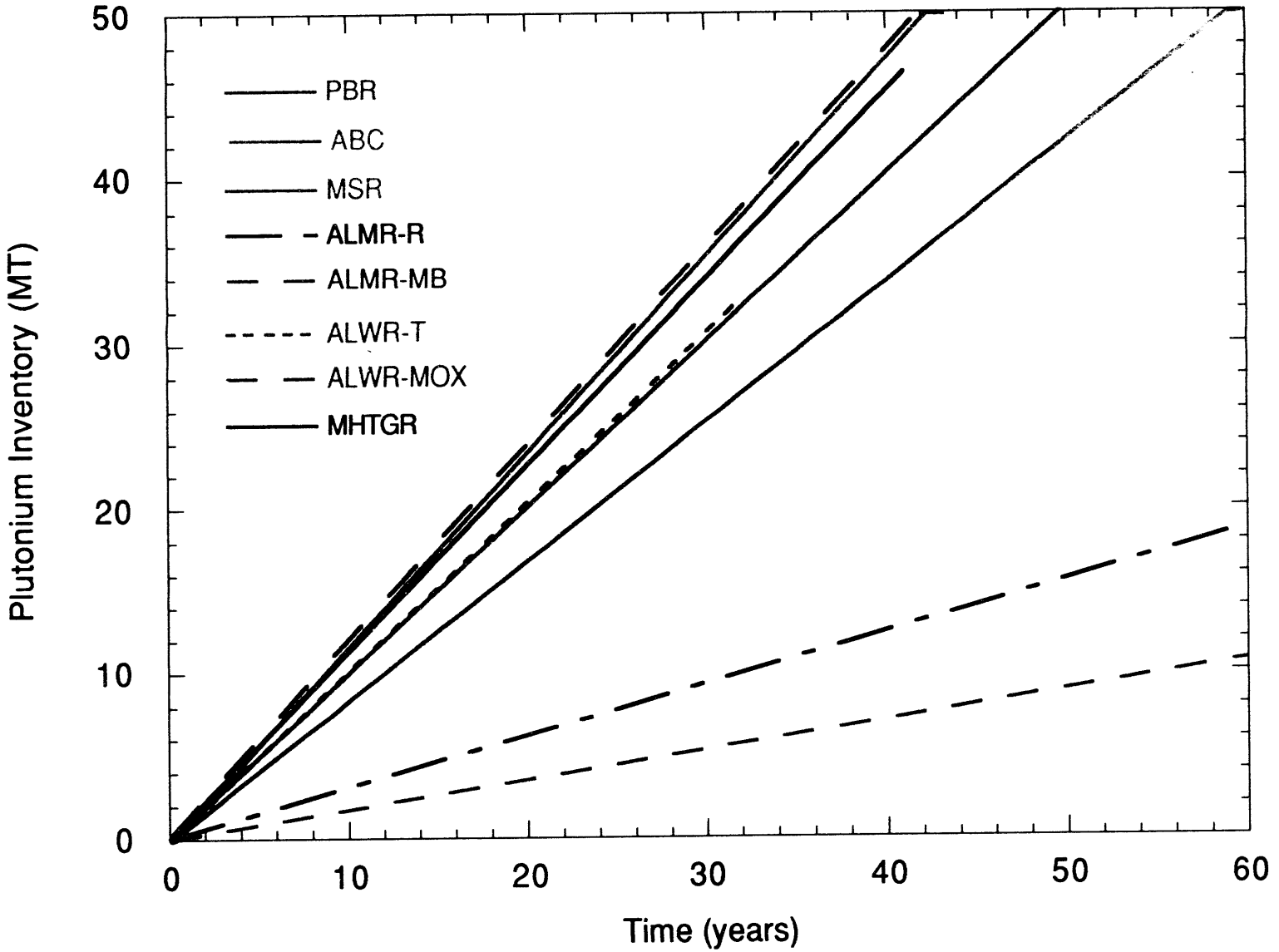

Figure S-1. Depletion plutonium inventories for plutonium disposition options.

Table S-1. Time required to annihilate various fractions of 50 MT Pu for reactor and accelerator systems, assuming reprocessing of the irradiated fuel.

Concept	MW(t)	Annihilation time (years)[a]			
		90%	95%	99%	99.9%
ABC	4,260	37.8	39.9	41.6	42.0
ALMR-R[b]	4,239	144.1	152.1	158.5	159.9
ALMR-MB[c]	4,239	37.9	40.0	41.7	42.1
ALWR-T[d]	3,636	NR	NR	NR	NR
MHTGR	4,050	39.8	42.0	43.8	U
MSR	3,030	53.2	56.1	58.5	59.0
PBR	3,600	44.8	47.3	49.3	49.7

U—This level of annihilation is unachievable.
NR—No reprocessing proposed.

a. 75% capacity factor assumed for each concept.
b. ALMR reference fuel cycle with 0.61 conversion ratio.
c. ALMR maximum burner fuel cycle with 0.02 conversion ratio.
d. ALWR ternary fuel with one-third reload fuel management scheme.

followed by the MHTGR and PBR. However, no method has been proposed for reprocessing the MHTGR or PBR fuel particles. In addition, the ABC, MSR, and PBR concepts are not highly developed and will require a significantly longer time period to implement and deploy.

Technology development issues must be resolved for all reactor and accelerator-based systems. Criticality and reactivity control during reactor operation must be examined when annihilating large percentages of plutonium. Development work on a reactor concept similar to the PBR option in the Commonwealth of Independent States indicates that substantial development work is required for the PBR. Specific technology development issues for the ABC, MSR, and PBR options were not identified because these concepts are in a preliminary stage of conceptual development. It is clear that significant issues relating to materials, design, and fabrication would have to be resolved before these concepts could be constructed.

The INEL believes that sponsor estimates for system development and construction costs and schedules are optimistic and that costs would be higher and schedules would be longer than predicted. Startup and operational costs are expected to be similar to those of currently operating reactor facilities on a per reactor basis.

Waste-Processing Status

A detailed technical assessment of the waste-processing area was not performed by the INEL. Waste characterization work will be necessary for all plutonium-based fuels. Waste processing is an integral part of the ABC, ALMR, and MSR. Technical development of the ALMR waste-processing system is under way at ANL. Waste processing for the ABC and MSR requires process and component development for plutonium-based fuels. The sponsor indicated that waste processing is not necessary for the PBR because no reprocessing is proposed and the particles would be packaged and sent to a waste disposal facility. Waste processing for

pletely evaluated. Oak Ridge National Laboratory examined disposal of uranium-based MHTGR fuel and concluded fuel elements could be safely placed in a repository without waste processing. Waste processing for the ALWR-T fuel would require more complex and time-consuming processes than the MOX fuel. Some benchtop development work for the uranium-based ternary fuel has been done.

Waste-Disposal Status

Several waste-disposal packages are possible using reactors or an accelerator. Each of the possible waste packages has disposal issues that must be considered in a comparative evaluation process, including:

- **Repository Availability.** The likelihood of waste from the reactor or accelerator-based concepts going to the first geological repository is very low and plans for a second repository have not been initiated. Monitored storage of plutonium, or its denatured form, could be required for several decades.

- **Repository Control.** There are two key variables in the control of material in a repository: control/containment of radioactive material and control of criticality. Criticality control is difficult to demonstrate to the general public because the fissionable material decays very slowly, which means material could be available to form a critical mass for long periods of time. In addition, there is no evidence that the containment material will last the long times necessary to prevent the fissionable material from migrating into a critical geometry.

- **Waste Forms and Characterization Programs.** Any new waste forms will require characterization and performance testing prior to acceptance at a future geologic repository.

Several option sponsors recognized these issues and it is clear adequate consideration of waste disposal needs in all options requires further investigation. Although it does not appear that any concept has a notable advantage in the waste characterization area, the ABC, ALMR, and MSR concepts all have decreased repository requirements such as radioactive lifetime of their final fuel form and minimal criticality control issues. Further study is required to characterize each option's waste streams and long-term storage implications.

General Rating of Plutonium Annihilation Options

The INEL produced ratings for the reactor and accelerator-based systems considering only options that have the capability to annihilate large fractions of weapons-grade plutonium. These ratings have a different basis than ratings produced by Lawrence Livermore National Laboratory (LLNL) because its ratings considered options that denatured plutonium as well as options that annihilated some or most of the plutonium. For the INEL's rating, three areas of comparison were selected: (1) operational capabilities of the fission options, (2) time required to deploy reactor or accelerator, and (3) cost estimates based on sponsor-supplied information. A brief discussion of the first two areas follows. Insufficient valid cost data are available to develop sound estimates for future use. Further work is required to produce defendable and comparable cost estimates.

The methodology for rating operational capability of the options followed the rating methodology developed by LLNL. Four stages of operational capability were defined as:

- **Concept Feasibility (CF).** The physical principles associated with the concept are well understood and general feasibility has been established.

- **Engineering Feasibility (EF).** The engineering system, subsystems, and major components have been identified and their performance has been generally established. A design basis, including design basis accidents and preliminary system response to such events, has been developed.

- **At-Scale Operation (ASO).** A successful operation, which is larger than bench scale and smaller than full scale exists.

- **Presently-Existing Capability (PEC).** Similar, but not necessarily identical, systems are currently operating successfully.

Table S-2 summarizes the rating of options. The most significant change in this table from the LLNL report is in rating of engineering feasibility for the ALWR. In examining the sponsor's information, the INEL concluded the ternary fuel requires additional development before its engineering feasibility becomes definite. Ratings of ABC, MSR, and PBR were listed as partial because engineering feasibility has not been proven for all systems components.

Categorization by time to deployment is based on a system capable of annihilation of plutonium. Information from the LLNL report was used in categorization, but different durations were assigned. Table S-3 shows options falling into three distinct groups. The ALMR-R uses fuel currently developed and replaces blanket material. Results show that it would take approximately 144 years to annihilate 90% of the assumed 50 MT of plutonium. The ALMR-MB, ALWR-T, and MHTGR would fall within the same deployment period because each requires fuel development and testing for plutonium-based fuel. The ABC, MSR, and PBR require additional development and design efforts.

Recommendations

Because insufficient time and information was available to the INEL to perform detailed comparisons of each concept, the INEL recommends further study of the what it believes are the top four concepts—ALMR-R (Advanced Liquid Metal Reactor with reference fuel cycle), ALMR-MB (Advanced Liquid Metal Reactor with maximum burner fuel cycle), ALWR-T (Advanced Light Water Reactor with ternary fuel), and MHTGR (Modular High Temperature Gas-Cooled Reactor).

Table S-2. Operational capability of plutonium annihilation options.

	ALMR-R	ALMR-MB	ALWR-T	MHTGR	ABC	MSR	PBR
CF	Yes	Yes	Yes	Yes	Yes	Yes	Yes
EF	Yes	Probably[b]	Probably[c]	Probably[d]	Partial[d]	Partial[e,f]	Partial[e]
ASO	Probably[a]	No	No	No	No	No	No
PEC	No	No	No	No	No	No	No

a. Assuming Fuel Cycle Facility operation.
b. Based on experience with uranium plutonium-based fuel, recognizing the need for additional fuel development work.
c. Based on experience with uranium-based ternary fuel, recognizing the need for additional fuel and reactor kinetics development work.
d. Assuming review of Peach Bottom PuO_2 TRISO particle tests will validate fuel performance and acceptable plutonium core accident response can be established.
e. Requires large scale-up of MSR.
f. Design basis, including design basis accidents, has yet to be developed.

Table S-3. Ranking of plutonium annihilation options based on time to deployment.

Group	Concept rankings	Comments
I (5–10 yr)	ALMR-R	Moderate extension of current technology, but a slow annihilation option
II (10–20 yr)	ALMR-MB	Technical development of plutonium-based fuel required
	ALWR-T	Technical development of plutonium-based fuel required
	MHTGR	Technical development of plutonium-based fuel required
III (20–30 yr)	ABC	Extensive technical development of concept required
	MSR	Extensive technical development of concept required
	PBR	Extensive technical development of concept required

ACKNOWLEDGMENTS

Information on which this report is based was provided by several organizations. Copies of the original material may be obtained from the contributors listed below. The INEL reviewed this information, but did not engage in an extensive attempt to verify its technical accuracy. For this, they must credit the suppliers of the original material and their contributions are greatly appreciated.

For supplying the original source information and for responding to the INEL's request for additional information, the INEL gratefully acknowledges:

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- James Powell, Hans Ludewig, and Michael Todosow of Brookhaven National Laboratory for contributions related to the PBR option.
- Edward D. Arthur and J. Wiley Davidson of Los Alamos National Laboratory for contributions related to the ABC option.

CONTENTS

FIGURES

TABLES

ACRONYMS

ABC	Accelerator-Based Conversion System
ALMR	Advanced Liquid Metal Reactor
ALMR-MB	Advanced Liquid Metal Reactor Maximum Burner Cycle
ALMR-R	Advanced Liquid Metal Reactor Reference Cycle
ALWR	Advanced Light Water Reactor
ALWR-MOX	Advanced Light Water Reactor with Mixed Oxide Fuel
ALWR-T	Advanced Light Water Reactor with Ternary Fuel
ANL	Argonne National Laboratory
ATR	Advanced Test Reactor
CANDU	Canadian deuterium-uranium (reactor)
CISAC	Committee on International Security and Arms Control
DOE	Department of Energy
ERWM	Office of Environmental Restoration and Waste Management
GWD/MT	gigawatt day/metric ton
HFIR	High Flux Integral Reactor
ICPP	Idaho Chemical Processing Plant
INEL	Idaho National Engineering Laboratory
LANL	Los Alamos National Laboratory
LLNL	Lawrence Livermore National Laboratory
LWR	light water reactor
MHTGR	Modular High Temperature Gas-Cooled Reactor
MOX	mixed oxide
MSR	Molten Salt Reactor
MT	metric ton
MTHM	metric ton heavy metal
MW(e)	megawatt electrical power
MW(t)	megawatt thermal power
NAS	National Academy of Sciences
NRC	Nuclear Regulatory Commission
OCRWM	Office of Civilian Radioactive Waste Management
ORNL	Oak Ridge National Laboratory
PBR	Particle Bed Reactor
SNL	Sandia National Laboratory
SRS	Savannah River Site
TREAT	Transient Reactor Test Facility
WIPP	Waste Isolation Pilot Plant

Comparison of Plutonium Disposition Options

1. INTRODUCTION

Nuclear weapons in both the United States and Commonwealth of Independent States will be dismantled as a result of recent nuclear arms reduction agreements. Plutonium removed from these weapons must be dispositioned in a manner that will prevent future use for hostile purposes. A wide range of methods for denaturing or annihilating the published 50 metric tons (MT) processed by the United States have been proposed. The Secretary of Energy requested the National Academy of Sciences (NAS) Committee on International Security and Arms Control (CISAC) to evaluate disposition options.

The Idaho National Engineering Laboratory (INEL) staff is supporting the CISAC Reactor Panel by providing technical analysis in three specific areas:

1. Assistance in evaluation of proposals submitted by sponsors of reactor and accelerator-based concepts by providing a basis from which objective comparisons can be made.

2. Evaluation of the feasibility of using plutonium fuels (without uranium) for disposal in existing commercial light water reactors (LWRs).

3. Preconceptual analysis for a reactor specifically designed for destruction of weapons-grade plutonium.

This volume presents the results of the INEL's activities to date in the first area. Separate volumes address the other two areas.

The CISAC Reactor Panel indicated particular interest in an outline of bounding cases for the disposition of plutonium. For reactor and accelerator options, the following bounding cases were proposed:

- If policy dictates burning up as much of the plutonium as possible with a once-through cycle (avoiding the cost and complexity of reprocessing), what would be the best approach?

- If policy dictates annihilating the plutonium (to 99.9%), including reprocessing, what would be the best approach?

At a meeting in Berkeley, California, between the INEL and the CISAC Reactor Panel, plutonium annihilation levels ranging from 90% to 99% were discussed. An additional request was received for a calculation of 99.9% (or perhaps parametric curves of different percentages versus time). A final request was received to identify potential technical issues that could delay implementation of reactor or accelerator options. These two questions and requests form the bounding cases for this study.

To assist in better understanding the bounds for these cases and examining technical issues, INEL personnel developed a set of questions to obtain information from the option sponsors. Questions were asked regarding the status of (a) fuel development and fuel fabrication facilities, (b) reactor or accelerator development and capability to denature or annihilate plutonium, (c) waste processing development and capabilities, and (d) the disposal of waste. A full set of questions is presented in Appendix A.

Option sponsors expressed several divergent opinions on which plutonium isotopes should be annihilated and the isotopic composition that would make weapons-grade plutonium unsuitable for further use in weapons. Some indicated that it should only be necessary to annihilate ^{239}Pu while others indicated that all plutonium isotopes should be annihilated. Los Alamos National Laboratory (LANL) transmitted Figure 1 to demonstrate a wide range of plutonium isotopes and other actinides, or combinations thereof, can become critical

within reasonable mass limits. LANL recognized critical mass was not the only factor for consideration in design of a nuclear weapon, but it was a very important factor. To accommodate a range of opinions on burnup, three questions on annihilation were asked on the capability of the options for (a) 90, 95, and 99% annihilation of initial ^{239}Pu and ^{241}Pu for systems with no recycling, (b) 90, 95, 99, and 99.9% annihilation of initial ^{239}Pu and ^{241}Pu for systems with recycling, and (c) 90, 95, 99, and 99.9% annihilation of all the initial plutonium isotopes.

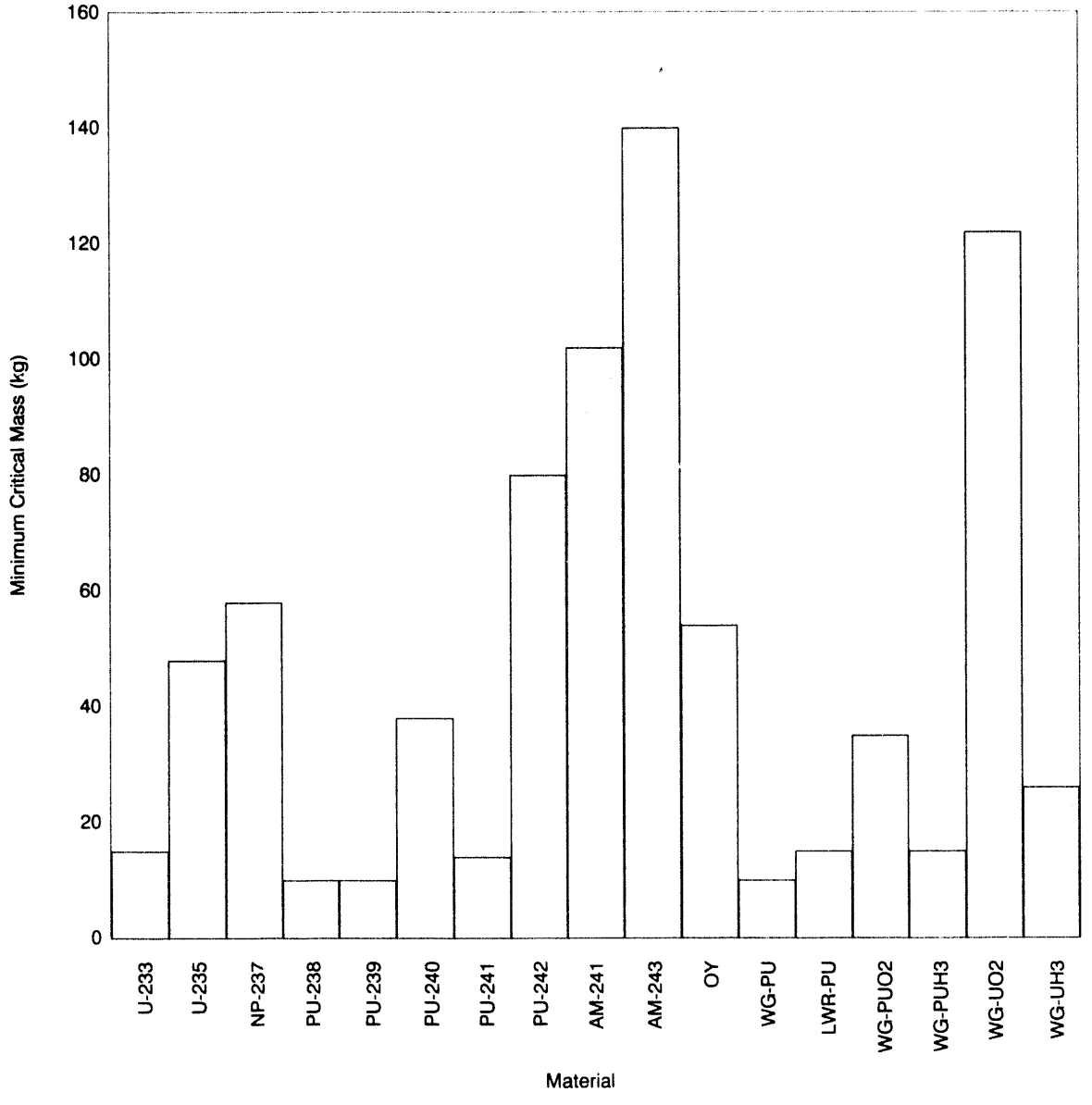

Figure 1. Minimum critical mass of actinide materials.

Sponsors proposed five different reactor types and one accelerator-based concept. The possibility of using different fuel types and fuel management schemes in several reactors increases the number of possible options to about 12. Responses did not discuss the use of commercial reactors and alternate fuels were only proposed for reactor types. Detailed information on the following eight options are presented (in alphabetical order according to their acronyms):

- Accelerator-Based Conversion System (ABC)
- Advanced Liquid Metal Reactor Maximum Burner Fuel Cycle (ALMR-MB)
- Advanced Liquid Metal Reactor Reference Fuel Cycle (ALMR-R)
- Advanced Light Water Reactor with Mixed Oxide Fuel (ALWR-MOX)
- Advanced Light Water Reactor with Ternary Fuel (ALWR-T)
- Modular High Temperature Gas-Cooled Reactor (MHTGR)
- Molten Salt Reactor (MSR)
- Particle Bed Reactor (PBR).

Contact was not established early enough to receive significant information on Canadian deuterium-uranium (CANDU) reactors. A letter from Atomic Energy of Canada, Limited, provided preliminary answers to the questions and is included in Appendix B. Because the CANDU uses natural uranium fuel, the INEL expects fuel development would include both development and irradiation efforts. It is not significantly different from the fuel development efforts for the other reactors. If the plutonium was mixed with the natural uranium, annihilation rates would be expected to be relatively low. Use of poisons in the fuel could possibly help achieve higher burnups by improving the kinetic properties (for example, Doppler coefficient) of the fuel. Online refueling offers an advantage in attaining high burnups, but reactivity control and neutron kinetics would have to be carefully examined to ensure stable, safe operation. Although the reactor technology is well developed and quite flexible in its capabilities, it is unclear that the CANDU has significant advantages over other reactor or accelerator concepts. Additional information on this concept may provide a better indication of its potential for plutonium destruction.

Responses developed by sponsors were evaluated by a team of INEL engineers, scientists, operators, and other personnel familiar with nuclear reactor types and issues to provide further insights on the applicability of each option. The remainder of this report summarizes information received from concept sponsors and presents the INEL's evaluation. A grouping of options is suggested based on capability to annihilate plutonium and current developmental status.

Conclusions and recommendations are followed by Appendices C–H providing sponsor-supplied information for each concept.

2. SPONSOR RESPONSES AND INEL EVALUATIONS

Sponsor responses to questions are presented in Appendices C–H and summarized for each question in this section. A discussion of INEL technical evaluations is presented when additional information is considered necessary for improved understanding of technical details. The presentation format first states the question, followed by INEL evaluation and a summary of sponsor responses.

Ratings on technological readiness from Lawrence Livermore National Laboratory (LLNL)[1] indicate that the ABC, MSR, and PBR are in early stages of development. Answers provided by sponsors of these options to some of the questions reflect this situation. The INEL believes there are large uncertainties in the remaining technical development required for these concepts and in cost and schedule estimates. In some cases, no attempt was made to further clarify the information supplied for particular options.

2.1 Fuel Status

Four questions were posed to the sponsors regarding fuel system requirements for fabrication of an acceptable fuel type. These questions related to quality of weapons-grade plutonium, requirements for additional technical development of fuel or fuel fabrication process, and cost estimates for development and implementation of the fuel system.

In general, development and fabrication of plutonium-bearing fuels will be the major hurdle for most options if policy dictates annihilation of between 90 and 99% of all plutonium isotopes within a 20- to 40-year period. Vital development work remains for plutonium fuels capable of the high burnup levels necessary and projections forecast completion within 5 to 10 years.

Sponsor estimates for fuel fabrication facility costs and schedules may be optimistic. The sponsors assumed experience with uranium oxide fuel could be directly extrapolated to plutonium-based fuel fabrication. The plutonium fuel facilities may have safety and environmental issues that will be complex and difficult to resolve. In addition, all fabrication and storage facilities will likely reside on a highly secure Department of Energy (DOE) reservation, imposing additional costs and schedule durations. A plutonium fuel production facility sufficiently large and secure is projected to cost more and take longer to design, build, and make operational than a uranium fuel facility. Operating costs and durations are expected to be greater than currently encountered for uranium-based fuel. However, operating costs are a small part of the overall costs of the plant. It is not clear that these differences are adequately considered in the sponsor estimates.

The next sections detail the questions on fuel development and fabrication and a summary of the sponsors' responses. Although the ALWR sponsor provided responses for the mixed oxide fuel (ALWR-MOX) option, both the INEL reviewers and the sponsor agree that this fuel is not well suited for annihilating large quantities of plutonium in reasonable time periods. Consequently, extensive comments are not provided on the ALWR-MOX option.

2.1.1 Fuel Feed Material. Question 1 was posed to the concept sponsors regarding the purity of the plutonium required for their fuel fabrication process:

> *1. Did you assume that plutonium (PuO_2 or Pu metal) used in the fuel would be free of contaminants (alloying metals and americium now in the nuclear weapon pits)? Will the fuel proposed be negatively impacted if plutonium is contaminated with these alloying metals and the americium?*

Table 1 summarizes the sponsor responses. Responses from two sponsors (ABC and ALMR) indicate fuel cycles could accept weapons-grade plutonium or oxide made directly from weapons-grade plutonium, without removal of contaminants. The MSR response indicates contaminants may not affect the fuel but should be examined.

Table 1. Sponsor responses on influence of contaminants in the plutonium on the fuel.

Concept	Response
ABC	No specific sponsor response was provided for this question.
ALMR	Plutonium with alloying metals and americium is acceptable for the metal fuel used. Plutonium cleanup can be accomplished in the electro-refiner, if needed.
ALWR-MOX	Alloying metals and americium would be removed from the plutonium before fuel fabrication.
ALWR-T	Although not specifically stated, the response on MOX fuel is assumed to apply.
MHTGR	Alloying metals and americium would be removed from the plutonium before fuel fabrication.
MSR	Alloying metals would not be expected to have a significant negative effect on the fuel, but they should be examined. Americium would be transmuted or taken out in the waste stream.
PBR	The process proposed for preparing fuel is relatively insensitive to contaminants. More information on the contaminants would be necessary to answer the question.

The INEL's evaluation indicates contaminants should be identified and an assessment made to determine effects on concepts using liquid fuels (ABC and MSR). Depending on the products of the chemical reactions between contaminants and salts, acceptable levels of contaminants may need to be established and fuel stability assessed.

Capability and availability of facilities for plutonium processing will affect availability and capacity of some options to produce fuel. Two sponsors (ALWR-MOX and MHTGR) show an early need for a fuel facility with capabilities to remove contaminants and other actinides from the weapons-grade plutonium and produce high grade plutonium oxide or metal.

2.1.2 Technical Development Requirements. Question 2 was posed to concept sponsors regarding the technical work requirements for completion of fuel development:

2. *Briefly describe the technical work scope necessary to complete development of the fuel and its estimated duration.*

Table 2 summarizes the sponsor responses to this question. The following is a discussion of the INEL evaluation of the sponsor responses. These evaluations are presented alphabetically by concept acronym.

Advanced Light Water Reactor—Ternary (ALWR-T). Response for ternary fuel indicates the effects of plutonium on Doppler feedback and the possibility of separation of plutonium oxide from other fuel constituents during severe accidents are identified for further investigation.

The INEL's staff believes additional developmental work will also be necessary. Using the reference given in the sponsor response, ternary fuels based on UO_2-ZrO_2-CaO have been used in several reactor applications including Shippingport, Power Burst Facility, and the light water breeder reactor. The addition of CaO and ZrO_2 lowers the melting point of UO_2. Thermal conductivity of the UO_2-ZrO_2-CaO system is about half that of UO_2. ZrO_2 undergoes a monoclinic to tetragonal trans-

Table 2. Sponsor responses on technical work scope for development of the fuel and its duration.

Concept	Response
ABC	No specific sponsor response was provided for this question.
ALMR	Both the standard fuel and the fuel for the moderate burner (0.6 conversion ratio) are considered developed. The maximum burner fuel (0.02 conversion ratio) requires some development work for fuel fabrication and performance testing. Preliminary evaluation of this fuel indicates no undue strain on the fuel cycle.
ALWR-MOX	There is essentially no development needed based on experience in Europe and Japan.
ALWR-T	It is fully developed with apparently no additional work required. It has been manufactured by Westinghouse Advanced Energy Systems Division and evaluated in U.S. and foreign programs.
MHTGR	Required development activities for plutonium fuel will directly parallel those defined for the NP-MHTGR highly enriched uranium fuel development program. Eighteen months would be added to the NP-MHTGR fuel development program. (See Appendix F for details.)
MSR	The fuel development constitutes the dissolving the metal with fluoride into a molten salt. There are various chemical possibilities. All of them do not require much development. The duration is not significant.
PBR	A process similar to that demonstrated for uranium would have a high probability of success. Three steps are outlined in Appendix H. The duration is estimated to be about five years with sufficient funding.

formation at about 1000°C. The addition of CaO at greater than 15 m/% stabilizes a face-centered cubic structure that is compatible with the UO_2 face-centered cubic structure. These effects emphasize the importance of knowing the thermal, mechanical, and physical properties of these alloys as a function of temperature and composition, particularly stoichiometry. No direct experience was found for PuO_2-ZrO_2-CaO fuels in the sponsor reference.

The addition of CaO and ZrO_2 lowers the melting point of UO_2. The magnitude is composition dependent, but an estimate for UO_2-ZrO_2-CaO is about 170°C lower than that of UO_2. The melting point of PuO_2 is 2425°C, 415°C lower than that of UO_2. Phase equilibria in PuO_2-ZrO_2-CaO cannot simply be extrapolated from UO_2-ZrO_2-CaO fuel. Melting points as a function of temperature (determination of phase equilibria) would have to be found in order to establish operating margins for the plutonium-based ternary fuel. Phase equilibria have been treated as a pseudo-ternary diagram, but actually the phase diagram is a four component system (Pu, Ca, Zr, and O) in which stoichiometry plays an important part in the phase equilibria, thermal properties, and mechanical properties.

Thermal conductivity of the UO_2-ZrO_2-CaO system is about half that of UO_2. Assuming that the PuO_2-ZrO_2-CaO system behaves similarly for establishing a trend which may not be a valid assumption, low thermal conductivity causes a higher fuel temperature than that for UO_2 for the same power level and heat transfer conditions. Because of lower melting points and higher fuel temperatures, the operating power level and hence plutonium burnup rate may have to be downgraded. This effect emphasizes the importance of knowing the thermal, mechanical, and physical properties of these alloys as a function of temperature and composition, particularly stoichiometry. The latter is why extrapolation from a UO_2 system

to a PuO_2 system is not warranted, because the stoichiometry will be different at operating power. Furthermore, removal of uranium may lead to a positive temperature coefficient. This issue also needs to be investigated.

The sponsor indicates reprocessing as unnecessary because fuel irradiation to 843 GWD/MT Pu would reduce total plutonium content approximately 93.4%. Based on INEL calculations, this average fuel burnup is comparable to current maximum burnup limits for commercial LWR fuel assemblies. However, the time that fuel assemblies are in the reactor to achieve these burnups (i.e., resident time) is a factor of two to four times greater than for current reactor experience (10–14 years versus 3–5 years). During normal operation, fretting of the fuel rod and stresses on spacer grids have a strong influence on fuel bundle lifetime. These phenomena are coolant- and flow-induced effects rather than neutronic. The INEL concludes that fuel assembly lifetime for long reactor resident times will be dictated by performance of assembly materials. Without some form of fuel assembly refurbishment, the irradiation period could be limited to something shorter than needed to achieve the desired levels of plutonium annihilation. Additional development and testing is needed to examine this issue.

Modular High Temperature Gas-Cooled Reactor (MHTGR). An accurate mechanistic fuel particle model, which can predict fuel failures, remains a hinderance in designing high performance fuel. Current fuel models do not include knowledge that would be obtained from separate-effects tests for both in- and out-of-pile experiments.

The incremental time period of 18 months reported by the sponsor to develop a plutonium TRISO fuel is believed to be optimistic by the INEL. A separate-effects and integral testing program should be pursued in parallel, so additional knowledge can be gained on component models and incorporated into models predicting integral test performance.

Particle Bed Reactor (PBR). Fuel technical issues focus on fuel particle integrity, fuel element integrity, and power/flow matching. Each of these broad issues will be briefly discussed and additional details are provided in Appendix I. Resolution of these issues would have to be accomplished through engineering design and fabrication and material process development.

Fuel Particle Integrity—Fuel tested to date is a uranium-based BISO fuel consisting of a uranium-carbide kernel, a porous graphite layer, a dense graphite seal coating, and a zirconium carbide outer coating. TRISO-coated particles can be employed if the silicone carbide layer of this fuel is replaced by a zirconium carbide layer.

Another issue for PBR fuel is the chemical compatibility of the fuel, coatings, fission products, and coolant with each other. For plutonium burning, diffusion may occur along the particle temperature gradient. Superimposed on the temperature gradient within a particle is the temperature gradient arising across the fuel bed annulus. This perturbation has not been evaluated or tested yet and the magnitude of gradient across the bed could be greater than that within an individual particle.

Fuel particles and cold and hot frit materials need to be chemically compatible for high temperatures and long durations. In previous nuclear testing, either temperatures became high enough to melt the hot frit or the thermal interaction between the fuel and the hot frit material caused the hot frit to liquify.

Retention of fission products under high temperature conditions leads to significant fuel swelling, on the order of 20 to 30%, for uranium-based fuel. This amount of swelling could easily lead to extensive fuel failure. Similar behavior would be expected in plutonium-based fuels.

Impurities in plutonium may effect development of plutonium carbide fuel fabrication processes and quality of the spherical particles produced. Extrap-

olation from uranium fuels fabrication technology to plutonium is not warranted.

Fuel Element Integrity—During thermal cycling the fuel particle bed in the annular core becomes very closely packed (locked) unless there is some provision to prevent it. The hot frit will expand more than the cold frit, placing the particle bed in compression and inducing bed lockup. This bed lockup constrains the hot frit from expanding further axially. Additional temperature rise after the yield point is reached will cause plastic deformation of the hot frit. As a result the hot frit becomes shorter with cycling, which could lead to fracture and loss of the fuel pellets. The number of cycles required for failure would depend on the mechanical properties of the material selected for the hot frit. Nuclear testing of two PBR fuel elements resulted in hot frit shortening and numerous cracks in one element. Good engineering design, appropriate material selection, and fabrication development may alleviate this problem, but currently this effect may limit the number of times the PBR fuel elements could be thermally cycled.

Power-Flow Matching—The cold frit will be fabricated from metal filters that contain about 30% interlinked porosity and pores 5 to 10 microns in diameter. The flow passages in the particle fuel bed may be only slightly larger, depending on the fuel particle size and its distribution. The thermal-hydraulic flow stability in these very small capillaries has been questioned and local flow instabilities may propagate to adjacent regions.

Analytical modeling of flow instability is extremely difficult, if not impossible, because of truncation and limits in numerical analysis schemes handling differential equations. Analysis of two particle bed experiments was performed and results were inconclusive because some coefficients in the analytical model need to be determined experimentally. Experimental measurements found that some values were 2.5 times higher than that predicted theoretically.

Experimental problems such as plugging of the cold frits with graphite were experienced. The Commonwealth of Independent States has tested a PBR fuel element and concluded that the operating temperature should be limited to 2300°C and low power levels on the order of kilowatt or fractions of kilowatts per liter because of flow instabilities. Engineering designs for a different fuel form to achieve the high surface area to volume ratios indicate the possibility of extensive fuel development for the PBR.

Question 3 was posed to the concept sponsors regarding any technical issues which could impede completion of fuel development:

3. *Identify technical issues that could impede fuel development and fabrication. For example, have all issues related to material lifetime, compatibility, etc., been resolved?*

Table 3 summarizes the sponsor responses to this question. The following is a discussion of the INEL evaluation of the sponsor responses. These responses are presented alphabetically by concept acronym.

Advanced Light Water Reactor—Ternary (ALWR-T). Additional development would be necessary for the high burnup cases for ternary fuel. As discussed in the previous question, the maximum reactor residence time of the current ALWR fuel assembly design will likely impede ternary fuel development.

Modular High Temperature Gas-Cooled Reactor (MHTGR). Post-irradiation examination and fuel process optimization studies are expected to provide the technical basis for determining required changes in fuel design and/or fuel process conditions. An open issue that remains is the possible volatilization of the fuel kernel in the coating process and the associative effects on the kernel. A review of the coating process is under way as part of MHTGR activities.

Table 3. Sponsor responses on technical issues that could impede fuel development and fabrication.

Concept	Response
ABC	No specific sponsor response was provided for this question.
ALMR	Fuel-processing development is ongoing and demonstration is scheduled for completion in 1996 by Argonne National Laboratory (ANL). Testing of material to its lifetime is ongoing and the design codes have been updated to examine fuel lifetime and compatibility.
ALWR-MOX	All issues have been resolved because MOX fuel is currently being used in other countries to produce electrical power.
ALWR-T	The sponsor did not address this question.
MHTGR	Seven issues are discussed ranging from design of high burnup plutonium fuel particles under peak core conditions to integral inpile testing. (See Appendix F for details.)
MSR	Fluid fuel reactors have no fuel fabrication. The feasibility of the fuel processing is considered resolved. Associated material problems are considered resolved in principle. Development work is necessary to establish full-scale operating processing.
PBR	Having not manufactured any plutonium-based fuel, it is essentially impossible to identify any outstanding issue that could impede fuel development. At this point, none are envisioned. Cost and environmental issues are currently seen as the primary impediments to fuel development.

The highly enriched uranium TRISO-coated particles tested in postirradiation heating tests referenced by the sponsor were fabricated with a different kernel than used in the MHTGR fuel. The type of kernel used in the particles tested would not be desirable for plutonium fuel particles. Drawing conclusions from this test for other fuel types can be misleading. In addition, the MHTGR program only dealt with highly enriched uranium that did not require glovebox handling, while the plutonium fuel development program will require glovebox handling.

Particle Bed Reactor (PBR). The PBR will require new or substantial modifications to existing facilities to test the fuel under prototypical conditions at 5 MW/l. Existing test reactors, such as the Advanced Test Reactor (ATR) at the INEL or the High Flux Integral Reactor (HFIR) at Oak Ridge National Laboratory (ORNL), may be able to test PBR elements up to 3 to 4 MW/l, but not 5 MW/l. This type of testing could verify design and manufacturing processes to resolve the technical feasibility issues for the PBR concept. The feasibility of looking at a closed loop supporting 5 MW/l would have to be evaluated.

Previous nuclear testing of a PBR element was done in the Annular Core Research Reactor at Sandia National Laboratory (SNL). This is a transient reactor that is limited in both power level and test time durations. The previous tests have been done in terms of seconds and not for the long times required for plutonium burning. The Transient Reactor Test Facility (TREAT) had been proposed for similar testing, but again being a transient reactor, the testing times are limited to less than 20 seconds.

The Air Force is currently considering a small test facility to test PBR elements for nuclear propulsion applications. This facility would test several elements as an open Brayton cycle engine using hydrogen. The final Environmental Impact Statement is nearing approval. A Record of Decision is expected in mid-June 1993 that will assist in defining whether this program will continue, whether the test facility will be built, and if it is built, which site will be selected. For applications

to plutonium burning, the design of this facility would have to be changed to accommodate a closed loop for plutonium fuel.

Current facilities for fabricating PBR fuel are based on uranium. Essentially no facilities exist for fabricating plutonium carbide fuels by the gelation process. LANL had been developing a cryochemical process to develop uranium-based carbide fuels, but LANL's plutonium facility would have to be modified to develop this process for plutonium-based carbide fuels.

2.1.3 Cost Estimates. Question 4 was posed to the concept sponsors regarding cost estimates for fuel deployment:

> *4. What are the current cost estimates for fuel development and for the fuel fabrication facility construction, startup, and operation? What estimating method was used (e.g., parametric, historical cost, unit cost).*

Table 4 summarizes the sponsor responses. The following is a discussion of the INEL evaluation of the sponsor responses.

The sponsor cost estimate responses used the historical cost estimating method. With a known large variance in data vintage and accuracy, these responses should be considered only as rough estimates rather than factual. As stated in the introductory paragraphs of this section, cost estimates for fuel development may be low. The INEL recommends a separate parametric cost analysis be completed to establish criteria. This cost analysis should be applied to the top three or four options, providing a firm cost baseline for future use.

Table 4. Sponsor responses on current cost estimates for fuel development and fabrication.

Concept	Response
ABC	No specific sponsor response was provided for this question.
ALMR	Current construction cost estimates for a fuel cycle facility for one plant [1,500 MW(e)] is $120 million. Estimated operating cost is $25 million per year, including hardware and staff. The estimating basis is historical data with upgrading of developed equipment for the batch-based pyroprocessing.
ALWR-MOX	Cost estimates are included for a Greenfield facility and for using existing facilities. The estimate for the Greenfield facility is $680 million. Costs were estimated using the Freiman Analysis of System Technique (FAST) program. (See Appendix E for details.)
ALWR-T	The sponsor did not address this question.
MHTGR	Plutonium fuel development costs are estimated at $261 million. This is an increment cost beyond commercial fuel development costs (about $50 million). Fuel fabrication facility construction costs are estimated at $260 million for a four-module plant. Operating costs are estimated at about $19 million per year. The basis for costs are given in Appendix F.
MSR	No current cost estimate available for fuel processing. In economical evaluations it must be considered that no large-scale, fuel-processing facility is necessary.
PBR	At this point in time, no reasonable answers can be given to this question.

Advanced Light Water Reactor—Ternary (ALWR-T). Although the sponsor did not identify any fuel development costs for the ternary fuel concept, these costs could be significantly higher than those given for the MOX fuel concept because of assembly burnup limitations.

2.2 Reactor or Accelerator-Based Systems

Six questions were posed to the sponsors regarding reactor or accelerator-based system requirements for annihilating weapons-grade plutonium, requirements for additional technology development, and cost estimates for development and implementation. The following subsections report sponsor responses to these questions and the INEL's evaluation of the sponsor responses.

2.2.1 Annihilation of Weapons-Grade Plutonium. Questions 1–3 were posed to the sponsors regarding concept capability to annihilate weapons-grade plutonium and the plutonium isotopic weight percent at the start and end of an equilibrium fuel cycle:

1. *If plutonium disposition policy is to annihilate ^{239}Pu and ^{241}Pu in a single fuel cycle, what is the total burnup or exposure (GWD/MT Pu) required to reduce the initial inventory of these two isotopes by 90%, 95%, and 99% (if possible)? For each of these cases, identify the weight percent of all plutonium isotopes in the initial fuel loading and those remaining in the spent fuel after an equilibrium fuel cycle. Also identify the cycle times.*

2. *If plutonium disposition policy is to annihilate ^{239}Pu and ^{241}Pu and irradiated fuel can be reprocessed to recover and recycle plutonium, what is the total burnup or exposure (GWD/MT Pu) required to reduce the initial ^{239}Pu and ^{241}Pu inventory by 90%, 95%, 99%, and 99.9%? If the option uses a batch mode fuel cycle, how many times would a core fuel load need to be recycled to reach each burnup percentage? For each of the four cases, identify the weight percent of all plutonium isotopes prior to initiation of irradiation and those remaining in the spent fuel.*

3. *If plutonium disposition policy is to annihilate all plutonium isotopes and irradiated fuel can be reprocessed to recover and recycle plutonium, what is the total burnup or exposure (GWD/MT Pu) required to reduce the inventory of all plutonium isotopes by 90%, 95%, 99%, and 99.9%? If the option uses a batch mode fuel cycle, how many times would a core fuel load need to be recycled to reach each burnup percentage? For each of the four cases, identify the weight percent of all plutonium isotopes remaining in the spent fuel.*

Because the sponsor responses to these three questions varied and did not always directly answer these questions, a concept-by-concept comparison was impractical and usually meaningless. Question 1 was not answered for the ALMR concept, because recycling is an inherent part of its process and without it annihilation of the plutonium by the amounts indicated by the question would be impractical if not impossible. Questions 2 and 3 were not relevant to the ALWR-T concept because this process assumes no reprocessing of the ternary fuel. Although reprocessing of fuel particles was not part of the original proposals for the MHTGR and PBR concepts, the sponsors did respond with reprocessing schemes in order to achieve annihilation percentages greater than 90% for the MHTGR and 99.7% for the PBR. The MSR concept did not answer any of the questions because, in the sponsor's words, "This question is not applicable to MSRs, there is no burnup in an MSR."

In addition, all concepts should denature the total plutonium at an equivalent rate when normalized to density of thermal power production (i.e., normalized to GWD/MT). Individual plutonium isotopics will vary at a given point because of different destruction and production rates of isotopes due to differences in the neutron spectrum of each concept. The overall plutonium destruction rate measured per GWD/MT Pu should be equivalent, however.

An alternative method for comparing denature and/or annihilation capabilities of these options was therefore proposed. Using the sponsor proposals, yearly estimates were made of the remaining weapons-grade plutonium, the plutonium inventory in the reactor, the amount of plutonium fissioned, and plutonium inventory either residing in spent fuel or in reprocessing. These calculations were performed assuming the weapons-grade plutonium was the only available fuel source for operating the reactors and the reactors operate at a 75% capacity factor. The number of reactors for each option used in these calculations was chosen to obtain an equivalent thermal capacity for each option. From the results of these calculations, the years to annihilate 90%, 95%, 99%, and 99.9% of the initial 50 MT of weapons-grade plutonium was obtained. The following sections present the results for each option and the facility characteristics and assumptions used in obtaining these results.

Before each option is discussed, however, an example of the plutonium inventories for a reactor without spent fuel reprocessing is presented in Figure 2. This figure does not represent an actual reactor option and is presented for clarification of the remaining figures in this section. The four plutonium inventories discussed above are presented in this figure. The first inventory is the remaining weapons-grade plutonium. This inventory appears as a downward stairstep representing the decrease in weapons-grade plutonium at the time of each reactor refueling. The second inventory is the amount of plutonium residing in the reactor. This inventory has a sawtooth shape representing the plutonium depletion during cycle operation and plutonium addition during refueling. The third inventory, plutonium burnup, appears as a steadily rising slope, which represents the steady annihilation of plutonium in the reactor. The fourth, and last, inventory is of the plutonium mass remaining in the discharged spent fuel. This inventory appears as a rising stairstep representing the increase in spent fuel at the time of reactor refueling and spent fuel discharge. If a reactor concept includes reprocessing of the spent fuel, this last inventory would be replaced with an inventory representing the amount of plutonium in reprocessing. The concepts are presented alphabetically by their acronym.

Accelerator-Based Conversion (ABC) System. ABC uses a high energy proton accelerator to generate an intense neutron source driving two subcritical reactor assemblies. Each assembly is rated at 2,130 MW(t) for a total rating of 4,260 MW(t). The stated ABC net electrical capacity is 1,040 MW(e).

ABC circulates a molten salt slurry containing plutonium through the subcritical reactor. A portion of the slurry is continuously withdrawn from circulation and processed to remove fission products. The slurry is then returned to the system for additional irradiation. Although the sponsor reports each subcritical reactor has a steady-state plutonium inventory of approximately 80 kg, no information is provided on the reprocessing rate. If a 75% capacity factor is assumed for reactor operation (instead of the 78.8% capacity factor used in the sponsor proposal), each subcritical reactor would fission 595 kg of plutonium per year.

Because insufficient data were available for estimating reprocessing inventories, only the plutonium depletion inventory is calculated. This information is presented in Figure 3. Complete annihilation of the weapons-grade plutonium can be accomplished over a 42-year period.

Advanced Liquid Metal Reactor (ALMR). The ALMR facility considered in this analysis is based on nine ALMR modules. Each module is rated at 471 MW(t) with a total facility rating of 4,239 MW(t). The total electrical capacity of this facility would be 1,440 MW(e).

The information presented in the original ALMR proposal and the sponsor's response to additional information requests indicate that two fuel cycles should be considered for comparison. The first fuel cycle is the reference cycle or moderate burner. This fuel cycle is 48 months long and has a conversion ratio of 0.61. The proposed fuel is metallic U-Pu-Zr in stainless steel cladding. Initial core loading for the reference fuel cycle is 6.4 MTHM, including 1.12 MT of plutonium and 5.28 MT of uranium. Refueling intervals are conducted every 12 months and one-fourth of the fuel elements is replaced and reprocessed at that time. Each core reload requires the addition of 38.8 kg of plutonium in makeup fuel during reprocessing.

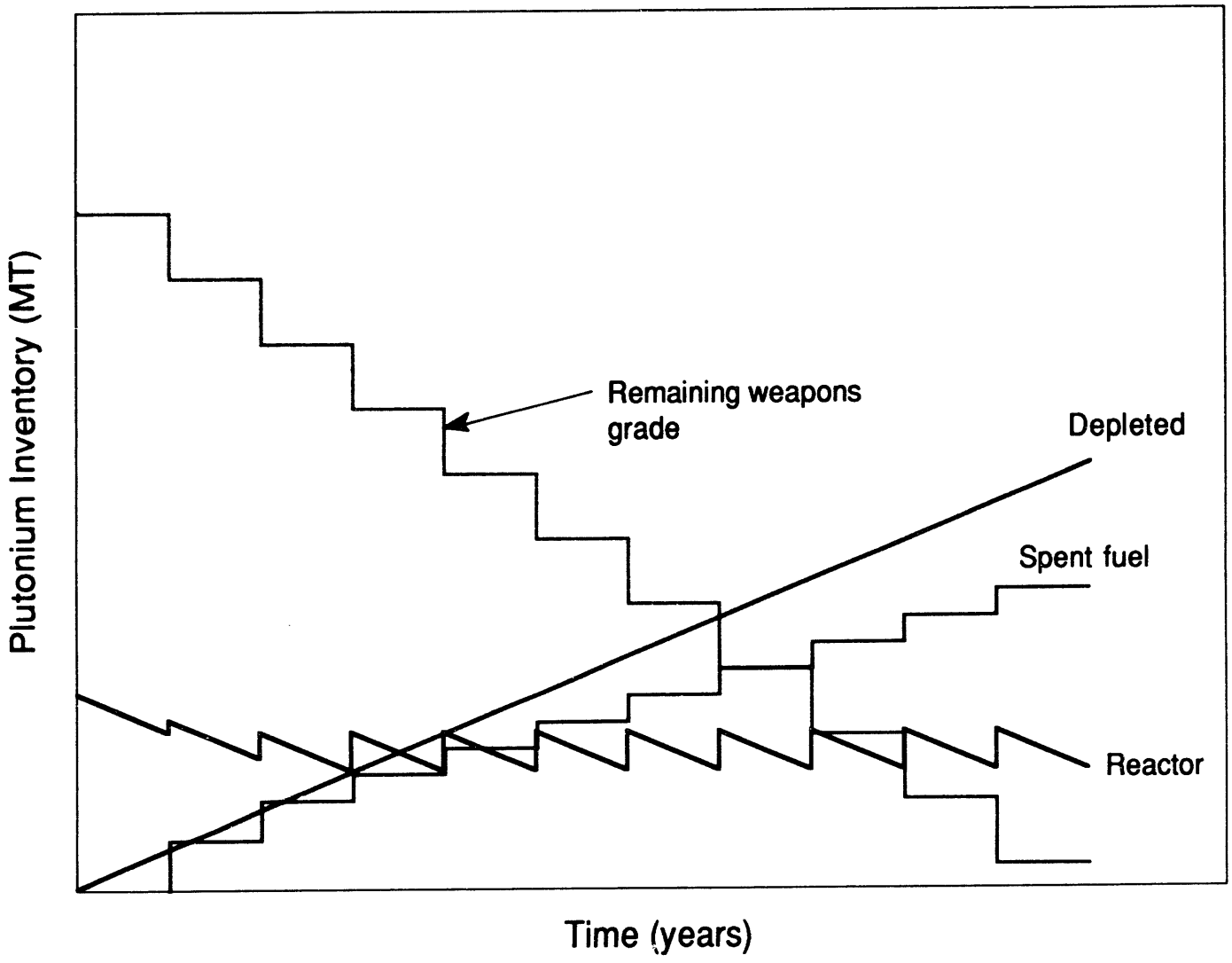

Figure 2. Example of plutonium inventories for a reactor with no reprocessing.

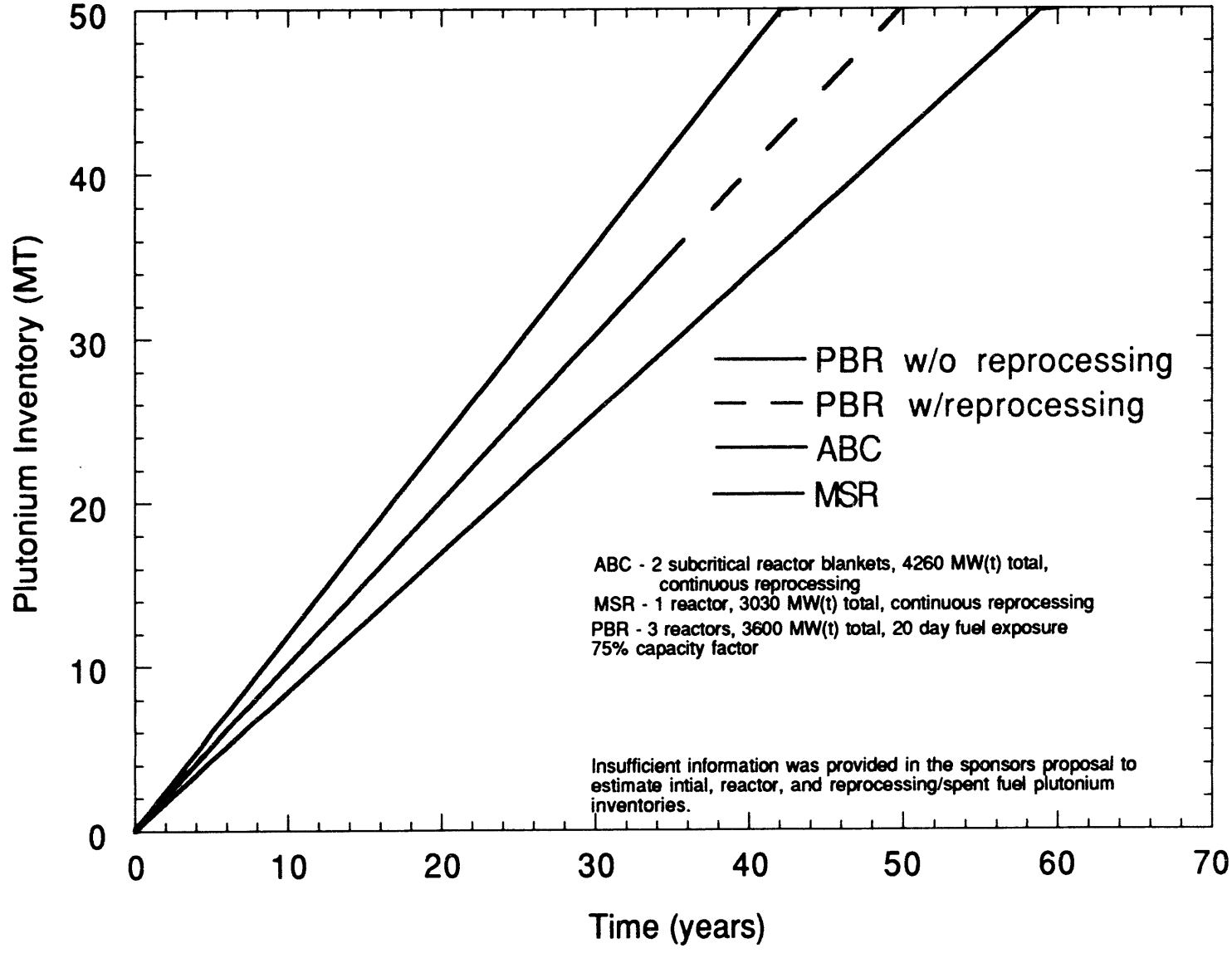

Figure 3. Plutonium depletion inventory for ABC, MSR, and PBR.

If a 75% capacity factor is assumed for reactor operation (instead of the 83% capacity factor used in the sponsor proposal), this concept would fission 34.7 kg of plutonium per year per module. Plutonium inventories for this fuel cycle with and without reprocessing are presented in Figures 4 and 5, respectively. The initial core inventory is 15.5 MT for this option. This includes one full core plus two reload batches, which are placed in storage positions above the reactor core for preconditioning.

If only the 50 MT of weapons-grade plutonium is available for operating the reactors, it becomes necessary after 106 years to begin decreasing the number of operating modules because insufficient plutonium inventory remains to refuel all modules. This decrease in operating modules continues until only one ALMR module remains. Because the plutonium inventory in the final core cannot be completely annihilated, the maximum degree of annihilation for this concept is 96.8% over a 283-year period.

If other fuel sources are available to maintain operation of all nine ALMR modules, complete annihilation of the weapons-grade plutonium can be accomplished in 160 years. In the reference fuel cycle with no reprocessing, only 10.5% of the weapons-grade plutonium will be annihilated over a 17-year period.

The second fuel cycle is referred to by the sponsor as the maximum burner fuel cycle. This fuel cycle is 24 months long and has a conversion ratio of 0.02. The proposed fuel is metallic Pu-Zr with a Zr-Hf sheath in stainless steel cladding. Initial core loading for the reference fuel cycle is 1.023 MT Pu. Refueling intervals are conducted every six months and one-fourth of the fuel elements is replaced and reprocessed at that time. Each core reload requires the addition of 147 kg of plutonium in makeup fuel during reprocessing.

If a 75% capacity factor is assumed for reactor operation (instead of the 83% capacity factor used in the sponsor proposal), this concept would fission 132 kg of plutonium per year per module. Plutonium inventories for this fuel cycle with and without reprocessing are presented in Figures 6 and 7, respectively. The initial core inventory is 19.5 MT for this option. This includes one full core plus four reload batches, which are placed in storage positions above the reactor core for preconditioning.

If only the 50 MT of weapons-grade plutonium is available, it becomes necessary after 26 years to begin decreasing the number of operating modules because insufficient plutonium inventory remains to refuel all modules. This decrease in operating modules continues until only one ALMR module remains. Because the plutonium inventory in the final core cannot be completely annihilated, the maximum degree of annihilation for this concept is 96.6% over a 77-year period. If other fuel sources are available to maintain operation of all nine ALMR modules, complete annihilation of the weapons-grade plutonium can be accomplished in 42 years. In the maximum burner fuel cycle with no reprocessing, only 23.6% of the weapons-grade plutonium will be annihilated over a 10-year period.

Advanced Light Water Reactor (ALWR). The ALWR facility considered in this analysis is based on two units. Each reactor is rated at 1,818 MW(t) with a total facility rating of 3,636 MW(t). The total electrical capacity of this facility would be 1,200 MW(e).

The information presented in the original ALWR proposal indicated that a MOX fuel cycle was the fuel cycle of choice. However, in the sponsor's response to the request for additional information, it was indicated that ternary fuel cycle is now the preferred fuel. Because of this sponsor comment and because of MOX fuel's capacity for breeding additional plutonium and the reprocessing methods required to purify the irradiated fuel, the MOX fuel cycle was not considered for comparison by the reviewers.

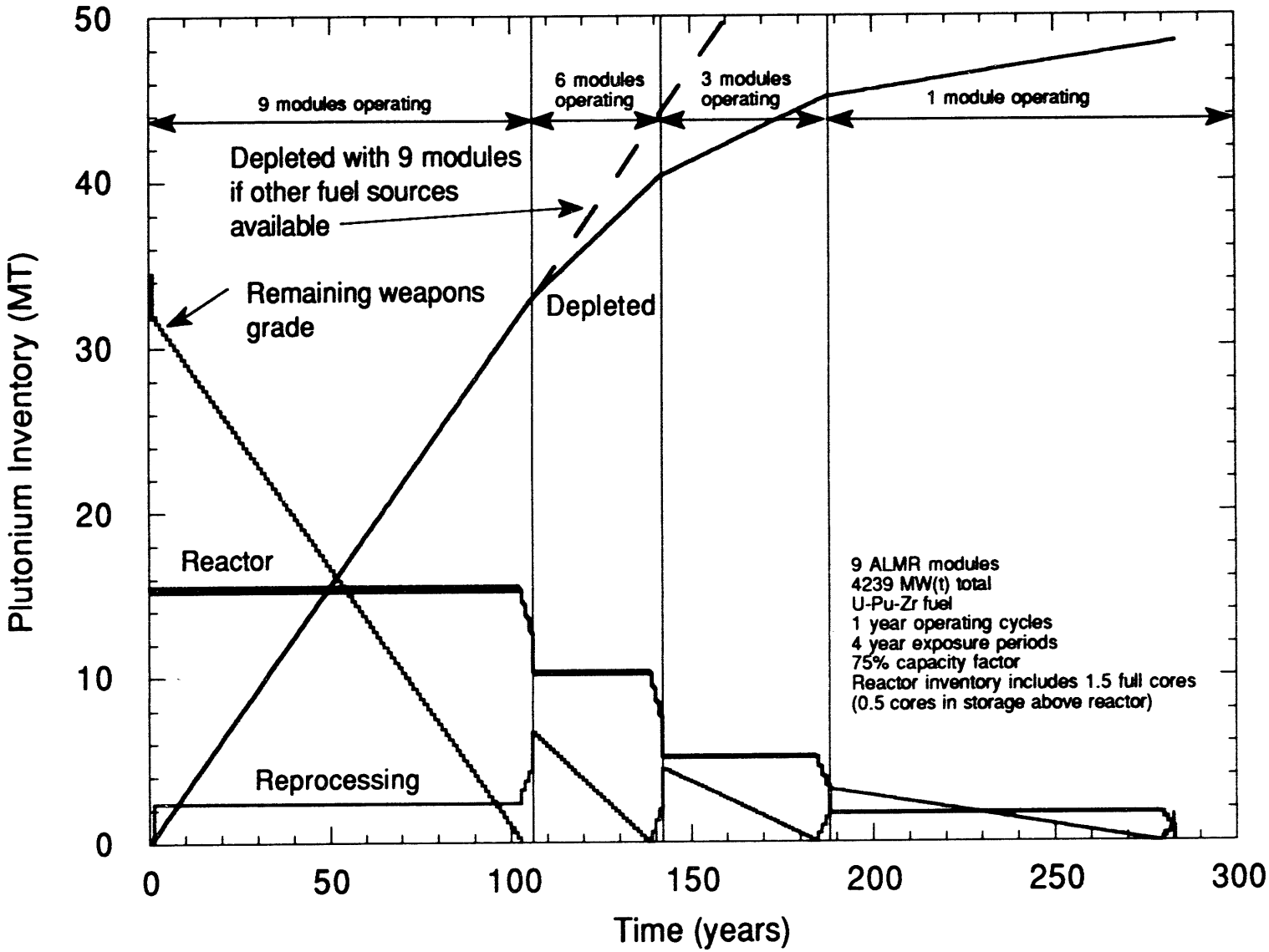

Figure 4. ALMR plutonium inventories for a reference fuel cycle with reprocessing.

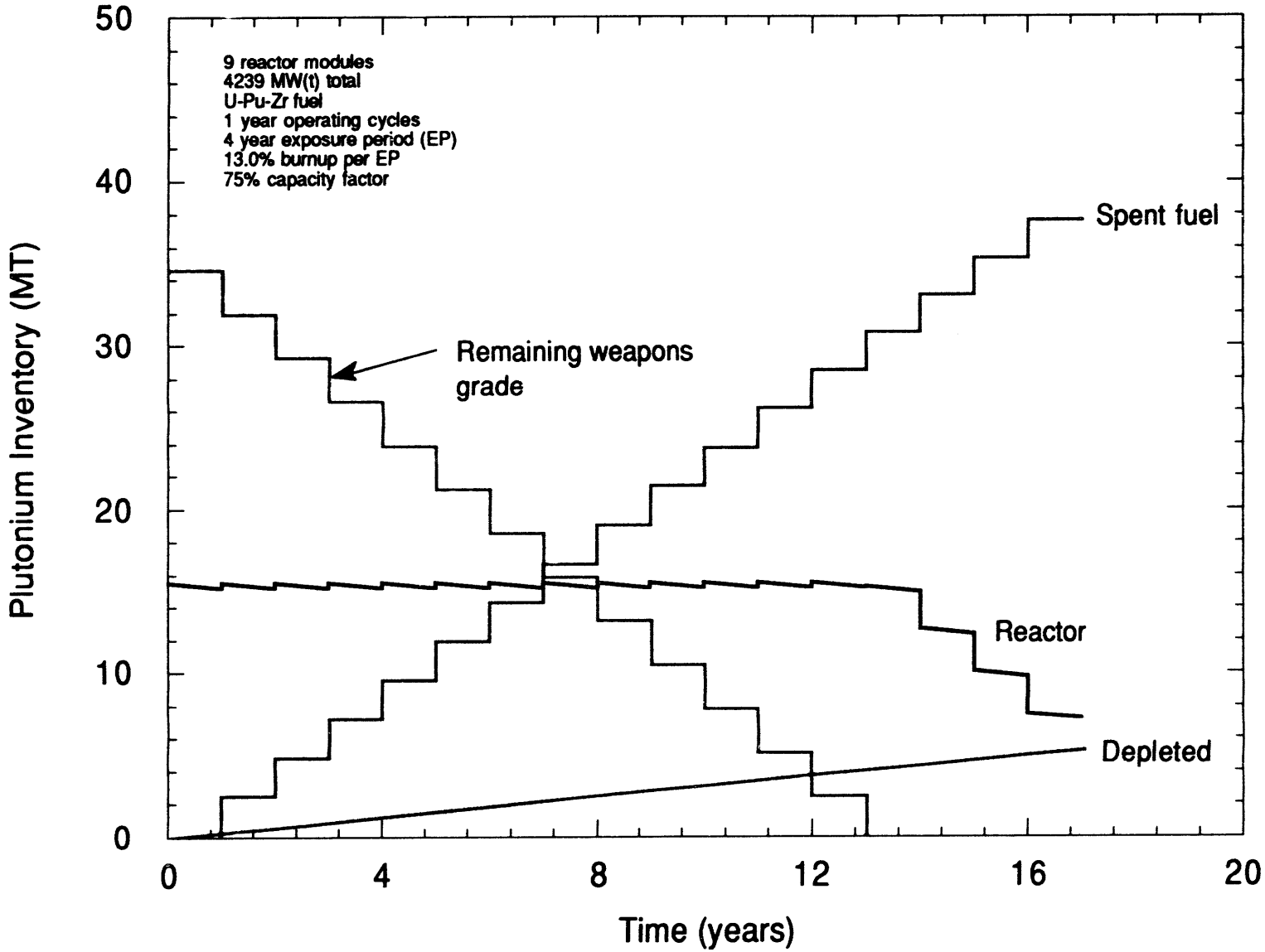

Figure 5. ALMR plutonium inventories for a reference fuel cycle with no reprocessing.

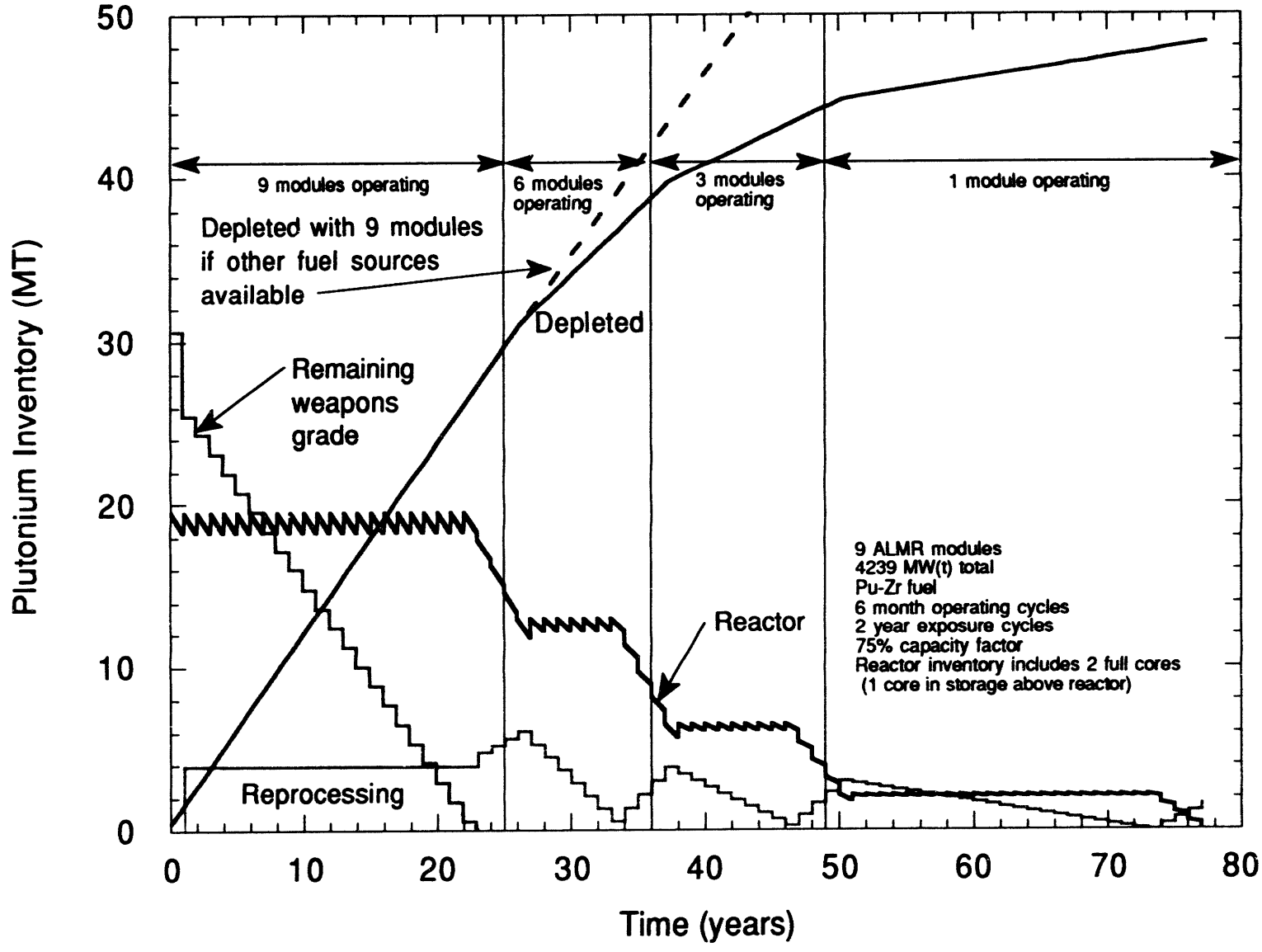

Figure 6. ALMR plutonium inventories for a maximum burner fuel cycle with reprocessing.

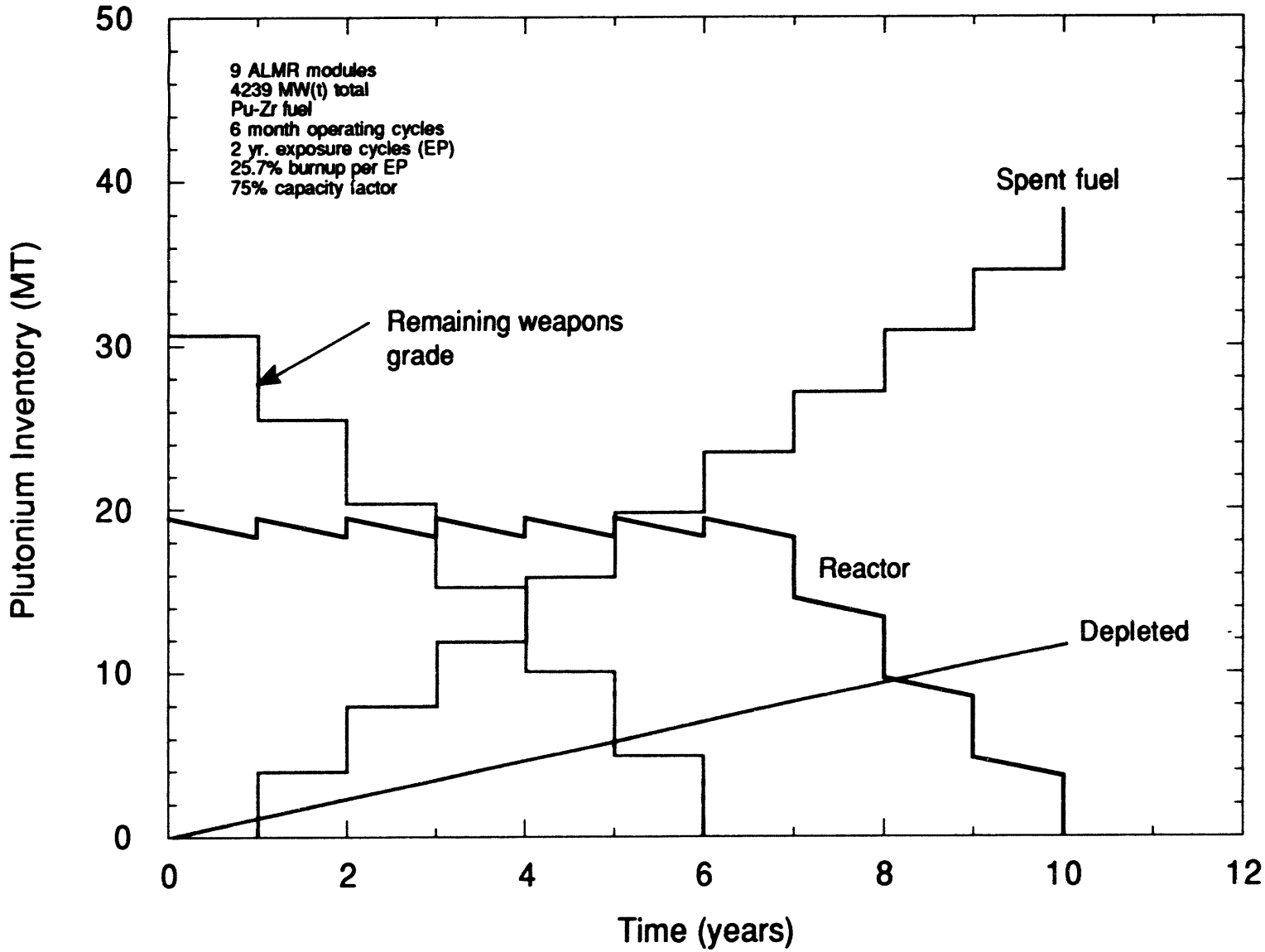

Figure 7. ALMR plutonium inventories for a maximum burner fuel cycle with no reprocessing.

The ternary fuel consists of solid or annular pellets of PuO_2, ZrO_2, and CaO. The composition of this fuel is assumed to be 14 wt% PUO_2, 77 wt% ZrO_2, and 9 wt% CaO. This fuel, as proposed, will undergo prolonged exposure to achieve 90% plutonium annihilation. The sponsor proposes a fuel lifetime of up to five cycles, or 11.5 years. The initial core loading for this fuel cycle is 4.14 MT Pu. Refueling intervals are conducted every 2.3 years when one-fifth of the fuel elements is replaced. Each core reload requires 828 kg of plutonium. At the end of a five-cycle exposure history, average fuel burnup is 822.3 GWD/MT Pu. To achieve 95% annihilation, the sponsor proposes a one-sixth core refueling every 2.3 years for a fuel element reactor resident time of 13.8 years. As stated in Section 2.1.2, it is doubtful that fuel assembly materials could survive fuel cell erosion for such long reactor resident times. Further investigation would be required to determine the feasibility of these fuel management schemes.

If a one-third core refueling scheme is employed and a 75% capacity factor is assumed for reactor operation (instead of the 66% capacity factor used in the sponsor proposal), each reactor would fission 778.8 kg of plutonium per cycle. The initial core inventory for each reactor is 3.115 MT with an additional 1.038 MT required every reload. Operating cycle length is 2.3 years and fuel exposure time is 6.9 years. Using this refueling scheme, a 75% plutonium annihilation is achieved over the fuel exposure period. No reprocessing information has been provided by the sponsor for the ALWR ternary fuel cycle.

The plutonium inventories for this fuel cycle are shown in Figure 8. Because of the plutonium inventory in the final core cannot be completely annihilated, the maximum degree of annihilation for this concept is 65.4% over a 32.2-year period.

Modular High Temperature Gas-Cooled Reactor (MHTGR). The MHTGR facility considered in this analysis is based on nine MHTGR modules. Each module is rated at 450 MW(t) with a total facility rating of 4,050 MW(t). The total electrical capacity of this facility would be 1,557 MW(e). Each MHTGR core is comprised of 840 hexagonal graphite fuel elements containing TRISO-coated fuel particles.

The information presented in the MHTGR proposal indicated that a single, once-through, two-year fuel exposure period would achieve 63% annihilation of all plutonium isotopes. In response to Questions 1 and 2, the sponsor has proposed shuffling one-half of the core into the reflector region after its two-year fuel exposure period for an additional year. According to the sponsor, this fuel scheme could achieve 90% annihilation of ^{239}Pu and ^{241}Pu and 73% annihilation of all plutonium isotopes. Annihilation fractions greater than this cannot be achieved without reprocessing the fuel particles.

Assuming a 75% capacity factor for reactor operations (rather than the 78% assumed in the sponsor proposal), this concept would fission 125.7 kg per year per module. Each module's initial core inventory is 344 kg and each reload would require and additional 172 kg. At the end of each fuel element's three-year reactor resident time, the average fuel burnup is 677 GWD/MT Pu. The plutonium inventories for this fuel cycle are shown in Figure 9. From this figure it is seen that over a 31-year period, 70.2% of the weapons-grade plutonium will be annihilated.

In response to Question 3, the sponsor has proposed a reprocessing/recycling fuel scheme to achieve 90% annihilation of all plutonium isotopes. Under this fuel scheme, of the 420 fuel elements normally replaced after a single, two-year exposure period, 336 fuel elements would be removed and the fuel particles reprocessed; 63 blocks would remain in the reactor for an additional two-year exposure period before they are removed and the fuel particles reprocessed; and the remaining 21 fuel elements would remain in the reactor for a total of six years before they are removed. Fuel particles exposed for six years would not be reprocessed. A total of 336 fresh fuel elements are inserted during each reload. However, the INEL believes the technical work required to develop a method for reprocessing the TRISO-coated fuel pellets will likely cause delays in the implementation of this concept.

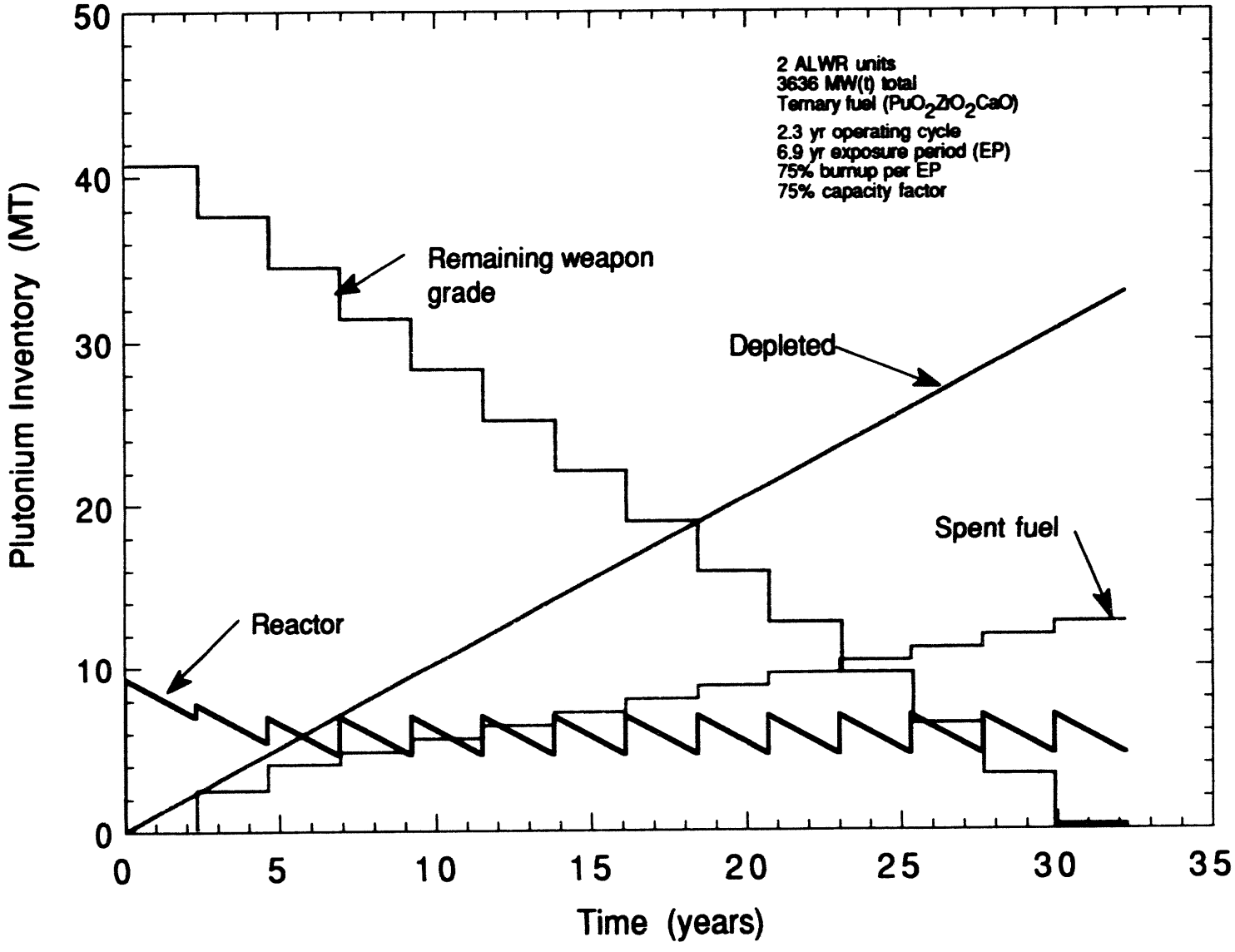

Figure 8. ALWR plutonium inventories for a ternary fuel cycle with no reprocessing.

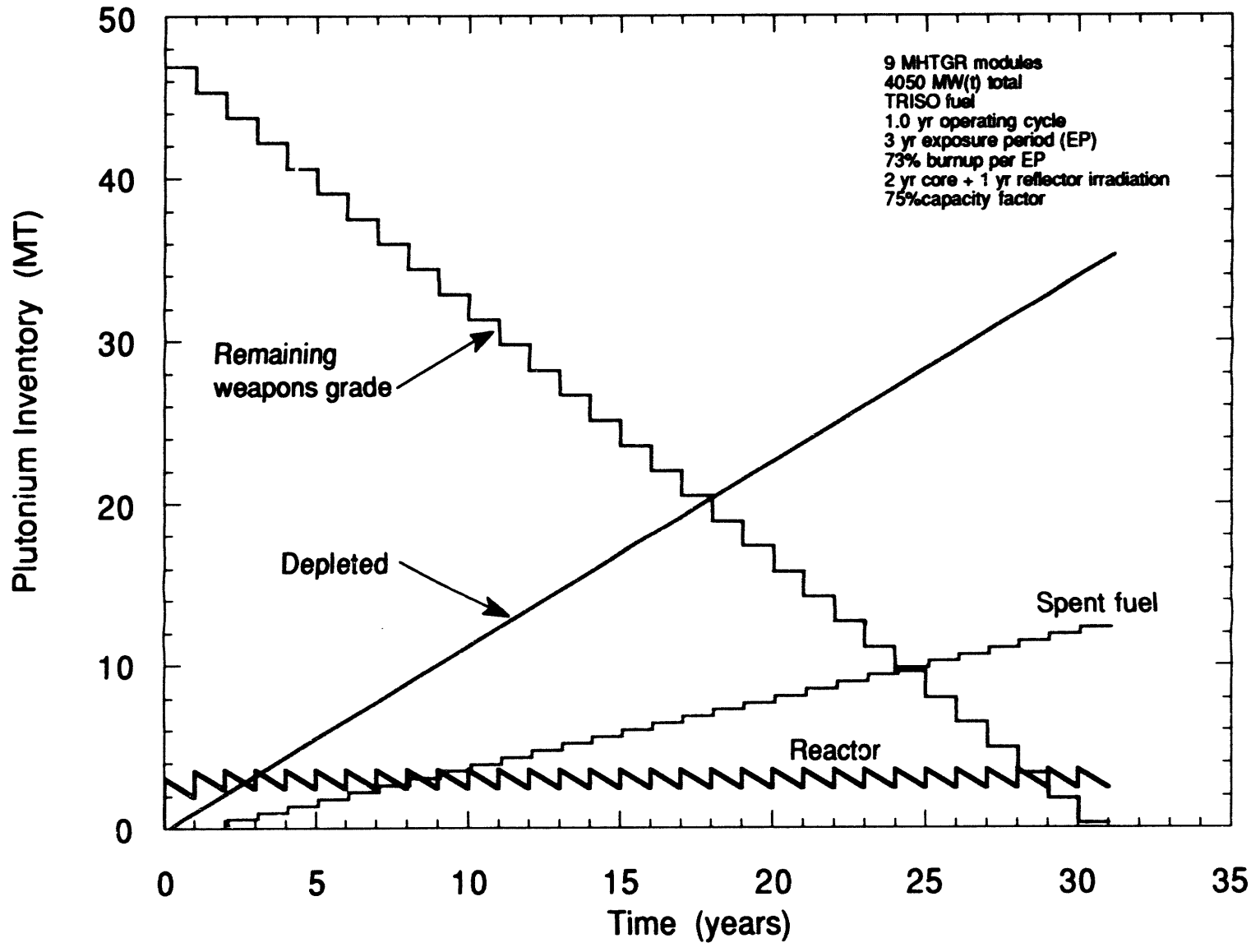

Figure 9. MHTGR plutonium inventories for a once-through fuel cycle with no reprocessing.

Assuming a 75% capacity factor for reactor operations (rather than the 78% assumed in the sponsor proposal), this reprocessing fuel scheme would fission 125.7 kg per year per module. Each module's initial core inventory is 368.8 kg and each reload would require an additional 125.7 kg. At the end of a six-year reactor resident time, the average fuel burnup is 813 GWD/MT Pu. The plutonium inventories for this fuel cycle are shown in Figure 10. From this figure it is seen that over a 41-year period, 92.8% of the weapons-grade plutonium will be annihilated. Because the sponsor does not propose reprocessing all the fuel particles, the maximum annihilation achievable for this fuel cycle is 99% over a 43.8-year period. However, this level of annihilation can only be achieved if other fuel sources are available.

Molten Salt Reactor (MSR). An MSR of approximately 1 GW(e) was projected in the sponsor proposal for plutonium disposition. Assuming a 33% efficiency rating, electrical capacity corresponds to 3,030 MW(t). At this thermal capacity and assuming operation at a 75% capacity factor, this concept would fission 846 kg of plutonium per year.

Because insufficient data were provided by the sponsor to adequately estimate the reactor and reprocessing inventories, only the plutonium depletion inventory will be presented. This curve is given in Figure 3 with the ABC depletion inventory.

Particle Bed Reactor (PBR). The PBR facility considered in this analysis is based on three PBR reactors. Each reactor is rated at 1,200 MW(t) with a total facility rating of 3,600 MW(t). The total electrical capacity of this facility would be 1,200 MW(e). This facility differs substantially from the one used for comparison of these concepts in the LLNL report. In this report, each PBR was rated at 1,905 MW(t) with an electrical capacity of 629 MW(e).

The PBR fuel cycle as discussed in the sponsor's proposals is 20 days long. The proposed fuel is BISO-coated fuel particles, which are used in the Air Force space propulsion program and similar to that used in HTGRs. The pellets will be contained in 127 porous hexagonal shaped annular fuel elements. Coolant will either flow from a central region, through the particle bed, and exit through the fuel element outer wall, or via the reverse flow path. Refueling is conducted every 1 to 1.5 weeks at which time one-third of the fuel particles is replaced. Each year, an additional 532 kg of plutonium is required to fuel each reactor (Reference 1 reports the larger PBR reactor as consuming 700 kg of plutonium per reactor per year). Total fuel particle lifetime is 20 days with an average fuel burnup of 500 GWD/MT Pu. Both once-through and reprocessing after one exposure cycle are considered for comparison purposes.

In response to Question 3, the sponsor implies a burnup of 500 GWD/MT Pu results in total plutonium inventory reduction of 72%. This is incompatible with an expected value of approximately 53%. Preliminary calculations at the INEL for a variety of neutron spectrums indicate little variation in the destruction rate of plutonium isotopics. Further verification and validation of sponsor data will be required.

If a 75% capacity factor is assumed for reactor operation (instead of the 96% capacity factor assumed in this proposal), this concept would fission only 335 kg of plutonium per year per reactor. Insufficient data were provided by the sponsor to adequately estimate the reactor and reprocessing inventories. Therefore, only the plutonium depletion inventory, with and without reprocessing, will be presented. This curve is given in Figure 3 with the ABC and MSR depletion inventories.

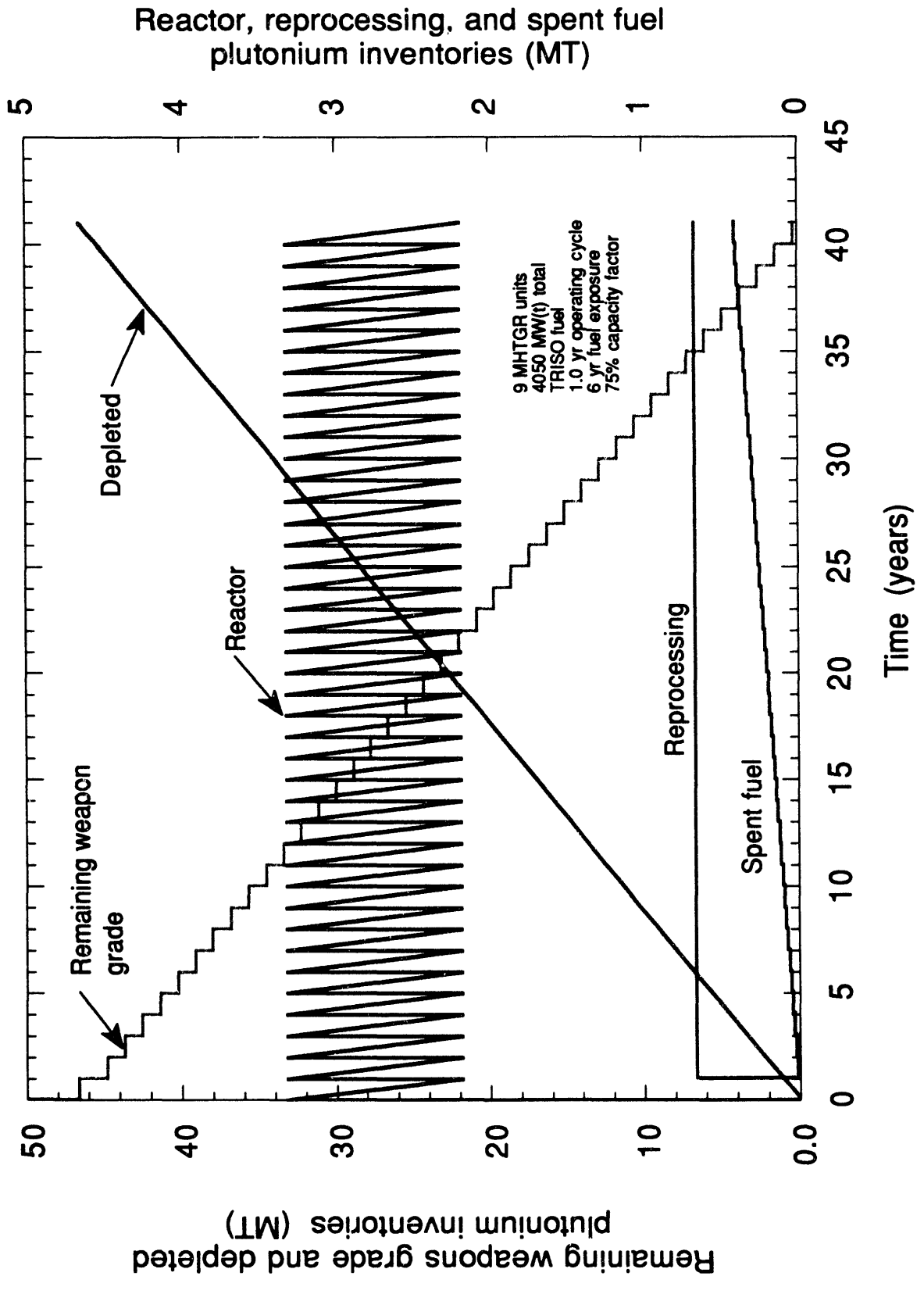

Figure 10. MHTGR plutonium inventories with reprocessing.

Table 5. Time required to annihilate various fractions of 47 MT ^{239}Pu and ^{241}Pu for reactor and accelerator systems, assuming no reprocessing of the irradiated fuel.

Concept	MW(t)	Annihilation time (years)[a]			
		90%	95%	99%	99.9%
ABC	4,260	NA	NA	NA	NA
ALMR-R[b]	4,239	U	U	U	U
ALMR-MB[c]	4,239	U	U	U	U
ALWR-T[d]	3,636	U	U	U	U
MHTGR[e]	4,050	32.3	U	U	U
MSR	3,030	NA	NA	NA	NA
PBR[f]	3,600	U	U	U	U

NA—Not applicable to this concept.

U—This level of annihilation is unachievable without reprocessing of the fuel.

a. 75% capacity factor assumed for each concept.

b. ALMR reference fuel cycle moderate burner, with 0.61 conversion ratio. The maximum achievable annihilation is 10.5% over 17 years for a once-through fuel cycle.

c. ALMR maximum burner, with 0.02 conversion ratio. The maximum achievable annihilation is 23.6% over 10 years for a once-through fuel cycle.

d. ALWR ternary fuel with one-third core reload fuel managment scheme. The maximum achievable annihilation level is 75% over 32 years for a once-through fuel cycle.

e. MHTGR 2-year, in-core plus 1-year, in-reflector fuel cycle.

f. The maximum achievable annihilation is 88% over 37 years for a once-through fuel cycle.

A summary of the times required to achieve the annihilation fractions requested in Questions 1 through 3 is presented in Tables 5 through 7, respectively. The information presented in these tables was obtained from the calculations supporting Figures 3 through 10. In Table 7, annihilation times were calculated assuming other fuel sources are available to maintain reactor operation after all of the initial weapons-grade plutonium has been inserted into the reactor.

2.2.2 Technical Development Requirements. Questions 4 and 5 were posed to sponsors regarding any technical development required to complete the concept:

4. *Briefly describe the technical work scope necessary to complete development of your reactor or accelerator system and its estimated duration.*

5. *Identify technical issues that could impede system development, design, construction, and startup. For example, have all issues related to material lifetime, compatibility, etc., been resolved?*

Tables 8 and 9 summarize the sponsor responses to these two questions, respectively. The following is a discussion of the INEL evaluation of sponsor responses.

Table 6. Time required to annihilate various fractions of 50 MT Pu for reactor and accelerator systems, assuming no reprocessing of the irradiated fuel.

Concept	MW(t)	Annihilation time (years)[a]			
		90%	95%	99%	99.9%
ABC	4,260	NA	NA	NA	NA
ALMR-R[b]	4,239	U	U	U	U
ALMR-MB[c]	4,239	U	U	U	U
ALWR-T[d]	3,636	U	U	U	U
MHTGR[e]	4,050	U	U	U	U
MSR	3,030	NA	NA	NA	NA
PBR[f]	3,600	U	U	U	U

IDP—Insufficient data provided by sponsor.

NA—Not applicable to this concept.

U—This level of annihilation is unachievable without reprocessing of the fuel.

a. 75% capacity factor assumed for each concept.

b. ALMR reference fuel cycle moderate burner, with 0.61 conversion ratio. The maximum achievable annihilation level is 10.5% over 17 years for a once-through fuel cycle.

c. ALMR maximum burner, with 0.02 conversion ratio. The maximum achievable annihilation level is 23.6% over 10 years for a once-through fuel cycle.

d. ALWR ternary fuel with one-third core reload fuel managment scheme. The maximum achievable annihilation level is 65% over 32 years for a once-through fuel cycle.

e. MHTGR 2-year, in-core plus 1-year, in-reflector fuel cycle. The maximum achievable annihilation is 73% over 32.3 years for a once-through fuel cycle.

f. The maximum achievable annihilation is 74% over 37 years for a once-through fuel cycle.

Advanced Light Water Reactor (ALWR). No sponsor response was provided on the ternary fuel concept, and it is unknown what the operation characteristics of such a reactor may be. The proposed cycle length for the ternary fuel concept is given as 2.3 years. Cycle length is dictated by the ability to control reactivity swings from the beginning to the end of the cycle and the core enrichment. Higher enrichment limits will allow longer cycle lengths but reactivity control becomes a greater problem. Additional study will be required to determine whether sufficient reactivity control is provided and if a one-fifth core refueling scheme would allow for 2.3-year cycles.

Of particular concern is whether an analysis of the prompt thermal feedback coefficients has been performed. It is possible the proposed fuel might have either a positive or only a marginally negative coefficient, especially if only boron is used for control purposes. This statement is based on preliminary calculations at the INEL using PuO_2-ZrO_2 fuel. The presence of calcium may help improve the thermal feedback behavior, but this needs to be verified with additional calculations.

Table 7. Time required to annihilate various fractions of 50 MT Pu for reactor and accelerator systems, assuming reprocessing of the irradiated fuel.

Concept	MW(t)	Annihilation time (years)[a]			
		90%	95%	99%	99.9%
ABC	4,260	37.8	39.9	41.6	42.0
ALMR-R[b]	4,239	144.1	152.1	158.5	159.9
ALMR-MB[c]	4,239	37.9	40.0	41.7	42.1
ALWR-T	3,636	NR	NR	NR	NR
MHTGR	4,050	39.8	42.0	43.8	U
MSR	3,030	53.2	56.1	58.5	59.0
PBR	3,600	44.8	47.3	49.3	49.7

U—This level of annihilation is unachievable.
NR—No reprocessing proposed.

a. 75% capacity factor assumed for each concept.

b. ALMR reference fuel cycle with 0.61 conversion ratio.

c. ALMR maximum burner cycle with 0.02 conversion ratio.

d. ALWR ternary fuel with one-third reload fuel management scheme.

Because no sponsor response was provided on the ternary fuel concept, it is unknown what technical issues may impede the development of this concept. Fuel assembly burnup limitations, criticality concerns, and reactivity control throughout 2.3-year cycles may increase technical development time for this concept.

Modular High Temperature Gas-Cooled Reactor (MHTGR). Other than the fuel development area, little discussion has been transmitted for the MHTGR detailing the stability and controllability of the all-plutonium core containing burnable poisons. The all-plutonium core must maintain a negative temperature coefficient throughout its cycle life for a 90% burnup once-through cycle. INEL reviewers did not perform a core physics analysis of this concept to verify feasibility. This effort should be pursued as a followup to this study.

Achieving annihilation levels greater than 63% would require shuffling previously irradiated fuel blocks into the reflector for an additional 1-year period. This would achieve only a 73% annihilation level. Further investigation would be required to determine the feasibility and economic practicality of such a fuel management scheme.

The sponsor has assumed reprocessing TRISO-coated fuel particles to achieve an annihilation level of 90%. However, the sponsor has not identified a method for reprocessing these fuel particles nor has this been identified as a technical development requirements. The time required to develop a reprocessing method that is economically feasible in a full-scale facility will likely impede concept implementation time.

Table 8. Sponsor responses on technical work scope for development of the system and its duration.

Concept	Response
ABC	No specific sponsor response was provided for this question.
ALMR	Fundamental technical questions have been resolved for the reactor. The design concept is well advanced and the essential safety approach and design features are completing acceptance in principle by the NRC.
ALWR-MOX	There has been extensive work done for the ALWR. The Advanced Reactor Corporation has funding to perform first-of-a-kind engineering by 1996. The NRC is expected to provide design certification by 1996.
ALWR-T	The sponsor did not address this question.
MHTGR	Five technical areas that pertain to a plutonium-fueled MHTGR are identified including fuel development, thermal-hydraulics development, reactor physics development, structural materials development, and component test development. Use of plutonium fuel would add 18 months to system development.
MSR	A proof of principle program based on completed development and with restricted processing could be available in five years. A full development program without about 12 steps is discussed in Appendix G. A lower-end estimate for time may be 10 years and a higher-end estimate may be 30 years.
PBR	A five-phase development program is discussed, including feasibility studies and identification of go/no go critical issues, preliminary system design and component development, engineering design and component validation, prototype construction and operation, and construction of full-scale plants. The development effort through completion of prototype demonstration is estimated to take 12–13 years.

2.2.3 Cost Estimates. Question 6 was posed to the sponsors relating to cost estimates for implementing their concept.

6. *What are the current cost estimates for system development and for construction, startup, and operation of the facility? What estimating method was used (e.g., parametric, historical cost, unit cost)?*

Table 10 summarizes the sponsor responses to this question. Most sponsor cost estimate responses used the historical cost estimating method. With a known large variance in data vintage and accuracy, these responses should be considered only as rough estimates rather than factual. The INEL recommends a separate parametric cost analysis be completed to establish criteria. This cost analysis should be applied to the top three or four options, providing a firm cost baseline for future use.

2.3 Waste Processing

Four questions were posed to sponsors regarding processing of radioactive waste streams. The first two questions were directed at technical work required to complete development of the waste processing system. The remaining two questions related to schedule and cost estimates associated with completing waste processing facilities. The following sections list sponsor responses to these questions.

Table 9. Sponsor responses on technical issues that could impede system development.

Concept	Response
ABC	No specific sponsor response was provided for this question.
ALMR	In-place technology programs need to be completed. Certain component detailed designs and proof testing need to be completed. Demonstration plant design, construction, and testing needs to be completed.
ALWR-MOX	If fuel loading is based on one-third MOX, no design changes would be required because this fuel would be within current safety analysis envelopes. If a full MOX core is used, control rod changes may be necessary and sensitivity studies would be needed to determine control rod worth requirements. Three control rod options are discussed in Appendix D. Aside from these changes, there would be no redesign necessary to accommodate a full MOX core.
ALWR-T	The sponsor did not address this question.
MHTGR	The critical path issue would be development and qualification of plutonium-based fuel. Fabrication processes must be adapted and the fuel must be qualified by irradiation testing.
MSR	At this time there are no known feasibility questions remaining. For the thermal option, solutions to limited graphite lifetime would have to be selected. Several solutions to material problems were accomplished near the end of the MSR program and have not been demonstrated. Fuel processing has been tied to laboratory-scaled tests only. A system integration and demonstration is needed. New remote and robotics technology needs to be adapted and demonstrated.
PBR	Two fuel development issues need to be addressed: (1) confirmation of fuel particle loading and unloading by hydraulic means and (2) fluid dynamics, heat transfer, and material compatibility tests. A wide range of testing will be needed up to testing of a prototype in a reactor.

2.3.1 Technical Development Requirements. The following two questions were supplied to concept sponsors regarding waste-processing development, facility design and construction, and startup. For these questions, waste is defined as spent fuel and byproduct waste streams from recycling spent fuel.

1. *Briefly describe the technical work scope necessary to complete development of a waste conditioning/processing flowsheet for your option and its estimated duration.*

2. *Identify technical issues that could impede development of the waste processing flowsheet, systems, or facilities.*

Tables 11 and 12 summarize the sponsor response to these questions. No INEL evaluation of these responses was made.

2.3.2 Schedule and Cost Estimates. The following responses were supplied by concept sponsors for cost and schedule estimates for waste processing. For these questions, waste is defined as spent fuel and byproduct waste streams from recycling spent fuel.

3. *What are the current estimates for the time required for construction and startup of the waste-processing facility?*

4. *What are the current cost estimates for waste-processing development and for construction and startup of a waste-processing facility?*

Table 10. Sponsor responses on current cost estimates for system development, construction, startup, and operation.

Concept	Response
ABC	No specific sponsor response was provided to this question.
ALMR	Estimated plant and fuel research and development costs are $325 million. Design costs are estimated to be $400 million. Construction costs are estimated to be $3.6 billion for a prototype plant and a nine-module plant. Operation and maintenance costs for the reactor are estimated to be about $110 million per year. The estimating method is historical adjusted for factory fabrication.
ALWR-MOX	First-of-a-kind engineering costs are being subsidized by the Advanced Reactor Corporation and the reactor vendors will bear the remainder of the development costs. Capital costs estimated for one 600 MW(e) ALWR and a MOX fuel fabrication plant is in the range of $1.5 to $2 billion. For three 600 MW(e) ALWRs and a MOX fuel fabrication plant, capital costs are estimated at $4 to $4.5 billion. Annual operating costs are estimated to be $110 million for one plant and $250 million for the three-reactor scheme.
ALWR-T	The sponsor did not address this question.
MHTGR	First-of-a-kind overnight plant capital costs for four modules are about $1.5 billion and for eight modules it is about $2.9 billion. Operating costs including fuel costs and decontamination and decommissioning costs for the reactor and fuel fabrication facility are about $120 million per year for four modules and about $220 million per year for eight. Costs were estimated by combinations of parametric, unit, and historical.
MSR	There are no current cost estimates for MSRs. A recent publication suggested they were within 5% of LWRs. There are economic advantages to the MSR because it closes the fuel cycle.
PBR	The estimates apply to three reactors and a fuel fabrication facility each at two government sites. The capital costs for two sites is estimated to be $11.5 billion. The operating cost for two sites is estimated to be $100 million per year.

Tables 13 and 14 summarize the sponsor response for these questions. The sponsor cost estimate responses used the historical cost estimating method. With a known large variance in data vintage and accuracy, these responses should be considered as rough estimates rather than factual. The INEL recommends a separate parametric cost analysis be completed to establish criteria. This cost analysis should be applied to the top three or four options, providing a firm cost baseline for future use.

2.4 Waste Disposal

Six questions were posed to concept sponsors regarding waste disposal technical development requirements, time and cost estimates for waste disposal system implementation, and self-protecting capacity of the final waste form. Before providing the sponsors' responses to these questions, the INEL reviewers viewed it necessary to provide background information and observation on plutonium waste disposal issues.

Table 11. Sponsor responses on technical work scope necessary to complete development of a waste flowsheet.

Concept	Response
ABC	No specific sponsor response was provided for this question.
ALMR	The waste processing is an integral part of the fuel cycle. The fuel cycle, involving multiple recycle and equilibrium fission products, will be demonstrated by ANL.
ALWR-MOX	Reprocessing would only be required where complete burnup of plutonium is desired. Reprocessing could be done at existing government facilities or at a new dedicated facility. Existing facilities include the Idaho Chemical Processing Plant (ICPP) at the INEL and the Savannah River Site (SRS) recycle facilities. Flowsheets of the process are presently in place. Details on the waste processes are included in Appendix E.
ALWR-T	There is no need to reprocess this fuel because adequate annihilation can be obtained by irradiating the fuel in multiple cycles. Recycling of fuel containing zirconium oxide is much more complex than reprocessing MOX fuel. Development of a process flowsheet for such fuel and qualification of a suitable waste form would require an extensive (likely >1 yr) research and development program, because plutonium-based fuels of this type have not been previously processed.
MHTGR	Based on an ORNL evaluation, the preferred option appears to be disposal of the spent plutonium fuel as whole blocks. ORNL concluded whole fuel elements could be placed in fuel waste containers similar to LWR containers. Development efforts are estimated to require about three years. Integral systems tests and demonstration are estimated to require an additional three years.
MSR	It is expected that no additional waste disposal or conditioning is required beyond that included in the design and development of the online system.
PBR	No waste processing is envisioned for this concept. The irradiated particles will be suitably packaged and stored.

2.4.1 Plutonium Waste Disposal Issues. The following background information and observations are provided to add perspective to the questions and the sponsor's answers.

Background. DOE is currently managing its radioactive waste via two distinct programs [i.e., the Office of Civilian Radioactive Waste Management (OCRWM), which originated as a result of the Nuclear Waste Policy Act of 1982, and the Office of Environmental Restoration and Waste Management (ERWM)]. Both organizations are charged to operate with scientific and technical excellence to ensure safe and cost-effective programs and the protection of the public health and the environment. Both programs also have significant institutional issues with the general public that are highly controversial and remain unresolved. Options for denaturing the plutonium inventory must be evaluated against these issues to enhance success with ongoing DOE programs.

Table 12. Sponsor responses on technical issues that could impede development of waste processing.

Concept	Response
ABC	No specific sponsor response was provided for this question.
ALMR	The basic fuel cycle, which includes waste processing, has been demonstrated with over three-year operation of an essentially full-scale electro-refiner system. By late 1993, a facility will be in operation for the demonstration of the closed fuel cycle.
ALWR-MOX	There are no outstanding issues regarding a flowsheet. If disposition of leached segments of cladding is necessary, a process for their encapsulation into a suitable waste form would have to be chosen and tested.
ALWR-T	The sponsor did not address this question.
MHTGR	No technical issues have been identified for the whole-block disposition option that would require extensive technology development.
MSR	There are no known technical issues that can impede the waste processing.
PBR	It is not expected to be technically difficult to find a suitable method for packaging the irradiated fuel particles. It is expected to be a relatively small step compared to development and construction of the PBRs.

Table 13. Sponsor responses on current estimates of time required for construction and startup of a waste-processing facility.

Concept	Response
ABC	No specific sponsor response was provided for this question.
ALMR	The waste-processing facility is contained within the fuel cycle facility.
ALWR-MOX	A new facility or modification to an existing facility would be constructed on a schedule comparable with a dedicated plutonium-burning reactor, that is, within six or seven years. Reprocessing would not be required for at least a year after first fuel discharge from the reactor. Reprocessing could be delayed up to 15 years before all the initial charge of plutonium is used.
ALWR-T	The sponsor did not address this question.
MHTGR	Facilities for packaging the spent fuel blocks are expected to be designed and constructed as an integral part of the spent fuel handling and storage facilities at the reactor. Therefore, the schedule is the same as for the reactor plant.
MSR	A separate waste-processing plant is not needed. No extra time or cost is required, so the schedule is the same as the reactor.
PBR	Design of a processing and packaging facility has not been carried out. It is not expected to be a controlling item in construction.

Table 14. Sponsor responses on current cost estimates for waste-processing development and facility construction.

Concept	Response
ABC	No specific sponsor response was provided for this question.
ALMR	The waste-processing facility is an integral part of the fuel-processing facility. The waste-processing developmental costs are covered in fuel development activities at ANL.
ALWR-MOX	Cost estimates have not been developed for a dedicated facility. Comparison with other facilities suggests that costs would be in excess of $1 billion. Modifications to an existing Savannah River facility can be done for about $110 million. Estimates for ICPP are not available.
ALWR-T	The sponsor did not address this question.
MHTGR	Cost estimates to design, construct, and start up the packaging facilities at the reactor site have not been defined. They are not expected to be a significant component of the total plant capital costs.
MSR	Development work is needed to optimize the waste processing and handling part of the fuel processing to current requirements and desires. This task is somewhat simplified by the absence of fuel in the waste.
PBR	No estimates have been made for a fuel particle packaging facility. Costs are not expected to be large compared to the other costs.

While the OCRWM and ERWM programs share common goals, there are significant differences. For example, OCRWM is funded by a fee levied on waste generators, is regulated by NRC, and will provide a system to handle packaged spent nuclear fuel (commercial) and vitrified high-level waste. Key parts of the OCRWM program include disposal of the previously mentioned inventories, development of a mined geologic repository, development of interim storage facilities, and transportation from points of origin to storage or disposal sites. The OCRWM program must obtain licenses from the NRC. There are currently about 25,000 MTHM of spent nuclear fuel in storage at various reactor sites around the country. The capacity of Yucca Mountain is set at 70,000 MTHM. The amount of defense waste placed in this repository is likely to depend on when the repository is opened and how much commercial fuel and defense waste will be ready for disposal at that time. In February 1993, OCRWM initiated a study, as directed by Section 803 of the Energy Policy Act of 1992, to examine the adequacy of its current waste program to manage additional volumes or types of nuclear waste that may be generated by new nuclear power or future defense facilities. Additional types of nuclear waste considered in this report (due to Congress in October of 1993) are rather extensive and include some materials that may be managed by the evolving ERWM program. Thus, it is currently unclear which program would control the disposal of the denatured plutonium.

ERWM is funded by general tax revenues, must comply with Environmental Protection Agency and DOE orders rather than the NRC, and may address a significantly wider range of radioactive waste and nuclear material for treatment and disposal. However, licensing for disposal may ultimately reside with the NRC. Until just recently, DOE spent fuel has been reprocessed. These operations recycled fissile materials and other by-product isotopes and isolated high-level waste for vitrification and shipment for disposal (disposal of high-level waste glass is under OCRWN). The recent DOE decision to discontinue reprocessing (because of the low demand for the recovered high-enriched uranium) has significantly changed the inventory of materials that is being addressed by

ERWM. Reprocessing produced one waste form from many different fuel types. Programs to evaluate disposal and treatment options for this previously processed spent nuclear fuel are under way and ERWM is also interested in developing new technologies that will minimize the future generation of all types of radioactive waste, especially high-level waste. If a common treatment system is not developed, then each of the basic fuel types may represent a separate waste form that will require characterization prior to disposal. Thus, key decisions on how to treat or condition different nuclear materials, as well as which materials will be treated in the ERWM program, are still under development and dependent on being integrated with the OCRWM plan.

Disposal/Repository Issues—Denaturing tactics under consideration will produce four basic waste packages: (1) spent nuclear fuel/targets that are partially burnt, (2) stainless steel canisters of vitrified defense waste mixed with small concentrations of plutonium, (3) spent nuclear fuel/targets that contain no remaining plutonium, and (4) plutonium diluted with transuranic waste for disposal at the Waste Isolation Pilot Plant (WIPP). Each of these candidate waste packages has some significant disposal issues that must be considered in a comparative evaluation process. The following subsections highlight these issues.

Repository Availability—The basic intent of the above candidate denaturing tactics is to denature plutonium and isolate it via placement in a geological repository as soon as possible. The two proposed proliferation protection barriers are self-protecting radiation fields and the isolation of a geologic repository. Each of these barrier concepts has a rather significant weakness. First, the concept of high radiation preventing proliferation can certainly be challenged by the willingness of people around the world to give their life for a particular cause. Crude shielding and remote handling concepts can quickly render radiation ineffective. Thus, the isolation aspect of a repository probably provides greater protection. However, the chances of this material going into the first repository (potentially Yucca Mountain) are believed to be extremely low and plans for a second geological repository have not been initiated. Monitored and controlled storage of the plutonium in its current or denatured form will be required for several decades until a geologic repository is open. An exception to this could be shipping the material to WIPP via transuranic-type waste packages (Option 4). However, significant political resistance would likely result, as this action would be viewed as a new mission that has never been addressed. This type of action would likely be embraced by the general public as an end-run tactic that could jeopardize the entire WIPP program.

Repository Control—The candidate denaturing proposals that would place significant amounts of highly enriched fissile material in a geologic repository must consider long-term criticality issues. These issues remain to be resolved for spent commercial fuels and have never been proposed for fissile materials with these potentially higher levels of enrichment. The difficulty of demonstrating criticality control for a geologic repository is discussed in the following paragraphs.

A fundamental performance issue for the geologic disposal is control and there are two key control areas: control/containment of fission products and criticality control for fissile material. The geologic repository concept must achieve some level of risk that is ultimately acceptable to the regulatory bodies and the general public. This acceptability issue is based on (a) the perceived hazards (chemical and radiological) of the waste form and (b) control of the hazards for as long as the hazards exist. Because radiological materials are decaying naturally, they do have a limited lifetime. The lower part of Figure 11 shows, in a gross comparative manner, that the lifetime of fissile materials (^{239}Pu and ^{235}U) will generally exceed the related lifetime of the controlling fission products. Thus, it is very difficult to demonstrate criticality control via a geologic repository for the lifetime of ^{235}U because its half-life is 704 million years. The half-life of ^{239}Pu is significantly less (25,000 years), but even this scenario is difficult to model.

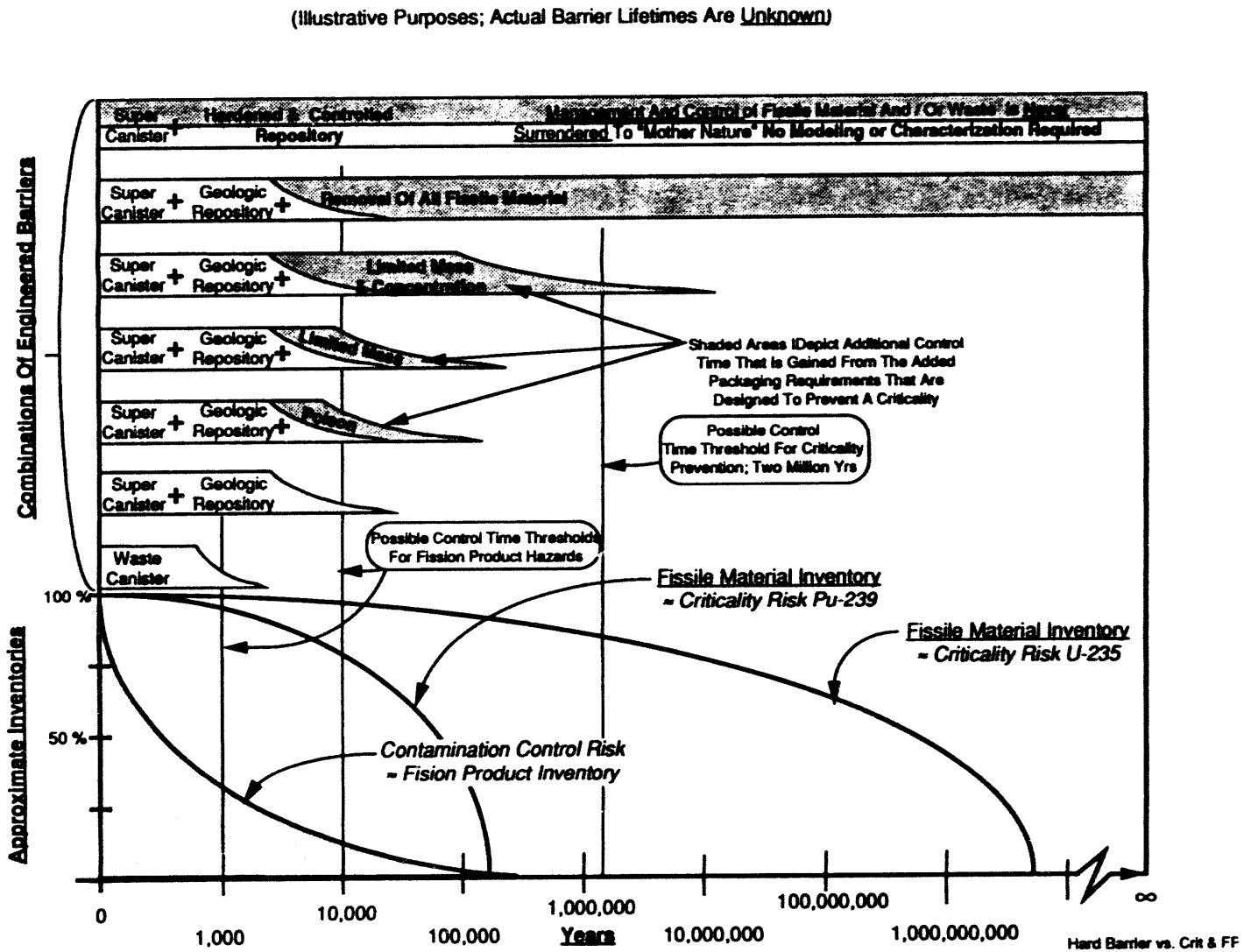

Figure 11. Control concepts versus waste hazard lifetimes.

The upper part of Figure 11 depicts different repository concepts that attempt to extend the control time via multiple barriers and other packaging options that help minimize the criticality risk via extended control times. If criticality control must be provided for as long as a risk exists, then introduction of fissile material requires configuration control of the repository and the waste form into geological time. It follows then that zero criticality risk for the geologic repository is only obtained when the fissile material is removed. If removal is not provided, then the next best situation results when the control of the fissile material is never surrendered and monitoring is maintained. The perceived risk (chemical and radiological) is generally linked to the processes that could cause the material to migrate or jeopardize confinement. These could include internal (e.g., chemical and radiological) or external (e.g., earthquakes, water intrusion) events. The potential to loose confinement becomes significantly greater with energy from an uncontrolled criticality and the inventory of fission products under these conditions includes the entire spectrum (short and long lived). Thus, the introduction of fissile material in a geologic repository increases the perceived risk and the necessary control time. High burnup proposals would help to minimize the criticality concern and complete burnup would eliminate the issue entirely. The complete burnup concept would have another advantage if fission products were treated and packaged like defense waste. This would greatly simplify characterization efforts.

Waste Forms and Characterization Programs—Any new waste form generated by these candidate proposals will require characterization and performance testing prior to acceptance at a future repository. This will require a research and development program with appropriate funding and time to complete. Funding and schedules for this type of activity can be minimized by producing waste forms that have been extensively characterized (i.e., defense waste glass and spent commercial fuel).

Mixing plutonium with defense waste at the Defense Waste Processing Facility will generate a new waste form that will require characterization and performance testing to be qualified for acceptance at a geological repository. This concept would also require a major redesign of Defense Waste Processing Facility process equipment to ensure criticality control for normal and abnormal plutonium blending operations. Facility ventilation modifications to ensure adequate alpha contamination control are likely and special nuclear material security features must also be implemented. Any physical modifications to the facility or equipment and changes in source terms and processing rates would also require revisions in the facility's Safety Analysis Report and National Environmental Policy Act documentation.

Future Strategy—The amount of conditioning or treatment that is likely to be required for denatured plutonium remains to be determined by evolution of waste acceptance criteria for some future repository concept. If it is a geologic repository concept, then criticality control is likely to be a key issue for those proposals that do not completely destroy the plutonium. Denatured plutonium from these programs may have to be poisoned, diluted, or removed to meet waste acceptance criteria. However, if a new hardened and controlled repository concept becomes available, then the waste acceptance criteria for this monitored and controlled disposal concept may allow contaminated fissile materials. Concepts requiring removal of fissile material will need to implement aqueous or nonaqueous recovery systems. These systems could be part of a DOE program that is operated to dispose of all of its nuclear material inventories (see Figure 12). Segregation of these inventories into their components would simplify the waste treatment program for much of the waste. Thus, a combination of modified repository concepts (geologic and engineered to maintain monitoring and control of the stored inventory) and alternative simplified waste packages (segregation of fissile and fission product material) could shorten the regulatory process, reduce costs, and enhance general public acceptance. New repository concepts may evolve from the ERWM and OCRWM programs.

Dispositioning strategies specifically tailored for plutonium must address some unique issues and

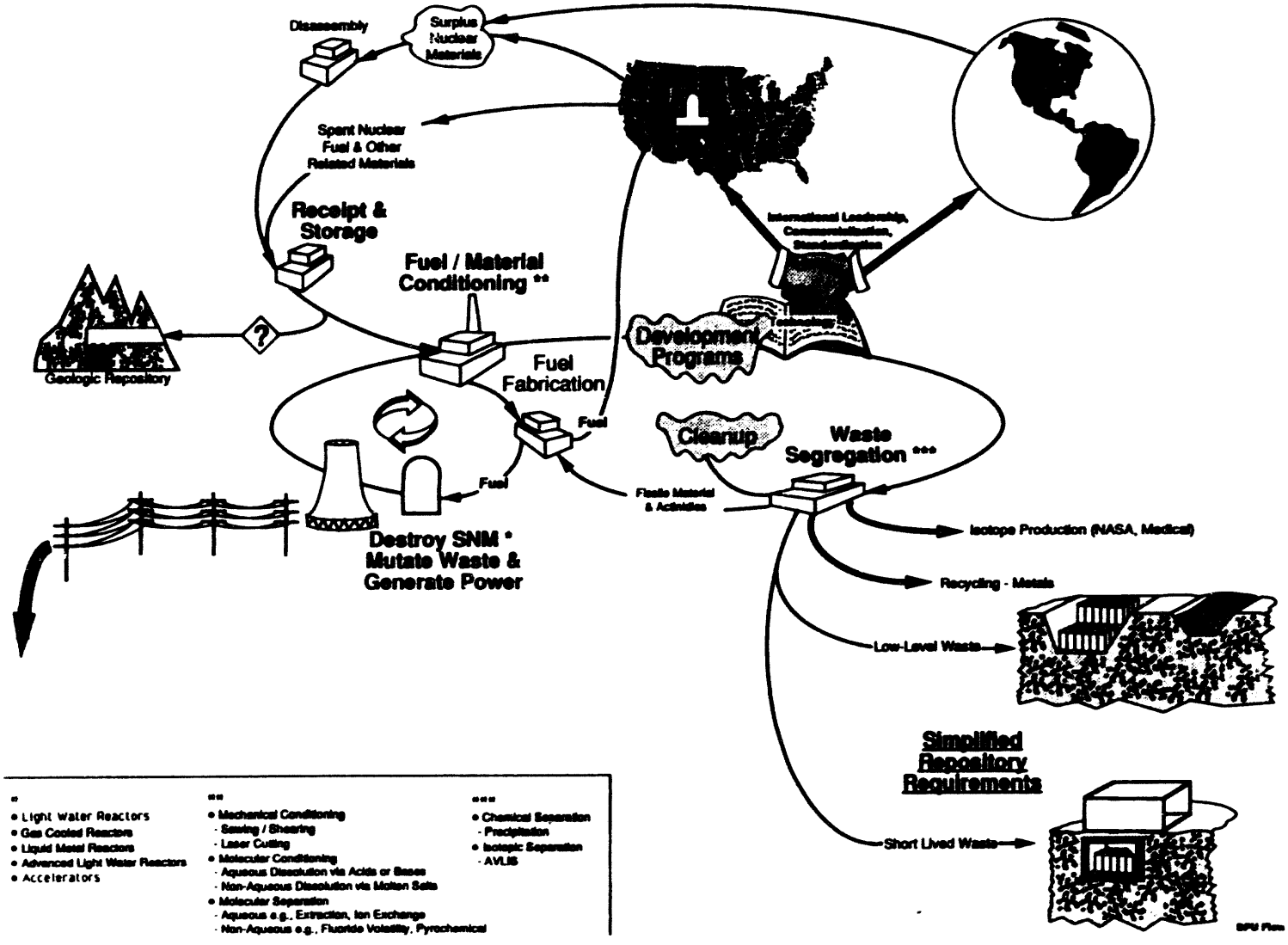

Figure 12. Systems diagram of radioactive waste system.

yet they must also embrace issues that are common for all special nuclear material (i.e., proposed dispositioning activities must be conducted in a manner that is consistent with current regulations via an overall strategy that is flexible enough to accommodate future policy modifications). These modifications will occur as a result of technology improvements, national policy issues, perceived public interest in the areas of waste minimization, environmental protection, nuclear safety, energy conservation, and general minimal risk for future generations. A no-action alternative (e.g., continued storage) serves as a useful benchmark if a different repository concept (e.g., monitored and controlled) is determined to be an acceptable solution. The resolution of these challenges requires a global, integrated, and structured plan across the DOE complex and all of its fissile materials. What may be best for plutonium denaturing should be part of the optimum solution for the entire DOE special nuclear material inventory.

2.4.2 Technical Development Requirements. Questions 1–3 concerning technical development requirements for waste disposal were asked and responses were obtained from the sponsors:

1. *Relative to the assumed acceptance of commercial fuel and defense waste in a geologic repository, is there waste characterization work that must be performed?*

2. *Briefly describe the technical work scope necessary to complete development of the waste disposal method and its duration. For example, are there any preconditioning or packaging requirements that must be satisfied for repository acceptance?*

3. *Identify technical issues that could impede the placement of waste from your option in a repository.*

Tables 15–17, respectively, summarize the sponsor responses for these three questions.

2.4.3 Schedule and Cost Estimates. Questions 4 and 5 were posed to the concept sponsors on their estimates for time schedule and costs for developing and implementing a waste disposal system and disposal of the waste in a repository.

4. *What are your estimates for the elapsed time prior to opening a suitable repository?*

5. *What are the current cost estimates for waste disposal system development and for disposal of the waste?*

Tables 18 and 19 summarize the sponsor responses to these questions. The sponsor cost estimate responses used the historical cost estimating method. With a known large variance in data vintage and accuracy, these responses should be considered only in general (versus factual) terms. The INEL recommends a separate parametric cost analysis be completed to establish criteria. This cost analysis should be applied to the top three or four options, providing a firm cost baseline for future use.

2.4.4 Self-Protection of Waste. Question 6 was posed to the sponsor on the self-protecting aspects of the final waste form to determine the waste's diversion resistance:

6. *Does radiation make your proposed waste package self-protecting (i.e., greater than 100 R/hr at 3 ft from the surface). If so, how long does it remain self-protecting?*

Table 20 summarizes the sponsor responses to this question.

Table 15. Sponsor responses on waste characterization work that may be necessary for repository acceptance.

Concept	Response
ABC	No specific sponsor response was provided for this question.
ALMR	The metal fuel pyroprocess produces two main waste streams: salt and metal, which do require performance and acceptance testing. However, these waste forms are being developed and tested and are expected to be acceptable for repository acceptance.
ALWR-MOX	The spent MOX fuel is to be sent to a geologic repository following a period of interim storage. Characterization studies would have to be factored into the spent fuel characterizations currently in progress for commercial oxide fuel.
ALWR-T	The sponsor did not address this question.
MHTGR	Based on ORNL conceptual evaluations, the fuel is disposed of as whole blocks, which means the plutonium fuel is permanently encased by large quantities of corrosion-resistant graphite. Characterization beyond the fuel and graphite appear to be minimal. Three technical issues are identified ranging from confirmation of the C-14 content of the fuel elements to oxidation rates of irradiated graphite. (See Appendix F).
MSR	The waste is fuel free. Because it is in a chemical processing plant, it can be adapted to the requirements of any repository. Side streams contain chemicals that will require characterization. It is expected they can be modified, classified, and separated according to requirements.
PBR	Because the waste packaging has not been explicitly defined, no precise answer can be given. The waste packages could be adapted to conform to a geological repository.

Table 16. Sponsor responses on technical development work necessary for waste disposal.

Concept	Response
ABC	No specific sponsor response was provided for this question.
ALMR	The preconditioning requirements should be covered in the testing described in the response to the previous question. Packaging is expected to be standard.
ALWR-MOX	Technical work scope is well defined. The major elements are spent fuel characterization, reactor basin for underwater cooling, program for dry storage, program to develop treatment facility to repackage fuel, definition of repository acceptance criteria, and fuel qualification procedures with emphasis on criticality prevention.
ALWR-T	The sponsor did not address this question.
MHTGR	ORNL concluded fuel elements could be placed in spent fuel containers and placed in a repository without significant preconditioning or processing. Confirmation will be needed by detailed engineering analysis and possibly validated by testing programs.
MSR	Waste treatment development is part of the entire fuel-processing development. One of the concepts envisions that the waste will be optimized in every respect. This can be done because the waste stream is in a liquid form.
PBR	Suitable packing methods will need to be defined. Because no chemical processing is required, it is expected to be a relatively small step compared to development and construction.

Table 17. Sponsor responses on technical issues that could impede placement of waste in a repository.

Concept	Response
ABC	No specific sponsor response was provided for this question.
ALMR	There are no known impediments for placement of ALMR pyroprocessing waste in a repository. The waste performance and acceptance testing is ongoing and is expected to be successful.
ALWR-MOX	There are waste form qualifications with particular emphasis on criticality prevention over geologic time periods. Safeguard assessments are needed for both surface and subsurface storage. Because commercial fuels will face these same issues, successful commercial fuel qualification will provide a means to qualify MOX fuel.
ALWR-T	The sponsor did not address this question.
MHTGR	No technical feasibility issues were identified by the ORNL assessment that are expected to impede placement of whole fuel elements in a repository. Minimizing space requirements may need to be addressed.
MSR	There are no known issues unique to this concept. The waste is fuel free. Side streams that contain elements such as fluorine or beryllium with radioactive material may require special treatment.
PBR	Currently no issues have been identified.

Table 18. Sponsor responses on current schedule estimates for opening a suitable repository.

Concept	Response
ABC	No specific sponsor response was provided for this question.
ALMR	It is assumed that the current repository schedule (e.g., opening about 2010) is compatible with the ALMR schedule.
ALWR-MOX	Only defense high level waste and commercial spent fuel are authorized for storage at Yucca Mountain. It is unlikely this situation will change. It is likely that MOX fuel disposal will have to wait for a second repository. It is a widely shared opinion that a second repository would not be available before 2030-2040.
ALWR-T	The sponsor did not address this question.
MHTGR	There is no incremental elapsed time beyond that for opening a suitable LWR fuel repository for commercial or defense HLW.
MSR	The MSR is not dependent on the opening of a repository for waste. The waste is fuel free and relatively small in quantity.
PBR	No estimate has been made of the time prior to opening a repository. Political issues could dominate this question.

Table 19. Sponsor responses on current cost estimates for waste disposal system development and for disposal.

Concept	Response
ABC	No specific sponsor response was provided for this question.
ALMR	The cost of disposal of ALMR processed fuel is expected to be less than the current 1 mill/kWh fee. It is expected that the cost on an equivalent basis would be one-half to three-quarters of a mill/kWh.
ALWR-MOX	Spent fuel disposal estimates included in WSRC-RP-92-1004[2] are estimated to be $65 million and additional development costs are estimated to be about $10 million. (See Appendix E for additional details.)
ALWR-T	The sponsor did not address this question.
MHTGR	A rigorous cost estimate has not been made, but it should be comparable to costs for commercial spent fuel disposal.
MSR	There are no estimates for waste disposal. Some new and additional steps will be needed, but they are not considered critical.
PBR	No cost estimate has been made of a particle processing facility.

Table 20. Sponsor responses on the self-protecting features of the waste package.

Concept	Response
ABC	No specific sponsor response was provided for this question.
ALMR	It will be self-protecting. See the curve provided in Appendix D.
ALWR-MOX	It will be self-protecting for about the same time period as the commercial spent fuel. The time has not been specifically calculated, but it is expected to be self-protecting for greater than 50 years.
ALWR-T	The sponsor did not address this question.
MHTGR	A graph shows it to be self-protecting according to the stated criteria for about 60 years.
MSR	The MSR waste does not contain any fuel, so it is the ultimate in self-protection.
PBR	No estimate has been made.

3. GENERAL RATING OF PLUTONIUM ANNIHILATION OPTIONS

The INEL produced ratings for the reactor and accelerator-based systems considering only options that have the capability to annihilate large fractions of weapons-grade plutonium. These ratings have a different basis than ratings produced by LLNL because its ratings considered options that denatured plutonium as well as options that annihilated some or most of the plutonium. For the INEL's rating, three areas of comparison were selected: (1) operational capabilities of the fission options, (2) time required to deploy reactor or accelerator, and (3) cost estimates based on sponsor supplied information. A brief discussion of the first two areas follows. Insufficient valid cost data are available to develop sound estimates for future use. Further work is required to produce defendable and comparable cost estimates.

The methodology for rating operational capability of the options followed the rating methodology developed by LLNL.[1] Four stages of operational capability were defined as:

- **Concept Feasibility (CF).** The physical principles associated with the concept are well understood and general feasibility has been established.
- **Engineering Feasibility (EF).** The engineering system, subsystems, and major components have been identified and their performance has been generally established. A design basis, including design basis accidents and preliminary system response to such events, has been developed.
- **At-Scale Operation (ASO).** A successful operation that is larger than bench scale and smaller than full scale exists.
- **Presently-Existing Capability (PEC).** Similar, but not necessarily identical, systems are currently operating successfully.

Table 21 summarizes the rating of options. The most significant change in this table from the LLNL report is in rating of engineering feasibility for the ALWR. In examining the sponsor's information, the INEL concluded the ternary fuel requires additional development before its engineering feasibility can be definite. Ratings of ABC, MSR, and PBR were listed as partial because engineering feasibility has not been proven for all systems components.

Categorization by time to deployment is based on a system capable of annihilation of plutonium. Information from the LLNL report was used in categorization, but different durations were assigned. In general, as with cost estimates, accurate schedule information is not available within the nuclear industry and will require further study. Table 22 shows options falling into three distinct groups. The ALMR-R uses fuel currently developed and replaces blanket material that would otherwise be used for plutonium breeding. However, results from Section 2.2.1 indicate that this concept would take approximately 144 years to annihilate 90% of the assumed 50 MT of plutonium. The ALMR-MB, ALWR-T, and MHTGR would fall within the same deployment period because each has fuel development and testing for plutonium-based fuel in process. The ABC, MSR, and PBR require additional development and design efforts for their entire concept.

A comparison of concept costs has been compiled in Table 23 from sponsor response information contained in Tables 4, 10, 14, and 19. These costs are not sufficiently complete to allow a ranking to be made. Electrical revenues were calculated based on the concept descriptions used in the calculations of Section 2.2.1.

Table 21. Operational capability of plutonium annihilation options.

	ALMR-R	ALMR-MB	ALWR-T	MHTGR	ABC	MSR	PBR
CF	Yes	Yes	Yes	Yes	Yes	Yes	Yes
EF	Yes	Probably[b]	Probably[c]	Probably[d]	Partial[d]	Partial[e,f]	Partial[e]
ASO	Probably[a]	No	No	No	No	No	No
PEC	No	No	No	No	No	No	No

a. Assuming Fuel Cycle Facility operation.
b. Based on experience with uranium-plutonium based fuel, recognizing the need for additional fuel development work.
c. Based on experience with uranium-based ternary fuel, recognizing the need for additional fuel and reactor kinetics development work.
d. Assuming review of Peach Bottom PuO_2 TRISO particle tests will validate fuel performance and acceptable plutonium core accident response can be established.
e. Requires large scale-up of MSR.
f. Design basis, including design basis accidents, has yet to be developed.

Table 22. Ranking of plutonium annihilation options based on time to deployment.

Group	Concept rankings	Comments
I (5–10 yr)	ALMR-R	Moderate extension of current technology but a slow annihilation option
II (10–20 yr)	ALMR-MB	Technical development of plutonium-based fuel required
	ALWR-T	Technical development of plutonium-based fuel required
	MHTGR	Technical development of plutonium-based fuel required
III (20–30 yr)	ABC	Extensive technical development of concept required
	MSR	Extensive technical development of concept required
	PBR	Extensive technical development of concept required

Table 23. Total concept costs for similarly sized facilities assuming a 40-year operating lifetime.

	Concept costs (in millions)					
	ABC	ALMR	ALWR	MHTGR	MSR	PBR
Research and development	IDP	325	IDP	261	IDP	208
Fuel Fabrication Facility	IDP	120	680	260	IDP	1,000
Fuel Fabrication Facility operations and maintenance	IDP	1,000	IDP	1,720	IDP	IDP
Reactor facility	IDP	3,400	3,600	3,260	IDP	5,250
Reactor facility operations and maintenance	IDP	4,400	10,000	9,920	IDP	2,000
Electrical revenue (calculated assuming 75% capacity factor and 0.06/kWh)	16,399	22,706	18,922	24,551	IDP	18,922

IDP—Insufficient data provided by sponsor.

4. CONCLUSIONS AND RECOMMENDATIONS

4.1 Conclusions

Based on information provided by concept sponsors and the INEL's evaluation of this information, the following conclusions and recommendations were reached.

4.1.1 Fuel Status. Previous fuel development work for many of the options concentrated on uranium-based fuel. As a result, plutonium-based fuel development has been limited and demonstration of full-scale fuel fabrication has not been made. Based on the current fuel status, the development and fabrication of plutonium-bearing fuels will be on the critical path if annihilation of a high percentage of the plutonium (90% or greater) is desired. Fuel development will consume much of the time required to design and construct any of the reactor or accelerator-based options.

Two fuel forms have been proposed for both the ALMR and ALWR options. A uranium-plutonium-based metal fuel has been proposed for the ALMR reference fuel cycle (ALMR-R) and a plutonium-based metal fuel for a maximum burner or annihilation cycle (ALMR-MB). A MOX fuel has been proposed for the ALWR (ALWR-MOX) and also a ternary fuel (ALWR-T) to provide more rapid plutonium annihilation.

Fuel development of ALWR-MOX has been completed. Irradiation testing of the ALMR-R fuel is under way at ANL. Vital developmental work remains for the ALMR-MB, ALWR-T, and MHTGR fuels. Significant development work will be required for the fuel and the components associated with the core region (frits, etc.) for the PBR. Insufficient information on fuel development needs was provided by sponsors of ABC and MSR but reviews by the INEL indicate that additional fuel development is required.

The INEL believes that sponsor estimates for fuel fabrication facility costs and schedules may be optimistic because experience with uranium oxide fuel cannot be directly extrapolated to plutonium-based fuel fabrication. The plutonium fuel facilities will have safety and environmental issues that are complex and difficult to resolve. In addition, all fabrication and storage facilities will likely be funded by DOE and reside on a highly secure DOE reservation, which could result in higher costs and longer schedule durations. A plutonium fuel production facility that is sufficiently large and secure is projected to cost more and take longer to design, build, and make operational than a similar uranium facility, especially if care is not taken in implementing design and safeguards requirements. Operating costs and durations are expected to be greater than currently encountered for uranium-based fuel. However, operating costs are a small part of the overall costs of the plant. It is not clear that these differences are adequately considered in the sponsor estimates.

4.1.2 Reactor and Accelerator System Status. The INEL believes that annihilation of plutonium is preferred over denaturing. Furthermore, because it is possible to construct a nuclear explosive device from a wide range of plutonium isotopic concentrations, the INEL believes that the capability to annihilate all five plutonium isotopes (^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, and ^{242}Pu) is a better measure of a concept's effectiveness than the capability to annihilate just the ^{239}Pu and ^{241}Pu isotopes. A discussion of the capabilities of the reactor and accelerator-based options to annihilate plutonium follows.

The original ALWR option description proposed using MOX fuel for disposition of plutonium. The ALWR sponsor response states that ternary fuel (PuO_2-ZrO_2-CaO) is now the preferred fuel. The INEL believes that the change in ALWR fuel choice is possibly because use of MOX fuel to annihilate 50 MT of plutonium would require long periods of time or large numbers of reactors. Numerous fuel reprocessing cycles would be necessary. Although the use of existing commercial reactors may be possible, reprocessing would

likely become the critical path for mission completion. Large amounts of spent fuel would have to be shipped to centralized reprocessing facilities, increasing the potential for diversion. In addition, the time and cost to license a large number of commercial LWRs for MOX fuel could be substantial. Because commercial LWRs are not standardized, it is likely that the NRC would have to consider the licensing of MOX fuel on a plant-by-plant basis.

ALWR-T and MHTGR options are both capable of annihilating high percentages of ^{239}Pu and ^{241}Pu without reprocessing. The ALWR sponsor proposes using a ternary fuel to achieve annihilation of a significant fraction of the total plutonium isotopes without reprocessing. ALWR-T high annihilation percentages can only be reached through in-reactor fuel assembly resident times significantly longer than current LWR fuel assemblies typically experience. Further investigation would be required to determine whether ALWR-T fuel assembly materials could survive long resident times without being refurbished. The MHTGR sponsor proposes employing a fuel management scheme that involves shuffling irradiated fuel blocks into the core reflector region to achieve annihilation of large fractions of all plutonium isotopes. Further investigation would be required to determine the practicality of such a fuel management scheme.

Calculations by the option sponsors indicate that concepts with reprocessing can achieve near total plutonium annihilation in a shorter duration than nonreprocessing options. As shown in Table 7, for the deployment of similar MW(t) facilities, the concepts that can annihilate total plutonium quickest are the ABC and ALMR-MB. These concepts are closely followed by the MHTGR and PBR. However, the ABC and PBR concepts are not highly developed and will require a significantly longer time period to implement and deploy. The annihilation levels of the MHTGR concept depend on a yet-to-be-proposed method for reprocessing the fuel particles.

Technology development issues must be resolved for all reactor and accelerator-based systems. Criticality and reactivity control during reactor operation must be examined when annihilating large percentages of plutonium. Development work on a reactor concept similar to the PBR option in the Commonwealth of Independent States indicates that substantial development work is required for the PBR. Specific technology development issues for the ABC, MSR, and PBR options were not identified because these concepts are in a preliminary stage of conceptual development. It is clear that significant issues relating to materials, design, and fabrication would have to be resolved before these concepts could be constructed.

The INEL believes that sponsor estimates for system development and construction costs and schedules are optimistic and that costs would be higher and schedules would be longer than predicted. Startup and operational costs are expected to be similar to those of currently operating reactor facilities on a per reactor basis.

4.1.3 Waste-Processing Status. A detailed technical assessment of the waste-processing area was not performed by the INEL. Waste characterization work will be necessary for all plutonium-based fuels. Waste processing is an integral part of the ABC, ALMR, and MSR. Technical development of the ALMR waste-processing system is under way at ANL. Waste processing for the ABC and MSR require process and component development for plutonium-based fuels. The sponsor indicated that waste processing is not necessary for the PBR because no reprocessing is proposed and the particles would be packaged and sent to a waste disposal facility. Waste processing for the MHTGR with reprocessing has not been completely evaluated. ORNL examined disposal of uranium-based MHTGR fuel and concluded that the fuel elements could be safely placed in a repository without waste processing. Waste processing for the ALWR-T fuel would require more complex and time-consuming processes than the MOX fuel. Some benchtop development work for the uranium-based ternary fuel has been done.

4.1.4 Waste-Disposal Status. Several waste-disposal packages are possible using reactors or an accelerator. Each of the possible waste packages has disposal issues that must be considered in a comparative evaluation process, including:

- **Repository Availability.** The likelihood of waste from the reactor or accelerator-based concepts going to the first geological repository is very low and plans for a second repository have not been initiated. Monitored storage of plutonium, or its denatured form, could be required for several decades.

- **Repository Control.** There are two key variables in the control of material in a repository: control/containment of radioactive material and control of criticality. Criticality control is difficult to demonstrate to the general public because the fissionable material decays very slowly, which means material could be available to form a critical mass for long periods of time. In addition, there is no evidence that the containment material will last the long times necessary to prevent the fissionable material from migrating into a critical geometry.

- **Waste Forms and Characterization Programs.** Any new waste forms will require characterization and performance testing prior to acceptance at a future geologic repository.

Several option sponsors recognized these issues and it was clear that adequate consideration of waste disposal needs in all options requires further investigation. It does not appear any option has a notable advantage in waste characterization area, and therefore should not be a discriminator in rating options. Three sponsor concepts, the ABC, ALMR, and MSR, have greatly reduced requirements for long-term monitoring of their final waste forms. These forms are expected to contain only small quantities of high-level nuclear waste and fissionable materials. Destruction of long-lived actinides are an inherent part of these concepts' processes and the effective lifetime requirement of a geologic repository is reduced from several hundred thousand years to several hundred years.

If a repository is not available for several decades, temporary onsite storage of the final waste should not pose any additional safeguards and security problems. These waste forms contain only trace quantities of plutonium and criticality control during storage in a repository becomes a low level issue. Further study of all the waste disposal issues for sponsor concepts is required.

4.1.5 General Rating of Plutonium Annihilation Options. The INEL produced ratings for the reactor and accelerator-based systems considering only options that have the capability to annihilate large fractions of weapons-grade plutonium. These ratings have a different basis than ratings produced by LLNL because its ratings considered options that denatured plutonium as well as options that annihilated some or most of the plutonium. For the INEL's rating, three areas of comparison were selected: (1) operational capabilities of the fission options, (2) time required to deploy reactor or accelerator, and (3) cost estimates based on sponsor supplied information. The ALMR-R rated highest in operational capabilities and time to deploy. Costs could not be used to provide adequate ranking because sufficient data were not available.

4.2 Recommendations

Based on the evaluation of the sponsor concepts, the following observations can be made:

- If annihilation of the weapons-grade plutonium is to commence as soon as possible, as the most technically developed concept, the ALMR-R can be deployed earliest. Total annihilation of 50 MT of weapons-grade plutonium over a 40-year period would require completion of the remaining technical development and the construction of approximately 48 ALMR modules over the next 5 to 10 years with the remaining 30 years dedicated to plutonium annihilation.

- If annihilation of the weapons-grade plutonium is desirable and reprocessing of the irradiated fuel is unacceptable, only two concepts are available, ALWR-T and MHTGR. There is currently no experience with plutonium-based fuels for the ALWR-T and limited experience for the MHTGR. Additional time would be required to complete technical development of both fuel types. ALWR-T high annihilation fractions can only be achieved through significantly longer fuel assembly resident times in the reactor than current LWR fuel assemblies. It is questioned whether ALWR-T fuel assembly materials could survive long resident times without being refurbished in some manner. It is also questioned whether criticality and reactivity control can be maintained throughout long cycle lengths. High annihilation fractions of the MHTGR can only be achieved through a modified fuel management scheme that replaces reflector materials with irradiated fuel elements.

  Approximately 73% annihilation of 50 MT of plutonium within a 40-year period would require completion of the remaining technical development and the construction of approximately eight ALWRs or 15 MHTGR modules over the next 10 to 20 years with the remaining 20 years dedicated to plutonium annihilation. Fuel development is required for each of these options, but insufficient time and information was available to chose one option over the other.

- If annihilation of weapons-grade plutonium is desirable and reprocessing irradiated fuel is acceptable, six concepts are available: ABC, ALMR-R, ALMR-MB, MHTGR, MSR, and PBR. Although some development time is still required for the ALMR-R and ALMR-MB fuel types, the ABC, MSR, and PBR concepts are not sufficiently developed and will require a significantly longer time period to implement and deploy, if successful at all. In addition, no method has been proposed for reprocessing the MHTGR or PBR fuel particles.

  For the ALMR-R, total annihilation of 50 MT of plutonium within a 40-year period would require completion of the remaining technical development and the construction of approximately 48 ALMR modules over the next 5 to 10 years with the remaining 30 years dedicated to plutonium annihilation. The ALMR-MB would require construction of 19 ALMR modules over the next 10 to 20 years with the remaining 20 years dedicated to plutonium annihilation. The MHTGR would require construction of 20 modules over the next 10 to 20 years with the remaining 20 years dedicated to plutonium annihilation. For the ABC or PBR, completion of the remaining technical development and the construction of approximately five ABC systems, six MSR reactors, or 15 PBR modules would require 20 to 30 years with the remaining 10 years dedicated to plutonium annihilation. Because of the large number of modules required for the ALMR-R and the technical development required for the ABC, MHTGR, MSR, and PRB, if reprocessing of the irradiated fuel is acceptable, the ALMR-MB is preferable.

Fabrication of the ALWR-T and MHTGR fuels will require weapons-grade plutonium be processed for removal of contaminants. Facilities exist for such processing, but transportation to the reactor site increases the potential for diversion. The ALMR-R or ALMR-MB fuel can employ weapons-grade plutonium directly into its fuel cycle, and the fuel cycle will reside at the same location as the reactors. The potential for diversion is diminished for the ALMR-R and ALMR-MB concepts.

Waste-processing and waste-disposal issues must also be considered for the above recommendations. Waste processing is an integral part of the ALMR concept and technical development of this

process is under way at ANL. Waste characterization work may be necessary for the MHTGR plutonium-based fuel and waste processing for the ALWR ternary fuel would require more complex and time-consuming processes than MOX fuel. Of the three concepts, the ALWR-T and MHTGR waste will be highly radioactive for hundreds of thousands of years. The ALMR waste will be highly radioactive for only several hundred years because most of the high level radioactive waste is recycled back into the ALMR fuel.

Because of time constraints and lack of detailed information available for this review, the INEL recommends further study of what it believes are the top four concepts—ALMR-R (Advanced Liquid Metal Reactor with reference fuel cycle), ALMR-MB (Advanced Liquid Metal Reactor with maximum burner fuel cycle), ALWR-T (Advanced Light Water Reactor with ternary fuel), and MHTGR (Modular High Temperature Gas-Cooled Reactor).

5. REFERENCES

1. R. P. Omberg and C. E. Walter, *Disposition of Plutonium from Dismantled Nuclear Weapons Fission Options and Comparisons*, Lawrence Livermore National Laboratory, UCRL-ID-113055, February 5, 1993.

2. M. R. Buckner and P. B. Parks, *Strategies for Denaturing the Weapons-Grade Plutonium Stockpile*, WSRC-RP-92-1004, Westinghouse Savannah River Complex, 1992.

Appendix A

Questions Asked for the Sponsor Options

Appendix A

Questions Asked for the Sponsor Options

Fuel

Please answer the following questions for fuel development, fabrication facility design and construction, and facility startup and operation.

1. Did you assume that plutonium (PuO_2 or Pu metal) used in the fuel would be free of contaminants (alloying metals and americium now in the nuclear weapon pits)? Will the fuel proposed be negatively impacted if plutonium is contaminated with these alloying metals and the americium?

2. Briefly describe the technical work scope necessary to complete development of the fuel and its estimated duration.

3. Identify technical issues that could impede fuel development and fabrication. For example have all issues related to material lifetime, compatibility, etc., been resolved?

4. What are the current cost estimates for fuel development and for the fuel fabrication facility construction, startup, and operation? What estimating method was used (e.g., parametric, historical cost, unit cost, etc.)?

Reactor or Accelerator System

Please answer the following questions for reactor or accelerator system development, facility design and construction, and facility startup and operation. Since the MSR and the ABC use a continuous fuel cycle, it is not necessary to answer the first question.

1. If the plutonium disposition goal is to annihilate ^{239}Pu and ^{241}Pu in a single fuel cycle, what is the total burnup or exposure (GWD/ MT Pu) required to reduce the initial inventory of these two isotopes by 90%, 95%, and 99% (if possible)? For each of these cases, identify the weight percent of all plutonium isotopes in the initial fuel loading and those remaining in the spent fuel after an equilibrium fuel cycle. Also identify the cycle times.

2. If the plutonium disposition goal is to annihilate ^{239}Pu and ^{241}Pu and irradiated fuel can be reprocessed to recover and recycle plutonium, what is the total burnup or exposure (GWD/MT Pu) required to reduce the initial ^{239}Pu and ^{241}Pu inventory by 90%, 95%, 99%, and 99.9%? If the option uses a batch mode fuel cycle, how many times would a core fuel load need to be recycled to reach each burnup percentage? For each of the four cases, identify the weight percent of all plutonium isotopes prior to initiation of irradiation and those remaining in the spent fuel.

3. If the plutonium disposition goal is to annihilate all plutonium isotopes and irradiated fuel can be reprocessed to recover and recycle plutonium, what is the total burnup or exposure (GWD/MT Pu) required to reduce the inventory of all plutonium isotopes by 90%, 95%, 99%, and 99.9%? If the option uses a batch mode fuel cycle, how many times would a core fuel load need to be recycled to reach each burnup percentage? For each of the four cases, identify the weight percent of all plutonium isotopes remaining in the spent fuel.

4. Briefly describe the technical work scope necessary to complete development of your reactor or accelerator system and its estimated duration.

5. Identify technical issues that could impede system development, design, construction, and startup. For example, have all issues related to material lifetime, compatibility, etc., been resolved?

6. What are the current cost estimates for system development and for construction, startup, and operation of the facility? What estimating method was used (e.g., parametric, historical cost, unit cost, etc.)?

Waste Processing

Please answer the following questions for waste processing development, facility design and construction, and startup. For these questions, waste is defined as spent fuel and by-product waste streams from recycling spent fuel.

1. Briefly describe the technical work scope necessary to complete development of a waste conditioning/processing flowsheet for your option and its estimated duration.

2. Identify technical issues that could impede development of the waste processing flowsheet, systems, or facilities.

3. What are the current estimates for the time required for construction and startup of the waste processing facility?

4. What are the current cost estimates for waste processing development and for construction and startup of a waste processing facility?

Waste Disposal

Please answer the following questions for waste disposal development for your option. For these questions, waste is defined as spent fuel and by-product waste streams from recycling spent fuel. You may benchmark your answers against the programs that are being developed for commercial fuel and defense waste.

1. Relative to the assumed acceptance of commercial fuel and defense waste in a geologic repository, is there waste characterization work that must be performed?

2. Briefly describe the technical work scope necessary to complete development of the waste disposal method and its duration. For example, are there any preconditioning or packaging requirements that must be satisfied for repository acceptance?

3. Identify technical issues that could impede the placement of waste from your option in a repository.

4. What are your estimates for the elapsed time prior to opening a suitable repository?

5. What are the current cost estimates for waste disposal system development and for disposal of the waste?

6. Does radiation make your proposed waste package self-protecting (i.e., greater than 100 R/hr at 3 ft from the surface)? If so, how long does it remain self-protecting?

Appendix B

Canadian Dueterium-Uranium Sponsor Response

AECL

AECL Technologies

9210 Corporate Boulevard
Suite 410
Rockville Maryland
20850 USA
1-800-USA-AECL
(301) 417-0047
Fax (301) 417-0746
Telex 403-442

March 15, 1993

Mr. Duane J. Hanson
EG&G Idaho Inc.
P.O. Box 1625
MS 2508
Idaho Falls, Idaho 83415

Dear Mr. Hanson,

I have received your FAX of March 11, 1993 providing a list of the questions requested by the National Academy of Sciences regarding reactor options for Plutonium disposition. You requested an answer by Monday, March 15. It is unfortunate that we did not receive this request until your study was almost completed. Obviously it is not possible to provide detailed answers in the time allotted to us. The questions warrant careful study and some analysis of fuel management schemes for our CANDU reactor line; there is not time enough for such study and analyses.

Nevertheless, I think it may be useful to INEL and the National Academy to consider the possibility of employing CANDU type reactors for the dual purpose of power generation and Plutonium annihilation, as they may have major advantages over the alternatives. I have summarized below some information on the current status of CANDU technology, and some of our initial thoughts regarding the application of CANDU reactors for disposing of excess Plutonium. This letter supercedes the draft letter which I FAXED to you on March 12, 1993.

STATUS OF TECHNOLOGY DEVELOPMENT - Questions 4, 5 and 6 under "Reactor Systems" ask about the status of development of the reactor technology being proposed. In fact, the CANDU technology is fully developed and proven. A Plutonium burning fuel system could be applied using available designs without changing the basic reactor configuration and control systems. The frequency of on-line refueling would have to be increased in some cases, but no change in design would be required. Thus additional technology development would not be required for the power plant or reactor control and safety systems. Detailed studies would be required on the fuel system to maximize the rate of plutonium consumption. Specifically:

1. The CANDU Pressurized Heavy Water Reactors have a proven record of safe, reliable, and economic performance which compares favorably with other proven reactor types such as Pressurized and Boiling Water Reactors. We have over 200 reactor years of safe and successful commercial operation; with 32 reactors currently in operation or under construction. Multiple unit stations provide much of the power for Ontario Hydro, and single unit CANDU stations are in operation in Quebec, New Brunswick, Korea, and Argentina. Additional plants are under construction in Romania, Korea, and Canada.

2. AECL is the developer and designer of the CANDU reactors. Manufacturing and construction is done by the utility or private

vendors. The 600 MWE CANDU 6 plants are stand alone plants currently producing power in Korea, Argentina, New Brunswick, and Quebec.

3. AECL has designed an improved version of the CANDU 6 reactor, known as the CANDU 3, with a rating of 450 MWE. This design has improved reliability, safety, and operability features, and has major improvements in constructability, with the result that economics compare favorably with a comparable size fossil power plant. Detailed design of CANDU 3 is about 80% complete (the Province of Saskatchewan is sharing in the funding for this design), with detailed regulatory review of the standard design underway by Canada's Atomic Energy Control Board. AECL Technologies, the U.S. arm of AECL, has applied to NRC for a U.S. license of the CANDU 3 design under 10CFR52; a pre-application review is underway and an NRC SER is scheduled to be issued in 1994. Discussions have been initiated with the Electric Power Research Institute to prepare a Utility sponsored CANDU Requirements Document similar to the ALWR Requirements Document recently completed.

4. Either the CANDU 6 reactors of the type currently in operation, or the improved CANDU 3 design now being licensed in Canada and the U.S., could be applied to the Plutonium annihilation program.

REACTOR SYSTEM - Questions 1 through 4 request information on the effectiveness of the proposed reactor system to fully annihilate the fissile Plutonium. In conversations with Woody Stroup, he emphasized the desire in this study to avoid the production of new plutonium from fertile materials. Because it uses on-line refueling, the CANDU system can achieve more complete annihilation of Plutonium without reprocesing, as compared to alternatives. The question of fuel design, including use of non-fertile dilutants, is discussed below.

In terms of the reactor system, present CANDU reactors use Natural uranium and achieve a burnup of about 8000 MWD/T generally in one pass through the reactor. However, many alternative fuel cycles using Thorium, slightly enriched Uranium, and Plutonium-Uranium Mixtures have been studied[1]. Some of these fuel cycles utilize multiple passes through the reactor to maximize a particular nuclear performance characteristic. This can be done in CANDU reactors with no loss of capacity factor because of the on-line refueling feature. In the case of plutonium annihilation, it should be possible to achieve virtually complete annihilation of the fissile plutonium in a single fuel bundle by judicious selection of the initial composition and multiple cycles through the reactor.

Recent work done for an overseas client has verified that CANDU 6 reactors can achieve near complete annihilation of the plutonium, without reprocessing, by using plutonia-beryllia fuel bundles in multiple passes through the reactor. Some increase in fuelling machine usage would occur, but this would be within the design envelope of the equipment. Also some change in the linear density of poison in the reactivity adjustor rods would be required, but this again is within the envelope of previous design and operating experience.

In one example studied, the fresh plutonium bundle would be initially inserted in the outer fuel channels where the neutron flux is lower. As the reactivity decreases, the bundle could be recycled into more central fuel channels which have higher neutron flux, to maintain the desired bundle power while continuing to burn the plutonium. The required

plutonium concentrations, total burnup, cycle burnup, and recharge locations would have to be determined based on detailed core analyses. It is conceivable that much higher burnups than the 8,000 MWD/t could be achieved on each bundle, thereby further reducing the total fuel cycle costs. CANDU fuel pins are capable of achieving burnup of 40,000 MWD/t and higher...the CANDU fuel pin is very similar to the fuel pin design used in light water reactors.

Question 5 asks for information on the technical work scope necessary to complete the development of the reactor system. In our case, that work is almost completed, except for the design and analysis of the plutonium core. We cannot estimate that work scope without some additional information from EG&G regarding your requirements.

FUEL SYSTEM - AECL has developed an improved fuel bundle design known as CANFLEX for use in advanced fuel cycles. This improvement allows higher thermal performance whilst assuring the fuel materials remain within the proven temperature-burnup envelope. However, most fuel work done in Canada and elsewhere has utilized fertile materials (U^{238} or Thorium) as diluent for the fissile materials. Mr. Stroup indicated that this study requires that no new plutonium be generated in fertile materials while the existing plutonium is being annihilated. While the plutonia-beryllia system mentioned previously would work very well from a neutronic point of view, it has never been proven in-pile.

A near term solution to this problem is to use ZrO_2 as a diluent. During the early 1960's I helped the AEC's Naval Reactors Branch manage a reactor development program for Shippingport Core 2 which developed and proved just such a fuel. Specifically, the seed portion of Shippingport Core 2 used a mixture of highly enriched UO_2 and ZrO_2 (up to 67 w/o) which performed very well to fairly high burnups. Although that design used flat plates, rather than fuel pellets, it did prove the irradiation performance of the material[2]. The cited reference notes that the thermal conductivity of the fuel, after irradiation, is not much different than that of UO_2 after irradiation (about 30% lower).

I therefore suggest that the best way to achieve total annihilation would be to confirm the acceptability of the PuO_2 - ZrO_2 fuel system, based on the earlier Shippingport work. I believe AECL has the capability to fabricate and test such fuel pellets in the Canadian Research reactors on an expedited basis.

WASTE PROCESSING AND WASTE DISPOSAL - The last two sets of questions deal with waste processing and disposal. There would be no difference in substance between the once through mode presently in use in Canada and that which would be required to handle and dispose of the spent fuel from a CANDU operating to annihilate plutonium. The higher burnup and the higher concentration of fission products and actinides would of course have to be accounted for in designing and analyzing the repository performance. This could be done.

DRY RECYCLE - Although not mentioned in your questionnaire, another fuel cycle is currently being developed by AECL and KAERI (Korea Atomic Energy Research Institute) which could also be applied to plutonium disposition. Specifically, KAERI and AECL are in the process of initiating a joint program to demonstrate the technology of The Dry Recycle of spent LWR fuel for direct use in CANDU reactors. A recent study by INEL[3] has been sent to Dr. Pigford of the National Academy of

Sciences. In this application, the plutonium would be mixed with declad LWR spent fuel and recycled into either a PWR or a CANDU reactor. This cycle would not annihilate all of the plutonium; in fact it would generate new plutonium while the old material is destroyed. However, it does render the plutonium useless for weapons programs, and it has the added advantage of reducing the quantity of civilian spent fuel, and at the same time reducing the concentration of Technetium and Radioiodine (per KWHR) in a repository, thereby reducing the ultimate hazard of HLW disposal. You should be made aware that the current development program being planned by KAERI and AECL calls for the full size demonstration of this technology in about 7 years.

I hope this brief note is helpful to you and the Academy in your study. We believe that the CANDU technology could be extremely valuable as a tool to help eliminate excess plutonium and at the same time generate safe and economic electricity. Please let us know how you would like to follow up on this matter.

Sincerely

Herbert Feinroth

Herbert Feinroth
Consultant to AECLT

REFERENCES:

1. "A Catalogue of Advanced Fuel Cycles in CANDU-PHW Reactors", Veeder and Didsbury, AECL-8641, June 1985

2. "The Irradiation Behavior of ZrO_2 -UO_2 Fuels," Berman and Bleiberg WAPD-264, October 1962

3. "Recycling of Nuclear Spent Fuel with AIROX Processing," Majumdar, et.al. DOE/ID-10423, December, 1992

Appendix C

Accelerator-Based Conversion System Sponsor Response

Duane Hanson
INEL

We are faxing you two additional documents that might be of use to you
In general, we are not able to give you cost information, since we do not have an integrated program for the development of all of the ABC system technologies. In addition, we do not have a system design at a significant enough level of detail to develop a total system cost. The development of a system design with enough fidelity to define the technology requirements, validate the system interface requirements, and provide the basis for a credible system cost estimate is one of the priority efforts that needs to be undertaken.
The information we sent you earlier shows the burnout of both total plutonium and of Pu-239 in the system. Based on an early system requirements, we used 40 years as our burnout time. This requires a system that has approximately half the power of our reference system. If it was desirable to burn the inventory in 20 years, we could do that with a system that is similar in size to our reference system.

We can handle any alloying material that is in US weapons in our fuel preparation and feed system. The americium would be feed directly into the system and burned also. There is no negative impact on the ABC system due to the nuclear weapons material form.

If you need any additional information, please give me a call.

TJ Trapp

DEVELOPMENT NEEDS FOR AQUEOUS-BASED CHEMICAL SEPARATIONS IN SUPPORT OF ACCELERATOR-DRIVEN TRANSMUTATION SYSTEMS

A. General Comments

Aqueous separations involving actinides and fission products can draw upon a wealth of literature developed as a result of decades of reprocessing throughout the world. Typical reprocessing plants have annual capacities of hundreds of tons of spent fuel. Accelerator transmutation systems would require significantly smaller plant capacities for their internal chemical separations. Because these are fluid-fueled systems there is no decladding required, and the removal of the matrix material (D_2O) is very straightforward. On the other hand, the ATW system must process relatively short-cooled material from the blanket while reprocessing plants typically deal with spent fuel that has been cooled for years. The processing of short-cooled material has been demonstrated at facilities such as the ORNL TRU spearations facility but presents chemical engineering challenges.

The development needs outlined here concentrate primarily on basic chemistry needs associated with the separations components that are listed below. The chemical engineering issues (equipment layout, sizing, control systems, etc) needed to make these processes work on a plant scale have not yet been explicitly addressed, although basic flowsheet components have been chosen, in many cases, based on technologies used in plant-scale environments. However what can be said now is that the chemical plant will be a highly shielded, remotely operated and maintained facility.

For example, we have proposed an aqueous actinide processing flowsheet that incorporates, to the extent possible, unit operations that have been demonstrated at or near such plant scales. Nevertheless, these unit operations must be demonstrated under the conditions as close as possible to those envisioned in the ATW system, and, eventually, it will be necessary to demonstrate the integrated flowsheet at pilot scale. Where possible, each unit operation would be backed up with alternative approaches, and newer technologies that could improve the overall performance should be evaluated. It should be emphasized that because the program is in an early stage of development, we have fallen back on proven technologies in an attempt to illustrate feasibility, not optimum performance.

B. The Base-Case Aqueous Flowsheet

This flowsheet is documented in the Los Alamos report, LA-UR-92-63, " Baseline Actinide Blanket Processing for the Accelerator Transmutation of Waste (ATW) Program."

One concern in developing the actinide processing flowsheet has been the effect of radiolysis due to the short cooling times prior to processing. A number of campaigns at the Oak Ridge TRU processing facility have involved processing of solutions having specific heats of 10-30 watts/L. The current flowsheet ensures an upper limit of 10 watts/L for all processing solutions. Recent Los Alamos tests using ^{238}Pu solutions have verified that the processing of such solutions is feasible (see Los Alamos report LA-UR-92-63)

Individual unit operations for actinide separations are discussed with an emphasis on development needs.

C. Actinide Separation Unit Operations: Development Needs

1. Solvent Removal

The removal of the D_2O solvent by evaporation should be straightforward. This solvent can be recycled to the transmuter, but tritium must be periodically removed. The same is true of the blanket moderator. The tritium removal system could be similar to those used with CANDU reactors. Volatile fission products, such as krypton, xenon, and iodine that are not trapped in the slurry particles must be scrubbed from the evaporator off-gas. The method for trapping the volatiles has not yet been determined, but the same procedures used in reprocessing plants should be acceptable. The amounts of these volatile fission products will depend on the particle size. We have not yet specified particle size or morphology although beginning assessments based on factors such as erosion, settling, and fission product trapping are beginning. Any nonvolatile fission products that escape the slurry particles will remain with the slurry after evaporation.

2. Oxide Dissolution

The oxide slurry should be easily dissolved in concentrated nitric acid, but any heels from the dissolution could be saved for more aggressive dissolution techniques. This operation is commonly done at the Los Alamos Plutonium Facility (see Los Alamos report, LA-3542 , "Plutonium Processing at Los Alamos Scientific Laboratory".) We have proposed the use of ozone to aid the dissolution and drive off ruthenium as the tetroxide. In addition to the RuO_4, the off-gas will contain Kr, Xe, I_2, Br_2, and NOx. Conventional techniques used in reprocessing plants for scrubbing the off-gas should be acceptable, but our system must reclaim and purify the iodine for transmutation. There are many other steps in the flowsheet that also produce NOx off-gases, and these will need to be scrubbed with provisions for nitric acid recycle. Significant engineering will be required in this area. Next, valence adjustment of the plutonium and neptunium to the +4 oxidation state is required for their subsequent removal. This is easily accomplished in the case of plutonium but may be more difficult in the case of neptunium. Again, development and demonstration will be required.

3. Liquid Anion Exchange

In the next step in the flowsheet, aliquat 336 is used to extract Pu, Np, Tc, and Pd. The use of such amines as extractants has been demonstrated for spent fuel on a large scale in the EUREX process developed in France and has been used for large-scale scrap recovery at Oak Ridge National Laboratory. For the particular case of aliquat 336, we have measured the k_ds for Pu Tc, and Pd. This work must be extended to Np with particular attention to the effect of oxidation state. We have observed third-phase formation under some circumstances, and further investigation may be required to find conditions that prevent third-phase or solids formation. The current flowsheet calls for the selective stripping of Pu and Np followed by the stripping of Tc and Pd. We will explore the possibility of stripping Tc and Pd separately. We also will examine other amines. Although aliquat 336 has good radiation stability (better than TBP), we believe that pyridinium-based analogs may be even more radiation resistant.

4. Pu/Np Thermal Denitration

The nitrate solution of plutonium and neptunium from liquid anion exchange will be spray calcined to produce oxide particles, slurried with D_2O, and returned to the transmuter. It is not known whether this process can produce acceptable particle size and morphology. Experience and data from the Oak Ridge slurry reactor experiments need to be assessed along with initiation of slurry flow experiments. Sol-gel processes could be a more feasible method for meeting slurry characterisitics requirements that warrent investigation.

5. Tc/Pd Separation

If we are not able to separately strip Tc and Pd from the liquid ion exchanger, they must be separated to produce a pure Tc feed stream to the transmuter. We have proposed a thermal denitration of the strip solution, perhaps in the presence of oxygen, to drive the Tc off as gaseous Tc_2O_7. This process is one of the few in our flowsheet that has not been demonstrated at large scale. Because the PdO heel will be sent to waste, it is crucial that it be relatively free of Tc. Experimental data is needed regarding separation factors for this process.

6. Formic Acid Denitration

The raffinate from the liquid ion exchange must be adjusted to a pH of 3 prior to the reverse-TALSPEAK extraction step. This is achieved using a formic acid denitration process. Although this process has been carried out at plant scale at many facilities, operation of the process in the radiation environment of ATW must be demonstrated.

Our initial material balance indicates that this step produces one of the largest waste streams in the flowsheet. It produces copious amounts of water that must be scrupulously cleaned of transuranics, RCRA metals, and anions such as nitrate so as to allow discharge to the environment. An alternative to this step, or the TALSPEAK process that requires the high pH, would have substantial payoff.

7. Reverse TALSPEAK

The reverse TALSPEAK process is used to extract the trivalent actinides and lanthanides. The actinides are then selectively back extracted, thermally denitrated to oxide (see discussion above regarding particle size control, etc.), and recycled to the transmuter. The lanthanides are then stripped, thermally denitrated, and sent to waste. The lanthanide fraction may contain some actinides and could be TRU waste. The raffinate from the TALSPEAK process will contain most of the fission products, except volatiles, lanthanides, Ru, Te, and Pd. This solution could be calcined. Development work will be required to find acceptable final waste forms.

The reverse TALSPEAK process has been demonstrated on a reasonably large scale in Sweden as a part of the CTH process. The process is quite sensitive to pH, and careful control is crucial. As mentioned earlier, the flowsheet might be improved substantially with an alternative process to TALSPEAK.

8. The Separation of Other Fission Products for Transmutation

The accelerator-driven system is not theoretically limited with respect to the suite of long-lived fission products that may be transmuted. The base-case system currently is configured to transmute only Tc and I because we believe any transmutation system will need to transmute these fission products as a minimum. Any decision to tackle additional radionuclides should be based on a cost/benefit analysis. If other fission products are targeted for transmutation, they must be separated in the actinide flowsheet. Cesium-135 is probably the next most likely candidate for transmutation. Transmutation of this isotope may require isotope separation from stable ^{133}Cs and the other short-lived isotopes. Among several possible approaches to such isotope separation, the plasma separation process (PSP) appears particularly attractive because Cs has a high vapor pressure and is easily ionized.

There are a number of possibilities for the separation of Cs from other fission products under acidic conditions. Possible approaches include the use of hexacyanoferrate, ammonium phosphomolybdate, zeolites, resorcinol/formaldehyde, crown ethers, or cobalt dicarbollides. Such unit operations could be incorporated into the flowsheet in a number of places, and even if it is decided not to transmute Cs, there may be benefits to separating Cs and Sr early in the flowsheet in order to reduce the radiation for subsequent processing. In addition, overall waste management for the system could be favorably impacted through the segregation of the Cs and Sr. Thus, any development program should include an effort aimed at Cs and Sr separations.

9. Alternative and Improved Processes for the Actinide Flowsheet

The flowsheet described in LA-UR-92-63 is a beginning effort to demonstrate components and performance. It is not optimized and would benefit from investigation and development of improved process components which are discussed here.

(a) Selective inorganic (carbonate/hydroxide) precipitation of actinides and fission products - Use of selective complexations/precipitations by varying the inorganic reagent could provide a rapid separation of actinides from the fission products. Further development includes determination of solubility properties of certain fission products (Mo, Ru, Pd, Cd) and actinides (Am, Cm). Interactive effects arising from mixtures of elements similar to that exiting from the transmuter requires investigation. Determination of separation factors under such mixture conditions is also required.
(c) Alternatives to the reverse TALSPEAK process- There are several possible approaches to the selective separations of trivalent actinides required for the transmutation system. One approach would be to first perform a group separation of the actinides and lanthanides and then follow that up with a separation of the actinides from the lanthanides. In this approach one could evaluate TRUEX or TRUEX alternatives and then perform the much more difficult actinide/lanthanide separations under conditions of reduced radiation and in the absence of many other interfering species. Although TRUEX has undergone substantial testing, difficulties have been identified such as third phase formation and difficult back extraction properties. Alternative extractants such as the diamides being developed in France and carbamoylmethylene phosphonates (CMP) could produce better extraction properties and less waste.
(c) Development of "soft" donor extractants for trivalent actinide/lanthanide separation - Development in this area could eliminate difficulties associated with the current flowsheet (significant process control conditions) or other methods (TRAMEX that uses concentrated salt condition). Sulfur containing donors can provide high separation of trivalent actinides from trivalent lanthanides. These systems require testing to determine their ability to function in processes where the extractant is cycled through many extraction steps. Both chemical and radiolytic stability of these sulfur compounds in the ATW environment require further investigation.
(d) Waste stream cleanup through use of water-soluble chelating polymers - This approach would use such polymers to bind with actinides and fission products for selective removal from aqueous streams by ultrafiltration. Testing of currently available polymer systems on synthetic waste streams will determine whether actinides and selected fission products can be removed to the degree required for discharge of the waste to the environment. Advanced water soluble polymers would also be synthesized and tested. Tests include pH sensitivity, capacity, and back extraction properties.

10. Integrated Flowsheet Testing

The integrated performance of major components of the actinide flowsheet would be tested under cold simulant conditions using the Advanced Testing Line for Actinide Separations (ATLAS facility) at Los Alamos. Use of this facility would also begin the process of component scaleup and sizing investigations. Beginning cold tests would be followed by testing under radiation environments and short-cooling times characteristic of ATW processing. These could be done using the Oak Ridge high-flux reactor (HFIR) for sample irradition coupled with the hot-cell processing lines associated with the ORNL TRU Separations Facility.

D. Fission Product Separations

1. Tc/Ru Separations

The base-case aqueous system includes a separate flow loop containing a D_2O solution of dissolved $^7LiTcO_4$ where Tc is transmuted. This loop is fed from any external waste (e.g., Tc from spent fuel) as well as the internally generated Tc from actinide burning. We envision pulling of a slip stream from this flow loop where the Ru transmutation product would be separated from the TcO_4^- solution before the TcO_4^- solution is returned to the transmuter. This separation is accomplished by sparging the solution with ozone to volatilize the Ru as RuO_4. The RuO_4 is trapped in a sodium hydroxide solution where it is converted to sodium ruthenate or perruthenate.

Although initial experimental validations of this process have been carried out at Los Alamos, it has not been demonstrated at large scale, and the development plan must include such demonstrations. Alternative approaches to the Tc/Ru have been examined including ion exchange, precipitation, magnetic separation, fluoride volatility, and solvent extraction. Several of these alternative approaches should be evaluated as a part of a comprehensive development plan.

2. Separations for Iodine Transmutation

We have proposed the transmutation of iodine using solid iodine as the target material. The gaseous xenon formed during transmutation would be allowed to diffuse from the solid and pass through a cryogenic trap to remove any untransmuted iodine before release. Preliminary calculations indicate that the low thermal conductivity of iodine may cause an unacceptable temperature rise in the iodine. A development effort should address this problem and develop alternative transmutation schemes. Other approaches include transmutation of an iodide salt or various D_2O solutions of iodine compounds.

3. Separations for Cesium and Other Long-Lived Fission Products

The current base-case aqueous system is not designed to transmute fission products other than Tc and I. Development plans must include evaluations of possible approaches to the transmutation of these other long-lived fission products. For example, Cs might be transmuted as a D_2O solution of $^{133}CsOD$. Under the proper conditions, the barium transmutation product could be precipitated from solution and filtered. As mentioned above, decisions regarding transmutation of these other fission products will be based on cost/benefit analyses. Such analyses will require scoping studies of possible approaches to the required separations.

Attachment 2

Development Needs for Aqueous Based Separations for ATW
Summary Table

	Status	Issue	Development
Solvent Removal			
Volatile trapping	Method requires explicit identification	None	Apply reprocessing plant procedures
Fission product escape into solvent	Beginning slurry particle assessment	Particle size and morphology requires definition	ORNL slurry reactor database examination, calculational assessment, slurry particle creation and testing

Oxide Dissolution			
Dissolution	Done routinely at the Los Alamos Plutonium Facility	None	
Ruthenium volatilization	Proposed		Needs demonstration
Scrubbing of off gases	Methods proposed based on reprocessing plant techniques	Iodine must be captured for insertion into transmuter	Needs demonstration
NOx gas trapping and recycle	Program at Pu facility to explore nitric acid recycle		Process demonstration and scaleup
Valence adjustment of neptunium		More difficult as compared with plutonium	Needs demonstration

Liquid Anion Exchange			
Aliquot336 use	Separation factors measured for Pu, Tc, Pd	Neptunium oxidation state, third phase effects, radiation stability	Further demonstration of aliquot needed, examination of other amines

Pu/Np and Am/Cm Thermal Denitration			
Slurry particle size	Spray calcination proposed	Particle size and morphology	ORNL database search and demonstration needed; sol-gel process requires investigation

Tc/Pd Separation			
Volatilization of Tc	Thermal denitration/volatilization proposed	Separation factors need to be determined	Demonstration needed

Formic Acid Denitration			
pH adjustment	Proposed based on plant scale experience	Control in radiation environment, waste volume generation	Demonstration in characteristic radiation environment, identification of alternate processes

Reverse TALSPEAK			
Performance	Demonstrated in Swedish CTH process	pH control and achievable separation factors	Demonstration needed in radiation environment
Waste forms	Specific forms not defined		Identify possible candidate forms and requirements

Other Fission Product Separations			
Candidate identification	Beginning neutronics performance analysis underway; ATW theoretically has suitable neutron economy for transmutation	Cost versus risks and benefits	Need tradeoff analysis to identify costs versus benefits

Actinide Flowsheet Improvement			

Inorganic precipitation methods for actinide/FP separations	Proposed, initial tests at small scale and for few element mixtures	Interactive effects due to mixtures	Needs tests using simulants for representative waste feed
Soft donor extractant development	Separation factors measured for sulfur-based donors	Performance in radiation and material recycle environments	Needs demonstration with representative waste forms and investigation of additional donor candidates
Chelating polymers for waste stream cleanup	Beginning polymers synthesized	Achievable separation factors for actindes and FP	Tests needed on synthetic waste streams to determine pH required, back extraction performance; synthesis of advanced water-soluble chelators

Fission Product Separations			
Tc/Ru	Ozonalysis method tested at small scale		Needs demonstration at large scale plus examination of alternative, backup approaches
I/Xe	Solid target with xenon diffusion	Heat generation in target	Identification of iodide salt and/or D_2O solutions of iodine compounds
Cs/Ba	No real processes yet proposed		Examination of cesium form, investigation of precipitation-based methods for separations

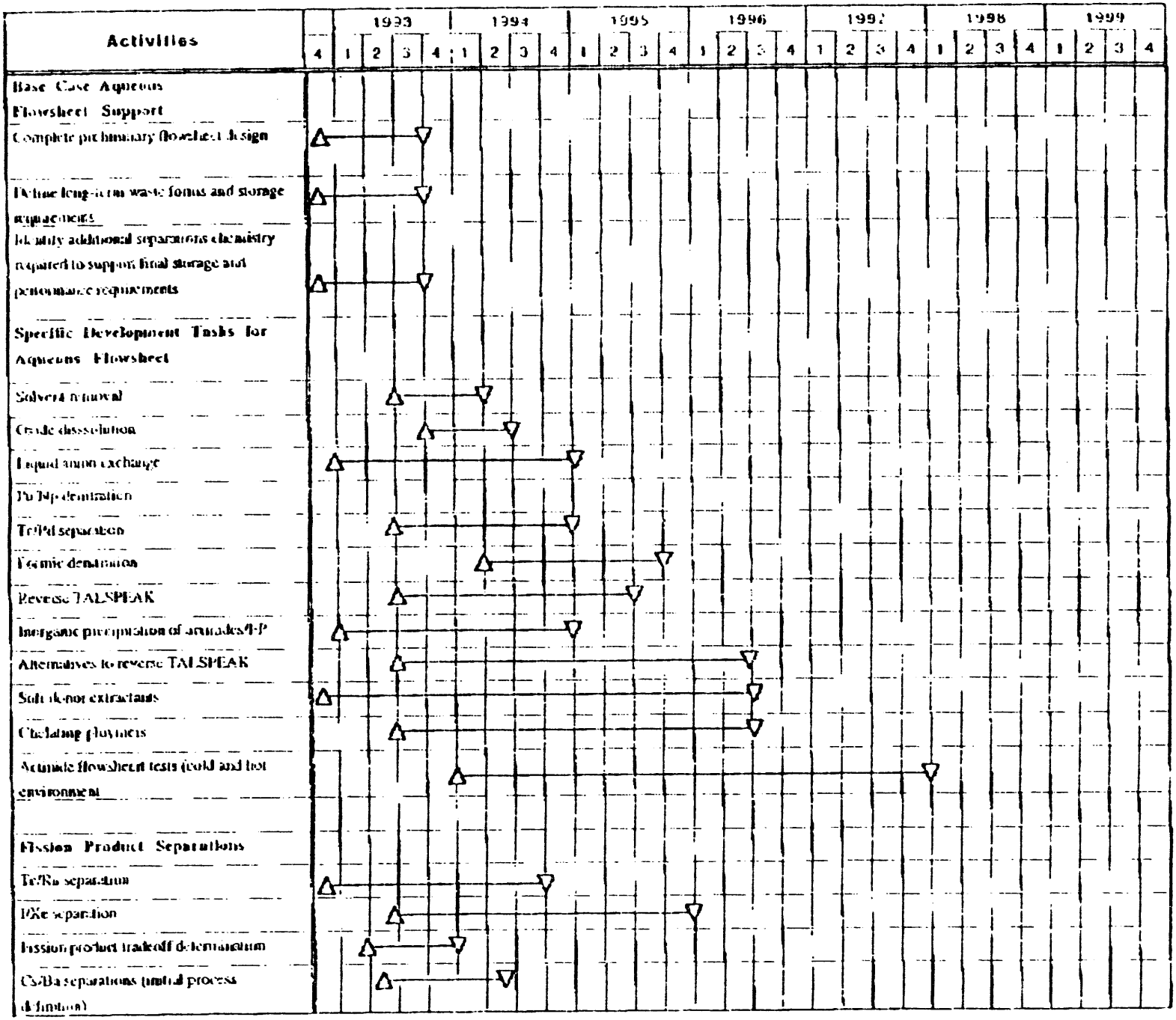
Activities
1993
1994
1995
1996
1997
1998
1999
1 2 3 4
Base Case Aqueous Flowsheet Support
Complete preliminary flowsheet design
Define long-term waste forms and storage requirements
Identify additional separations chemistry required to support final storage and performance requirements
Specific Development Tasks for Aqueous Flowsheet
Solvent removal
Cesium dissolution
Liquid anion exchange
Pu/Np determination
Tc/Pd separation
Technetium determination
Reverse TALSPEAK
Inorganic precipitation of actinides/LP
Alternatives to reverse TALSPEAK
Soft donor extractants
Chelating polymers
Actinide flowsheet tests (cold and hot environment
Fission Product Separations
Tc/Ru separation
I/Xe separation
Fission product tradeoff determination
Cs/Ba separations (initial process definition)

Table II Blanket Neutronic Performance Summary

LWRs Supported (actinide)	2.3
LWRs Supported (Tc and I)	2.3
Average Blanket Flux Level (n/cm^2-s)	1.54×10^{15}
Neutron Multiplication	12.5
k_{eff} for Target/Blanket Assembly	0.92
Slurry Loading (g/l)	500
Actinide Blanket Inventory (kg)	1150
Tc and I Blanket Inventory (kg)	800

The inventory of actinides in the system is larger compared to that (~890 kg) for the 20%-efficiency blanket design. The fission product (Tc-99 and I-129) inventories are larger as well. The higher power per slurry tube that results from the increased slurry loadings require a slurry velocity of ~12 m/s compared with ~7 m/s for the 20%-efficiency design (i.e., for the same flux, slurry velocity is proportional to the slurry loading).

This ATW reference system with four target blanket modules would transmute the actinides, technetium, and iodine in the spent fuel discharged from ~9.2 LWRs based on a LWR annual discharge of 33.3 tonnes (HM) of spent fuel. Using a 40-year ATW system life, a single reference ATW system could transmute these specified waste nuclides in ~12,250 tonnes (HM) of spent fuel. Approximately five such ATW systems would be required to transmute the actinides, technetium, and iodine in the 60,000 tonnes (HM) of spent fuel slated for the geologic repository. The economics of this system is addressed below.

Parametric Costing for the 30%-Thermal-Efficiency ATW System

In order to obtain credible cost information for an ATW system, a conceptual design is needed. In the absence of such conceptual design information, beginning system models and costing relationships have been developed, primarily for system optimization and parameter tradeoffs. These models, as embodied in the ATW Systems Code (ATWSC), also allow initial system costs to be estimated.

The parametric costing of an ATW that burns spent fuel, generates net electric power, and which incorporates the equilibrium neutronics performance described above has been performed using ATWSC. For the purpose of the ATW Systems Code, the key neutronic parameters are neutron multiplication, $k_{eff} = 0.92$, neutron yield per fission, $\nu = 3.02$, and capture-to-fission ratio, $\alpha = 1.62$. Recent changes in the systems analysis include an increase in the plant life time (from 30 years to 40 years) used to estimate present-worth costs and related unit costs and a ~10% reduction in the target neutron yield per incident proton (i.e., differences between an idealized pure-lead target *versus* an engineered

tungsten-lead composite target)

The cost-optimized ATW case shown in Table III is based on supporting 9.2 1000-MW_e LWRs, which for a four-blanket ATW corresponds to ~2.3 LWR/blanket. It is emphasized that these "strawman" parameters are generated from input to ATWSC that has little basis in detailed engineering design. Additionally, in some cases the parameters used in the ATWSC analysis do not correspond exactly with the accelerator and system design presented in April. The beam current and energy (465 mA and 940 MeV, respectively) are system optimized values which maintain the approximate beam power of the 250 mA and 1600 MeV, respectively, reference accelerator design. Also, the system thermal power used includes both the beam power deposited in the target and the heating in the moderator; energy recovery of these system components was not discussed for the earlier ATW system design.

Table III ATW Parameter Summary Used in the ATW Systems Code

Accelerator neutron yield (moles/yr)	3,300
Number of 1,000-MW_e LWRs supported	9.2
Beam current (mA)	465
Beam energy (MeV)	940
Beam power (MW)	437
Accelerator power, (MW)	971
Total thermal power (MW_{th})	8,394
Total electric power (MW_e)	2,518
Net electric power (MW_e)	1,547

Other than that above-noted changes, the economic models used are those described at the April 15th STATS Subpanel meeting (i.e., NUS cost accounts, EEDB costing methodologies, private utility funding groundrules, highly integrated cost estimating relationships for each NUS account, including the chemical-plant equipment, etc.). Specifically, the Cost Estimating Relationship used in ATWSC for the chemical plant equipment account is based on a curve fit to actual and projected capital cost *versus* capacity (tonne (HM)/yr) for a half-dozen fuel reprocessing plants. The costing of the ATW chemical plant equipment uses this cost estimating relationship and the mass of fission-product throughput appropriately converted to "heavy metal equivalent" through an assumed burn-up fraction. This procedure is used as an approximation of the ATW chemical processing cost, and does not specifically include front-end processing (i.e., fuel reprocessing) costs. Generally, processing costs along with the target/blanket capital and operational costs represent areas of large uncertainty and in need of further design-based resolution, although in both chemical plant equipment and target/blanket systems areas, the ATWSC costing is considered to be conservative. The costing for the reference 30%-efficiency ATW system is summarized in Table IV.

Table IV ATW System Costing Summary

Direct Costs	$M	%
Land and Privileges	10	-
Site	110	2
Accelerator	1810	36
Target-Blanket Systems	1550	31
Turbine Plant Equipment	610	12
Reactor Plant Equipment	240	5
Miscellaneous Plant Equipment	120	2
Chemical Plant Equipment	530	12
Total Direct Cost	4990	100
Total Cost (Direct plus Indirect)	8430	169
Annual Charges	**$M/yr**	
Capital	350	
Operating	250	
Total Annual Charges	600	
Present Worth of Charges	**$B**	
Total Capital	14	
Total Operating and Maintenance	10.3	
Decontamination and Decommissioning	.2	
Total Present Worth of Charges	24.5	
Present Worth of Revenues (@ COE = 50 mill/kW$_e$H)	20	
NET COST per ATW Unit	4.5	
Electricity Cost (cost recovery) (mill/kW$_e$h)	54.5	

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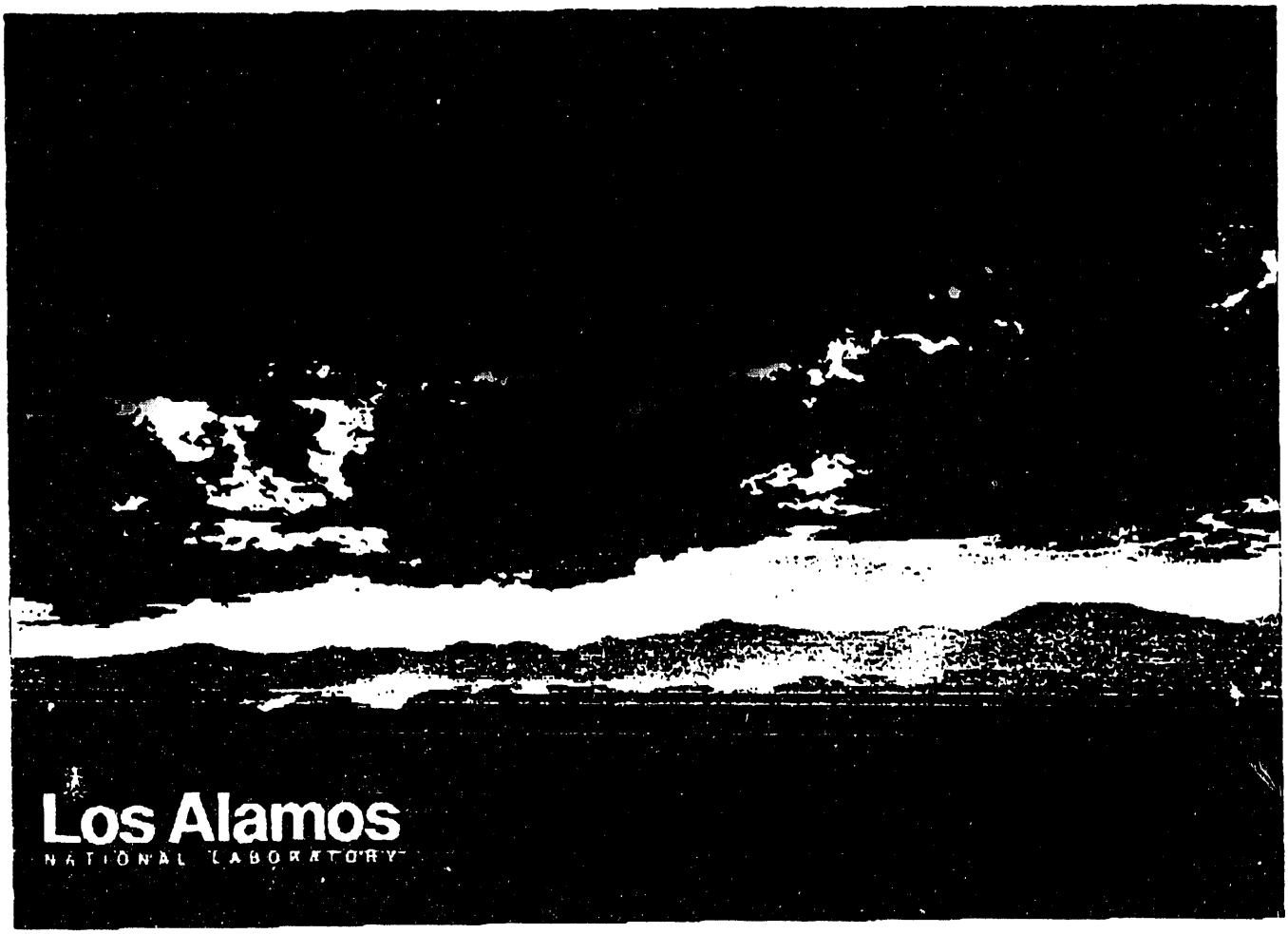
Baseline Actinide Blanket Processing for the Accelerator Transmutation of Waste (ATW) Program



Photograph by Chris J. Lindner

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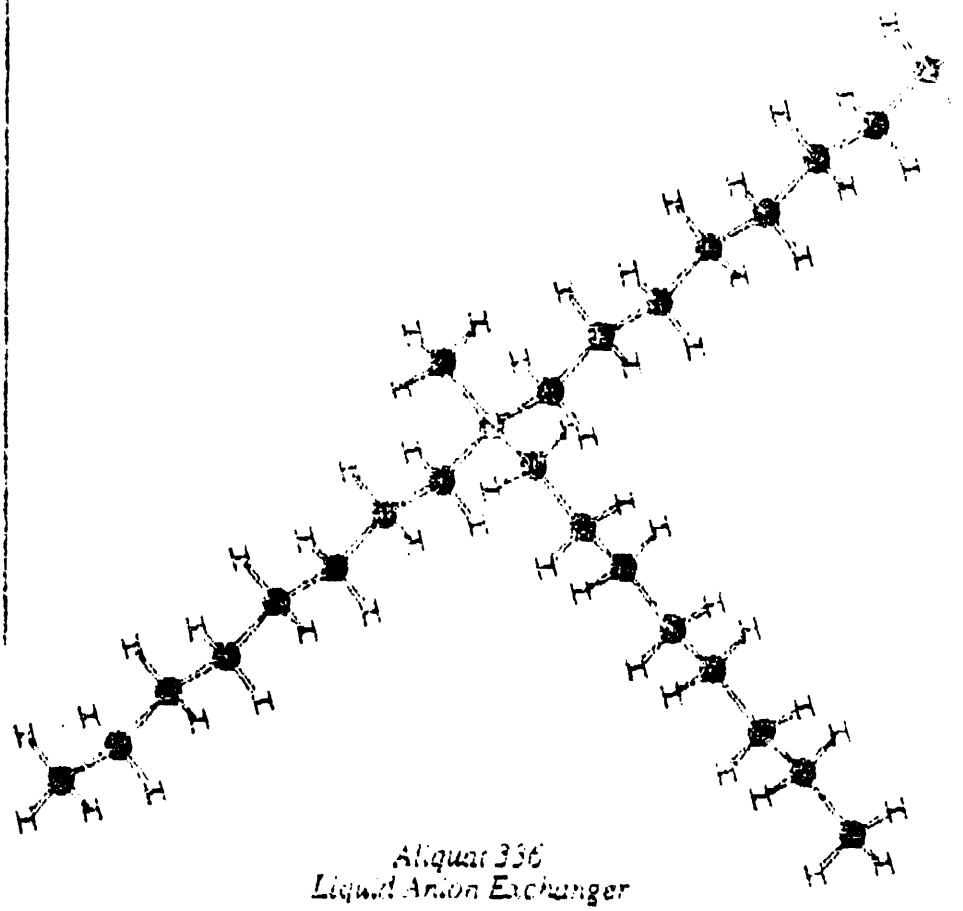
Baseline Actinide Blanket Processing Flowsheet for the Accelerator Transmutation of Waste (ATW) Program



Aliquat 336
Liquid Anion Exchanger

Los Alamos

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Revised: April 1992

Baseline Actinide Blanket Processing Flowsheet for the Accelerator Transmutation of Waste (ATW) Program

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R. Villarreal
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Los Alamos National Laboratory
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S.L. Yarbro, M. Yates,

Los Alamos National Laboratory, P.O. Box 1663, Los Alamos, NM 87545

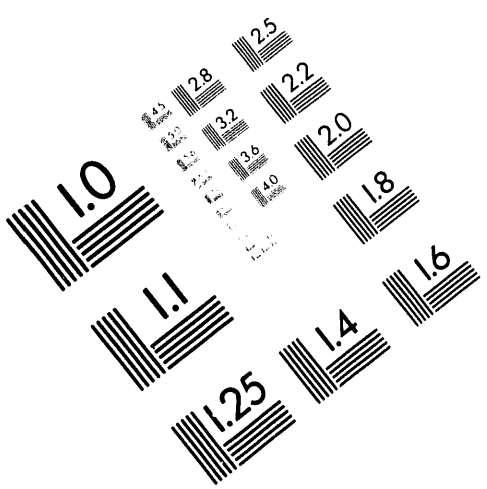
Abstract

A baseline flow sheet for chemical processing of actinide and fission product materials has been developed. Because of the need to keep the actinide concentrations constant and the fission product concentrations low for efficient transmutation, and to minimize the actinide inventory in storage after transmutation, it is necessary to process the material within the blanket soon after discharge from the transmuter. The high radiation and thermal load from the irradiated material places severe constraints on the processes that can be used. After dissolution in nitric acid, a radiation-stable and selective liquid anion exchanger (Aliquat 336) is proposed to separate neptunium, plutonium, technetium and palladium from the bulk of the process stream. A reverse-TALSPEAK process employing a liquid cation exchanger (di-2-ethylhexyl phosphoric acid) will separate the trivalent actinides from the remaining fission products. The proposed processes were chosen as the baseline because they have been successfully tested with high-level radioactive wastes or fuels in gram to kilogram quantities. These steps but potentially improved processing steps are discussed as advanced options relative to the baseline case.

Introduction

To establish a high performance level in the blanket, a constant fissile concentration with a low neutron poison inventory must be maintained. This requires that the purified plutonium and neptunium be returned to the blanket with a short five day cool-down time. The initial processing step must be robust and selective for plutonium and neptunium. To accomplish this, a liquid anion exchange separation using a quarternary amine was chosen. It has the features of high selectivity for plutonium and neptunium, very low affinity for typical neutron poisons and the extractant is more radiation stable than tributyl phosphate (PUREX) solvents. Also, the degradation products are weaker extractants than the original ligand and therefore do not extract fission products such as zirconium. Furthermore, the thermal decomposition products of the extractant are non-radioactive gases and do not add to the waste. A further advantage of this process is that technetium produced during fission is well-extracted and can be easily sent to the technetium transmutation loop.

The americium, curium, lanthanides and higher actinides produced during transmutation will be processed to separate the lanthanides, which are neutron poisons, from the transmutable transplutonium elements. Because of the high heat produced from the decay of fission product lanthanides, this fraction will be separated from the plutonium and neptunium and cooled for an additional 90 days. At this point, a reverse TALSPEAK process will be used to separate the lanthanides from the transplutonium elements. Thermal



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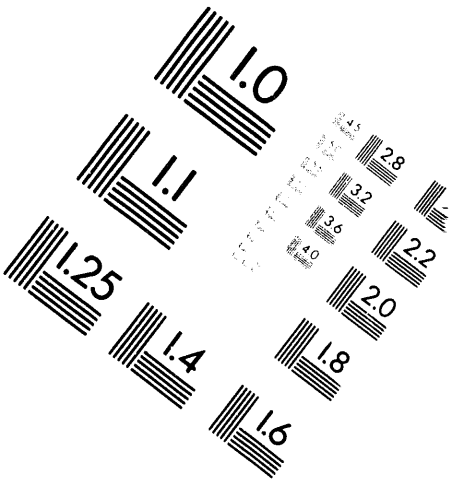

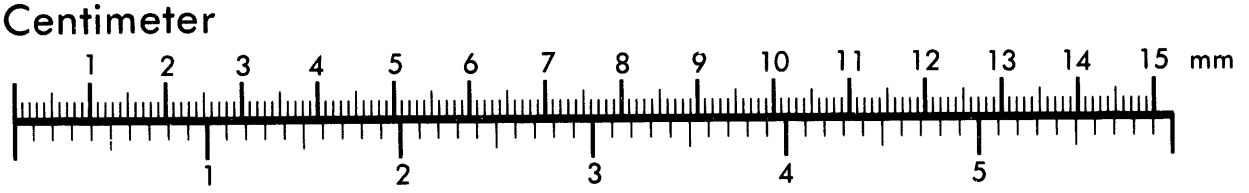

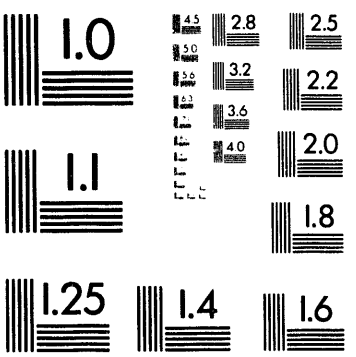

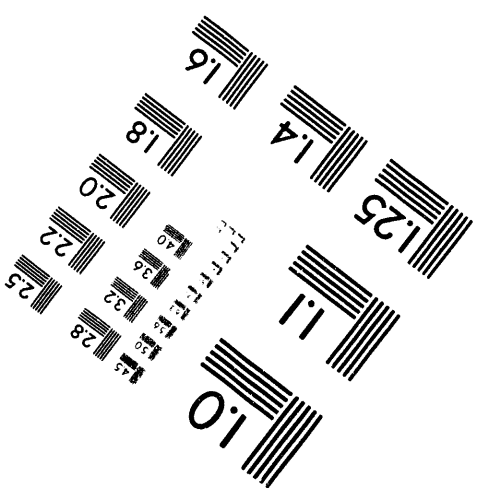

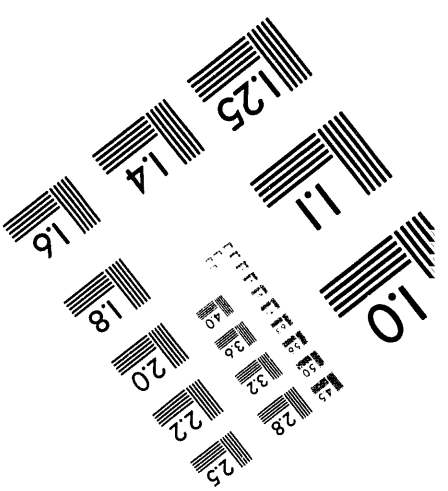

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denitration to produce product oxide for return to the transmuter and spray calcination for the waste streams were chosen because they do not add inerts to the final solid waste stream.

All of the components of the base case flowsheets have been successfully tested with either high-level wastes or spent fuels in gram to kilogram quantities. All of the features of the flowsheet have been documented in the literature.

The high radiation and thermal load of the irradiated material places severe constraints on processes that can be used. A baseline flow sheet (Appendix I) has been developed to meet these goals. It was chosen as the baseline because all the components have been successfully tested with high-level radioactive wastes or fuels in gram to kilogram quantities. All of the flowsheet steps have been documented in the literature. Less tested but potentially improved processing steps are proposed as alternatives to the baseline case.

The baseline actinide blanket feed can consist of ~400 kg of actinides as a low-fired-oxide slurry in "heavy water" or D_2O. After a residence time of 15 days in the transmuter the oxide slurry will be removed and processed after five days cool-down time to recover the plutonium, neptunium, and technetium for return to the transmuter. This process may also be done as a 7% per day slipstream on the blanket slurry. The raffinate containing the remaining radionuclides will be stored for 90 days before it is further processed for trivalent actinide recovery, mostly Cm and Am, which will also be returned to the transmuter. Because our goal is transmutation and not fuel fabrication, the purity specifications of the material that is sent to the transmuter are less stringent (Stoller et al. 1961), requiring different criteria for metal ion separation factors. The fission product impurities are maintained at a level where they do not interfere substantially with the neutron economy of the transmuter. The overall goal of the processing system is to minimize TRU waste streams and return all actinides to the transmuter. The details of the flow sheet are discussed below.

Dissolution/off-gas treatment

The transmutation must be performed in D_2O rather than normal water to more efficiently thermalize the neutron flux. Slurry feeds allow easy removal of actinides and fission products from the D_2O carrier. Using oxide slurries avoids using acid solutions in the transmuter eliminating the production of neutron activation products and redox products of the acid anion. This minimizes the breakdown of the D_2O in the blanket. The oxide slurry concentration of about 50-75 g/L is more readily handled than the 1500 g/L ThO_2 slurries used by ORNL researchers on early Th-U breeder fuel cycles (Lane 1960). The slurry preparation will involve spray calcining for particle size control and slurry stability (Long 1978). Further particle optimization, if warranted, could be

accomplished with a sol gel process (Mannice et al. 1969), realizing that more waste will be generated. As most of the fission product oxides are insoluble, the majority of the fission products are expected to remain intact in the particles and can be filtered from the D_2O. Those that are anticipated to be released from the slurry are the gaseous products such as Xe, Kr, I_2, Br_2, and RuO_4, which will be handled by an off-gas system (Benedict 1981, McKay et. al.1984). The bulk of the D_2O will be evaporated from the oxide and recycled back to the manufacturer. The concentrated slurry will be evaporated several times with fresh D_2O and helium sparged for complete tritium removal. At periodic intervals over the lifetime of the process, the D_2O will have to be replaced or purified from tritium. There is a transition from a D_2O slurry system to H_2O processing at this point. The wet slurry will be dissolved in concentrated HNO_3 without the aid of HF (Lerch 1979; Harvey 1947; Cleveland 1979, p. 573). Further outgassing is expected and will be vented to a gas-handling system. RuO_4 volatilization will be enhanced by O_3 sparging during dissolution (Stoller 1961; Baetsle et al. 1981) and collected. Because of the low-fired nature of the slurry, the majority of the material will be easily dissolved and any residue will be filtered and saved for more stringent dissolution techniques (Dahlby et al. 1975; Weigel et al. 1986). The acid is adjusted to 2 M HNO_3 and the neptunium and plutonium valences adjusted with $NaNO_2$. The total volume of the solution is scaled to approximately 10 watts/L to allow manageable heat loadings for the solutions. Processing experience at the ORNL TRU Processing Facility and recent Los Alamos experience with concentrated ^{238}Pu solutions support this heat loading value (Lueze 1962; Yarbro, 1993). In recent Los Alamos tests, a 10 watt/L solution of ^{238}Pu in 2 M HNO_3 was prepared. The acid concentration and Pu valence was measured every day for seven days with no detectable changes occurring. Therefore, based on ORIGEN2 calculations of the specific decay heat (Figure 1) this volume adjustment gives a solution containing approximately 1 g/L of actinides that is reasonable to process for Pu/Np/Tc recovery.

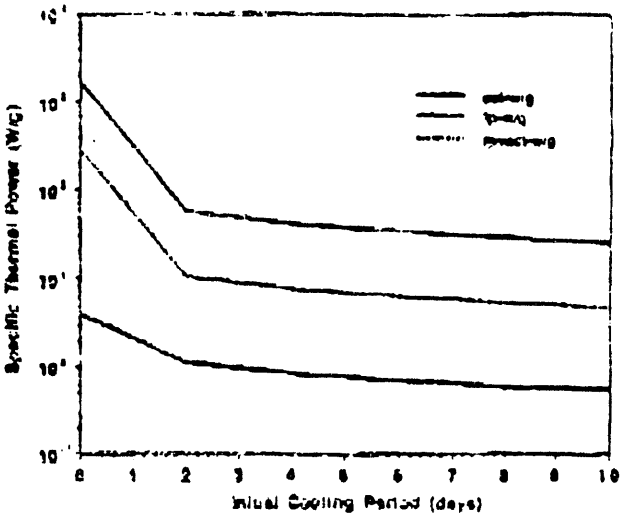

Figure 1. ORIGEN2 Calculation Results for the Specific Decay Heat of the Np/Pu Loop after Transmutation

Pu/Np/Tc Recovery

The Pu/Np/Tc recovery system employs a liquid anion exchanger (Aliquat

336) with centrifugal contactors or possibly pulse columns as the baseline process. Centrifugal contactors could minimize solvent contact time with the highly radioactive aqueous phase, thus extending the solvent's lifetime. However, potential third phase or solids formation would reduce their effectiveness. The subsequent extraction and back-extraction processes will use pulse columns as they allow for longer contact time which will facilitate separations. The liquid anion exchanger was chosen because of its stability in the presence of the high radiation fields. It also has high extraction values and selectivities for Pu and Np (Chesne et. al. 1963; Cleveland 1979, p. 220, Cleveland 1979, p. 473; Coleman 1963) over fission products. Aliquat 336 has higher radiation stability than tributylphosphate (TBP), which is used in the PUREX process(Leuze 1969). Since little uranium is produced in the transmuter, Aliquat 336 can readily replace TBP when only Np and Pu require selective removal.

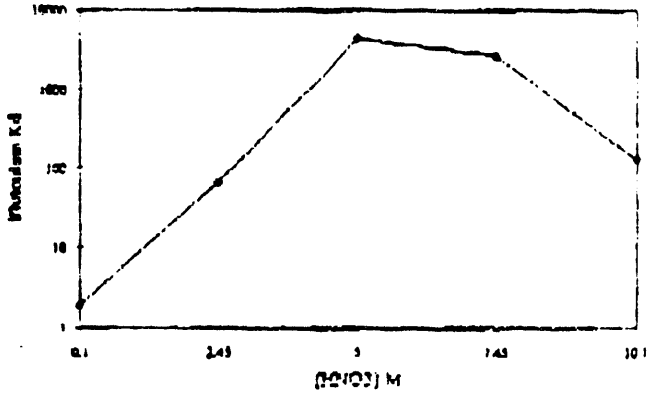

Figure 2. Plutonium Distribution Coefficients in 0.2 M Aliquat 336 (Yarbro 1992)

Assuming that Np and Pu behave similiarly (Maeck et. al. 1961), over 99.9% of Np and Pu can be extracted into the organic phase along with 99.9% of the Tc and Pd (Coleman et al. 1960). Since Pu and Np extract so well the process was sized for Tc removal.

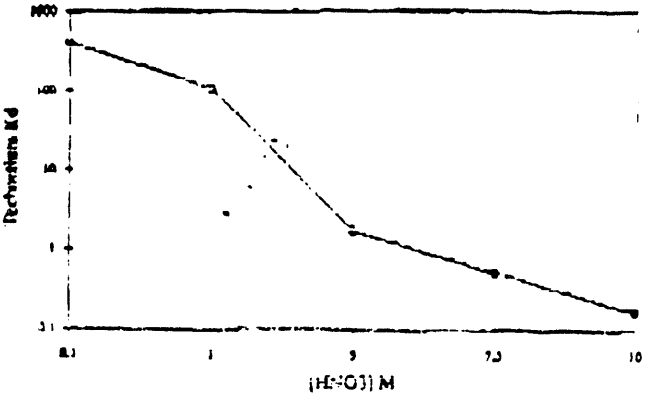

Figure 3. Technetium Distribution Coefficients by Aliquat 336 (Schroeder 1992)

Sizing the column based on the least extractable component ensures that enough stages are available to properly extract the elements for return to the transmuter. The Pu and Np are back-extracted or stripped with 0.05 M HNO_3 leaving Tc and Pd in the organic phase (Cleveland 1979, p. 220). Thermally denitrating (Long 1978) the aqueous strip solution will produce NpO_2 and PuO_2 suitable for return to the transmuter. Small amounts of fission products are expected to be entrained and sent back to the transmuter. This can be tolerated. This separation should give a waste stream that is non-TRU (<100 nCi/g) in Np and Pu. The high acid aqueous stream containing the

remainder of the fission products and trivalent actinides will be stored with external cooling for 90 days to allow for substantial decay to facilitate further actinide separations. The Tc and Pd are stripped with 1 M ammonia solution.

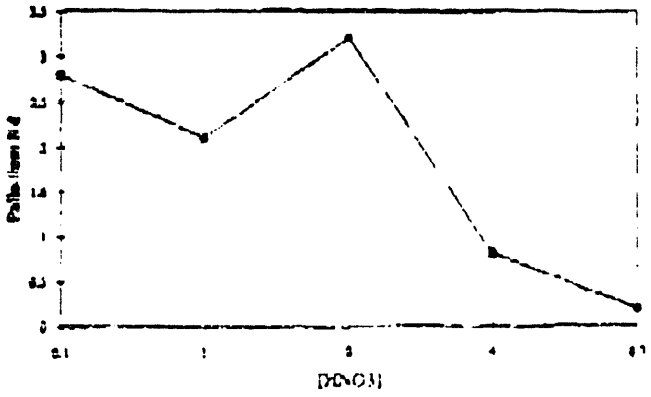

Figure 4. Distribution Coefficients for Palladium for 0.2 M Aliquat 336 in Diisopropylbenzene/5% Octanol (Smith 1992)

This stripping also serves as a extractant cleaning step to remove some organic degradation products. It is expected that substantial degradation will occur at these high radiation fields (Baroncelli et. al. 1963). An additional filtration step for the organic extract may be required as solid polymers have been observed. The aqueous strip is denitrated with volatilization of Tc_2O_7 (Kinkead et al. 1991; Smith et al. 1953) which is collected for Tc recycle to the transmuter. Palladium oxide is the major solid product from this step. This material can be combined or kept for separate disposal. The organic phase will be recycled for reuse in the next batch. Solvent makeup from degradation losses will be necessary. It is also possible that there will be such high losses that all new solvent solutions will be needed.

Cm/Am Recovery

After 90 days to allow for the decay of some of the very radioactive lanthanides and other fission products, the aqueous stream will be processed to recover Am, Cm, and higher actinides. Acid adjustment to pH 3 with 20 M formic acid will be required for proper extraction in the reverse-TALSPEAK extraction step. Because of its low concentration ($<10^{-4}$ M), Zr will not interfere with the TALSPEAK separation process (Bond and Leuze 1980).

The baseline process, reverse-TALSPEAK, was chosen because it is the best system currently demonstrated for radioactive waste treatment (Weaver and Kappelmann 1964; Weaver 1974). It has been demonstrated that Am and Cm can be separated from the lanthanide fission products probably with sufficient purity for reentry into the transmuter. The first step, extraction of the trivalent actinides and lanthanides from the rest of the fission products with 1M HDEHP (di-2-ethylhexylphosphoric acid), is critical because it determines if the aqueous waste stream is TRU or non-TRU. It is this aqueous stream that can be further treated for Cs and Sr recovery if required. This waste stream contains the majority of the fission

products that make up the HLW for short term on-site storage (500 years). In the Swedish CTH-process reasonable separations were accomplished but were not sufficient to make this waste stream non-TRU (Persson et al. 1984). The conditions used in this step were not those that have been reported in other processes, such as the CLEANEX process used at ORNL (Bigelow et al. 1980) that use the same solvent/ligand system. It has been demonstrated that at equilibrium hydrogen ion concentrations of 0.02M, instead of 0.1M, extraction coefficients are greatly improved. The organic-to-aqueous flow ratios were not optimized in the CTH process. With some modifications on the reverse-TALSPEAK as demonstrated by the CTH process and the CLEANEX process, this aqueous stream can be class C waste or better for TRU elements. Greater than 99.9% of the trivalent actinides and lanthanides are initially extracted into the organic phase. The aqueous raffinate stream from the initial extraction will be spray calcined. This is a possible solid waste form for storage in short-term, engineered storage. After the HDEHP is loaded with actinides and lanthanides, the actinides will be selectively back extracted from the organic phase with an aqueous phase containing 1.0 N lactic acid and 0.05 M diethylenetriaminepentaacetic acid (DTPA). Because the radiation levels are so high from the presence of Cm, DTPA and lactic acid will be severely degraded and will not be available for recycle (Bourges et al. 1980). This aqueous back-extraction stream will be directly denitrated to give CmO_2, AmO_2, and higher actinide and lanthanide oxides for recycle to the transmuter. The 1 M HDEHP organic stream from which the actinides have been stripped is contacted with 6M nitric acid to recover the lanthanide fraction. This lanthanide fraction may contain some trivalent actinides and could be TRU waste. Because the acid concentration is high, it will be recovered by distillation. The HDEHP solvent will be washed with 0.5 M ammonium carbonate and recycled after makeup for degradation. The solid waste stream from combining the distillate bottoms and the organic wash will be spray calcined. The Np and Pu transmuter loop will have a residence time of 15 days, whereas the Am and Cm loop will recide in the transmuter for 90 days. This difference will require a separate dissolution for the Am/Cm loop as shown in Appendix 1. This feed will have low levels of Np and Pu and can enter the baseline process at the reverse-TALSPEAK step for processing. The total material balance for the baseline process is given in Appendix 1. It is reported on an elemental weight basis as opposed to an oxide weight basis. After the majority of the material is returned to the transmuter the remainder of the wastes consists of TRU solids and non-TRU liquid or gaseous wastes with the majority being water. The need for rapid processing after transmutation will generate a larger volume of aqueous waste. Water will be recycled to various points within the

reprocessing plant. Gaseous products of thermal denitration (CO_2, N_2) will be filtered and scrubbed as necessary before release.

Materials control and accountability (MCA) is integrated into the process. Fissile material amounts at all points in the entire system at any time should be known for both MCA and criticality purposes. For an aqueous based process system this is particularly easy to accomplish. The current determinations for reactor systems uses manual isotope dilution mass spectrometry (IDMS) techniques. At a variety of points in the processing, online verification of actinide materials will be developed as advanced concepts. Advanced process diagnostics development is desirable.

The final waste forms are yet to be determined. Since there are no licensed waste forms at this time and the current licensing concepts are based on longer-term underground storage requirements (10,000 years) as contrasted to short-term storage (200-1000 years), it is desirable to consider other forms besides vitrification. Calcined forms currently used at the Idaho National Engineering Laboratory (INEL) may be acceptable. Further waste form development is required and should be considered in the context of short-term on-site engineered storage facilities.

Advanced Concepts for Actinide Processing

Molten Salt Slurry

An important advanced concept is the replacement of the D_2O-oxide slurry system with a molten (m.p. 360°C) LiF-BeF_2 salt-oxide-slurry. This modification would enhance the neutron economy through decreased neutron capture and reduced structural requirements on the present design of the transmuter. Less tritium would be produced and perhaps the requirements for such rapid recycle time could be relaxed, lessening the severe demands on shielding and processing. This concept would enhance the energy production aspects of the transmuter.

Many of the actinide oxides are insoluble in molten salts (Porter 1965). It would have to be established that the nongaseous fission products had similar insolubility's or remained intact in the original oxide particles. Thus, ready separation could be accomplished from the molten salt that would be recycled with another batch of actinides back to the transmuter. Though it has been accomplished in the past, improved technology would include advanced physical separation techniques of solids from molten salts at high temperature. The actinide oxide solids would be dissolved in acid and could be processed using the scheme as described in the baseline above.

TRAMEX with high nitrate solutions

If all the actinides could be recovered simultaneously using one solvent system, the baseline flow sheet could be greatly simplified. In high-chloride extraction systems with liquid anion exchangers such as trialkylamines or tetraalkylammonium salts, both the tetra- and trivalent actinides can be extracted and separated from most other fission products including the trivalent lanthanides. Processing in chloride medium is undesirable because of corrosion problems and neutron absorption problems from chloride impurities in the blanket. Studies have indicated separations of actinides from fission products could be accomplished (Lloyd 1963). This idea would require further verification as there is conflicting data in the literature. The advantages are the use of one solvent system that has high stability, removes the presence in the flow sheet of phosphorus-containing extractants that give difficulty in waste management, and reduces the number of required processing steps. The disadvantage is the use and subsequent disposal or recovery of high concentrations of nitrate salts.

Direct Trivalent Actinide Extraction

Another advanced approach over the baseline flowsheet is to find high stability solvent extraction systems that can directly remove trivalent actinides from the trivalent lanthanides and other fission products with high specificity to replace the TALSPEAK process. Studies have demonstrated this actinide-lanthanide separation approach (Musikas et al.1980; Ensor et al. 1988; Hannink et al. 1991). Though these extractants give the desired specificity, they have not had extensive testing for stability or recycle. The advantages of replacing the TALSPEAK process would be a decrease in the number of processing steps with an ensuing waste reduction, especially if the phosphorous containing extractants could be replaced.

Carbonate flow sheet

An alternative flowsheet might be based on selective precipitation of actinide and lanthanide carbonate complexes. This alternative could provide a greatly simplified processing scheme for rapid separation of the bulk of the actinides from the fission products (Kovlev 1955; Martella et al. 1984; Hagan and Miner 1976) for recycle to the transmuter. The strategy relies on the concept of selective precipitation of the smaller impurity fraction (lanthanides) from the bulk of the actinides and fission products. Once the interfering lanthanides have been selectively removed, all the actinides could be recovered simultaneously by a selective carbonate precipitation under different conditions leaving the fission products. The advantages would be more rapid turnaround of processing streams without the accompanying organic solvent degradation and ease of redissolving carbonates. This

would allow for lower process [illegible] for the actinides. A disadvantage would be that further processing would be required for high TRU decontamination factors. But these wastes would be dilute and could be more easily handled after appropriate cool-down times.

Conclusion and Summary

The current baseline flowsheet is feasible and can be acheived with available technology. It has performance factors that make it attractive for supporting transmutation options for waste treatment. Judicious use of spray calcination reduces solid waste volumes and the selective nature of the ion exchange extractants allow for efficient recovery of the targeted radionuclides. Issues, such as trivalent actinide/lathanide separation have been identified and potential solutions have been developed. Also, the flowsheet has the flexibility to be adapted for spent fuel recovery or possibly tank waste remediation, if required.

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Appendix 1

Flowsheets and Material Balance Information

Table 1

Loop	% Fissioned per Pass	Process Time Days	Process Cap L/hr	Heat Load W/L
Np/Pu	9.4	15	1600	10
Am/Cm	8.0	90	1760	1

Summary Processing Information

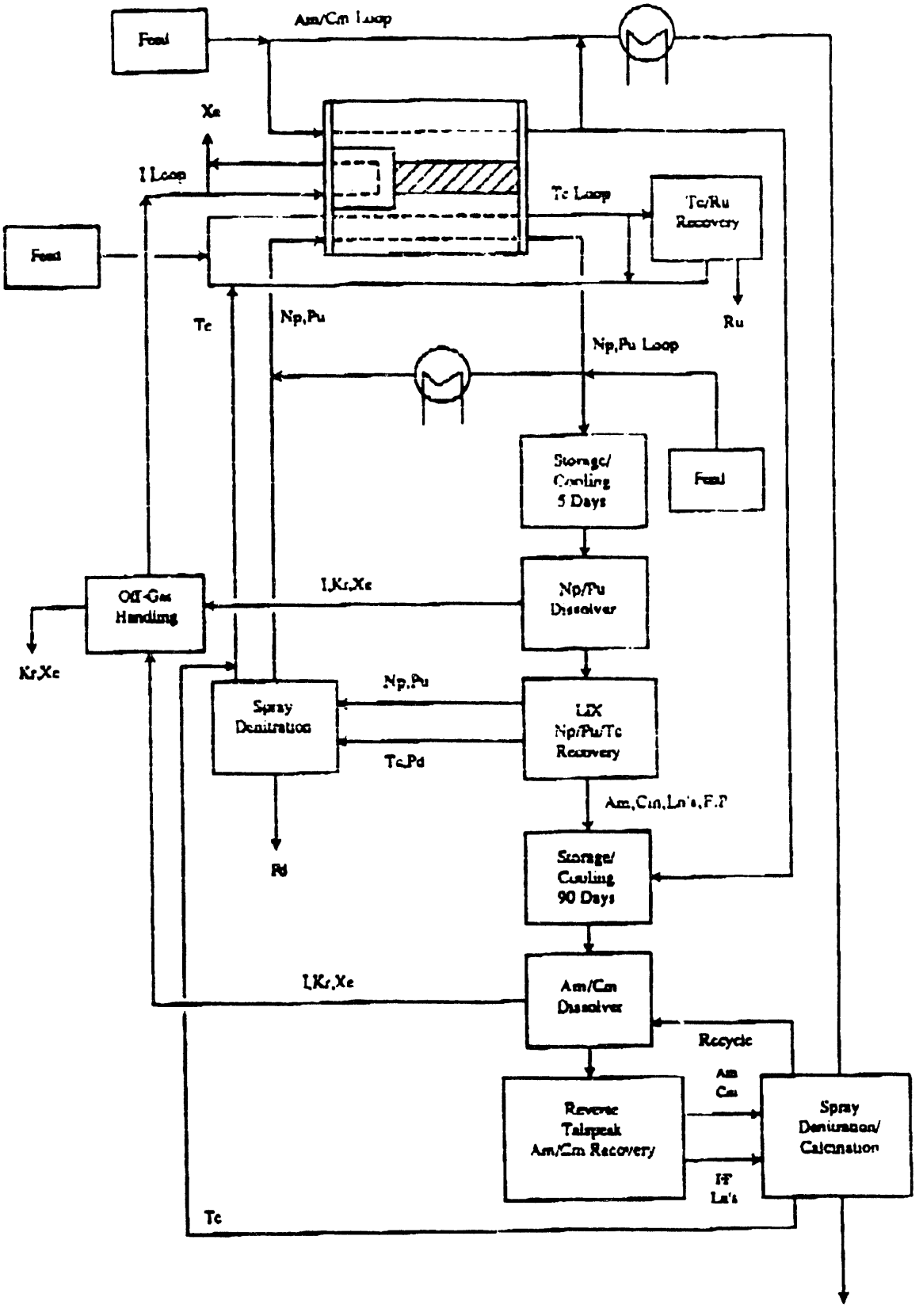

Figure 1. Overall Processing Flow Diagram for the ATW Program

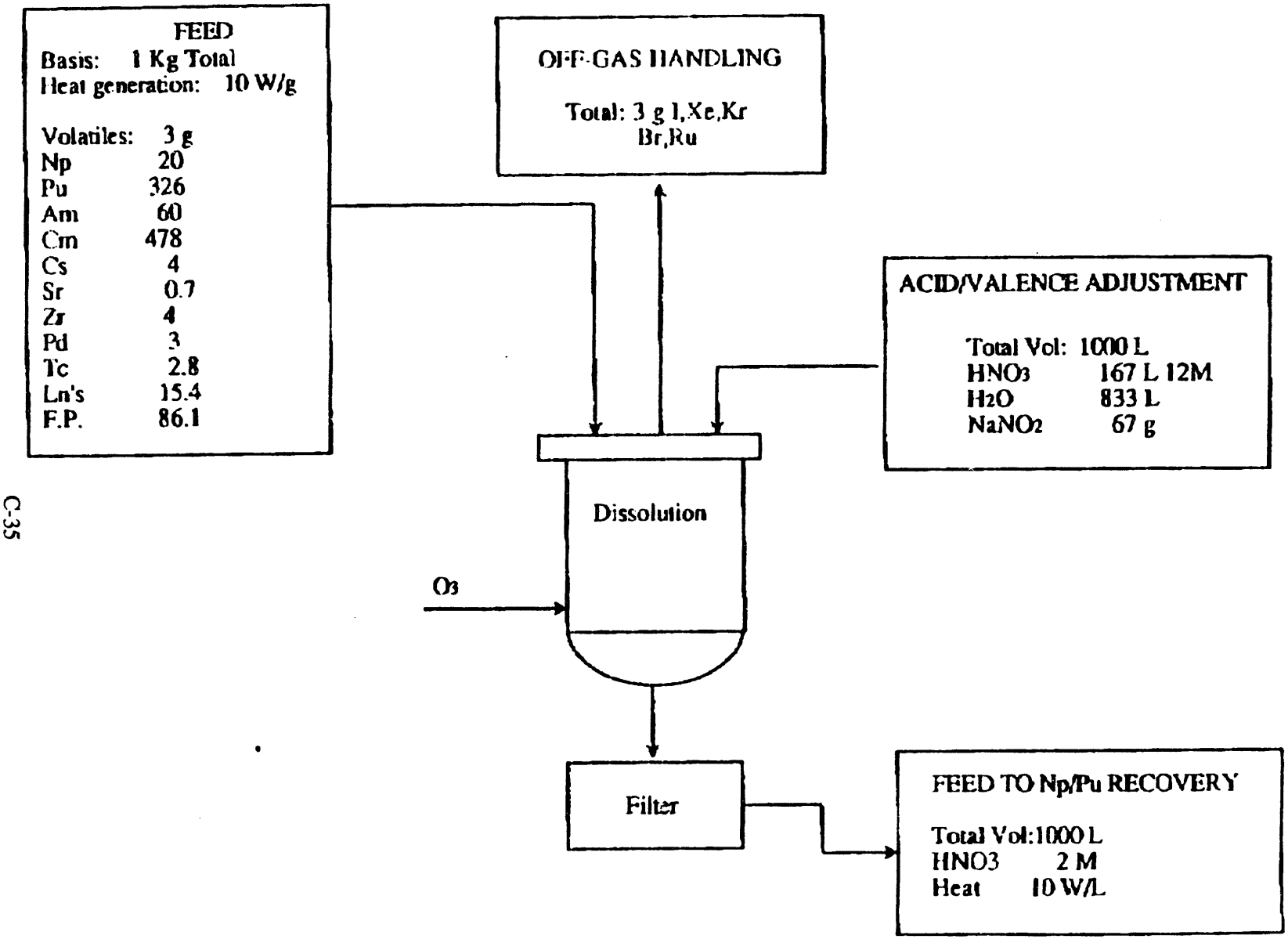

Figure 2. Material Balance for Head-end Dissolution of the Np/Pu Blanket

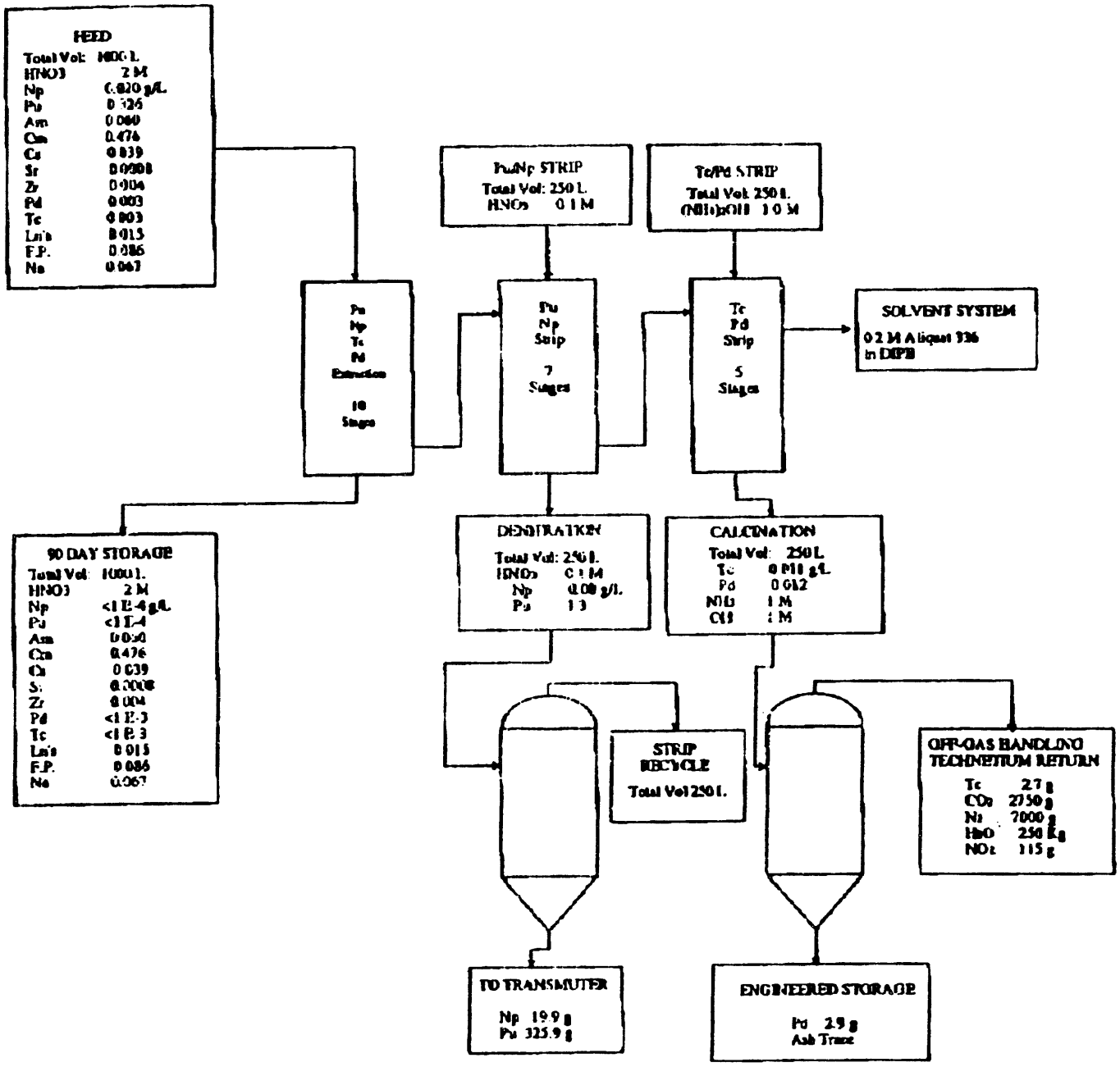

Figure 3. Material Balance Np/Pu Fast Return Loop

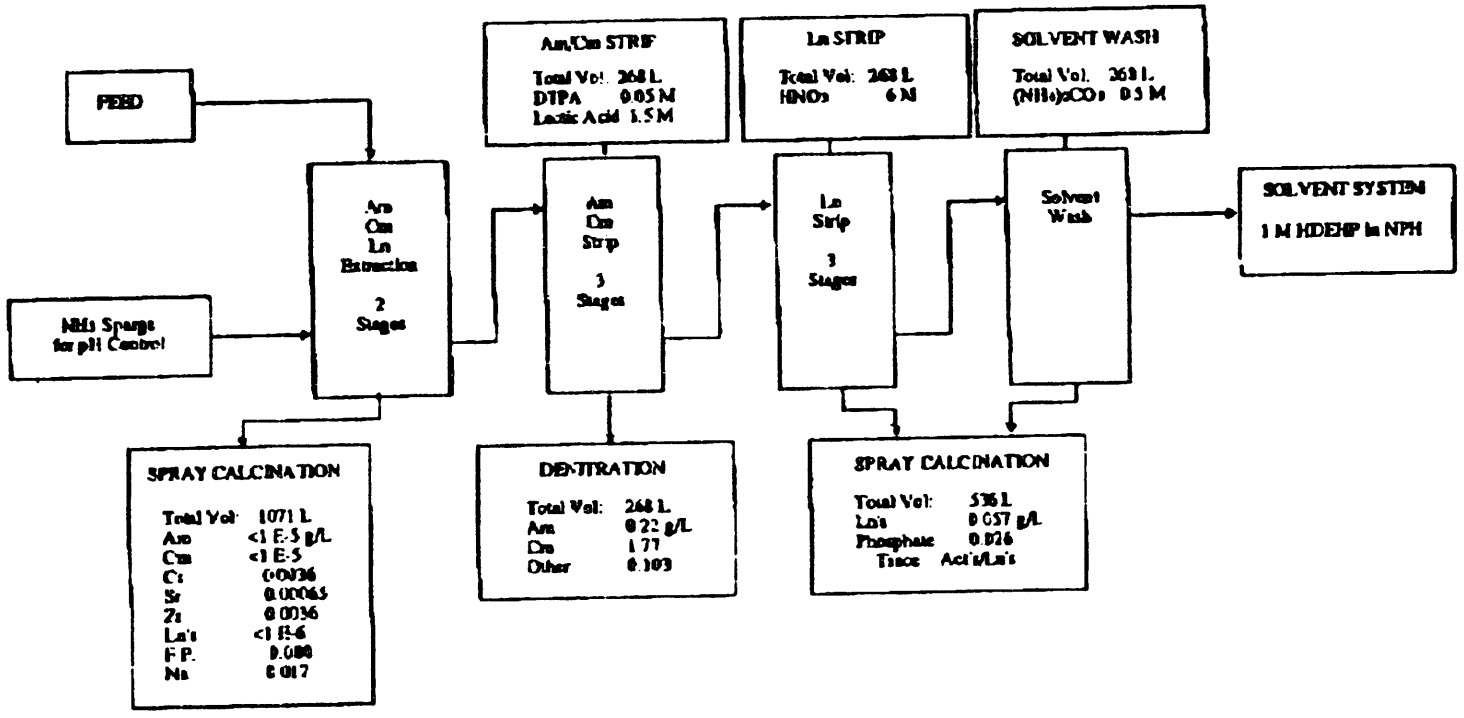

Figure 4. Material Balance for the Actinide/Lanthanide Separation Process

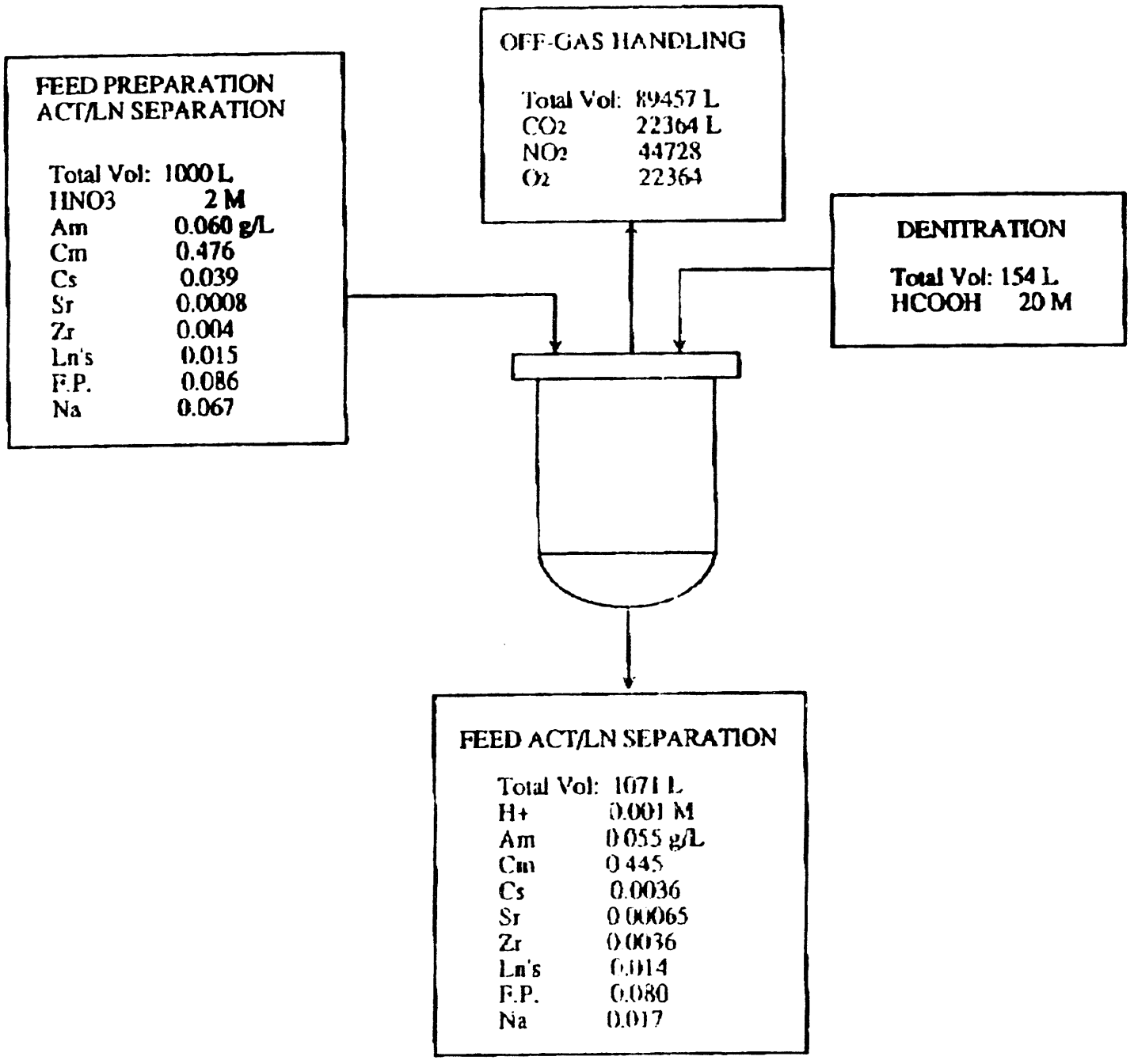

Figure 5. Material Balance for Actinide/Lanthanide Separation Feed Preparation

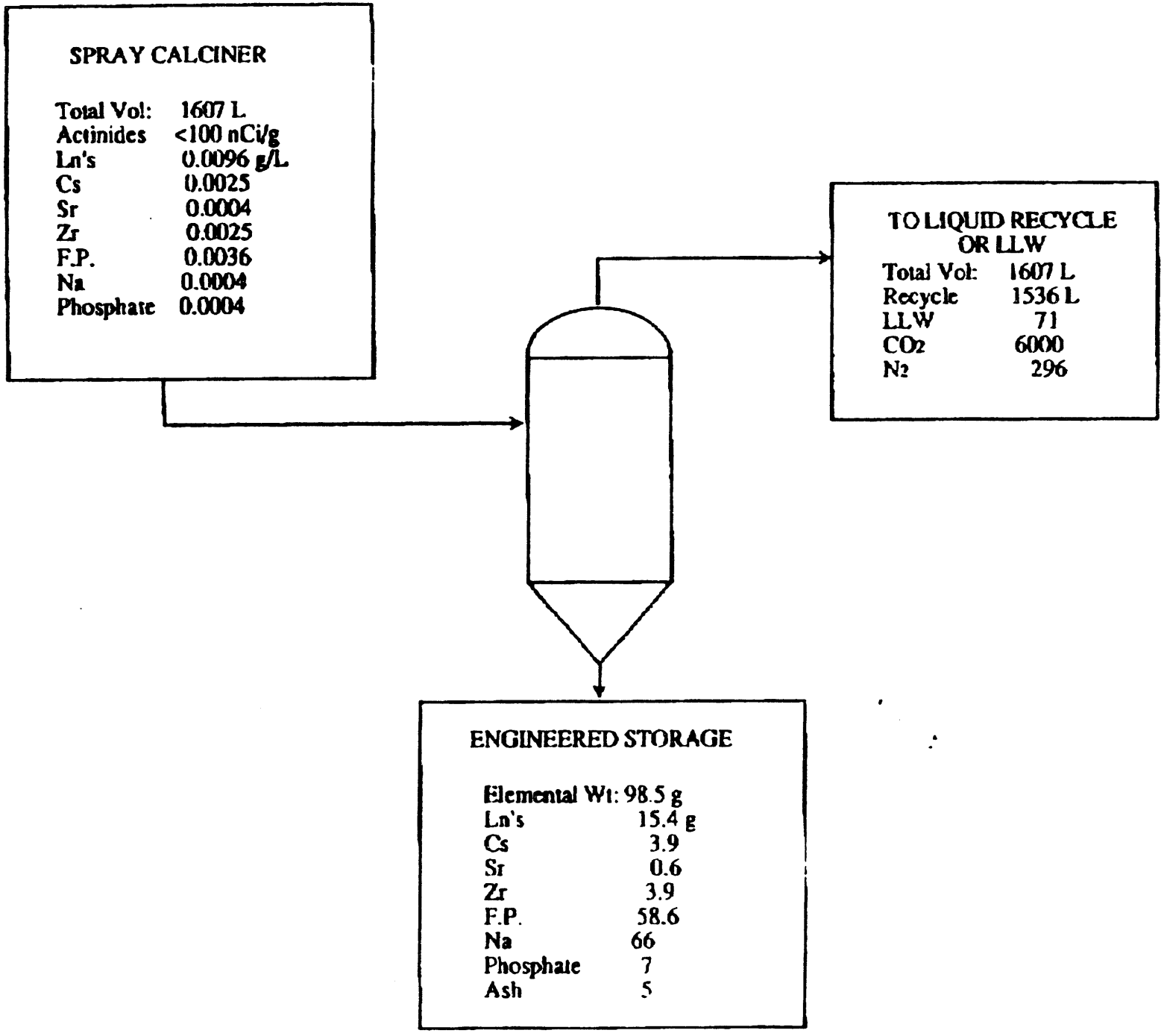

Figure 6. Material Balance for Spray Calcination Waste Treatment

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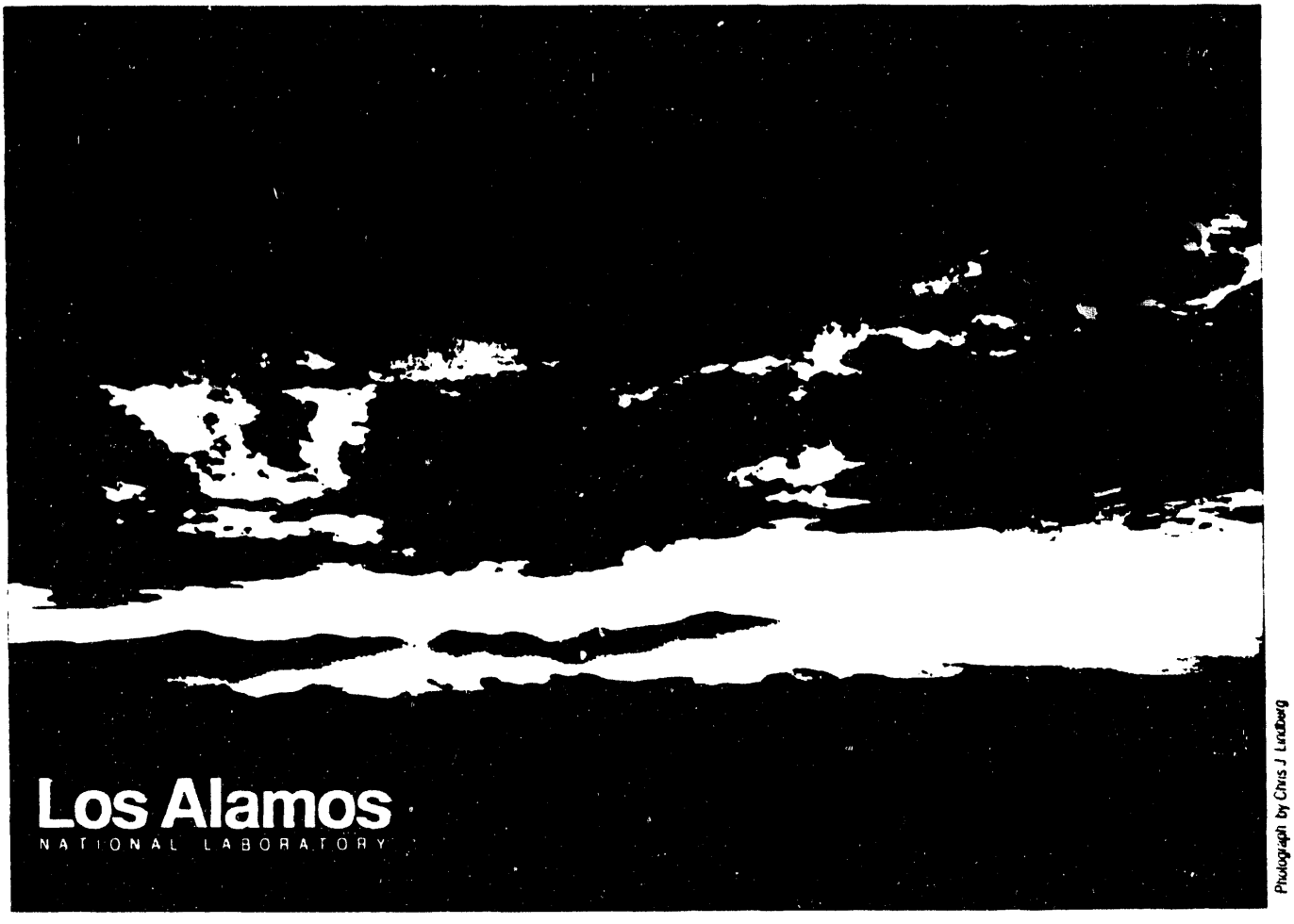
Accelerator Technology for Los Alamos Nuclear-Waste-Transmutation and Energy-Production Concepts



Photograph by Chris J Lindberg

ACCELERATOR TECHNOLOGY FOR LOS ALAMOS NUCLEAR-WASTE-TRANSMUTATION AND ENERGY-PRODUCTION CONCEPTS*

G. P. Lawrence, R. A. Jameson, and S. O. Schriber
Los Alamos National Laboratory, MS H811
Los Alamos, New Mexico 87545
(505) 667-9349

ABSTRACT

Powerful proton linacs are being studied at Los Alamos as drivers for high-flux neutron sources that can transmute long-lived fission products and actinides in defense nuclear waste, and also as drivers of advanced fission-energy systems that could generate electric power with no long-term waste legacy. A transmuter fed by an 800-MeV, 140-mA cw *conventional* copper linac could destroy the accumulated ^{99}Tc and ^{129}I at the DOE's Hanford site within 30 years. A high-efficiency 1200-MeV, 140-mA niobium *superconducting* linac could drive an energy-producing system generating 1-GWe electric power. Preliminary design concepts for these different high-power linacs are discussed, along with the principal technical issues and the status of the technology base.

BACKGROUND

Present U.S. plans for disposal of high-level defense wastes, namely vitrification and long-term storage in deep geologic repositories are meeting with public skepticism and opposition. A principal concern is that migration probabilities for the long-lived fission products (^{99}Tc and ^{129}I) in these wastes may not satisfy long-term confinement criteria for the environment of the proposed repository. Current studies at Los Alamos[1] suggest that an accelerator-driven intense *thermal* neutron source could transmute all the ^{99}Tc and ^{129}I accumulated at the DOE Hanford site (about 2000 kg) to stable or short-lived products within about 30 years, eliminating them from the waste stream, and overcoming a serious environmental objection to the repository plans. Higher actinides, such as ^{237}Np and ^{242}Am, could also be rapidly burned by such a system if desired. Neutron sources driven by high-power proton accelerators have been studied previously for waste transmutation and other nuclear process applications,[2] but the technology base has only recently reached the point that the feasibility of such machines is assured.

In the Los Alamos scheme[3] for accelerator transmutation of waste (ATW) a heavy-metal target is used to produce a high flux of spallation neutrons with an incident medium-energy high-current proton beam. The primary neutron spectrum is moderated to yield an intense thermal flux (10^{15} to > 10^{16} n/cm^2-s) in a D_2O blanket surrounding the target. Material to be converted is transported through the neutron field by continuously flowing aqueous or molten-fluoride-salt carrier loops. Precision chemical partitioning removes transmuted material from the carrier flow while the residue is returned to the blanket for continued irradiation. The Los Alamos scheme differs from other transmutation ideas in that it employs a *thermal* neutron spectrum. Other schemes, both reactor-based and accelerator-based, have relied on a *fast* spectrum which is inefficient for burning fission products. With the right fractional loading of actinides, the tranmuter can generate enough fission energy to power the accelerator. Details of the Los Alamos ATW scheme and its advantages in comparison with previously described systems are discussed in several companion papers in these proceedings.

If fertile material (^{232}Th or ^{238}U) is added to the D_2O blanket, the ATW concept can be configured as an accelerator-driven subcritical converter/burner. The fertile material is converted by neutron capture to fissile fuel (^{233}U or ^{239}Pu), which is then burned directly in the blanket to produce power. Preliminary studies reported at this meeting[4] suggest that such a system has the potential to generate electricity at competitive prices, while producing enough excess neutrons to convert its own high-level waste to stable or short-lived products. This concept could lead eventually to a new safe fission-energy system fueled by abundant fertile resources and requiring no off-site waste management.

Driver accelerator requirements for a defense-waste transmuter and for an advanced energy production system are somewhat different, both in terms of performance goals and development needs. Because disposal of defense wastes is a near-term concern, we consider conventional linac technology as the appropriate design approach for an ATW. By conventional we mean a linac in which the radiofrequency (RF) accelerating cavities are fabricated from copper and are water cooled. For an energy production accelerator, on the other hand, preliminary studies show that high power efficiency will be critical, and that a more advanced approach would be the best solution. For this longer-range but potentially higher-impact application, we consider a design in which the high-energy portion of the linac is made up of superconducting (niobium) cavities where RF losses are negligible.

ATW ACCELERATOR REQUIREMENTS

Neutron transport calculations suggest that the primary source strength for a defense-waste transmuter should be approximately 2×10^{19} n/s, based on a plan for destroying Hanford site wastes within 30 years. Figure 1 plots the calculated neutron yield versus energy for protons axially incident on a 0.5-m-diameter, 2-m-long cylindrical lead target. Also shown are the proton current and beam power needed to produce this neutron source strength. The required beam power is nearly constant at 100 MW above 1000 MeV, so that current can be traded inversely for beam energy. Below this energy the neutron yield drops rapidly, and more beam

* Work performed under the auspices of the U.S. Department of Energy with Los Alamos National Laboratory program development funds.

launcher (comprising two dc injectors, two 350-MHz RFQs, and two 350-MHz DTLs) funneling beams at 20 MeV into a 700-MHz CCL. Figure 3 sketches the configuration. For the beam parameters selected above, each leg of the ATW beam launcher would provide a 70-mA beam.

The CCL would be a 1-km-long 800-MeV side-coupled linac, carrying 140-mA cw current. It would be divided into six sections, each made up of modules consisting of n accelerating cells, a quadrupole magnet, and a diagnostic station. The number (n) of coupled cells per module increases from 2 to 10 as the proton energy increases from 20 MeV to 800 MeV. The average accelerating gradient is relatively low 1 MV/m to minimize RF structure power losses, and the CCL aperture is large (3 cm to 7 cm) to achieve a very high ratio of aperture to rms beam size (9 to 22). This high ratio assures the extremely small fractional beam losses (< 10^{-6}/m) required for hands-on maintenance. The CCL cavities are somewhat more efficient than those in the APT design, providing an RF efficiency of 0.70. Because of the lower beam current and smaller beam size in ATW, smaller CCL apertures may be tolerable, which could push the RF efficiency up to 0.75.

Table 2 lists design values for the RFQ, DTL, and CCL. Other features of the design, including the avoidance of permanent magnets in the DTL drift tubes (because of the radiation threat), transition to a CCL structure at the low energy of 20 MeV, possible use of emittance filtering, and strong transverse focusing are similar to those in the APT study, and are discussed in Ref.6.

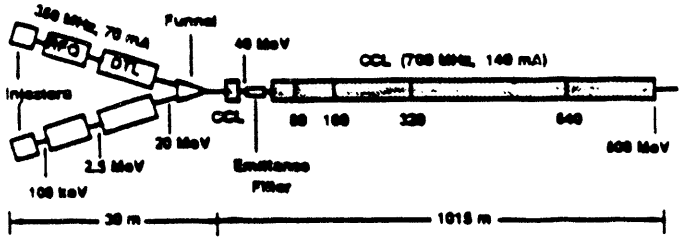


Fig. 3. Reference cw linac concept for ATW

Table 2
ATW Linac Parameters

	RFQ	DTL	CCL
Frequency (MHz)	350	350	700
Energy (MeV)	0.1 to 2.5	2.5 to 20	20 to 800
Synchr. phase (deg)	-90 to -37	-40	-60 to -40
Radial aperture (cm)	0.4 to 0.3	0.8	1.4 to 3.5
Beam current (mA)	150 to 140	140	140
Length (m)	3.4	11.3	1015
Accel. grad. (MV/m)		1.1 to 3.1	1.0 (avg)
Copper power (MW)	0.4x2	1.3x2	47.6
Beam power (MW)	0.2x2	1.2x2	108.9
Total power (MW)	1.2	5.0	156.5
Beam loading	0.33	0.48	0.70
No. of klystrons	2 (1-MW)	6 (1-MW)	82 (2-MW)
Accel. structure	4-vane	2βλ	side-coupled

For the nominal ATW current specification, funneling is not an absolute requirement. A current of 140 mA could be obtained from a single 350-MHz RFQ and DTL, which would simplify the accelerator front end. However, funneling allows a significantly lower emittance in the CCL for the same total current, and reduces the charge-per-bunch by a factor of two. This can be translated into smaller cavity apertures and improved CCL RF efficiency. The cost of RF power (both capital and operating) dominates the transmuter facility cost to such a degree that there is a premium in designing for as high an efficiency as practical. This factor alone appears to outweigh the extra complication introduced by funneling. In addition, the ion-source current demand would be reduced by a factor of two in a funneled sytem.

RF power for the ATW RFQs and DTLs would be provided by existing, commercially available, 1-MW cw 350-MHz klystrons; eight tubes are needed. For the CCL it would be necessary to develop a new high-power RF amplifier tube at 700 MHz. In order to reduce capital costs and improve system reliability, we propose a power level of 2 MW per unit or greater.

TECHNOLOGY ISSUES AND TECHNOLOGY BASE

Accelerator technology improvements in the past few years and advances in understanding of high-current beam behavior provide high confidence that a machine of the ATW power level can now be built and operated. The major technical concerns for a high-power proton linac are 1) beam-loss activation of machine components, threatening hands-on maintainability; 2) RF system efficiency and capital costs; 3) reliability and longevity of components; and 4) operability of an integrated cw system.

The APT point design study addressed the above technical issues in detail [5] for a 4 x more powerful machine. It included complete beam simulations with matching errors, a machine configuration layout, engineering assessment of critical components, and an analysis of off-normal conditions and beam/target safety issues. The design codes have been benchmarked in the relevant energy and charge-density regimes through simulation of high-current behavior on the Los Alamos NPB Accelerator Test Stand (ATS), and by an end-to-end simulation of LAMPF that predicts measured emittance values as well as beam loss locations and approximate magnitudes.

A number of accelerator systems have operated at or near ATW -level parameter values. Existing ion source designs appear capable of delivering the needed proton current with the desired brightness. Performance requirements are not as demanding as those for the NPB program. A 267-MHz 0.6-MeV proton RFQ at CRNL has operated at 70 mA cw, [7] and peak H^- currents of 100 mA have been demonstrated in a 7-MeV ramped-gradient 425-MHz DTL at Los Alamos. Beam funneling in the relevant current and frequency range has been successfully demonstrated at Los Alamos.[8] A coupled-cavity accelerating structure at NIST has operated cw with a 1-MV/m gradient, at 4 x the ATW frequency.

Experience with existing research linacs that have operated for years with high availability as beam "factories" has provided a strong foundation for making extrapolations to the ATW performance regime. Because of its high average current (1 mA), operational experience at LAMPF is especially relevant, and also directly addresses the important beam-loss issue. For most of the LAMPF CCL length, the beam loss fraction is estimated to be < 2×10^{-7}/m, and radiation levels after shutdown allow unlimited-access hands-on maintenance. Because all CCL RF buckets contain charge in the ATW concept and the duty factor is 1.0, compared with LAMPF's 1-in-4 bucket filling and 0.06 duty factor, the charge/bunch in ATW is only 2.5 times greater than in LAMPF. Therefore, even though the average beam power is 140 times greater in ATW, the beam dynamics in a well-understood range. Given the very large aperture-to-beam-size ratio in the ATW CCL and the high quality input beam, we can be confident of achieving the low fractional beam loss (1/10 that of LAMPF) needed for hands-on maintenance.

power is needed. The relations in Fig. 1 provide inputs to a simple cost model that has been used to help select the linac parameters that would produce a minimum-cost ATW system.

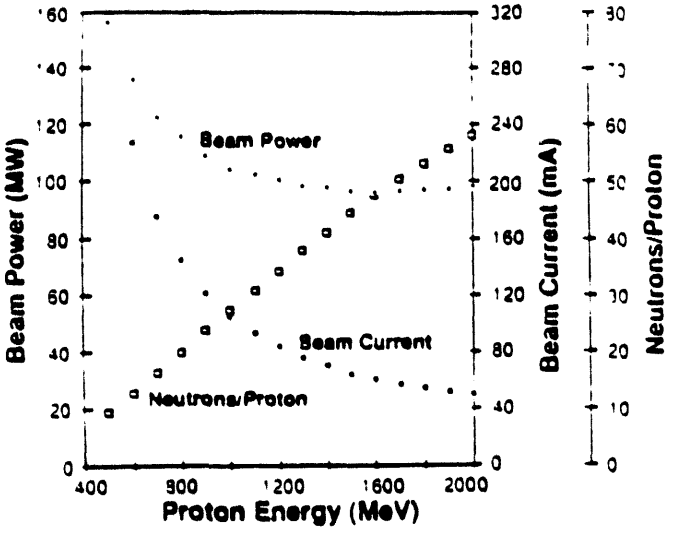


Fig. 1. Energy dependence of neutron yield from lead target. Also linac beam power and current for specified ATW source strength.

COST/PERFORMANCE MODELING FOR ATW

A linac for ATW will be similar to the concept developed in the recent study of an accelerator for production of tritium (APT).[5] Parameters for APT were 1600 MeV and 250 mA, cw. At that high current level much more RF power is delivered to the beam than is lost in the accelerating structures, resulting in a high RF efficiency – nearly 0.8 in the high-energy portion of the linac.

Since the ATW beam power requirement is only 1/5 to 1/4 that of APT, the current, energy, and duty-factor tradeoffs must be re-examined to determine the best design space. Power-efficiency is critical because of the very high cost impact of the RF power system. This criterion could lead to either: 1) a lower-energy high-current cw machine; or 2) a pulsed high-energy machine with high peak current. In order to obtain a first-order quantitative comparison of these two possibilities, simple accelerator cost models have been constructed. These models assume a common machine architecture similar to APT, with dual RFQ/DTL 350-MHz beam inputs funneled into a 700 MHz CCL. Because most of the accelerator cost is contained in the CCL, these models treat the linac front ends simply as fixed sums, and focus on algorithms that parameterize the CCL costs. While the models are incomplete in terms of structural detail, the principal cost factors are included, along with the usual multipliers for contingency, project management, ED&I, etc. Results are displayed in Fig. 2, which shows the estimated construction cost and annual cost versus beam energy for cw and pulsed linacs.

Table 1 lists values of the key model parameters for each kind of linac. Average accelerating gradient for the cw linac was chosen as 1.0 MV/m, a value that is close to minimizing construction cost , but slightly above the annual cost minimum. The pulsed linac duty factor was taken as 0.25 at 1600 MeV, which would require 240 mA peak current. The cost-optimized gradient for a pulsed machine with that duty factor is about 1.5 MV/m. An RF system (installed) capital cost of $2/watt was assumed for a cw machine, based on about 85 2-MW power modules. For a pulsed machine, with the high duty factor and pulse length that

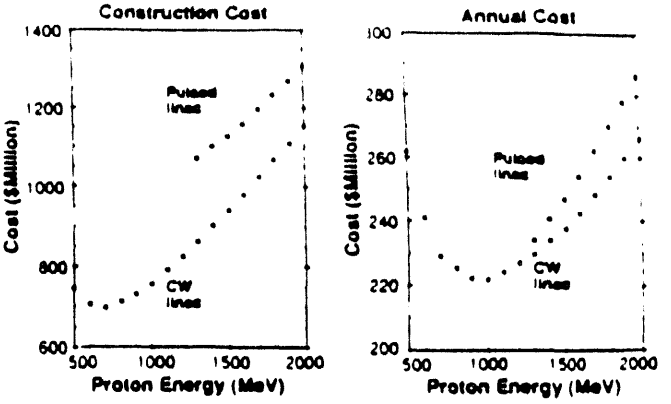


Fig. 2. ATW construction cost and annual cost versus proton energy, for cw and pulsed linac models.

Table 1
ATW Linac Cost Model Parameters

	CW	Pulsed
CCL real estate gradient (MV/m)	1.0	1.5
Duty factor	1.0	0.25
RF unit capital cost ($/avg. watt)	2.0	4.0
CCL structure cost (M$/m)	0.100	0.100
CCL shunt impedance, avg. (Mohm/m)	32.5	23.8
Cost of electric power ($/kWh)	0.05	0.05
Time-on fraction	0.75	0.75
Number of operating staff	200	200
Power conversion efficiency (rf/ac)	0.60	0.60

would be required to keep peak current at or below APT levels, the capital cost (per average watt) was doubled, based on preliminary comparisons of cw and pulsed RF system costs.

Figure 2 shows that construction costs for a cw linac minimize near 700 MeV, while annual costs minimize closer to 1000 MeV. Annual costs are dominated by electric power (at $0.05/watt) and the capital charge (at 10%/year). The position of the annual cost minimum is remarkably insensitive to moderate variation of the principal model parameters. Fifty-percent changes in accelerating gradient, average CCL shunt impedance, CCL structure cost, and RF power unit cost shift the cost minimum less than 50 MeV.

For a pulsed linac the energy cannot be decreased much below 1400 MeV without incurring excessive peak current levels in the CCL (> 300 mA) or duty factors greater than 30%. Figure 2 shows that at 1400 MeV the construction cost of a pulsed linac would be significantly higher than for an 800-MeV cw machine, but the annual cost is nearly the same as that for a cw system, within the credibility of the model. This cost result, which on balance favors the cw system but not overwhelmingly, leaves the real choice to technical considerations. A cw linac would simplify RF control aspects, eliminate modulators and energy storage from the RF system, and permit substantially lower peak currents in the accelerator, with lower resultant beam-losses. These are important advantages. We therefore propose that a linac for the ATW application should be a cw machine, with energy and current selected as 800 MeV and 140 mA.

ATW ACCELERATOR POINT DESIGN

A first approach to a point design for a cw ATW could be based on the APT architecture,[6] and would thus consist of a beam

RF TECHNOLOGY, RF SYSTEM COST IMPACT

High power cw RF tubes (klystrons) in the 0.5 to 1.0-MW class are available at frequencies near 350, 500, and 1000 MHz. Operating lifetime information for 1-MW cw tubes is sparse, but vendors are confident that 50,000 hours is a reasonable expectation. The tube longevity is somewhat addressed by LAMPF operating statistics, which show the average lifetime of the 1.25-MW peak-power 805-MHz klystrons (up to 12% duty factor) as > 50,000 hours, with many tubes surviving for > 80,000 hours.

Major leverage for reducing the cost of an ATW linac could come from reducing the unit capital costs of the CCL RF power system, and/or from development of higher efficiency RF generators. The capital cost (per watt) of installed RF capacity is expected to scale inversely as the square root of the module output power, so there should be an advantage in going to larger tubes than the 1-MW cw generators now available. A smaller number of tubes should also improve overall accelerator reliability. Candidates for ATW use are the klystron, klystrode, and the magnicon.

The klystron, which operates by velocity modulation of an electron beam, represents mature high-power technology. Development of a new 1-MW cw klystron for service at 700 MHz would be well within the explored design space and a straightforward enterpise. It is thought that 2 MW is probably the practical upper power limit for klystrons at this frequency. The klystrode, a relatively new device, produces RF power through amplitude (grid) modulation. Pushed by SDI program requirements, high power klystrodes (up to 0.5 MW) are being developed at ATW-relevant frequencies. The power limit is thought to be about 1 MW, due to grid heating, but the tube has the advantage of compactness and retains high efficiency (0.70) over a large output range. Although there is no lifetime data for the new high power tubes, experience with the 50-100 kW klystrodes widely used in television transmitters is good. The magnicon, a new RF tube invented in the USSR, produces RF power by using circular deflection of the electron beam to produce a rotating electromagnetic wave.[9] It may be capable of generating 4 MW cw at very high efficiency. However, a cw high power version not been demonstrated and a significant development program will be needed to assess the promise of this technology.

ACCELERATOR FOR POWER-PRODUCTION

Initial estimates for an accelerator-driven power-producing system specified at 1000-MWe generating capacity call for a neutron source strength somewhat greater than required for a defense-waste transmuter, about 3.3×10^{19} n/s. This translates to about 160 MW of beam power for proton energies in the linear spallation neutron yield range (1200-2000 MeV). To generate electric power at competitive prices, the accelerator efficiency must be as high as possible and the capital and operating costs as low as possible. Initial studies using a cost model similar to that for ATW suggest that these objectives can best be achieved with a linac whose high-energy section (above 20 MeV) consists of superconducting RF (SCRF) accelerating cavities. The ac-to-beam power efficiency could be > 0.65. The best that can be achieved with a conventional machine is about 0.45. SCRF niobium cavity technology, developed over the past 20 years, has reached a high level of maturity, culminating recently in major (electron) accelerator projects at several high-energy physics laboratories (CEBAF, CERN, KEK). Standard accelerating gradients achievable within the accelerating structures are in the range 5 to 8 MV/m, and cavity fabrication costs, initially high, have come down to $200K/m, with further decreases anticipated.

The simple ATW linac cost model was extended to accomodate a superconducting CCL. While RF power losses in the accelerating cavities become very small, there are significant refrigeration requirements to handle them as well as the ambient heat leaks. Table 3 lists the relevant parameters, including refrigeration assumptions, included in the model. The structure gradient was chosen as 5 MV/m, even though costs appeared to be somewhat lower at higher gradients, in order to avoid an excessively large RF drive power per unit length. SCRF (cw) linac construction and annual costs are compared in Fig. 4 with costs for a room-temperature (RT) cw linac as a function of beam energy.

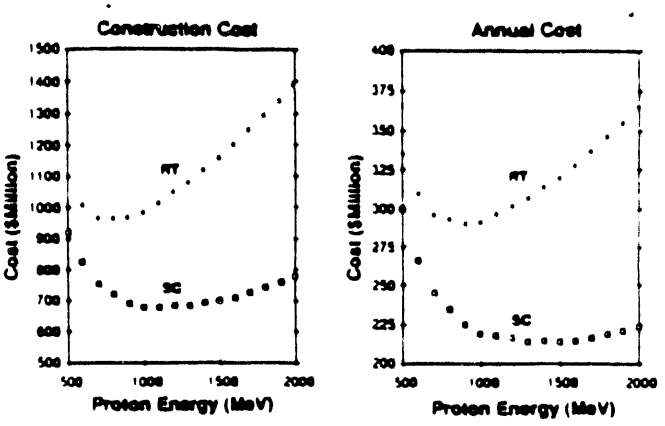


Fig. 4. Construction cost and annual cost versus proton enrergy for superconducting and room-temperature cw linac models.

Table 3
Energy-Production Linac Cost Model Parameters

	RT	SCRF
CCL real estate gradient (MV/m)	1.0	3.0
CCL structure gradient (MV/m)	1.3	5.0
Duty factor	1.0	1.0
RF unit capital cost ($/avg. watt)	2.0	2.0
CCL structure cost (K$/m)	100	200
RF structure losses (W/m)	72,500	20
Refrigerator efficiency		0.002
Cost of electric power ($/kWh)	0.05	0.05
Time-on fraction	0.75	0.75
Number of operating staff	200	200
Power conversion efficiency (rf/ac)	0.60	0.60

The construction cost and annual cost dependence on beam energy for the SCRF linac appear to have considerably broader minima than those for a room-temperature system, and the cost optimum is at higher energy. This effect is due to the elimination of cavity RF power consumption in the CCL, which is only partly offset by the refrigeration requirements. The cost comparisons suggest that a superconducting linac for energy production could be 25 to 30% less expensive to build and operate than a room-temperature system. Because SCRF cavities can operate cw at higher gradients than RT copper structures, a superconducting CCL can be much shorter than its room-temperature equivalent, even if the SCRF linac has a higher output energy. Another advantage of a superconducting CCL is that beam apertures in coupled SCRF cavities can typically be much larger than those in RT cavities of the same frequency, allowing lower beam losses. Since negligible RF power is lost in SCRF cavities, there is no design imperative to reduce apertures in order to maximize the shunt impedance. On the contrary, apertures are made large in order to provide adequate on-axis coupling for the fundamental accelerating RF mode and to prevent trapping of destructive beam-excited high-order modes.

A possible accelerator for driving an energy-producing system might have an architecture as sketched in Fig. 5. Table 4 summarizes some of the expected machine parameters.

Table 4
Parameters for an Energy-Production Linac

	RFQ	DTL	CCL
Frequency (MHz)	350	350	700
Energy (MeV)	0.1 to 2.5	2.5 to 20	20 to 1200
Radial aperture (cm)	0.4 to 0.3	0.8	5 to 8
Beam current (mA)	75 to 70	70	140
Length (m)	3.4	11.3	515
Accel. grad. (MV/m)		1.1 to 3.1	3.0 (avg)
Copper power (MW)	0.4x2	1.3x2	0.006
Beam power (MW)	0.2x2	1.2x2	165.2
Total power (MW)	1.2	5.0	165.2
RF efficiency	0.33	0.48	1.00
No. of klystrons	1 (1-MW)	4 (1-MW)	87 (2-MW)
Accel. structure	4-vane	2βλ	axis-coupled
Refrigerator power (MW)			5.6

The beam launcher for this machine could be a room-temperature funneled system identical to that described for ATW. The coupled cavity linac, from 20 MeV to 1200 MeV would consist of multicell superconducting niobium cavities, with the number of coupled cells per module increasing from 2 to about 5 as the energy increases. If an average packing factor of 0.6 can be achieved (as at CEBAF), and assuming a structure gradient of 5 MV/m, the real-estate gradient would be 3 MV/m, which leads to a CCL length of only 0.5 km.

Beam performance for the superconducting linac should be very similar to that estimated for the ATW linac. Transverse and longitudinal emittance are determined essentially in the beam launcher. Only a small growth is anticipated in the CCL. The ratio of structure aperture to rms beam size in the SCRF CCL could be 2 x larger than for the RT machine, with the machine aperture limit probably determined by the quadrupole bores.

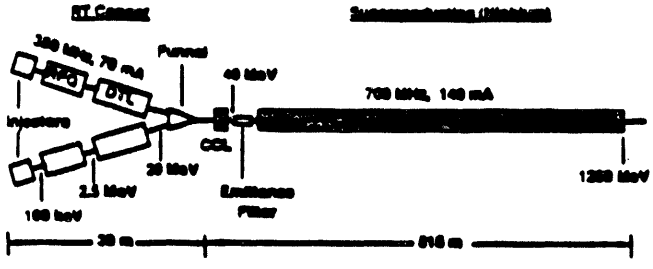


Fig. 5. Superconducting linac for energy production.

The SCRF cavities for an energy-production linac would be cooled at 4.2 K by a refrigeration system comparable in scale to those now in use at CERN, DESY, and KEK. Estimated residual RF losses in the niobium cavities (at 5 MV/m) will be about 20 W/m. This load and the static heat load to the cryostat (also about 20 W/m) must be rejected at room temperature by the refrigeration plant, which would require about 6 MW of ac power, assuming an overall efficiency of 0.0022. This is to be compared with the 80 MW of ac power saved by eliminating CCL RF power losses.

With a 140-mA cw beam, the CCL RF power input requirement averages 700 kW/m; at the 1200 MeV point this implies a 700 kW power coupler feeding each 5-cell module. This high power coupling requirement constitutes one of the technology challenges for development of a superconducting linac. The practical level that has been reached is 100 kW per feed (at 500 MHz), but Cornell University is now developing a 500-kW coupler (also at 500 MHz). Additional areas that need to be addressed in an R&D program for high-power SCRF linacs include the sensitivity of niobium cavities to radiation damage, cavity Q-degradation due to adsorbed residual gas layers, handling of beam-excited high-order RF modes and other control issues, and development of cavity designs appropriate for the large range of proton velocities in the linac (v/c = 0.2 to 0.9).

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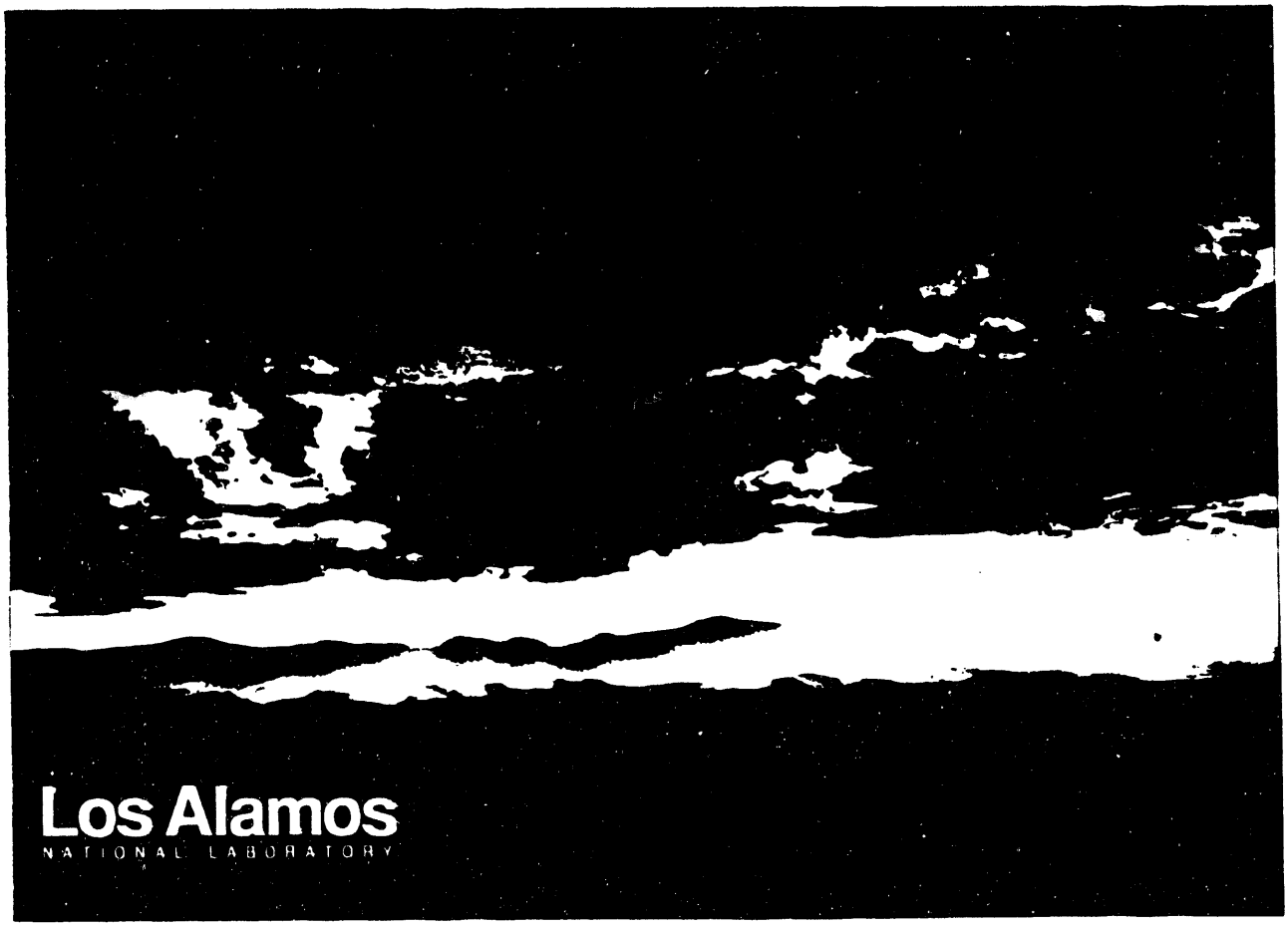
Accelerator Transmutation of Iodine-129



Photograph by Chris J Lindberg

Abstract

Accelerator Transmutation of ^{129}I

Moses Attrep, Jr.
Isotope and Nuclear Chemistry
Los Alamos National Laboratory
Los Alamos, New Mexico 87545

Iodine-129 is one of several long-lived reactor products that is being considered for transmutation by the Los Alamos Accelerator Transmutation of Waste (ATW) program. A reasonable rate of transmutation of ^{129}I is possible in this system because of the anticipated high neutron flux generated from the accelerator. This report summarizes previous papers dealing with the transmutation of ^{129}I where reactor technologies have been employed for neutron sources. The transmutation process is considered marginal under these conditions. Presented here are additional information concerning the final products that could be formed from the transmutation process in the ATW blanket. The transmutation scheme proposes the use of solid iodine as the target material and the escape of product xenon from the containers after van Dincklange (1981). Additional developmental plans are considered.

Accelerator Transmutation of Iodine-129

Moses Attrep, Jr.
Isotope and Nuclear Chemistry Division
Los Alamos National Laboratory
Los Alamos, New Mexico 87545

1. Introduction.

Iodine-129 ($T_{1/2}$= 15.7 Ma), produced from the fissioning of actinides, is one of the longest lived fission products. This nuclide with ^{99}Tc ($T_{1/2}$ = 0.213 Ma) and the long-lived waste actinides (Pu, Np, Am, Cm, etc.) is the topic of concern for the proper management and disposal. Over the years a number of options have been advanced for the proper management of these nuclides. These include (a) low risk factor long term storage, (b) extra-terrestrial disposal, (c) use of nuclear devices to transmute and vitrify nuclear waste in underground explosions, (d) direct storage of irradiated fuel pins in a repository, and (e) transmutation by neutron absorption (neutrons from reactors, accelerators or controlled thermonuclear devices) to produce stable and/or manageable products. Most all of these options requires some degree of chemical partitioning before treatment and disposal. Each has its draw-backs touching on objections which are social, political, economical, scientific and/or environmental. This interim report focuses on the application of accelerator transmutation of the fissiogenic iodine isotopes as integrated into the Los Alamos accelerator transmutation of waste (ATW) plan.

2. Iodine Inventories.

It is estimated that the amount of ^{129}I in the Hanford tanks (defense waste) is ~33 Ci (Wodrich, 1991). This equivalent to ~190 kg ^{129}I or as much as ~240 kg fission produced ^{129}I + ^{127}I assuming no additional stable iodine had been added to the Hanford tank waste. This is considered to be in ~25 million gallons of double shell tank liquid. The focus of iodine recovery was on spent fuel dissolver off-gas recovery where the concentrations may have been 1,000 to 10,000 greater than in the existing waste. In addition to the presently US stored nuclear waste associated with the nuclear defense, it should be noted

that there are estimations that cumulative amounts of fissiogenic $^{129}I + ^{127}I$ will be 15,200 kg in the year 2000 for the Euratom countries (MacKay et al., 1984). This amount is many times more for world-wide inventories. The Los Alamos scheme presently envisions an ATW system to transmute actinides, ^{99}Tc and iodine isotopes from 8 light water commercial reactors. Regardless of the inventories and the load requirements, the amount of fissiogenic iodine to be transmuted is considerable and conventional technologies and methods of tranmutation by reactors probably cannot support the disposal of such inventories. Thus, high neutron flux accelerator technology becomes an option for rapid transmutation.

3. Nuclear Considerations of Iodine.

Iodine-129 is a β^- emitter (E_{max} = 0.606 Mev), emits a weak gamma at 39.6 keV, and is considered not to be very radiotoxic (MacKay et al.,1884). The fission yield for ^{129}I from the thermal fission of ^{235}U is 0.75%. In addition, ^{127}I (stable) is also produced as a fission product with a fission yield of 0.126%. From a prompt ^{235}U fission system the total iodine produced would be composed of approximately 86% ^{129}I and 14% stable ^{127}I. This composition would change if different fissile materials were used, if the energy spectrum of neutron is changed, and if the residence time of the iodine in the actinide fuel were increased. The latter reflects *in situ* transmutation which will be related to their respective neutron capture cross sections, etc.

The central issue for the successful transmutation of radiogenic ^{129}I is the rate of that transmutation. This is dependent on two factors: the neutron absorption cross section (σ) and the neutron flux (ϕ). The thermal neutron cross sections for the following reaction of ^{129}I are 20 + 10 barns (to isomeric and ground states of ^{130}I, respectively). Neutron capture consideration must be given for the stable ^{127}I that would also be present (thermal neutron capture cross section = 6.2 b). Its presence automatically reduces the ^{129}I transmutation efficiency by consuming neutrons. These reactions are illustrated as follows and are the primary reactions of consideration.

$^{129}I(n,\gamma)\ ^{130}I$ (β^- emitter) to ^{130}Xe (stable)
and

^{127}I (n,γ) ^{128}I (β^- and β^+ emitter) to ^{128}Xe (stable) and ^{128}Te(stable).

Multiple (n,γ) reactions and (n,2n) (especially where harder neutron spectra are used) on starting materials and intermediate irradiation species must be considered for ultimately defining the waste stream and estimating the radiation fields during and post irradiation. This is addressed in the section on Transmutation Products and the Waste Stream.

4. ^{129}I Transmutation Proposals.

Iodine-129 was considered for transmutation by von Dincklange (1981) in an accelerator system. A requirement of 10^{16} n cm^{-2} sec^{-1} was considered necessary for the system to work effectively. Considering the energy release from the formation of ^{130}I and the β^- Q energy release, he deduced that I_2 could be irradiated in rods of approximately 0.5 to 1 cm radius to maintain temperatures where elemental iodine would not vaporize appreciably. Figure 1 illustrates the van Dincklange proposal. Rods containing ^{129}I are loaded and placed in the transmuter. Stable ^{130}Xe gas produced from neutron capture on ^{129}I and subsequent β.decay would escape to the plenum. (Iodine-127 was not considered in this proposal, hence other products from ^{127}I are not considered.) Any iodine released from the rods would be captured in the de-sublimer and eventually returned to the transmuter whereas the xenon is released to the environment having passed through a filter to capture any remaining iodine. The proposal envisions the de-sublimer functioning as a cryopump for the iodine. This proposed scheme has many attractive features and will provide a starting point for consideration of the transmutation of iodine in this report.

Serious consideration of reactor neutron transmutation have been given. Wacher and Croff (1980) discussed a transmutation system for iodine (as NaI for calculational purposes) whose isotopic composition was 75% 129I and 25% 127I. Origen calculations for a PWR estimated that transmutation would produce 1.1 g ^{128}Te; 18.1 g ^{128}Xe; and 56.2 g ^{130}Xe per kg iodine treated per full-power year. Their conclusion was that the low transmutation in the system planned is marginally feasible. The CURE (1989) concept addresses a wide range of applications of transmutation of fission products from

commercial light-water reactor fuel. Included in the considerations is ^{129}I. The report discusses the transmutation products, the collection of iodine from the reporcessing of the fuel and transmutation rates, etc. The report is optimistic about the control of inventories of these materials using transmutation as a reasonable option. Specifics to the flow sheets for iodine in the transmutation process are not presented.

5. Target Considerations.

The Los Alamos ATW actinide and ^{99}Tc loops are designed as continuous flow systems where the target material is either a slurry or in solution. Circulating iodine containing compounds/solutions in systems create some serious materials, chemical and target density problems. This is due to the multiple oxidation states of iodine, the uncertainty and complexity of the radiolysis products of iodine, and the known corrosive properties of iodine and iodine compounds. The corrosive nature of iodine on the proposed containers has been raised by Logan et al. (1980) and van Dincklange (1981). Although the optimal iodine compound or species has not been finalized for this system, elemental iodine is attractive and is adopted as the target material for this study on the basis of simplicity related to competing neutron absorbing elements, chemical speciation and radiolysis issues.

6. Transmutation Products and Waste Stream.

Besides the off-gas treatment of iodine, there are not many, if any, "baseline" solution/flow chemical process technologies available for adoption in a transmutation system. Considering the nature of the complexity of the chemistry of iodine and its corrosiveness, the system of irradiating solid I_2 has been adopted as previously mentioned as the starting or "baseline" system.

High chemical purity iodine from reactor waste or nuclear defense waste is to be prepared and loaded into rods for placement in the accelerator blanket. After irradiation the rod and contents are removed and time is allowed for the radioactive decay. The resulting products are processed and disposed and the un-transmuted iodine is reprocessed for further

transmutation. Unlike the actinide transmutation system where there are many elements formed requiring complex chemistry, the iodine system, somewhat like the technetium system, may be considered "simple." The system is similar to that of van Dincklange (1981), but other products, etc., will have to be evaluated.

To determine the parameters and characteristics associated with the transmutation of iodine in high neutron fluxes for the purpose of designing strategies for handling and treatment, calculations for the transmutation of iodine and the production of products were performed. The following conditions were set for ORIGEN2 calculations. A single rod (1 cm diameter x 1 meter) much like that described by van Dincklange (1981) was filled with 200 g of ^{129}I and ^{128}I in a ratio described by Wacher and Croff (1980). A neutron flux of 10^{16} n cm^{-2} sec^{-1} with an he energy spectrum of the neutrons was similar to that of the CANDU reactors was assumed. The transmutation times were set for 1 hour, 1 day, 10 days and 100 days to provide a wide range of reaction periods in order to see the development of other products. The reaction vessel was assumed to be sealed unlike that of the previous considerations in which a plenum was provided for xenon gas escape. These were set to maximize higher order element formation as "worse case" scenarios. The percent ^{129}I transmutated and the approximate activity at end of transmutation for each irradiation period are given in Table I. Under this high neutron flux it is seen that remarkably high quantities of the ^{129}I can be transmutated. Caution should be exercised in accepting these transmutation values. Fluxes at 10^{14} to 10^{15} may be more realistic to determine iodine transmutation values. The activities at end of the transmutation period have been listed to provide some guidance as to the level of radiation fields that the target material and the encapsulation assembly will experience during the irradiation. The radiation level will have decreased significantly also after the 180-day cool period. In all cases, the radiation levels are dominated by iodine isotopes (128, 130m, 130g, 131, and 132) with the 8-day ^{131}I controlling the cooling time.

To help define the chemical waste stream for the system, the elements produced (quantities) have been tabulated for the four irradiation periods immediately following irradiation and for a 180-day cooling period. (A 90-day cooling period would probably be satisfactory based on the ^{131}I.) These numbers are given in Tables II and III, respectively. For the 1-hour, 1-day

and 10-day irradiation periods the products are Te, I, and Xe with very little contribution of barium, cesium or other elements. For the 100-day transmutation period, Te, I and Xe again are the primary products with small contributions of Cs (0.13 g) and Ba (3.8 mg) and trace quantities of lanthanum and cerium. For purposes of chemical treatment the products of these elements are assumed to be TeI_4, BaI_2 and CsI. In the 100-day inventory, these three elements would bond approximately 3.2 g of iodine. The waste stream at first approximation following the transmutation phase appears to be quite simple consisting of un-transmuted iodine, tellurium (TeI_4), xenon gas, barium (BaI_2) and Cs (CsI).

It is necessary and desirable to consider a continuous release of product xenon gas. When this is done, the amounts of higher elements (Ba, Cs, etc.) would be reduced since further neutron capture and decay in the transmuter would be reduced. By removing the xenon the build-up of ^{135}Xe which has a very large neutron absorption cross section (poisoning the system) would be minimized. However, those elements being formed from the β^- decay of xenon isotopes would be physically located in the filter system and/or filter.

7. Flow Diagrams of Waste Streams.

There does not appear to be many flow sheets for the treatment of iodine in a tranmutation system. However, to our advantage there must be considerable experience in handling and dealing with off-gas iodine from the reprocessing of fuel rods (Maeck et al., 1968). The following treatment and waste stream scheme is summarized in figure 2. Because it is not practical or safe to have sealed containers where the buildup of Xe gas pressure would increase dramatically, the scheme incorporates the release of xenon from the reaction rods as van Dincklange (1981) indicated in his system for the transmutation of iodine. Waste management for this system includes the preparation of the target rods or pins, transmutation and collection of products, removal of rods and processing of untransmutated iodine and transmuted products.

The numbers in figure 2 assume an approximate 17% transmutation of ^{129}I (corresponding to calculation results previously described for estimations of a 10-day irradiation) to estimate the mass balance of the waste stream

products. No estimation of the fraction of xenon that would escape the reaction rods. If higher transmutation rates are permitted, then the amounts of Te, Ba, and Cs will become higher (see Table III). Deposition of these elements in other components of the iodine assembly probably would not pose a serious radiation handling problem but would present challenges of recovery if this were necessary.

If needed, chemistry for the collection of the iodine chemically bound to the other elements should not be difficult. For example, the following steps could easily be employed. To the iodides of Te, Ba, Cs, etc., dilute hydrochloric is added and the mixture is heated. Sodium nitrite is added slowly and the iodides are converted to I_2 which are trapped and collected for further reprocessing and transmutation. This process could be done in a glove box. The major concern with regards to long lived transmuted products would be those of the cesium isotopes. In a worse case situation (long irradiation periods in sealed tubes) the cesium isotopes would begin to build up. Using the 100-day irradiation as an example, the amount of the 2.065-year ^{134}Cs would be the greatest activity producer at about 20 Ci. Thirty-year ^{137}Cs is produced at a level of $\sim 4 \times 10^{-5}$ g (~0.004 Ci). In either case, cesium does not appear to present a problem in radiation handling or disposal. None of the barium, tellurium, xenon, or higher iodine isotopes appear to present either radiation handling/storage or disposal problems.

8. Summary.

This document provides the review and additional transmutation calculations that are useful to define the products, maximum radiation levels, and information for developing strategies for transmutation of fissiogenic iodine isotopes by accelerator transmutation. It is apparent that a number of additional exercises and developmental plans will have to be performed. Some of these include, for example, the following:

-refined calculations for the transmutation of ^{129}I in the final (near-final) blanket design;

-investigation of the feasibility of isotopically separating ^{129}I from ^{127}I;

-the study and selection of materials (corrosion resistant) capable of handling iodine and disposal fate of these.;

-determination of the degree of chemical purity of the iodine for transmutation;

-removal (if necessary) of products (Te, Cs, Ba) other than iodine from filters (assuming the Ag-mordenite type);

-further investigation of a possible cyrogentic system to move I_2 in and out of the transmuter;

-the behavior of I_2 in the rods and the xenon release system, continued calculations (for transmutation rates and product formation) of iodine using neutron fluxes and neutron spectra which are evolving from new blanket designs;

-incorporation of advanced techologies to specialize design of target iodine in the blanket; and

-refined details of the chemical treatment of the products.

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J. W. Wachter and A. G. Croff, "Actinide Partitioning-Transmutation Program Final Report. III. Transmutation Studies." ORNL/TM-6983, pp 37-39 (July 1980).

D. D. Wodrich, Westinghouse Hanford Co, January 1991 Presentation to LANL visiting group; "Disclaimer: The numerical values in this briefing package are estimates and will likely change with further work. However, the values are adequate to provide a perspective."

Table I. Percent ^{129}I Transmuted at End of Transmutation (EOT) Activity Levels for a 200-g Iodine Sample in a Flux of 10^{16} n cm^{-2} sec^{-1}.

Irradiation Length	Percent ^{129}I Transmuted	Activity at EOT (mega Curies)
1 hour	0.059	1.72
1 day	1.83	4.36
10 days	16.6	6.55
100 days	83.7	2.99

Table II. Elements Produced (Grams) from Four Irradiation Periods at EOT for 200 g Target Iodine.

Element	1-hour	1-day	10-day	100-day
Tellunum	2.11E-4	9.59E-3	9.57E-2	0.7545
Iodine	2.00E+2	1.98E+2	1.72E+2	45.5
Xenon	1.11E-2	1.61	28	156
Cesium		3.91E-11	9.94E-4	0.11
Barium		1.45E-17	4.48E-10	2.02E-3
Lathanium				4.44E-10
Cerium				2.02E-11
Praseodymium				2.42E-14

Table III. Elements Produced (Grams) from Four Irradiation Periods at EOT for 200 g Target Iodine Following 180-day Cooling Period.

Element	1-hour	1-day	10-day	100-day
Tellurium	2.11E-4	9.83E-3	9.59E-3	0.7545
Iodine	1.99E+2	1.97E+2	1.71E+2	45.11
Xenon	1.36E-1	3.235	29.94	155.3
Cesium	6.02E-16	1.74E-9	4.56E-5	0.1337
Barium		1.88E-14	4.88E-8	3.77E-3
Lathanium				5.00E-10
Cerium				3.00E-14
Praseodymium				1.00E-14

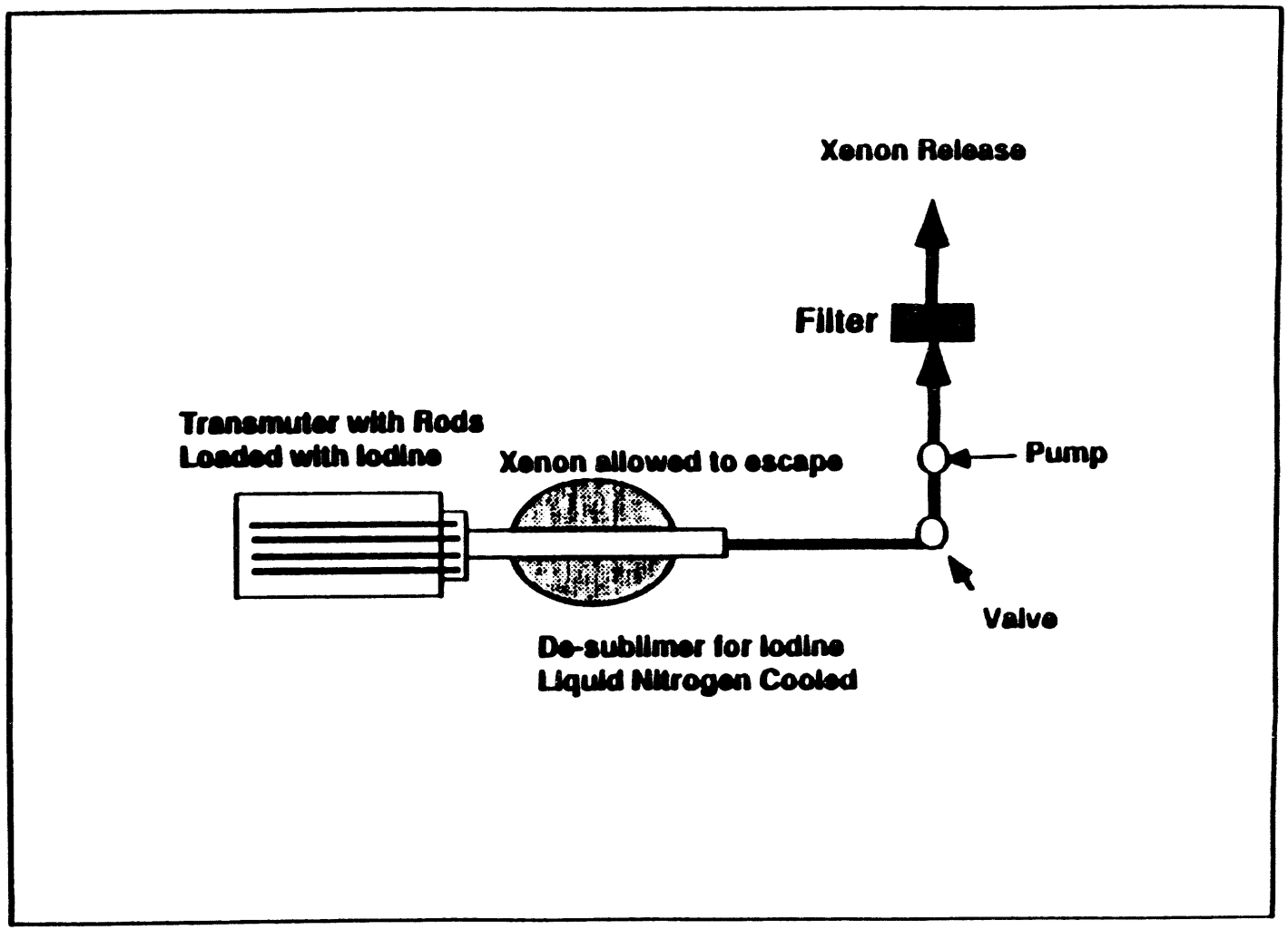


Figure 2. Represnetation of components of transmutation apparatus for Iodine after van Dincklange (1981).

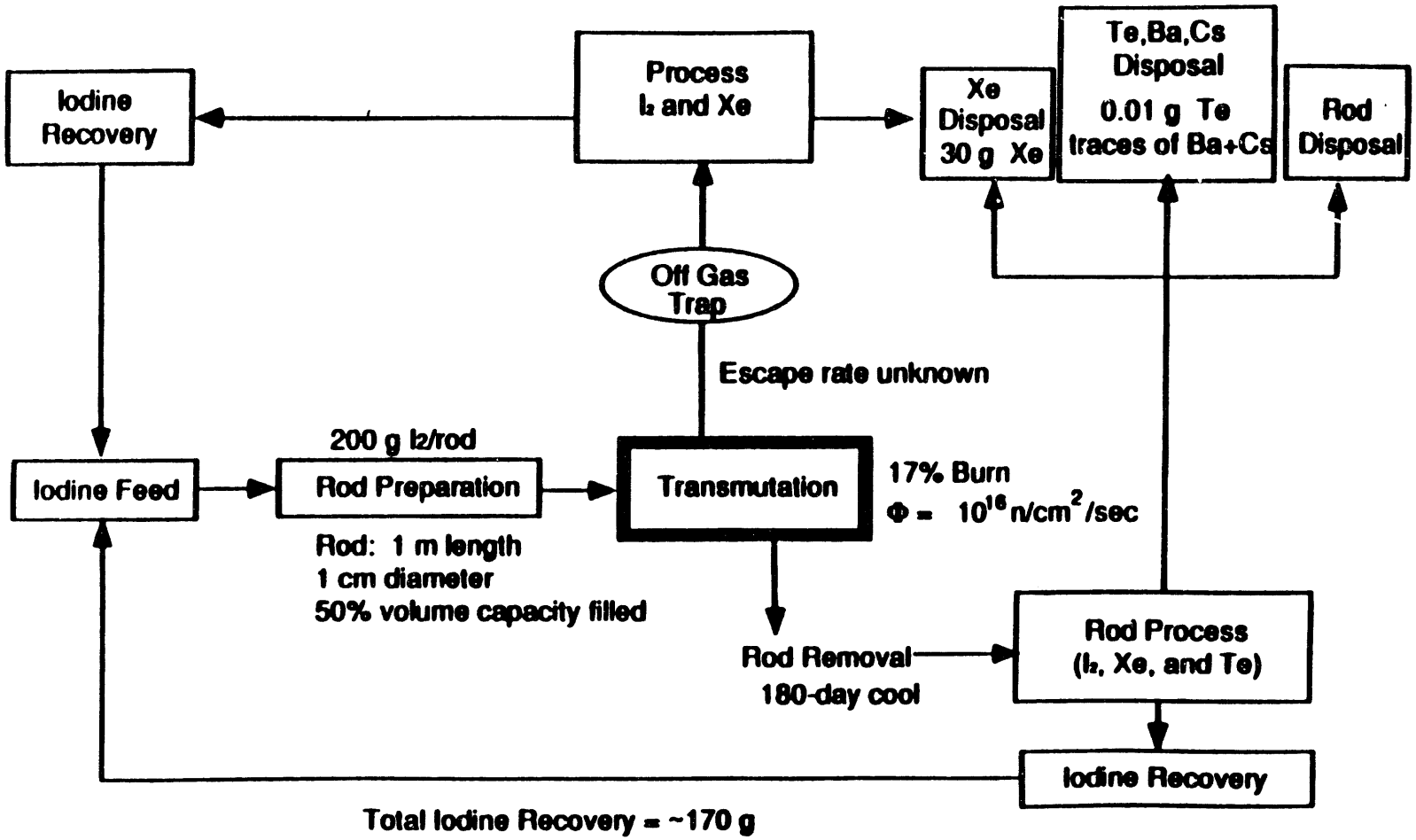


Figure 2. Diagram for iodine ATW and disposition of products.

Accelerator Production of ^{238}Pu

Edward D. Arthur
Los Alamos National Laboratory

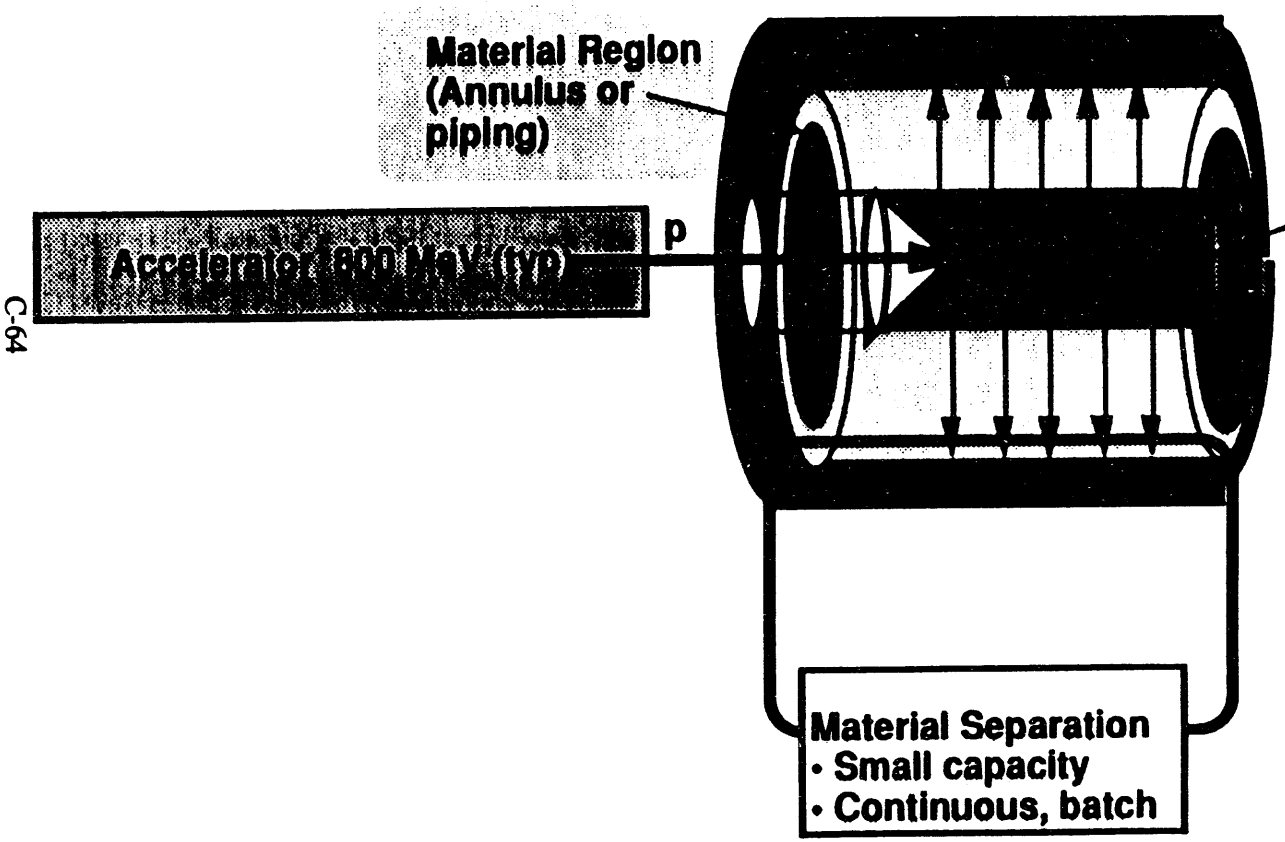
Presentation to DOE-NE
Radioisotope Facility Planning Task Force

February 12, 1992

Outline

- Introduction - Accelerator-Based Systems
- Three Systems
  - Stand Alone - APP
  - LAMPF Upgrade
  - Add-on to APT
- Technology Status
- Technology Demonstration
- Conclusions

General Features of An Accelerator Based System



Heavy Metal Target

- 18 to 20 neutrons/proton (typ)
- Fission-like neutron source spectrum with high-energy tail
- Highly thermalized spectrum in surrounding moderator
- Minimal radioactivity per source neutron

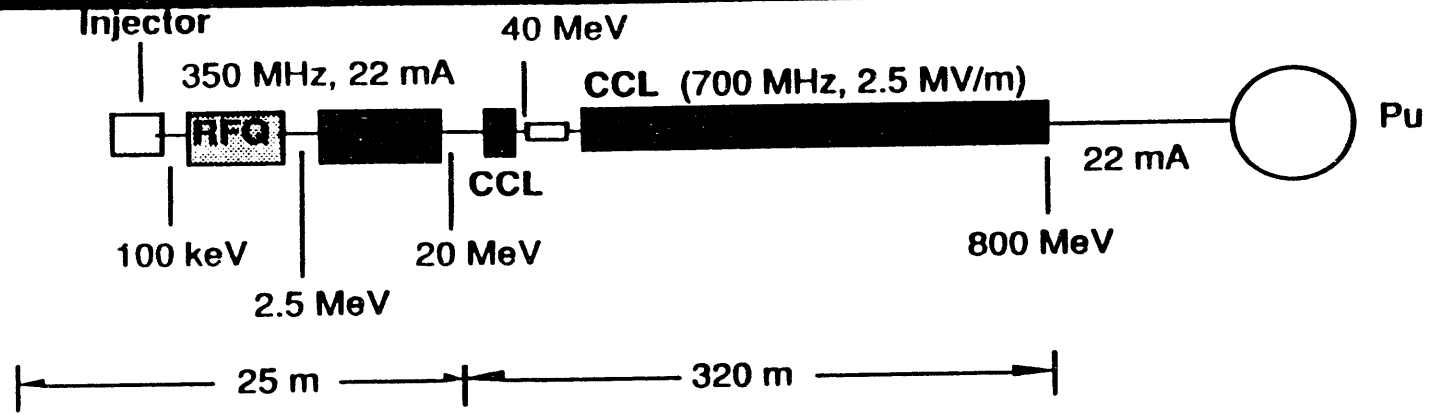
Technology Applied To A Spectrum of National Needs

Simplest — Technology Base — More Complex

Materials Production	Defense Waste	Commercial Waste	Energy Production
Accelerator			
Target			
Blanket			
Aqueous Chemistry			
		Power Production	
		Molten Salt Chemistry	

Accelerator-Driven ^{238}Pu Production System

Stand-Alone System (APP)



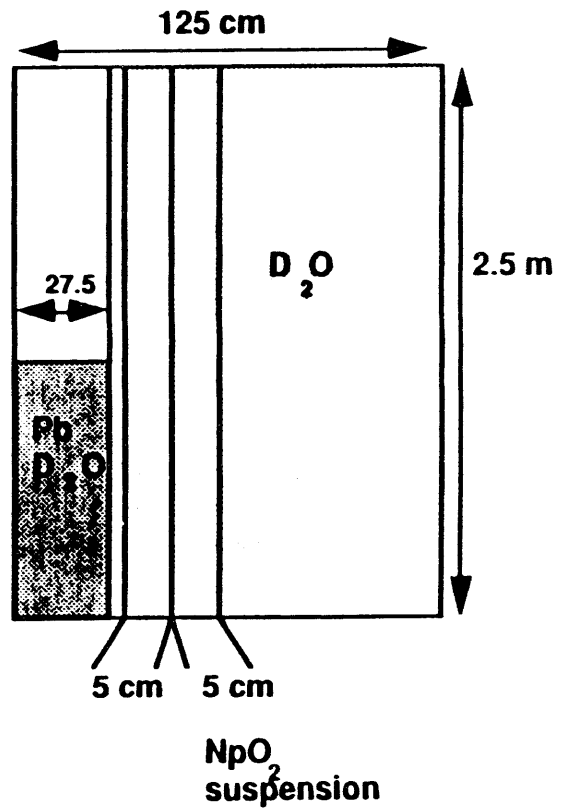
- **Components**
  - 22 ma, 800 MeV accelerator (pulsed, 18% DF)
  - Pb target and D_2O blanket
  - 23 kg inventory for 15 kg/yr production

- **<u>Rough</u> cost estimate**

- Accelerator -	200 to 250 M
- Target/Blanket -	25 to 50 M
- Processing-	75 to 100 M
- Operating	50 to 65 M

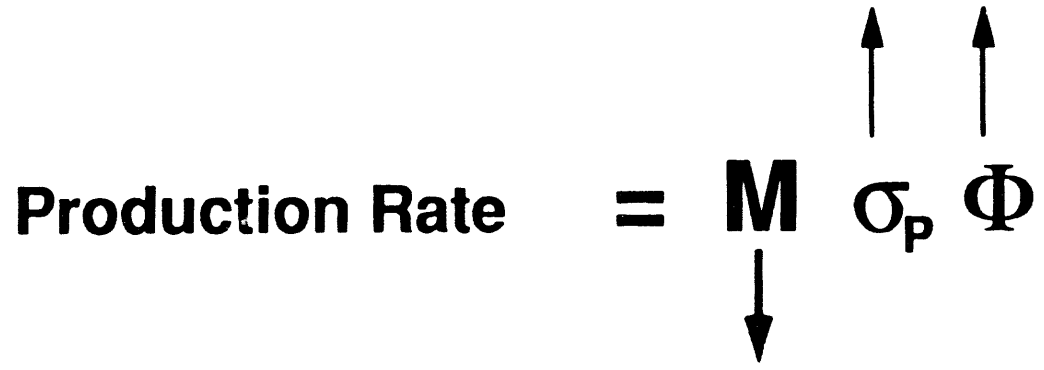
Initial Performance Model Results for 15 kg/yr Production

- 23 kg ^{237}Np inventory
- NpO_2 slurry (suspension) @100 g/liter
- 30 day irradiation, 20 day cooling
- Average production sigma = 100 b
- 2 x10^{-6} ^{236}Pu
- 2% ^{239}Pu
- 1-2 MW_T blanket power
- 800 MeV, 22 mA accelerator



Blanket Environment Allows Production Conditions to Be Optimized

In core inventory of feed material can be minimized



- Location in blanket allows use of maximum effective production cross section
- Choice of accelerator current range allows increase in flux
- Processing strategy chosen to produce minimal parasitic absorption, contaminant production

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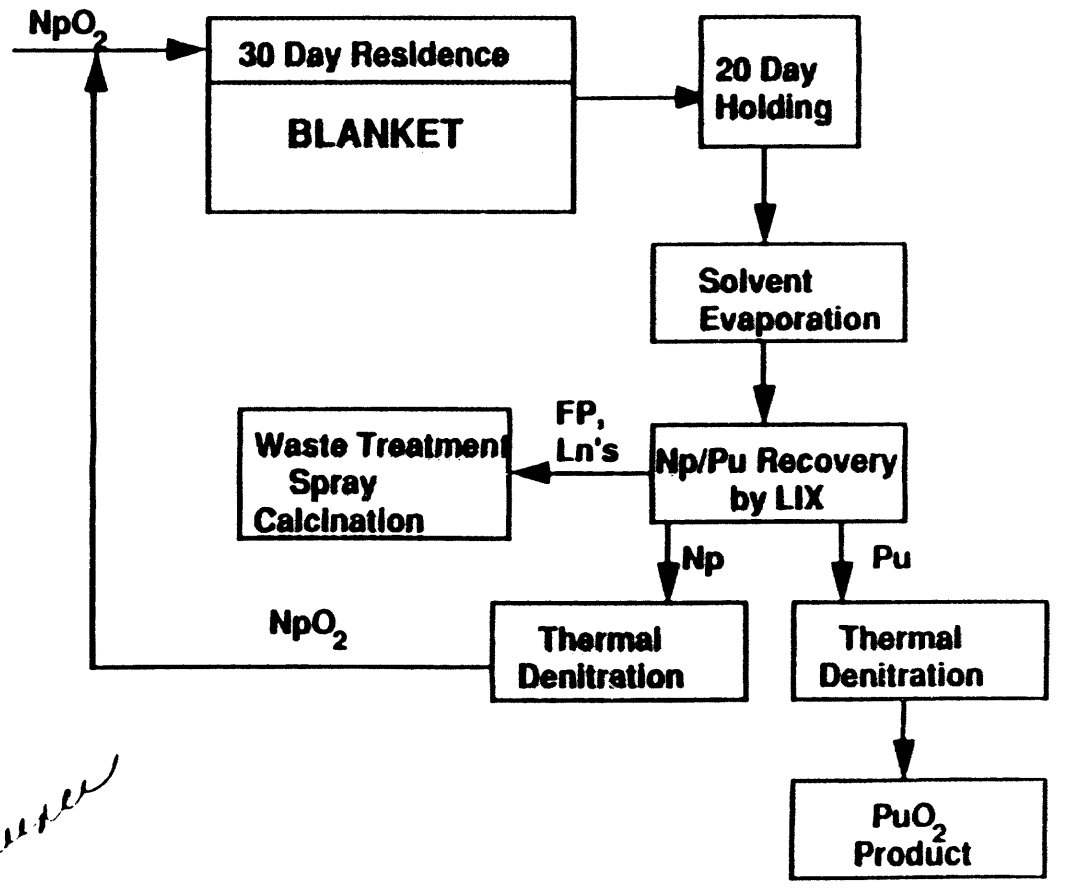
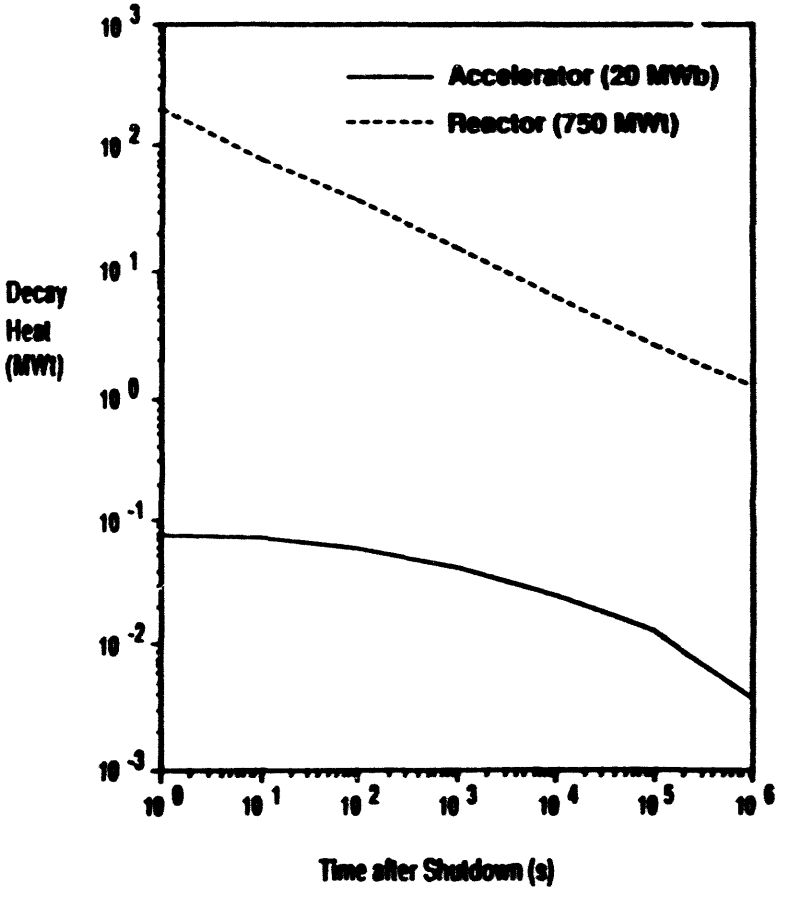
Processing Flowsheet

- NpO_2 suspension can be preproduced
- Small capacity processing system
- High Np, Pu separation
- Large (> 5000) decontamination factors for FPs
- TA-55 experience with Np target processing



Accelerator System Has Minimal Radioactivity

- Decay power is 1000 times less than an equivalent production reactor
- Small production of $\tau_{1/2}$ > 30 yrs radionuclides in target
- About 30 g of long-lived FP produced/yr



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Additional Options - LAMPF Upgrade

- **LAMPF presently operates at 1mA (can be increased to 2 mA)**

- **Upgrade to 10-15 mA**

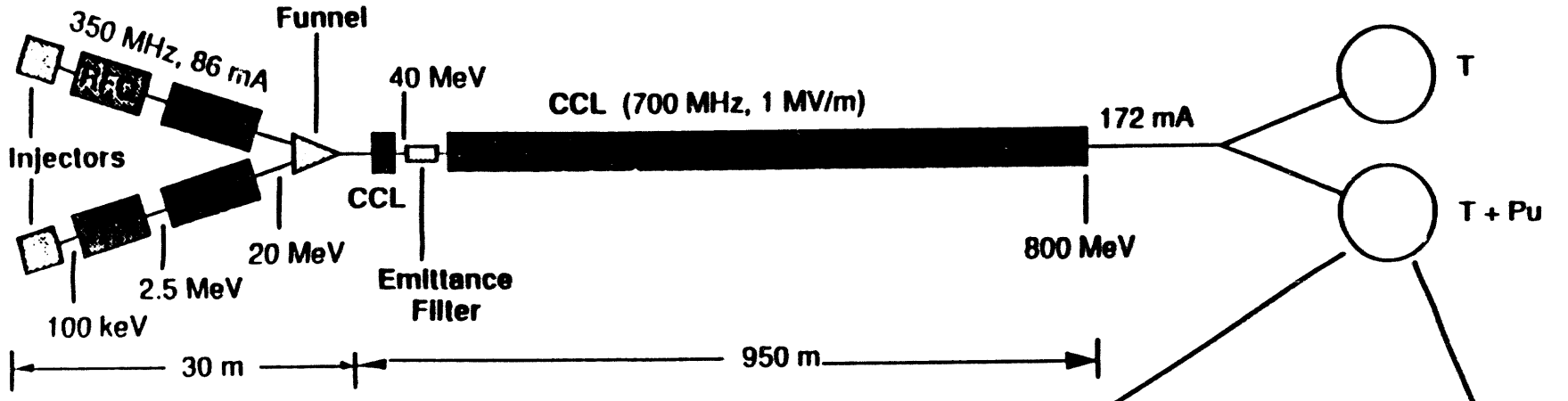
  - Requires refit of first 100 MeV of present accelerator

  - Takes advantage of existing facilities and support crew

  - Could produce 7 to 10 kg/yr if run on year-round basis

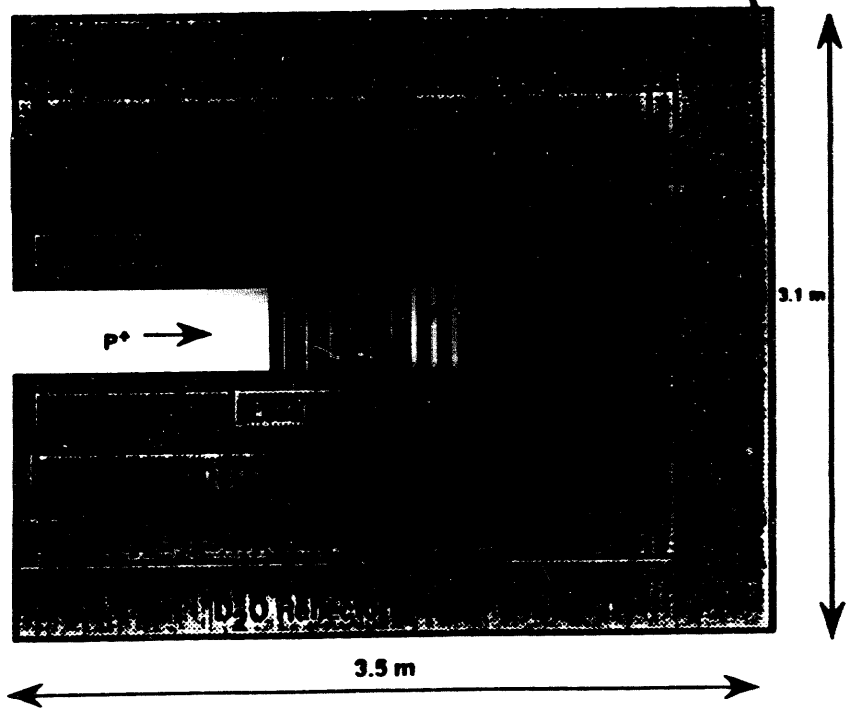
  - Very rough capital cost estimate (accelerator, target,processing) is $200 to 250 M

Additional Options
Add-On to APT System



- Inventory equal to stand-alone system

- Rough estimate of costs assignable to ^{238}Pu mission
  - Capital 175-200 M
  - Operating 25-30 M



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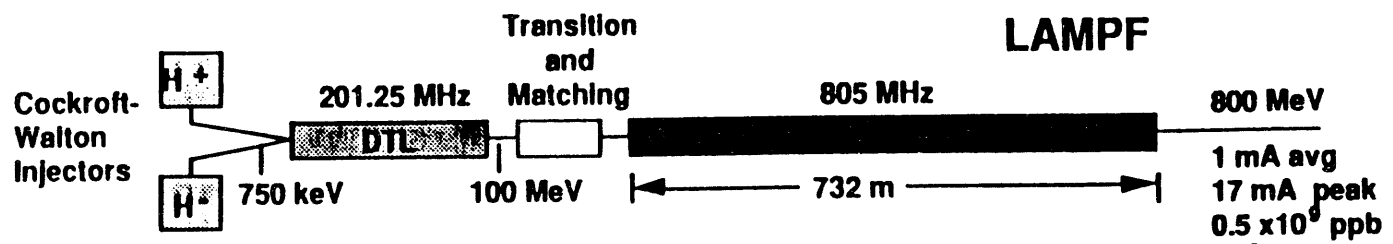
Technology Status of Principal Accelerator-System Components

High Power Accelerators Can Be Constructed Using Current Technology



RFQ Radio-Frequency Quadrupole
DTL Drift-Tube Linac
CCL Coupled-Cavity Linac

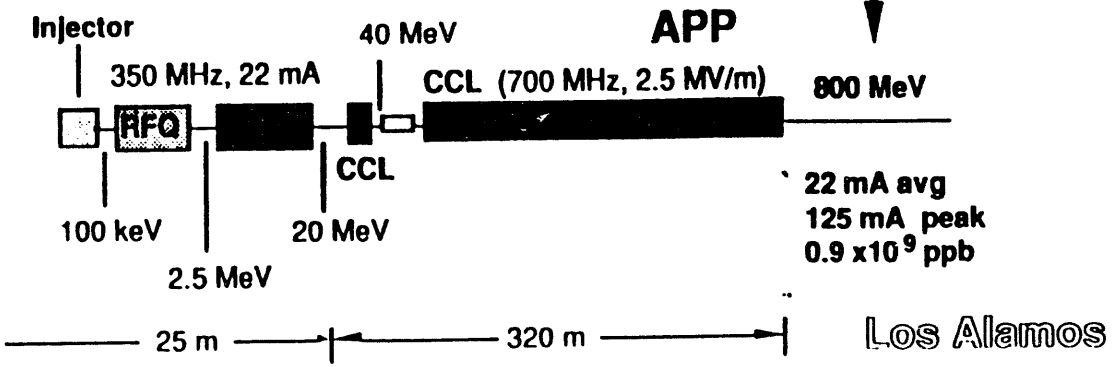
20 Years Technology Development

ATS

100 mA peak
1.5 x10⁹ ppb

GTA

APP



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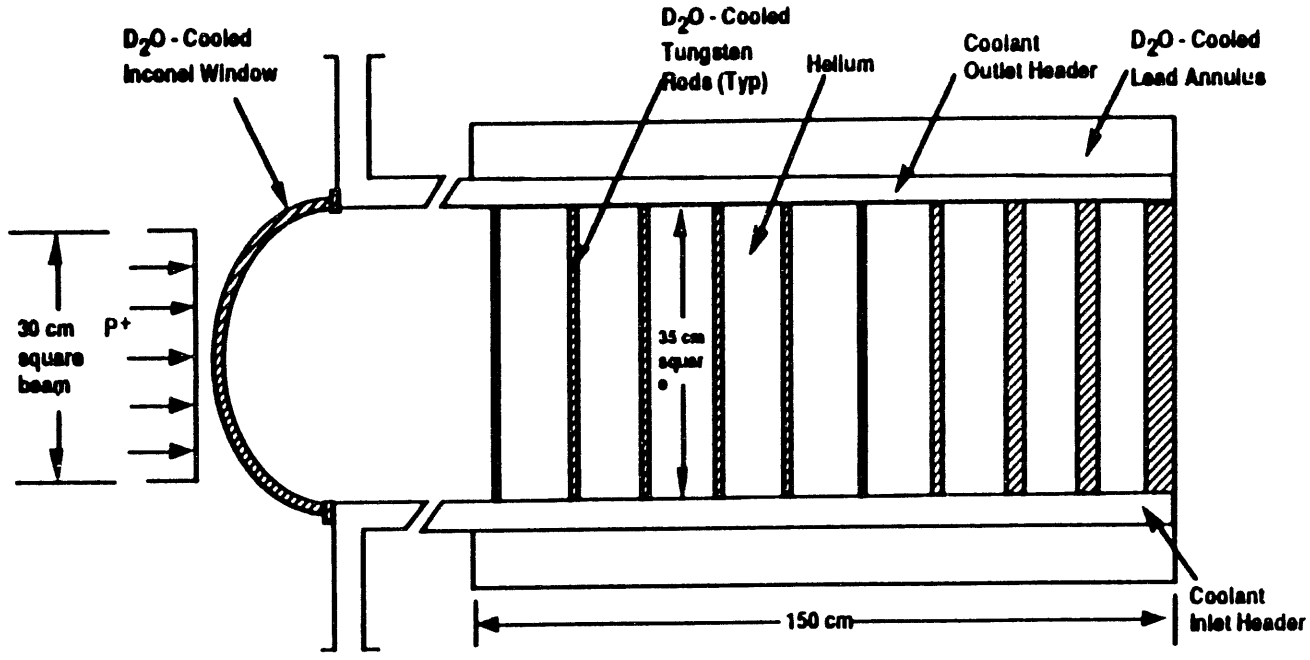
Current Technology Will Allow Construction of Reliable Neutron Target

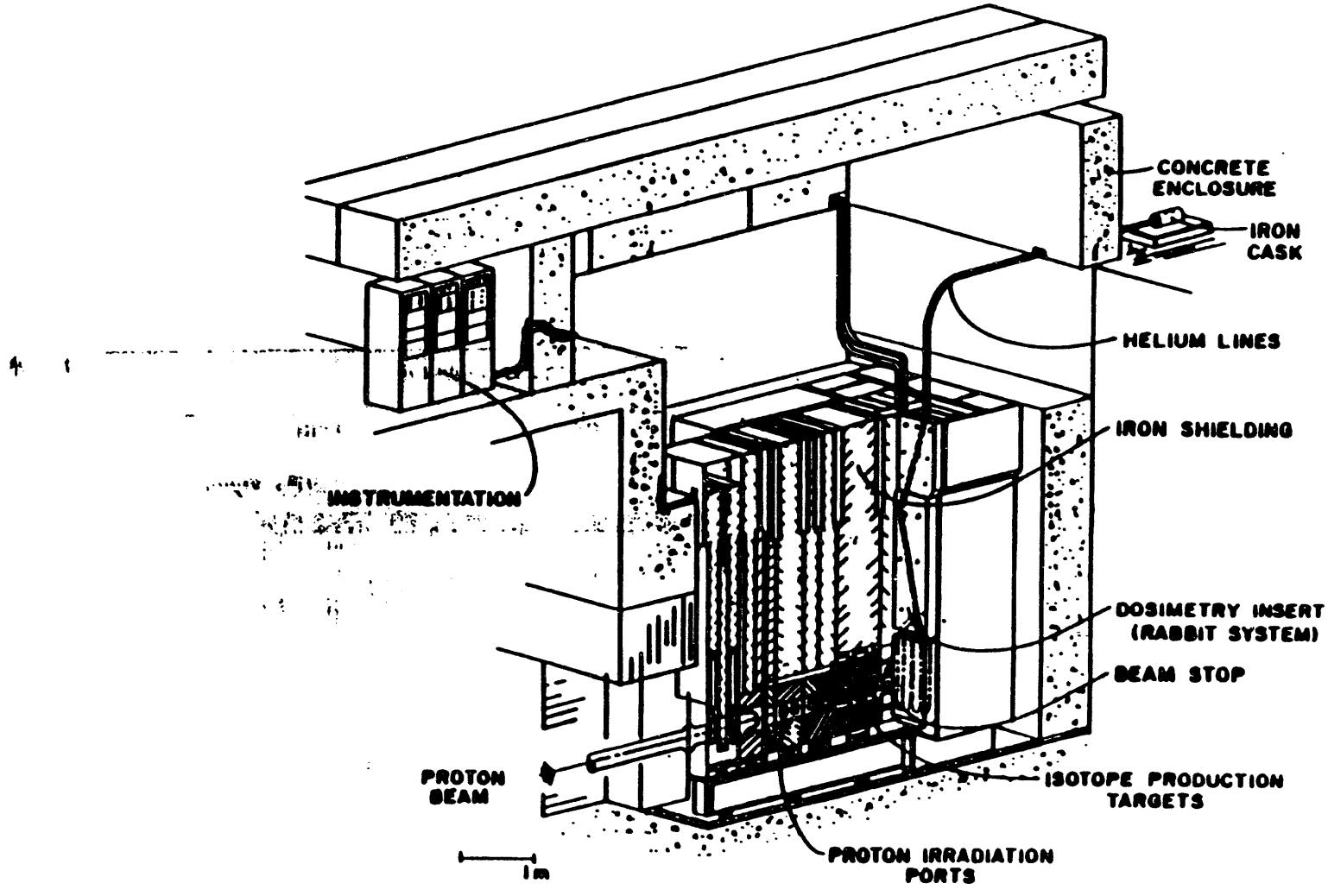


	APP	LAMPF
Proton Flux ($\mu A/cm^2$)	24	30
Fluence ($p/cm^2 \times 10^{21}$)	3.5 (/yr)	13
Power density (MW/m^3)	<2	1.6

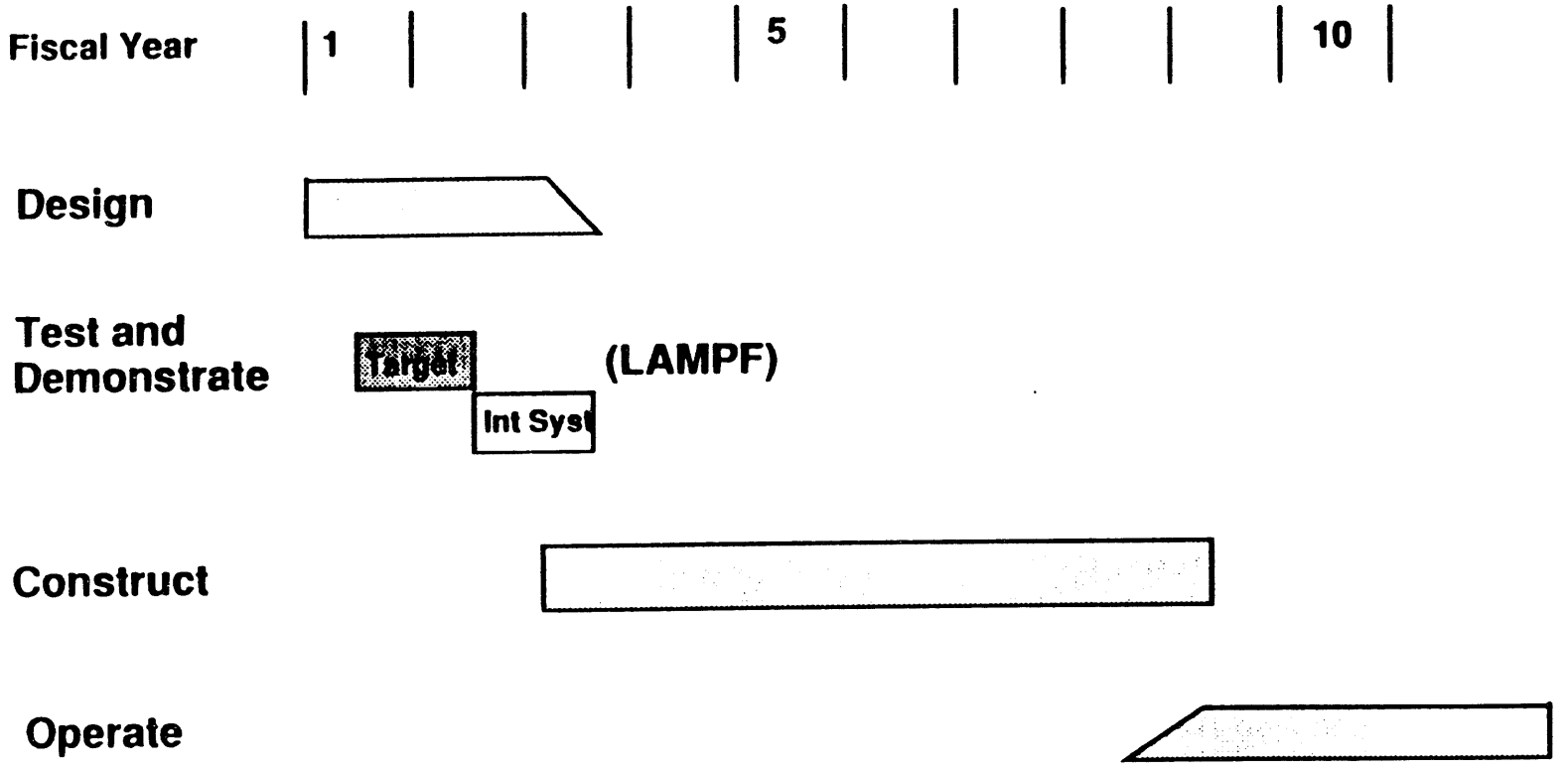
Component Demonstrations

- LAMPF is a key resource for
  - Full target power demonstrations
  - Production rate verification (^{238}Pu/proton)
  - Scaled integrated system demonstration

LAMPF Beamstop Provides Test or Production Demo Environment



APP Schedule



- Operation would begin within 10 years of project initiation
- LAMPF tests would demonstrate target operation (and possibly integrated system performance)

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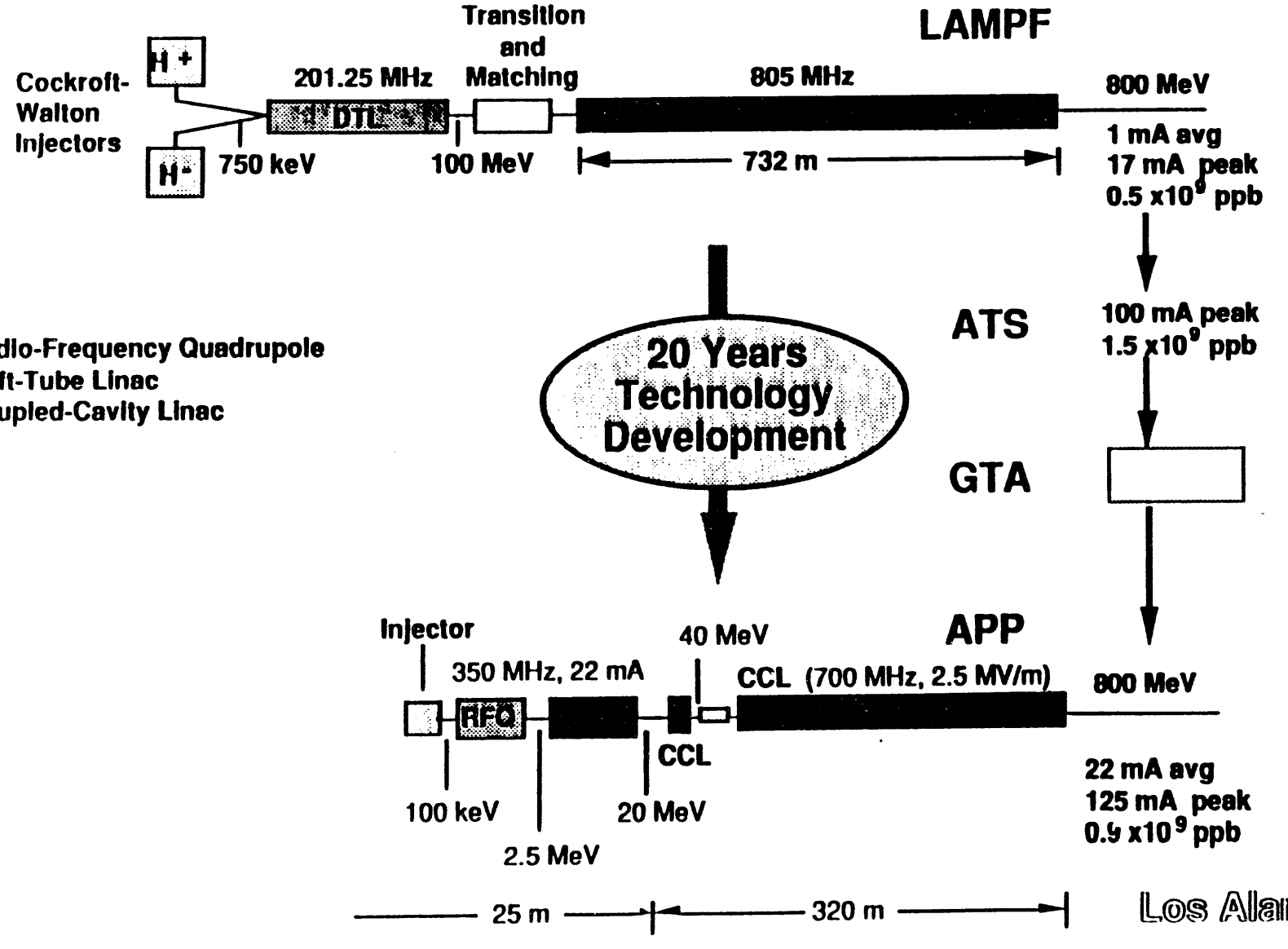
Technology Status of Principal Accelerator-System Components

High Power Accelerators Can Be Constructed Using Current Technology



RFQ Radio-Frequency Quadrupole
DTL Drift-Tube Linac
CCL Coupled-Cavity Linac

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Appendix D

Advanced Liquid Metal Reactor Sponsor Response

GE Nuclear Energy

XL-270-930070
March 9, 1993

Mr. Duane J. Hanson
EG&G Idaho, Inc.
P.O. Box 1625
MS 2508
Idaho Falls, ID 83402

Re: Your FAX transmittals dated 2/17/93 and 2/22/93 requesting responses to several questions related to plutonium disposition.

Dear Mr. Hanson:

We are pleased to respond to the questions you transmitted to me in the referenced FAX's. Subsequent to receipt of the FAX's I transmitted a copy of the ALMR presentation to the National Academy of Sciences, Committee on International Security and Arms Control, Panel on Reactor-Related Options and the ALMR Plutonium Disposition Study report to John Herczeg, DOE-HQ. A few of the following responses refer to the report to Herczeg.

The questions in your FAX's are shown below with the responses following each question. Please note that these responses are preliminary; a more comprehensive DOE-sponsored plutonium disposition study is currently underway in which most of these questions will be addressed in significantly more detail.

If you have any additional comments or questions, please let me know.

Sincerely,

Marion L. Thompson

Marion L. Thompson
(408)365-6481
FAX: (408)365-65643

cc: P.M. Magee
J.E. Quinn
I.N. Taylor
T. Wu
W. Hannum (ANL)

Information to be obtained from Option Sponsors for NAS

The National Academy of Sciences (NAS) has requested the INEL to obtain answers to the following questions from sponsors of reactor options for plutonium disposition. Answers are requested by February 24, 1993.

1. What is the irradiation time required for the fuel to reach a value of 0.10 and 0.20 for the ratio of Pu-240 to Pu-239? (Based on the average core power.)

Response:

See the attached viewgraph with plutonium isotope data; these values change somewhat with various core designs but represent nominal values for the irradiation times indicated.

2. If the goal is to annihilate the fissionable plutonium and processing of the irradiated fuel can be used to recover and recycle plutonium, how many times would a core fuel load need to be recycled and what is the total irradiation (GWD/MT) needed to reduce the initial core plutonium inventory by 90%, 95%, and 99%?

Response:

See the response to question No. 1 under the Reactor or Accelerator System section below.

The NAS requested that INEL obtain answers to the following questions from sponsors of reactor options for plutonium disposition.....please send written responses by 3/4/93.....

Fuel

1. Did you assume that plutonium (PuO_2 or Pu metal) used in the fuel would be free of contaminants (alloying metals and americium now in the nuclear weapon pits)? Will the fuel proposed be negatively impacted if plutonium is contaminated with these alloying metals and the americium?

Response:

No, it was not assumed the Pu would be free of contaminants. For the ALMR, metal fuel is the reference fuel form, and plutonium with alloying metals and americium are acceptable. Pu cleanup can be readily accomplished in the fuel process electrorefiner, if needed. Also, the americium serves as fuel in the fast neutron spectrum of the ALMR, therefore the Pu fuel would not have a negative impact on the ALMR fuel system.

2. Briefly describe the technical work scope necessary to complete development of the fuel and its estimated duration.

Response:

For the standard ALMR fuel and ALMR fuel for the moderate burner, (e.g., 0.6 conversion ratio) the fuel is considered to be developed. The developed fuel for the maximum burner (e.g., minimal or no uranium in fuel so as to achieve a 0.02 conversion ratio) requires some development work for both fuel fabrication and performance testing. Preliminary evaluation of this fuel, however, indicates no undue strain on the ALMR actinide recycle system.

3. Identify technical issues that could impede fuel development and fabrication. For example, have all issues related to material lifetime, compatibility, etc., been resolved?

Response:

Fuel process development is ongoing and demonstration of this pyrometallurgical process is scheduled to be completed by 1996 by Argonne National Laboratory (ANL). Compatibility (e.g., with cladding, etc.) is considered to be established. Testing of material to its lifetime is ongoing and the design codes have been updated to provide design information related to fuel lifetime and compatibility.

4. What are the current cost estimates for fuel development and for the fuel fabrication facility construction, startup, and operation? What estimating method was used (e.g., parametric, historical cost, unit cost, etc.)?

Response:

The current construction cost estimate for a fuel cycle facility to support one ALMR plant (1500 MWe) is about $120M. The operating cost is estimated at about $25M per year, including manpower and fuel hardware. The estimating method is primarily based on historical costs of similar facilities, but with upgrading of the developed process equipment as required to perform the batch based pyroprocess.

Reactor or Accelerator System

1. If the plutonium disposition goal is to annihilate Pu-239 and Pu-241 in a single fuel cycle, what is the total burnup or exposure (GWD/MT Pu) required to reduce the initial inventory of these two isotopes by 90%, 95% and 99% (if possible)? For each of these cases, identify the weight percent of all plutonium isotopes in the initial fuel loading and those remaining in the spent fuel after an equilibrium fuel cycle. Also identify the cycle times.

Response:

In the ALMR, average burnup is approximately 10%, thus in theory, any given fuel load would require recycle about 10 times for annihilation. However, the fuel load would be supplemented each recycle with makeup plutonium from outside the system (for a burner) and from breeding in U238 (for a breeder) to sustain irradiation. The main points to be made for the ALMR are that (a) with its recycle capability and hard neutron spectrum it burns the higher isotope TRU which saturates thermal neutron systems and (b) the only TRU disposed to waste from the ALMR are the process losses (estimated to be 0.1%); major advantages over once-through systems. Please see the attached viewgraph for estimated Pu isotopic values at various irradiation times.

2. If the plutonium disposition goal is to annihilate Pu-239 and Pu-241 and irradiated fuel can be reprocessed to recover and recycle plutonium, what is the total burnup or exposure (GWD/MT Pu) required to reduce the initial Pu-239 and Pu-241 inventory by 90%, 95%, 99%, and 99.9%? If the option uses a batch mode fuel cycle, how many times would a core fuel load need to be recycled to reach each burnup percentage? For each of the four cases, identify the weight percent of all plutonium isotopes prior to initiation of irradiation and those in the spent fuel.

Response:

Burnup for the ALMR is expected to have an average of about 100 GWD/MT. But high burnup is not necessarily the answer for Pu annihilation in the ALMR. Since the fuel is recycled, all but about 0.1% of the actinides are retained or burned in the system. And the 0.1% is all that goes to the waste stream. (See the response to No. 1 above).

3. If the plutonium disposition goal is to annihilate all plutonium isotopes and irradiated fuel can be reprocessed to recover and recycle plutonium, what is the total burnup or exposure (GWD/MT Pu) required to reduce the inventory of all plutonium isotopes by 90%, 95%, 99% and 99.9%? If the option uses a batch mode fuel cycle, how many times would a core fuel load need to be recycled to reach each burnup percentage? For each of the four cases, identify the weight percent of all plutonium isotopes remaining in the spent fuel.

Response:

(Same response as shown in No.s 1 & 2 above)

4. Briefly describe the technical work scope necessary to complete development of your reactor or accelerator system and its estimated duration.

Response:
See Section 2.4, Technical Availability, in the Herczeg Report.

5. Identify technical issues that could impede system development, design, construction, and startup. For example have all issues related to material lifetime, compatibility, etc., been resolved?

Response:
The system is based on known technology; only feature testing remains; thus, there appear to be no serious impediments to system development.

6. What are the current cost estimates for system development and for construction, startup, and operation of the facility? What estimating method was used (e.g., parametric, historical cost, unit cost, etc.)?

Response:
See Section 2.5, Economics, in the Nerczeg Report. The estimates are primarily based on historical costs upgrading to module (learning curve) fabrication in factories.

Waste Processing

1. Briefly describe the technical work scope necessary to complete development of a waste conditioning/processing flowsheet for your option and its estimated duration.

Response:
See Sections 2.1.2, IFR Description, and 2.4.2, Fuel and Fuel Cycle in the Herczeg Report.

2. Identify technical issues that could impede development of the waste processing flowsheet, systems, or facilities.

Response:
See response to No.1 above.

3. What are the current estimates for the time required for construction and startup of the waste processing facility?

Response:
The waste processing facility is a relatively small contribution to the fuel cycle facility and is contained within the fuel cycle facility. See response to question No. 4 in the Fuel section.

4. What are the current cost estimates for waste processing development and for construction and startup of a waste processing facility?

Response:
See response to No. 3 above. The waste processing development costs are covered in the ongoing fuel process development activity at ANL.

Waste Disposal

1. Relative to the assumed acceptance of commercial fuel and defense waste in a geologic repository, is there waste characterization work that must be performed?

Response:
The current repository waste references are spent fuel and processed high level waste in glass. The metal fuel pyroprocess produces two main waste streams: salt and metal, which do require performance and acceptance testing. However, these waste forms are being developed and tested and are expected to be acceptable to the repository.

2. Briefly describe the technical work scope necessary to complete development of the waste disposal method and its duration. For example, are there any preconditioning or packaging requirements that must be satisfied for repository acceptance?

Response:
The preconditioning requirements should be covered in the testing described in the response to No. 1 above. Packaging is expected to be "standard."

3. Identify technical issues that could impede the placement of waste from your option in a repository.

Response:
The waste performance and acceptance testing is ongoing and is expected to be successful. There are no known impediments for placement of waste from the ALMR pyroprocess in the repository.

4. What are your estimates for the elapsed time prior to opening a suitable repository?

Response:
It is assumed that the current repository schedule, for example, opening in about 2010, is compatible with the ALMR system development and deployment schedule.

5. What are the current cost estimates for waste disposal system development and for disposal of the waste?

Response:
The waste disposal system being developed by DOE for the first high-level waste repository is considered to be acceptable and therefore the cost is the cost of that program. The cost of disposal of ALMR processed fuel is expected to be less than the current 1 mill/kwh fee. For example, it is expected that the cost on an equivalent basis would be a half to three-quarters of a mill/kwh.

6. Does radiation make your proposed waste package "self protecting" (i.e., greater than 100 R/hr at 3 ft from the surface)? If so, how long does it remain self protecting?

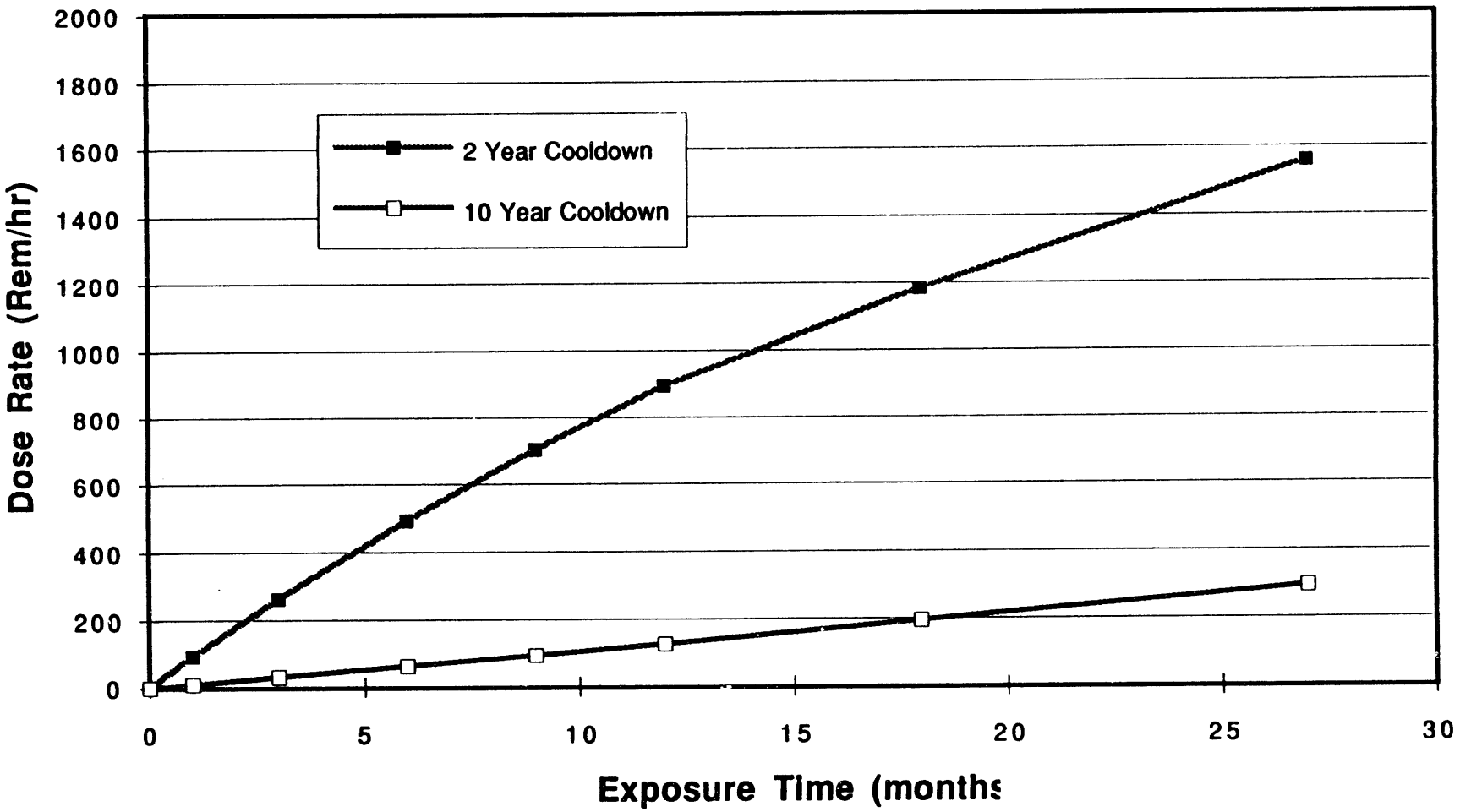
Response:

Yes, please see the attached preliminary figure, Discharge Assembly Dose Rate, for the estimate of the exposure obtained and decayed over time, with two relatively short ALMR fuel burnup times.

Plutonium Isotopic Data

	Estimated Defense Pu	Estimated "Denatured" Value of Defense Pu After One 5-year Cycle	Estimated ALMR "Equilibrium" After Several cycles
Pu-238	---	0.03	0.40
Pu-239	94.00	85.60	72.44
Pu-240	6.00	13.65*	23.28
Pu-241	---	0.69	2.66
Pu-242	---	0.03	1.22
	100.00	100.00	100.00

* Estimated to be 8.6% after one 20-month cycle and 11.2% after two 20-month cycles; a 5-year cycle represents three 20-month cycles and the normal life of fuel in the ALMR.


2000
1800
1600
1400
1200
1000
800
600
400
200
0
Dose Rate (Rem/hr)
2 Year Cooldown
10 Year Cooldown
0
5
10
15
20
25
30
Exposure Time (months

Appendix E

Advanced Light Water Reactor Sponsor Response

Plutonium Disposition Replies for Duane Hanson of INEL

Following are the replies requested by Duane Hanson of INEL as input to the NAS questions about plutonium disposition:

Question 1. What is the irradiation time required for the fuel to reach a value of 0.10 and 0.20 for the ratio of Pu-240 to Pu-239 (Based on the average core power)

A parametric study was conducted by SRS personnel to deduce the Pu-240 build-up rate for Weapons Grade (WG) plutonium that is employed as a mixed oxide (MOX) fuel in pressurized water reactors. A 600 MWe reactor with a specific thermal power rating of 12.5 MWt per assembly was used as a basis. One part of the study was conducted with only one-third MOX fuel loading (to reflect the current practice with Westinghouse designed, European licensed reactors), while the second part of the study used full core loading of MOX fuel.

The calculated Pu-240 build-up rate is given in Figures 1 and 2, for the one-third and full MOX fueled cores, respectively. Plutonium enrichment of the MOX was varied from 2% to 9% and each cycle was burned until 20% Pu-240 was attained or it was obvious that a practical cycle length was too short to meet this criterion. Cycle lengths assume a 75% capacity factor (i.e., 1 year on the time scale represents 9 months of full power operation).

Of note is the inability of high plutonium loadings to reach 20% Pu-240. In these cores the fissile loading is relatively high while the fraction of Pu-239 burned is relatively low, and thus the fractional isotopics in the WG plutonium do not change significantly over practical cycle lengths. The lower plutonium enrichments reach about 10% Pu-240 in about 3-4 months and 20% Pu-240 in about one year, respectively, for the one-third MOX fueled core. The full MOX loaded cores require about twice those lengths.

Figure 1

Plutonium Denaturing For One-Third MOX Core

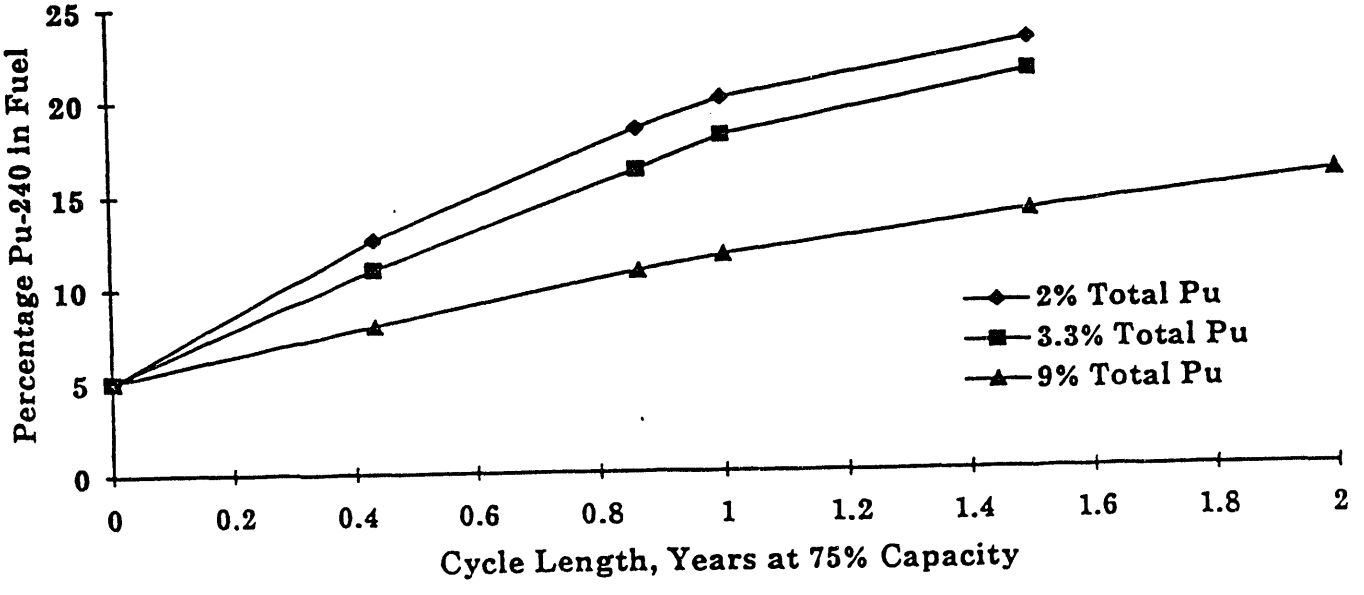
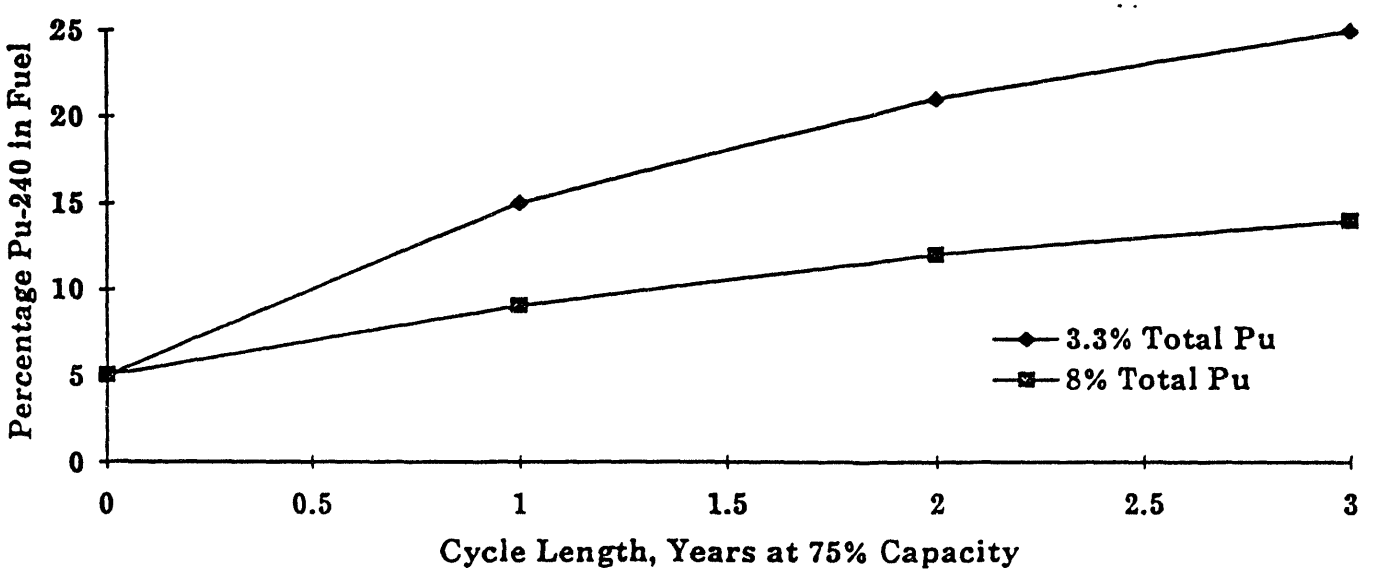


Figure 2

Plutonium Denaturing For Full MOX Core



Plutonium Disposition Replies for Duane Hanson of INEL

Question 2. If the goal is to annihilate the fissionable plutonium and processing of the irradiated fuel can be used to recover and recycle plutonium, how many times would a core fuel load need to be recycled and what is the total irradiation (GWD/MT) needed to reduce the initial core plutonium inventory by 90%, 95%, and 99%?

For this option, it is desirable to contain the plutonium in a fuel containing no U^{238} so that no new plutonium would be produced. The preferred fuel is likely to be "Ternary Fuel" in which PuO_2 is contained in a matrix of ZrO_2 and CaO. At present we are assuming about 14 wt % PuO_2, 77 wt % ZrO_2 and 9 wt % CaO. There is extensive irradiation experience with this type of fuel, both with highly enriched uranium and plutonium as the fissile material, and fuel performance has been very good. The fuel pellets would be annular surrounding a core of ZrO_2 - CaO. Boron would likely be added to either the core or the fuel to serve as a burnable poison.

Reactor operation would have about a two-year operating period, after which one-third of the core would be loaded with fresh fuel. Thus a fresh fuel assembly would remain in the reactor about six years. At the end of this time 80-90% of the fissile plutonium would have been destroyed. If it is desired to increase this fraction, the burned fuel could remain in the reactor longer, but other fuel would have to be discharged and stored for later use. That is, the one-third core reload fuel management would be disrupted. Possibly one-quarter core reload could achieve destruction fractions greater than 90%. If much higher destruction fractions are desired, chemical reprocessing of the burned fuel would probably be required.

The (GWD/MT initial fissile Pu) needed to reduce the initial fissile plutonium inventory by 90%, 95% and 99% are given below.

Fraction Fissile Pu Destroyed	GWD/MT Of Initial Fissile Pu
90%	781
95%	817
99%	845

Note that these exposures are defined entirely differently from GWD/MTHM that are conventionally quoted.

RESPONSES TO NAS PLUTONIUM DISPOSITION OPTION QUESTIONS

Q1. Did you assume that plutonium (PuO_2 or Pu metal) used in the fuel would be free of contaminants (alloying metals and americium now in the nuclear weapon pits)? Will the fuel proposed be negatively impacted if plutonium is contaminated with these alloying metals and the americium?

Answer:

Gamma activity builds up continuously after purification from the decay of Pu241 to Am241. Each year about 5% Pu241 decays. Some of the weapons grade plutonium has been around for awhile and will contain americium but at concentrations much less than 1%. Because Am241 is a gamma emitter, there may be difficulty in maintaining and achieving the DOE requirement of keeping exposure to personnel as low as reasonably achievable (ALARA) during fabrication and handling. Thus, returns will typically be reprocessed to remove any Am241 and other contaminants.

According to the process outlined in Strategies for Denaturing the weapons-Grade Plutonium Stockpile, WSRC-RP-92-1004, (pp 4-1 to 4-3), pits will be dissolved in sulfamic acid and passed through solvent extraction which will remove americium and other contaminants before the oxide is precipitated. However, some pits may have to be processed first by a hydride technique. They will be exposed to hydrogen gas to remove plutonium and eliminate other impurities before further processing. Plutonium oxide used to make fuel elements will be free of americium and contaminants.

Q2. Briefly describe the technical work scope necessary to complete development of the fuel and its estimated duration.

Answer:

For MOX (PuO_2-UO_2) fuel, there is essentially no development needed for fuel manufacturing. Similar processes are currently being used in Europe and Japan to make MOX fuel elements. The concept has also been evaluated by Westinghouse as discussed in the above report (p. 2-1). The basic techniques of pelletizing, sintering, grinding and assembly are similar to processes used in the U.S. to make UO_2 power reactor fuel.

"Ternary Fuel" has been proposed for complete destruction of plutonium, and it is fully developed with apparently no additional work required. The fuel form consists of solid or annular pellets of PuO_2, ZrO_2 and CaO_2. This special fuel has been manufactured by Westinghouse Electric Corporation, Advanced Energy Systems Division and has been evaluated in U.S. and Foreign programs.[1]

However, ternary fuel will require further evaluation. With highly enriched UO_2 as the fuel material in ternary, there is essentially no Doppler feedback. This will have to be evaluated for PuO_2 as the fuel material. Also, ternary fuel has performance concerns for beyond design basis accident scenarios. For fuel meltdown, the 3 ternary fuel components separate and recriticality of the PuO_2 becomes a concern. This would have to be investigated for the particular features of the ALWR concept.

Q3. Identify technical issues that could impede fuel development and fabrication. For example have all issues related to material lifetime, compatibility, etc., been resolved?

Answer:

None known at this time. All issues have been resolved because MOX fuel is currently being used in other countries to produce electrical power.

Q4. What are the current cost estimates for fuel development and for the fuel fabrication facility construction, start-up, and operation? What estimating method was used?

Answer:

Cost estimates for facility construction are estimated in the document WSRC-RP-92-1004. Costs are included for constructing a "Greenfield" facility and for using existing WSRC and AGNS facilities. The costs are given on page 4-3 to 4-5 and were estimated using the PC-based Freiman Analysis of System Technique (FAST) which has been used by DOE to provide uniform cost estimating for reconfiguration studies (p. 4-1).

These estimated cost values are also compared to published costs for MOX plants in Germany and England (page 4-3). The high spot estimate for the WSRC "Greenfield" plant was $680 million (~70 MTHM) while the cost for Germany's plant was $500 million (120 MTHM) and the cost for BNF's plant was $400 million (100 MTHM). The differences in WSRC costs are assumed to be due to rigorous safety standards imposed by DOE for new construction of nuclear facilities.

The operating costs was estimated to be about $30 million per year. The startup costs will be higher, but no estimates are available at this time.

Reference:

1. WSR-84-252, "Ternary Fuel Performance Data for the Special Water Reactor," Westinghouse Electric Corporation -AESD.

Reactor System

Q1-Q3. Combined Response regarding annihilation options:

It appears that satisfactory annihilation of Pu239 and Pu241, with resulting Pu unusable in weapons, can be achieved by irradiating fuel assemblies in multiple fuel cycles without chemical reprocessing. Therefore, only question 1 out of questions 1-3 will be addressed.

For this plutonium disposition goal, it is desirable to contain the plutonium in a fuel containing no U^{238} so that no new plutonium would be produced. The preferred fuel is likely to be "Ternary Fuel" in which PuO_2 is contained in a matrix of ZrO_2 and CaO. At present we are assuming about 14 wt.% PuO_2, 77 wt.% ZrO_2 and 9 wt.% CaO. There is extensive irradiation experience with this type of fuel with highly enriched uranium as the fissile material, and fuel performance has been very good.[1] The fuel pellets would be annular surrounding a core of ZrO_2-CaO. Boron would likely be added to either the core or the fuel to serve as a burnable poison.

The reactor operation which has been calculated thus far would have a 2.3-year operating period, after which a fraction of the core would be loaded with fresh fuel. This fraction was assumed to be 1/3 initially and thus a fresh fuel assembly would remain in the reactor for 3 operating periods, or seven years. Irradiation for longer periods was calculated by simulating leaving the most irradiated fuel in the reactor for one or more additional operating periods while simulating 1/3 core reload for the other 2/3 of the reactor. If some particular number of operating periods (say 5) produced irradiated fuel deemed acceptable, then a 1/5 core reload scheme would be designed to produce this material on a continuous basis. Results of the more approximate calculations are given below for 3, 4, 5 and 6 operating periods.

Number Of Operating Periods	Years In Reactor	Pu239 + Pu241 Initial Pu239 + Pu241 %	Total Pu Initial Total Pu %	GWD MTPu (initial)
3	6.9	13.59	24.88	697.7
4	9.2	7.04	15.23	778.9
5	11.5	3.31	9.61	822.7
6	13.8	1.40	6.59	843.3

	Wt. % Pu Isotope In Total Pu				
Nbr. Of Operating Periods	238	239	240	241	242
0	0	94.00	5.70	0.30	0
3	2.88	20.92	29.90	30.58	15.72
4	6.99	10.49	18.29	33.10	31.13
5	12.53	6.16	6.46	26.29	48.56
6	17.34	4.97	2.85	15.01	59.83

The above tables cover the range of interest implied by question 1. The values of GWD/MTPu (initial) for the specific points in question 1 were obtained by interpolation and extrapolation and are 741 (90%), 801 (95%) and 850 (99%). Furthermore, it is clear that the material after 6 operating periods could not be used in a weapon. We feel confident that the Weapons Labs would reach this conclusion based on the Pu239, Pu241 and Pu242 relative contents alone. But there is a further reason. The high Pu238 content (from decay of Cm242) would result in any recovered plutonium metal having quite a high temperature sitting in air. At the time when various non-proliferation schemes for civilian plutonium were being studied, spiking plutonium with Pu238 of this or lower content was proposed as a method of rendering it unusable in weapons.

Six irradiation periods may well be needlessly conservative. If three irradiation periods were adequate (using 1/3 core reload), material throughput estimates can be given now. One-third of a fresh core would contain 1.38 MT of weapons Pu. Since the operating period is 2.3 years, the average yearly fueling requirement would be 0.60 MT/reactor year. If only 15 years of reactor operation is available, it would take eleven 600 MWe reactors devoted to this purpose to accept the 100 MT of plutonium. It is implicit here that beginning irradiation satisfies the requirement of taking the plutonium out of the weapons stockpile. If longer irradiation is required, the number of reactors required would be greater than 11.

Q4. Briefly describe the technical work scope necessary to complete development of your reactor or accelerator system and its estimated duration.

Answer:

The advanced light water reactor (ALWR) has an extensive history of planning and engineering that goes back to the early 1980's (i.e., more than 700,000 engineering man-hours on the AP-600 as of 11/91). Recently, two ALWR designs won the financial support of a 16-utility consortium called the Advanced Reactor Corp. (ARC), which has control of over $150 million in

detailed engineering design funds. The winning designs are the Westinghouse AP-600, a 600 MWe passive design pressurized water reactor, and the General Electric ABWR, a 1300 MWe evolutionary design boiling water reactor. The ARC money will be used to perform "first-of-a-kind" engineering for both designs, making them essentially complete by 1996. Both G.E. and Westinghouse also expect to receive design certifications from the Nuclear Regulatory Commission (NRC) in early 1996 and late 1996, respectively. Design certification is equivalent to having a licensed reactor design, which will allow utilities to avoid the long, drawn-out process that was formerly required for each plant on a case-by-case basis. Construction of the first new reactor could presumably start on the day that NRC certification is received.

Q5. Identify technical issues that could impede system development, design, construction, and startup. For example, have all issued related to material lifetime, compatibility, etc. been resolved?

Answer:

There are no real issues that could impede reactor system development, design, etc. for the ALWR. If fuel loading were based on a 1/3 MOX core, no design changes would be required because a core with 1/3 MOX fuel has characteristics that are still within the safety analysis envelope of a full UO_2 core. If a full MOX core were used, sensitivity studies would have to be performed to determine the total control rod worth requirements under various conditions (e. g. rod ejection accident, shutdown margin). Options that could be used to meet the 100% MOX core criteria include:

- Increased number of control rods to compensate for the reduction in rod worth due to a 100% MOX core (might have to add 4 to 8 rods for an AP-600).
- Increased rod worth by using enriched Boron 10 control rods.
- Proper choice of a fuel management scheme to address shutdown margin concerns.

Aside from changes associated with the options listed above, there would be no redesign of the ALWR to accommodate MOX fuel. This assessment is based on the extensive experience with MOX fuel worldwide since the 1960's.

Q6. What are the current cost estimates for system development and for construction, startup, and operation of the facility? What estimating method was used (e. g. parametric, historical cost, unit cost, etc.)?

Answer:

As noted in the answer to Question 4, "first-of-a-kind" engineering costs are being heavily subsidized by ARC. The remainder of these "development" costs will be borne by the reactor vendors. Capital cost estimates for a scheme involving one 600-MWe ALWR and an associated MOX fuel fabrication plant

are estimated in the range of $1.5 to $2 billion. For a scheme involving three 600-MWe ALWRs and an associated MOX fuel fabrication plant, the capital costs are estimated to be $4 to 4.5 billion. Annual operating costs are about $110 million for the single reactor scheme and about $250 million for the three reactor scheme. These 1992 dollar estimates were derived in Reference 2 and are based on commercial industry and SRS data, using the Freiman Analysis of System Technique (FAST). The FAST technique, which is also used by DOE, involves parametric cost estimating.

References:

1. WSR-84-252, "Ternary Fuel Performance Data for the Special Water Reactor," Westinghouse Electric Corporation -AESD.

2. M. R. Buckner et al., "Strategies for Denaturing the Weapons-Grade Plutonium Stockpile, WSRC-RP-92-1004, Aiken, SC, October 1992.

Waste Processing

This is a response to the questions from the National Academy of Sciences regarding waste processing requirements for MOX fuel.

General Assumptions:

1. Waste processing is only required where fuel is reprocessed. This would be the case, for example, if complete burnup of the Pu is desired.

2. Reprocessing options would include either use of an existing government facility or construction of a new dedicated facility. Existing government facilities include the INEL chemical reprocessing plant and the SRS recycle facilities. Use of decommissioned government facilities at the Hanford site is not considered. DOE has committed to the Idaho government to shut down the ICPP operation without further reprocessing, but as this plant is still operable, it is considered here.

3. Reprocessing of fuel containing zirconium oxide (the so-called ternary fuel option) is much more complex than reprocessing uranium-plutonium oxide fuel. Highly corrosive acidic fluoride solutions are required to dissolve the fuel for reprocessing. Development of a process flowsheet for such fuel and qualification of a suitable waste form would require an extensive (probably >1 year) research and development program, because plutonium-based fuels of this type have not been reprocessed previously.

   The costs of constructing a new facility or converting a Savannah River facility for reprocessing ternary fuel would also be higher because of the need for special corrosion-resistant piping and vessels. However, the ICPP already has

a head end facility suitable for use with acidic fluoride solutions. The responses below are for MOX fuel only, and do not consider the additional costs associated with ternary fuel.

As is indicated in the responses on the reactor system, there is no need for reprocessing the ternary fuel since adequate annihilation can be obtained by irradiating fuel assemblies in multiple fuel cycles.

Responses to Questions:

Q1. Briefly describe the technical work scope necessary to complete development of a waste conditioning/processing flowsheet for your option and its estimated duration.

A. Fuel reprocessing by the PUREX process was described in detail during the Geneva conferences in the 1950's, and has been practiced extensively since. In the United States this technology has been used at Hanford, Savannah River, Idaho, and in the commercial reprocessing plant at West Valley, New York. The concentrated high level wastes from PUREX reprocessing can (after an appropriate period of cooling) be fed directly into calciners for making glass waste forms. This technology has been used in France for over fifteen years. Similar technology has also been developed in Germany, the United Kingdom, and Japan for the disposition of wastes generated in their reprocessing operations. A chemical treatment and concentration flowsheet was developed and used in this country for the West Valley wastes. These will be converted to glass during the next few years. The Defense Waste Processing Facility at Savannah River uses similar technology; the principal difference is that this operation requires several preliminary process steps to remove the large quantities of aluminum in SRS wastes. For a new facility the technical work associated with waste processing would primarily involve non-radioactive tests of equipment and components to optimize operating conditions and ensure adequate capacity. These would be performed with the individual components as they become available, and in cold tests of the system after installation.

B. Reprocessing at the Savannah River Site would require installation of a shear-leach head end facility in an existing plant. It has been estimated that such a head-end facility could be installed in a Savannah River canyon for approximately $110,000,000.[1] It might be possible to do without shear-leach dissolution at Idaho, because dissolution with acidic fluorides is possible there. However, at the Idaho plant it would be necessary to provide a second plutonium solvent extraction cycle, as well as facilities for concentrating plutonium and converting it to oxide; cost figures for such new facilities are not available, but by comparison with other recent construction within DOE, the costs would be in the range of hundreds of millions of dollars.

Flowsheets and equipment are presently in place at SRS for converting high-level wastes from reprocessing to glass waste forms for disposition in a repository. Flowsheets and equipment are presently in place at the ICPP for

converting similar waste to dry calcine for storage. A new facility will be required for converting this waste calcine, and the large amount of existing waste calcine now stored at Idaho, to a resistant waste form suitable for disposition to a repository.

In summary, little development would be needed to derive flowsheets for waste processing. However, there is no existing facility that can reprocess LWR mixed oxide fuel and recover plutonium without significant modification.

Q2. Identify technical issues that could impede development of the waste processing flowsheet, systems, or facilities.

As the response to the previous question indicates, there are no outstanding questions regarding a waste processing flowsheet. If disposition of leached segments of cladding is necessary, a process for encapsulating these into a suitable waste form would have to be chosen and tested.

Q3. What are the current estimates for the time required for construction and startup of the waste processing facility?

A new facility (or major addition to an existing facility) would be a line item project and could be constructed on a schedule comparable to that of a dedicated nuclear reactor for plutonium burning, that is, within six or seven years. Reprocessing and waste disposal facilities would not be required until at least a year after the first discharge from the reactor. They could be further deferred for some years if desired, because it would be at least fifteen years before all of the initial charge of plutonium is used.

As noted above, waste processing facilities are presently available at the Savannah River Site. They will eventually be required at ICPP to process existing calcined waste.

Q4. What are the current cost estimates for waste processing development and for construction and startup of a waste processing facility?

Approximate costs for converting an existing government facility are discussed above. Cost estimates have not been developed for a dedicated facility of the relatively small size required to support these operations. Comparison with other facilities suggests that any such facility would cost in excess of a billion dollars.

Reference:

1. J. M. McKibben, "Disposition of Non-Processible Fuels at SRS (U)," WSRC-RP-92-1242, November, 1992.

Waste Disposal

Q1. Relative to the assumed acceptance of commercial fuel and defense waste in a geologic repository, is there waste characterization work that must be performed?

1. No recycling of the irradiated fuel is contemplated. The spent MOX fuel is to be sent to a geologic repository following a period of interim surface storage. Characterization studies for the MOX fuel would have to be factored into the spent fuel characterizations currently in progress for the commercial oxide fuel.

Q2. Briefly describe the technical work scope necessary to complete development of the waste disposal method and its duration. For example, are there any preconditioning or packaging requirements that must be satisfied for repository acceptance?

2. The technical work scope supporting MOX spent fuel is not well defined at this time. We expect that major elements would include:

   a. Spent fuel characterization.
   b. Reactor basin for underwater cooling.
   c. Program for dry storage pending geologic repository storage.
   d. Program to develop treatment facility to repackage fuel following dry storage period in preparation for the geologic repository.
   e. Definition of repository acceptance criteria and fuel qualification procedures with particular emphasis on criticality prevention.

Q3. Identify technical issues that could impede the placement of waste from your option in a repository.

3. The principal technical issues that could impede the placement of spent MOX fuel in the geologic repository are:

   a. Waste form qualifications with particular emphasis on criticality prevention over geologic time periods.
   b. Safeguard assessments for both surface and subsurface storage.

   It should be noted that <u>any</u> spent fuel form (specifically including commercial spent fuel) will have to contend with these issues. A successful commercial fuel qualification program will, no doubt, provide the means to qualify spent MOX fuel for they have very similar characteristics.

Q4. What are your estimates for the elapsed time prior to opening a suitable repository?

4. There is considerable question as to the possibility of storing MOX fuel in the first repository, Yucca Mountain. The Nuclear Waste Policy Act of 1982

specifically authorizes only defense high-level waste and commercial spent fuel for storage in the first repository. All of the space in Yucca Mountain is already reserved for the above two waste forms.

Several high ranking officials in OCRWM have voiced strong opposition to the storage of anything but defense high-level waste and commercial spent fuel in Yucca Mountain on the grounds that the public confidence in the geologic repository program would be severely shaken by the addition of a third waste type, not previously disclosed. A congressional act could resolve all of these questions, but it is not considered likely in the present political climate.

A much more likely scenario is that MOX fuel would have to wait for a second geologic repository. The 1987 amendment to the Nuclear Waste Policy Act specifically forbids DOE from requesting a second repository before the year 2007. In view of the lengthy program involved in licensing and constructing the Yucca Mountain repository, a widely shared opinion among those concerned with the disposal of spent fuel is that a second repository would not be available before 2030-2040.

Q5. What are the current cost estimates for waste disposal system development and for disposal of the waste?

5. The waste disposal costs for MOX fuels were summarized in WSRC-RP-92-1004, Table 6.1 (p. 6-2) with bases as discussed in the text. Additional development work that could be required includes:

   a. MOX fuel waste form characterization might take 3 years at a cost of ~$1 million per year.
   b. Adaptation of a MRS facility design to the dry storage of MOX fuel, 2 years at ~$2 million per year.
   c. Investigation of needs for spent fuel treatment and repackaging prior to entombment in geologic repository, 5 years at ~$0.5 million per year.

Q6. Does radiation make your proposed waste package "self protecting" (i.e., greater than 100 R/hr at 3 ft from the surface). If so, how long does it remain self protecting?

6. The MOX spent fuel form will be "self protecting" for essentially the same time period as the commercial spent fuel form. They will both have the same exposure, if the primary aim is to generate electricity. The "self protecting" period of MOX fuel has not been specifically calculated, but based on similar studies for much shorter irradiations of PuO_2-Al fuels in K-reactor at SRS, that period is expected to be >> 50 years. The radiation field surrounding the spent assemblies will be more than ample to deter diversion while the spent MOX fuel is most vulnerable-during the surface storage period.

Appendix F

Modular High Temperature Gas-Cooled Reactor Sponsor Response

GENERAL ATOMICS
RESPONSES TO EG&G Pu DISPOSITION OPTION QUESTIONS

Fuel - Question #1

Did you assume that plutonium (PuO_2 or Pu metal) used in the fuel would be free of contaminants (alloying metals and americium now in the nuclear weapon pits)? Will the fuel proposed be negatively impacted if plutonium is contaminated with these alloying metals and the americium?

Response to Fuel Question #1:

The fuel fabrication process for the MHTGR fuel can accept a variety of plutonium feedstock forms. If the feedstock contains alloying metals or americium, they will be removed during a purification step (solvent extraction or ion exchange) prior to kernel fabrication.

Fuel - Question #2

Briefly describe the technical work scope necessary to complete development of the fuel and its estimated duration.

Response to Fuel Question #2:

As summarized in Reference 1, the required development activities for plutonium fuel for the MHTGR will directly parallel those defined for the NP-MHTGR HEU fuel development program (Reference 2). Key elements of a Pu TRISO-coated fuel development program are:

- Design of a TRISO-coated Pu fuel particle capable of achieving high burnup under peak MHTGR core service conditions predicted for normal operation and postulated accidents. The key product of this activity is the fuel product specification.

- Demonstration of Pu fuel particle and compact fabrication capability, including process development, equipment design, and process scaleup. The key products are fuel process specifications and equipment specifications.

- Single-effects testing, both in- and out-of-pile, to facilitate the development of component models to predict the performance of TRISO-coated Pu fuel particles and fission product transport under the service conditions predicted for normal operation and postulated accidents. The key product of this activity is a Fuel Design Data Manual.

- Integral testing, typically in-pile, under representative MHTGR conditions to independently validate the design methods and codes (which incorporate the component models described in the previous bullet) used to predict Pu fuel performance and fission product transport during normal operation and postulated accidents. The key product of this activity is a suite of computer codes which have been verified and validated to NQA-1/NQA-2 standards under plutonium-fueled MHTGR specific conditions.

The total duration of the NP-MHTGR fuel development program was nine years (Reference 2) ; it has been estimated that the Pu fuel development program would take an additional 18 months (Reference 1).

Fuel - Question #3

Identify technical issues that could impede fuel development and fabrication. For example have all issues related to material lifetime, compatibility, etc., been resolved?

Response to Fuel Question #3:

As described in Reference 1, TRISO-coated Pu fuel particles have been successfully fabricated and irradiated in the past in the USA and Europe to burnups in excess of those proposed. The performance of these particles provides a sound technical basis for concluding that there are no technical feasibility issues; nevertheless, significant technology development and testing will be required to fully qualify Pu fuels for use in the MHTGR. Similar tests with highly enriched uranium particles also indicate the practicality of these high burnups.

The generic technical issues for high-burnup, TRISO-coated particle fuel are defined in the NP-MHTGR HEU fuel development program (Reference 2). These issues generally apply to Pu fuel as well, but several Pu-specific issues are also anticipated (e.g., remotizing the fuel fabrication process). A preliminary list of key technical issues to be addressed in the development and qualification program for Pu fuels follows:

- Design of a TRISO-coated Pu fuel particle capable of achieving high burnup under peak MHTGR core service conditions predicted for normal operation and postulated accidents. Key elements will be specifying the kernel composition to suppress kernel migration and CO formation and specifying the coating system design.

- Modifications to the fuel particle design and/or fuel fabrication processes to improve the level of fuel performance over that exhibited in recent NP and NE fuel irradiation capsules (NPR-1, NPR-2, NPR-1A, and HRB-21). On-going PIEs and fuel process optimization studies being conducted under the commercial MHTGR program and the NPR closeout program are expected to provide the technical basis for determining the required changes in fuel design and/or fuel process conditions. Detailed fuel process and fuel performance data from the highly successful German TRISO fuel development program may also prove helpful in this regard.

- Remotizing the fuel fabrication processes and equipment to the extent necessary to safely manufacture TRISO-coated Pu fuel meeting the fuel product specification.

- Scaleup and integration of unit operations for kernel fabrication, particle coating, and fuel rod compacting to accommodate required fuel manufacturing throughputs.

- Demonstration of acceptable irradiation performance of TRISO-coated Pu fuels to high burnup (>90% destruction of Pu-239) under peak MHTGR core service conditions predicted for normal plant operation.

- Demonstration of acceptable performance of high-burnup TRISO-coated Pu fuels during dry and wet post-irradiation heating tests that bound the service conditions predicted for postulated MHTGR accidents, including depressurized core conduction cooldown transients. High-burnup (>75% FIMA), HEU TRISO-coated particles have performed well in such post-irradiation heating tests to temperatures well excess of the peak fuel temperatures predicted for the MHTGR (the peak temperature for a bounding core conduction cooldown to ground is expected to be <1600 °C). Nevertheless, analogous tests with high-burnup Pu fuel will be needed.

- Integral testing, typically in-pile, under representative MHTGR conditions to validate the design methods and codes used to predict the source terms for a Pu-fueled MHTGR during normal operation and postulated accidents. In this regard, particular attention must be given to confirming that the Pu isotopes will be essentially retained in the core under all credible service conditions. The release and transport of silver isotopes, especially Ag-110m, will also receive special attention.

As part of preconceptual design, design data needs (DDNs) for the development and qualification of TRISO-coated Pu fuel will be systematically defined, and the scope, schedule and cost of the attendant testing programs will be estimated. During a proposed follow-on study, this initial planning would be refined and extended; the end result would be a comprehensive draft of a Fuel Development Program Plan for TRISO-coated Pu fuels.

Fuel - Question #4

What are the current cost estimates for fuel development and for the fuel fabrication facility construction, startup, and operation? What estimating method was used (e.g. parametric, historical cost, unit cost, etc.).

Response to Fuel Question #4:

The current cost estimates for fuel development and for the fuel fabrication facility construction, startup, and operation were provided in Section 3.5 of Reference 1. Plutonium fuel development costs ($261 million) were developed based on historical cost and experience with the NP-MHTGR program. This is an incremental cost beyond the fuel development cost (about $50 million) for the commercial MHTGR program presented in Reference 1. Fuel fabrication facility construction costs ($260 million for a 4-module MHTGR) were derived from cost estimates developed by Fluor-Daniel for the NP-MHTGR Fuel and Target Fabrication Facility, with adjustments for use of plutonium fuel instead of highly enriched uranium and for deletion of target fabrication process equipment costs. These costs were developed based on labor, material, and commodity unit costs. Startup costs have not been calculated, but are expected to be small compared to construction costs. Operations costs are included in the plutonium fuel cost and represent about ⅓ of the fuel cost (or, $750 million over 40 years). Operations costs are estimated based on historical costs and parametric variations related to throughput and are estimated to be about $60,000 per kilogram of plutonium.

Reactor System - Question #1

If the plutonium disposition goal is to annihilate Pu-239 and Pu-241 in a single fuel cycle, what is the total burnup or exposure (GWD/MT Pu) required to reduce the initial inventory of these two isotopes by 90%, 95% and 99% (if possible)? For each of these cases, identify the weight percent of all plutonium isotopes in the initial fuel loading and those remaining in the spent fuel after an equilibrium fuel cycle. Also identify the cycle times.

Response to Reactor System Question #1 :

The current MHTGR reference design for achieving high burnup uses a 450 MW(t) core in which one-half (420) of the total core complement of 840 fuel elements is replaced each year to achieve a two year fuel exposure at discharge. At discharge this design achieves 90% burnup of the initial Pu-239, 80% of Pu-239 plus Pu-241, and 63% of the initial total plutonium charged, with an average burnup of 560,000 MWD/MT Pu. We believe that these discharge isotopics, with attendant high radiation dose from any recovered plutonium - and which would require processing many blocks to achieve a critical mass - make this fuel unattractive for weapons purposes. This design also meets all safety-related criteria for shutdown margin and power stability, has a large negative temperature coefficient of reactivity at all times in cycle, and displays safety characteristics comparable to those of the commercial MHTGR that has been reviewed by NRC.

Figure 1 shows plutonium burnup as a function of MWD/MT burnup out to an extended exposure of 813,000 MWD/MT burnup, where greater than 99% Pu-239 + Pu-241 burnup has been reached. This plot beyond ~550,000 MWD/MT was generated by merely extending the nominal two years exposure loadings to an exposure of ~2.5 years without regard to maintaining criticality.

A very promising potential option for achieving these high burnups and high Pu-239 and Pu-241 destruction in the actual reactor without recycle has been identified, and preliminary radial one-dimensional burnup calculations for this option have been completed. In this design the fuel is irradiated for a total of 3 years rather than the current 2 years. After 2 years of "in-core" exposure the discharged elements are moved into replaceable reflector locations next to the active core for an additional one year irradiation in the "high" thermal flux characteristic of the reflector. Data points for this case are also given on Figure 1 and show that without any reprocessing Pu-239 burnups of 97%, Pu-239 plus Pu-241 burnups of 90%, and total Pu burnup of 73% respectively are obtained. Results are also summarized in Table 1. To recover, for example, about 25 Kg of plutonium of this discharge mixture

would require diversion and reprocessing of more than 200 fuel elements.

Higher burnups, up to 99% Pu-239 + Pu-241, are considered feasible with this design; however, analytical results have not yet been obtained, and additional fuel development would be required.

Reactor System - Question #2

If the plutonium disposition goal is to annihilate Pu-239 and Pu-241 and irradiated fuel can be reprocessed to recover and recycle plutonium, what is the total burnup or exposure (GWD/MT Pu) required to reduce the initial Pu-239 and Pu-241 inventory by 90%, 95%, 99% and 99.9%? If the option uses a batch mode fuel cycle, how many times would a core fuel load need to be recycled to reach each burnup percentage? For each of the four cases, identify the weight percent of all plutonium isotopes prior to initiation of irradiation and those remaining in the spent fuel.

Reactor System - Question #3

If the plutonium disposition goal is to annihilate all plutonium isotopes and irradiated fuel can be reprocessed to recover and recycle plutonium, what is the total burnup or exposure (GWD/MT Pu) required to reduce the inventory of all plutonium isotopes by 90%, 95%, 99% and 99.9%? If the option uses a batch mode fuel cycle, how many times would a core fuel load need to be recycled to reach each burnup percentage? For each of the four cases, identify the weight percent of all plutonium isotopes remaining in the spent fuel.

Responses to Reactor System Questions 2 and 3:

The answers to questions 2 and 3 have been combined. As discussed in the response to Reactor System Question #1, the MHTGR can achieve high Pu-241 burnup in a three year fuel exposure in a fuel cycle while not requiring any fuel reprocessing. However, we have made estimates of the isotopic content as a function of fuel particle burnup for the assumption of reprocessing, using a simplified model, to provide preliminary results related to Questions 2 and 3. To date, using this model, the isotopic data for the reprocessing assumptions in Questions 2 and 3 are only available for 3 passes through the core, i.e., a 6 year fuel exposure.

At equilibrium conditions, this "3 pass" recycle mode of operation would result in ~80% of the 420 fuel blocks loaded per year being fresh fuel blocks, ~15% being first recycle (or second pass) fuel blocks, and 5% being second recycle (or third pass) fuel blocks.

Figures 2, 3 and 4 illustrate the Pu-239 burnup, the Pu-239 + Pu-241 burnup, and the total Pu burnup - expressed as % remaining - as a function of the MWD/MT Pu burnup in the fissile particle during each pass through the core. For the third pass the burnup is extended to ~460,000 MWD/MT by using a selective fuel placement strategy where the blocks are placed in high thermal flux locations.

After 3 passes through the core, with selective fuel placement in the last pass, the total burnup of the discharged recycle plutonium relative to the initial loading is 814,000 MWD/MT Pu. At this point 99.9% of the Pu-239, 97.5% of the Pu-239 + Pu-241 (relative to initial Pu-239), and 90.7% of total plutonium has been consumed after 6 years exposure. These burnup rates at 814,000 MWD/MT are in very close agreement to the values shown in Figure 1.

As noted earlier, the no reprocessing cycle with added exposure in the reflectors (innovative once-through design) achieves 97% Pu-239 burnup. The added cost, environmental impacts, and complexity of reprocessing MHTGR fuel would have to be considered in light of the modest increases in fissile burnup that result from reprocessing and recycling in the MHTGR. Recycle of the fuel elements that have been exposed for an additional cycle in the reflectors should also be possible, but plutonium destruction rates for a second pass of these elements have not been calculated.

Table 1 summarizes the burnup rates and the discharge isotopics for each of the cases previously discussed.

Added reprocessing cycles would be required to achieve $\geq$ 99% Pu239+Pu241 burnup. From the curves it is estimated that this would be reached after 5 or 6 cycles. Because of the relatively low thermal cross section of Pu-242, it would be difficult to achieve 99% burnup of all plutonium isotopes without a very large number of recycle steps.

Reactor System - Question #4

Briefly describe the technical workscope necessary to complete development of your reactor or accelerator system and its estimated duration.

Response to Reactor System Question #4:

The technical work scope necessary to complete development of the MHTGR is summarized in the NP-MHTGR Engineering Development Plan (Reference 3). The plan addresses five major technical areas that pertain to a plutonium-fueled MHTGR.

- Fuel development
- Thermal hydraulics development
- Reactor physics and shielding development
- Structural materials development
- Component test development

With funding such as was planned to be provided on the NPR program, the duration of the development activities would have been about nine years. It is estimated that use of plutonium fuel instead of highly enriched uranium would add 18 months to the development program.

Reactor System - Question #5

Identify technical issues that could impede system development, design, construction, and startup. For example have all issues related to material lifetime, compatibility, etc., been resolved?

Response to Reactor System Question #5:

The critical path technical issue that could impede design, development, and startup of a plutonium-fueled MHTGR is plutonium coated fuel particle development and qualification. Fabrication processes must be adapted to use of plutonium, and the fuel product and its fabrication processes must be qualified by irradiation testing.

Reactor System - Question #6

What are the current cost estimates for system development and for construction, startup, and operation of the facility? What estimating method was used (e.g. parametric, historical cost, unit cost, etc.).

Response to Reactor System Question #6:

The requested information is provided in Section 3.5 of Reference 1. These costs were estimated by combinations of parametric, unit, and historical costs.

Waste Processing - Question #1

Briefly describe the technical workscope necessary to complete development of a waste conditioning/processing flowsheet for your option and its estimated duration.

Response to Waste Processing Question #1:

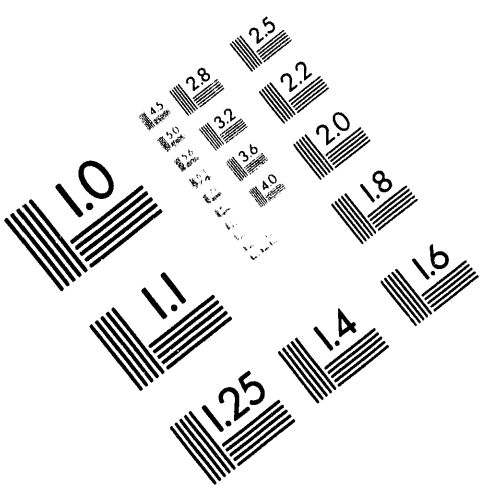
Based upon a recent ORNL conceptual evaluation of potential disposal options for spent HTGR fuel elements (Reference 4), the preferred option appears to be disposal of the spent Pu fuel as whole blocks. ORNL further concluded that whole HTGR fuel elements could be placed, without any significant preconditioning or processing, in spent fuel waste containers (essentially the same as planned for use with LWR discharged fuel). While this conclusion needs to be confirmed by detailed engineering analysis, no significant technology development is anticipated in this area for the whole-block disposal option.

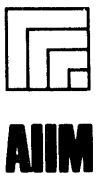
The fuel, highly depleted in plutonium, will remain in the fuel element just as they were irradiated in the reactor. The packaging will involve handling needed to remove the fuel elements from their shielded storage and to load them into their storage containers. Minimal new waste forms will be generated by this transfer operation.

The development steps would be part of the development of the overall fuel handling system. It would involve the design of the spent fuel block transfer and packaging system and facility, and integrating it with the interfacing systems. This overall effort is estimated to require three years. The system components will be built and integrated into the overall transfer and packaging system, and integrated system tests would then be conducted. This final demonstration testing is estimated to require an additional three years.

Waste Processing - Question #2

Identify technical issues that could impede development of the waste processing flowsheet, systems, or facilities.





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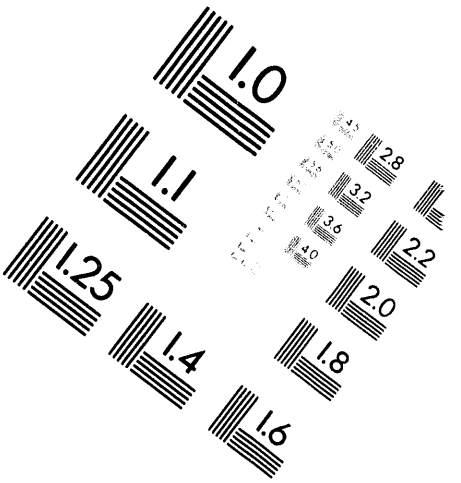
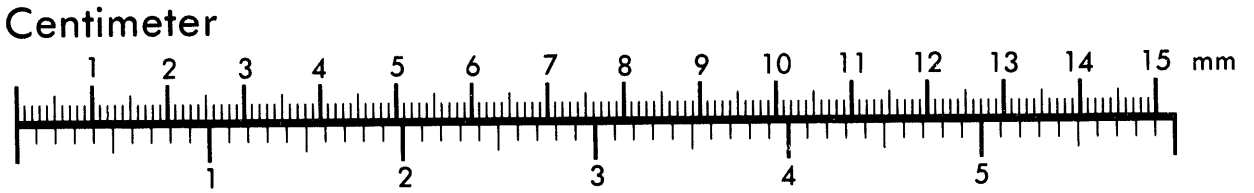
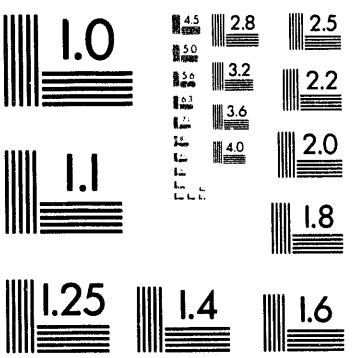
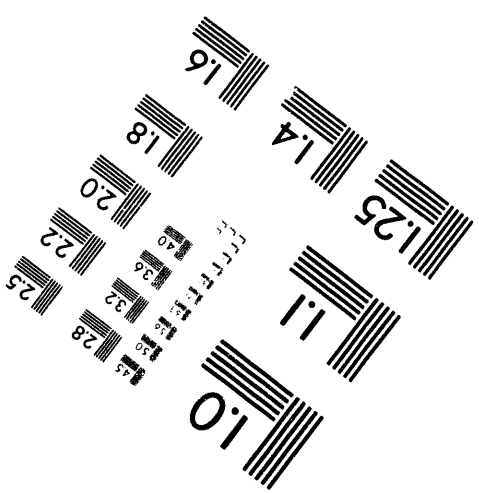
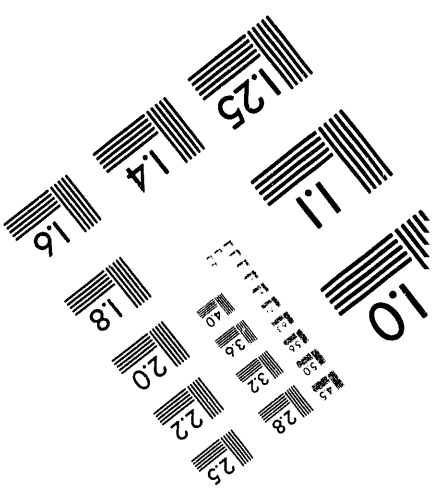










3 of 3

Response to Waste Processing Question #2:

No technical issues have been identified for the whole-block disposal option that would require extensive technology development. It is anticipated that the packaging of the spent fuel blocks for final disposal would be accomplished at the reactor site using currently available technology.

The equipment and facility will utilize components less sophisticated than the MHTGR in-reactor fuel handling equipment and the fuel handling equipment used at Fort St. Vrain. Development is not expected to present any insurmountable impediments although technology for neutron shielding, dust containment and system leaktightness will be added to the demonstrated technology as required.

Waste Processing - Question #3

What are the current estimates for the time required for construction and startup of the waste processing facility?

Response to Waste Processing Question #3:

The facilities for packaging the spent fuel blocks for final disposal are currently expected to be designed and constructed as an integral part of the spent fuel handling and storage facilities at the reactor. Therefore, the schedule for design, construction, and startup is essentially the same as the schedule for the reactor plant, which is presented in Section 3.4 of Reference 1.

Waste Processing - Question #4

What are the current cost estimates for waste processing development and for construction and startup of waste processing facility?

Response to Waste Processing Question #4:

As described above, minimal technology development is anticipated for the whole-block disposal option, and no separate waste processing facility is planned. The costs to design, construct, and startup the packaging facilities at the reactor site have not yet been defined. Nevertheless, they are not expected to be a significant component of the total plant capital costs.

Waste Disposal - Question #1

Relative to the assumed acceptance of commercial fuel and defense waste in a geologic repository, is there waste characterization work that must be performed?

Response to Waste Disposal Question #1:

Based upon a recent ORNL conceptual evaluation of potential disposal options for spent HTGR fuel elements (Reference 4), the preferred option appears to be disposal of the fuel as whole blocks, in which case the highly depleted Pu fuel will be permanently encased by large quantities of highly corrosion-resistant nuclear graphite. For this option, the required waste characterization - beyond the characterization of the fuel and graphite needed to qualify it for reactor application - would appear to be minimal, but this subject has not been exhaustively investigated. Several technical issues are immediately evident:

- The C-14 content of spent HTGR fuel elements would need to be confirmed since it has been identified as a key radionuclide in assessing the acceptability of whole-block disposal. Preliminary analysis by ORNL indicates that while HTGR spent fuel blocks contain more C-14 than LWR spent fuel, the release rate of C-14 from HTGR fuel will be lower (Reference 4, page 44).

- The leach rates of radionuclides from spent HTGR fuel elements with varying levels of fuel particle failure would need to be estimated. Such leaching studies with spent FRG fuel spheres from the AVR prototype HTR at Juelich, FRG, have been in progress for more than a decade, but differences in the fuel form and fuel materials need to be considered.

- The oxidation rates of the nuclear grade graphites used in HTGR cores (e.g., H-451) are very low at credible repository temperatures (Reference 4, Appendix C). However, these predictions are extrapolations of measurements made on unirradiated graphites. Certain fission metals, including Sr, Ba, and Ce, are known to catalyze graphite oxidation at higher temperatures. Therefore, the concentrations of fission metals in spent graphite blocks should be determined, and the oxidation rates of irradiated graphites with representative fission product loadings should be measured.

Waste Disposal - Question #2

Briefly describe the technical workscope necessary to complete development of the waste disposal method and its duration. For example, are there any preconditioning or packaging requirements that must be satisfied for repository acceptance?

Response to Waste Disposal Question #2:

As stated above, the preferred option appears to be disposal of the fuel as whole blocks; in fact, ORNL concluded that "...the HTGR fuel assembly may be a superior waste form with repository performance characteristics significantly better than conventional waste forms...[including spent LWR fuel].... (Reference 4)." They further concluded that whole HTGR fuel elements could be placed, without any significant preconditioning or processing, in spent fuel waste containers (essentially the same as planned for use with LWR discharged fuel) and placed into a permanent repository. While such conclusions are highly encouraging, it is recognized that the ORNL study was conceptual in nature, and that their conclusions need to be confirmed by detailed engineering analysis and, possibly, validated by testing programs.

Waste Disposal - Question #3

Identify issues that could impede the placement of waste from your option in a repository.

Response to Waste Disposal Question #3:

No technical feasibility issues were identified in the ORNL assessment (Reference 4) that are expected to impede the placement of whole HTGR fuel elements in a permanent repository; in fact, they concluded that whole HTGR fuel elements were a more suitable form than spent LWR fuel. Consequently, the issues of potential concern are the same political issues (e.g., states vs. federal rights) and sociological issues (e.g., NIMBY) that are currently impeding the disposal of commercial and defense nuclear waste in the US.

The one disadvantage of whole block disposal identified in the ORNL study was high volume of repository space per unit of heavy metal in the spent fuel because of the large volume of graphite and low power densities inherent to HTGR fuel. However, larger (taller) waste containers and alternate emplacement strategies were suggested as ways to mitigate the cost impact associated with the larger waste volume. (Alternate emplacements strategies include decreased distances between boreholes for HTGR fuel elements in recognition of the much lower volumetric heat generation rates associated with the much lower power densities

compared to LWR and LMR fuel assemblies.)

Waste Disposal - Question #4

What are your estimates for the elapsed time prior to opening a suitable repository?

Response to Waste Disposal Question #4:

Whole-block disposal of spent MHTGR plutonium fuel will be suitable for a LWR spent fuel repository. Accordingly, there is no incremental elapsed time beyond that for opening a suitable LWR fuel repository for commercial or defense high level waste.

Waste Disposal - Question #5

What are the current cost estimates for waste disposal system development and for disposal of the waste?

Response to Waste Disposal Question #5:

A rigorous cost estimate has not been made to date. However, engineering judgment suggests that the costs for whole element disposal of MHTGR fuel would ultimately be comparable to those for commercial LWR spent fuel disposal (Reference 1). That judgment is strongly influenced by the ORNL study (Reference 4), which concludes that repository design, size and cost are controlled primarily by heat generation rate. The very low volumetric heat generation rate of MHTGR spent fuel and the capability to optimize the repository design for that lower rate are projected to result in an overall repository size - hence, cost - which is comparable to that for other systems.

Waste Disposal - Question #6

Does radiation make your proposed waste package "self protecting" (i.e., greater than 100R/hr at 3 feet from the surface). If so how long does it remain self protecting?

Response to Waste Disposal Question #6:

The discharged MHTGR graphite fuel element is suitable for disposal "as is" in the standard LWR high-level waste package container (Reference 1). The MHTGR package would consist of seven such fuel elements.

Figure 5 shows the dose rate at 3 feet from this waste package for the high burnup options presented in Reference 1. As can be seen from the figure, the waste container remains self protecting for about 60 years.

References

1. "Excess Plutonium Disposition, Modular High Temperature Gas-Cooled Reactor (MHTGR)," Draft Report Prepared for the Fission Working Group of the DOE Plutonium Disposition Task Force, Rev. 4A, General Atomics, December 4, 1992.

2. "NP-MHTGR Fuel Development Program Plan," EGG-NPR-8971, Rev. C, EG&G Idaho, September 1992.

3. "NP-MHTGR Engineering Development Plan," Rev. 0, U.S. Department of Energy, October 17, 1991.

4. Lotts, A. L., et. al., "Options for Treating High-Temperature Gas-Cooled Reactor Fuel for Repository Disposal," ORNL/TM-12027, Oak Ridge National Laboratory, February 1992.

Table 1

SUMMARY OF BURNUP AND DISCHARGE ISOTOPICS*

	CURRENT REFERENCE 2 YR EXPOSURE NO RECYCLE	INNOVATIVE DESIGN 2 YR IN-CORE 1 YR IN-REFLECTOR NO RECYCLE	REPROCESSING AND RECYCLE (3 PASSES)
Burnup MWD/MT	560,000	677,000	813,000
Fuel Exposure, yrs.	2.0	3.0	6.0
# Passes Through Reactor	1.0	2.0	3.0
% Pu-239 Remaining	10.0	3.0	0.1
% 239 + 241 Remaining	20.5	10.0	2.5
% Pu Total Remaining	37.0	27.0	9.3
Isotopic Content at Discharge			
% Pu-239	25.5	12.5	1.0
% Pu-240	39.1	45.6	21.5
% Pu-241	27.0	23.8	24.6
% Pu-242	8.4	18.1	52.9

*In all cases WGPu feed is 94% Pu-239, 6% Pu-240

Figure 1

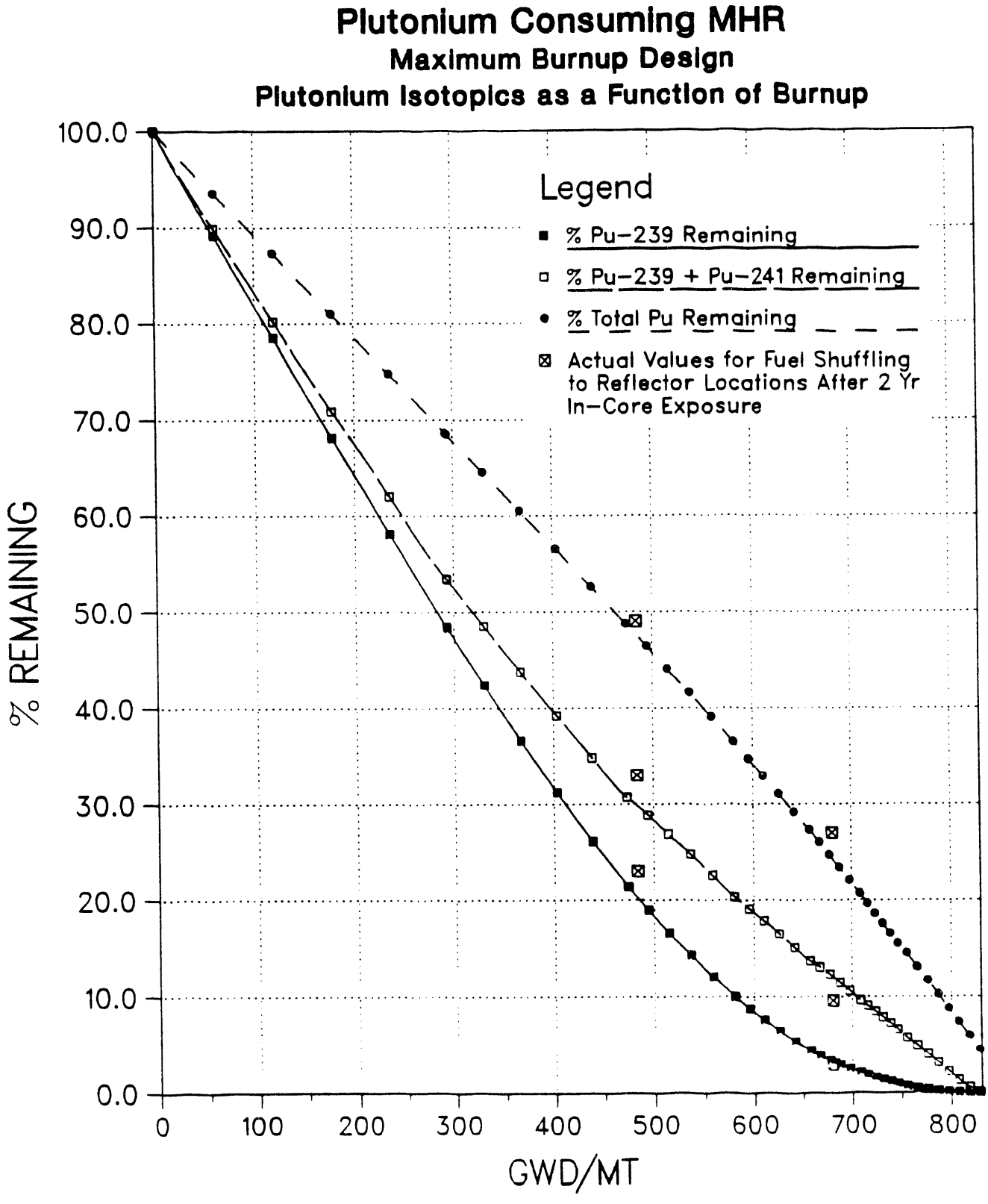

Plutonium Consuming MHR
Maximum Burnup Design
Plutonium Isotopics as a Function of Burnup
Legend
% Pu-239 Remaining
% Pu-239 + Pu-241 Remaining
% Total Pu Remaining
Actual Values for Fuel Shuffling to Reflector Locations After 2 Yr In-Core Exposure
% REMAINING
100.0
90.0
80.0
70.0
60.0
50.0
40.0
30.0
20.0
10.0
0.0
0
100
200
300
400
500
600
700
800
GWD/MT

Figure 2

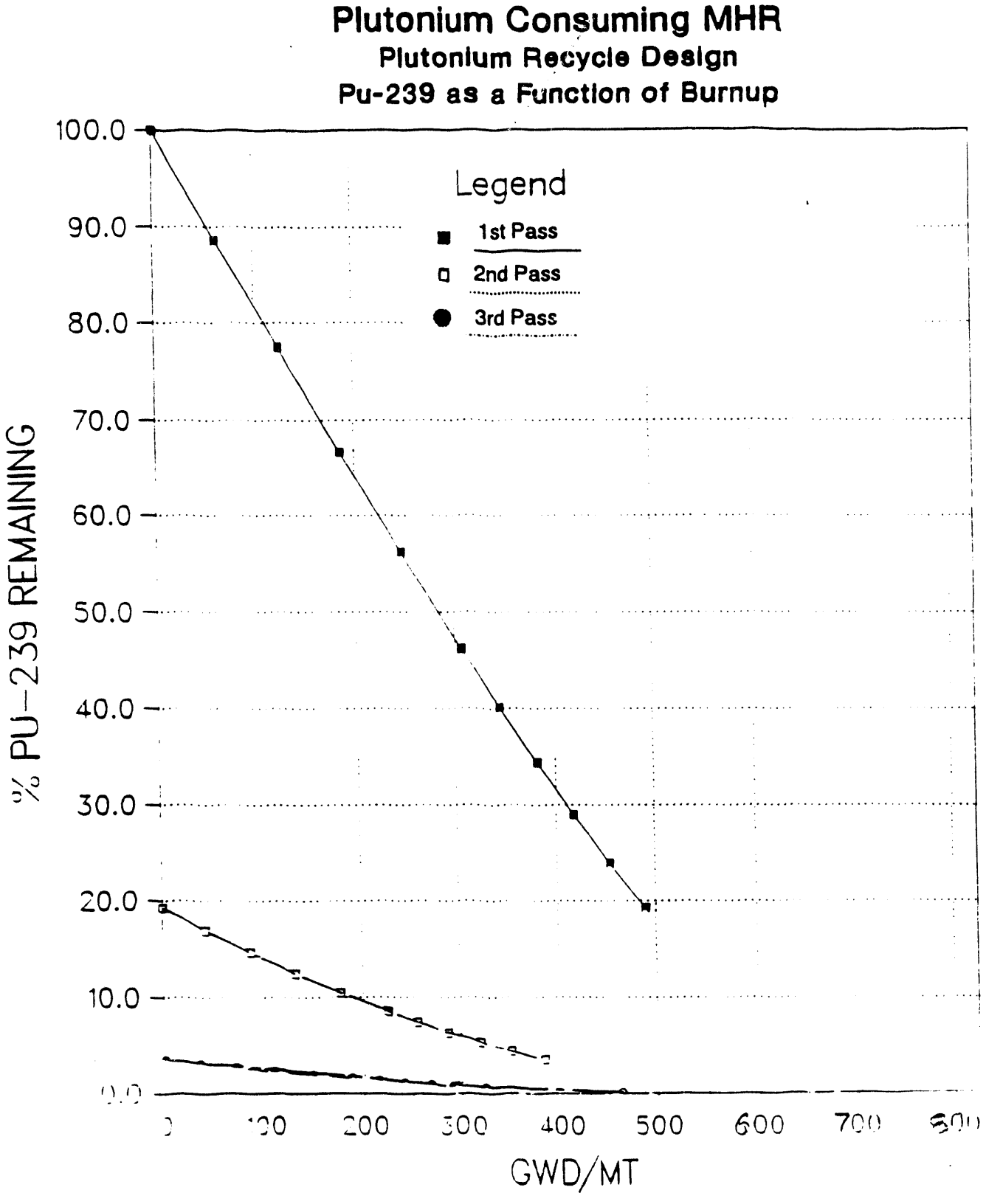

Plutonium Consuming MHR
Plutonium Recycle Design
Pu-239 as a Function of Burnup
Legend
1st Pass
2nd Pass
3rd Pass
% PU-239 REMAINING
100.0
90.0
80.0
70.0
60.0
50.0
40.0
30.0
20.0
10.0
0.0
0
100
200
300
400
500
600
700
800
GWD/MT

Figure 3

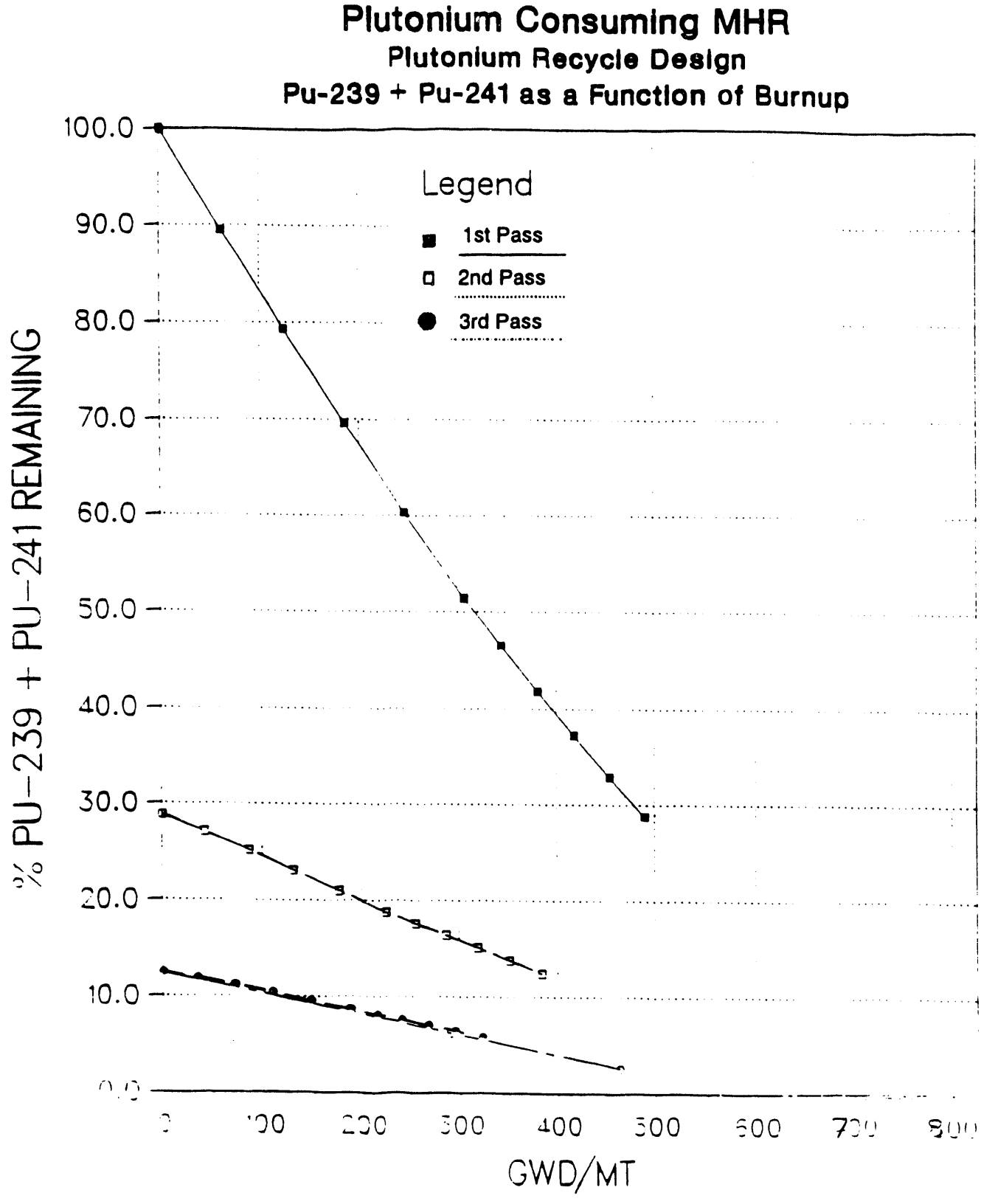

Plutonium Consuming MHR
Plutonium Recycle Design
Pu-239 + Pu-241 as a Function of Burnup
Legend
1st Pass
2nd Pass
3rd Pass
% PU-239 + PU-241 REMAINING
100.0
90.0
80.0
70.0
60.0
50.0
40.0
30.0
20.0
10.0
0.0
0
100
200
300
400
500
600
700
800
GWD/MT

Figure 4

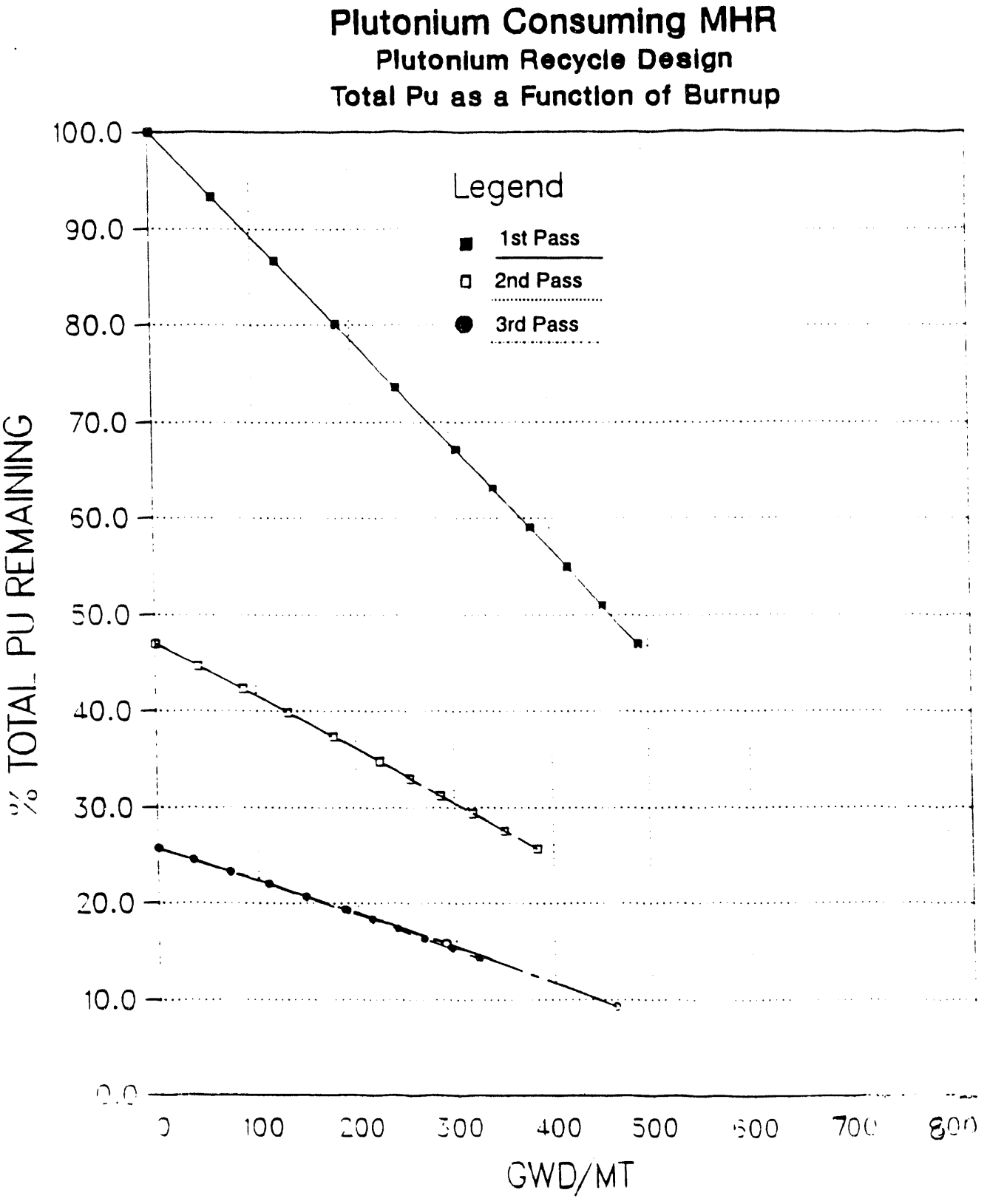
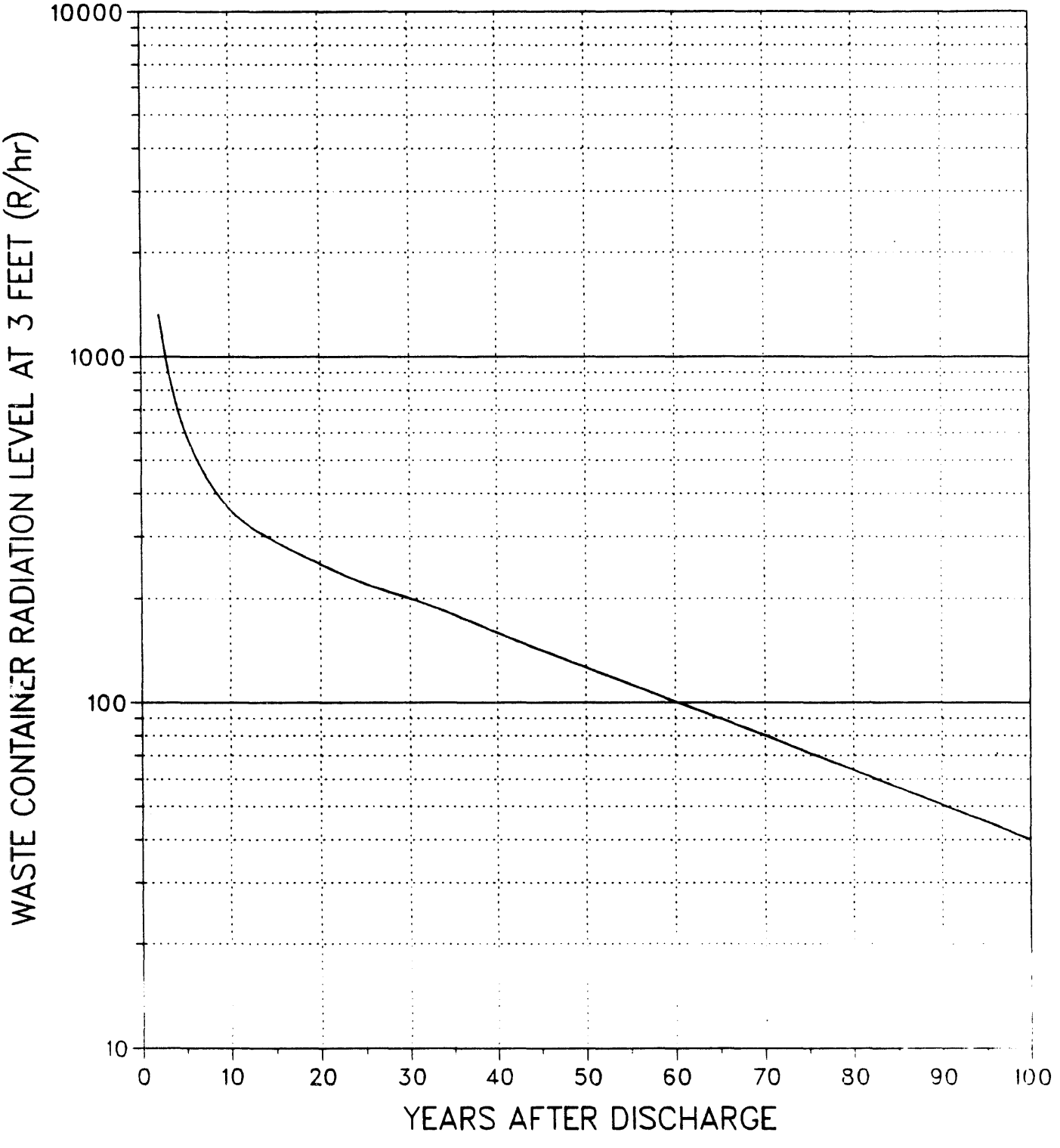

Plutonium Consuming MHR
Plutonium Recycle Design
Total Pu as a Function of Burnup
Legend
1st Pass
2nd Pass
3rd Pass
% TOTAL PU REMAINING
100.0
90.0
80.0
70.0
60.0
50.0
40.0
30.0
20.0
10.0
0.0
0
100
200
300
400
500
600
700
800
GWD/MT

FIGURE 5

RADIATION LEVEL IN MHTGR WASTE DISPOSAL PACKAGE



Appendix G

Molten Salt Reactor Sponsor Response

OAK RIDGE NATIONAL LABORATORY
OPERATED BY MARTIN MARIETTA ENERGY SYSTEMS, INC.
FOR THE U.S. DEPARTMENT OF ENERGY

POST OFFICE BOX 2008
OAK RIDGE, TENNESSEE 37831-6399

March 1, 1993

Mr. Duane J. Hanson
EG&G Idaho Inc.
P. O. Box 1625
MS 2508
Idaho Falls, Idaho 83415

Dear Mr. Hanson:

Plutonium Disposition Option

Following is the response to your questions on the subject matter. A general comment is that fluid fuel reactors are fundamentally different, as you recognized, and the questions are oriented toward conventional reactors. Also, there is no active program on molten salt reactors and, hence, we can not make any additional calculations, no matter how simple they are.

Fuel

1. The molten salt concept utilizes fluorides (there is a chlorides option). The plutonium would be used as PuF_3. While a detailed calculation is needed to determine the exact effect of any additives, it is not expected that alloying metals will have a significant negative effect. Some of the alloying metals may simply stay out because they will not dissolve as a fluoride. The molten salt reactor is also being proposed as an actinide transmuter. It is expected that any actinides would become part of the salt and either fission or transmute or be taken out in the processing waste stream. No particular difficulties are anticipated from plutonium alloying when considered in the reactor design and operation.

2. The fuel development for the molten salt reactor option constitutes the dissolving of the metal with fluoride into a molten salt. There are various chemical possibilities to accomplish that, all of them do not require much development. The usual safety precautions regarding quantities, criticality, spills, etc., associated with hazardous and fissionable material must be complied with. The duration for this head-end development is not significant.

   In the processing facility, accommodation for the additional components for the plutonium are necessary. The MSR processing development reached the stage of individual steps in laboratory scale. There is need to integrate the steps and up the scale. This is not unlike other fuel processing schemes, and the inclusion of plutonium is not expected to have a big impact.

3. Fluid fuel reactors have no fuel fabrication. The feasibility of the fuel processing of MSRs is considered resolved. Associated material problems are considered resolved in principle. The fuel processing for MSRs is, relatively to common fuel processing facility, small scale. Much of the lifetime questions are basing on components and pipes exchanges as needed,

providing for an essentially unlimited life of the facility. The remote technology needed is limited to such exchanges, which can be considered relatively simple in today's state of the art. The process itself is handling a fluid in contained components and pipes. Development work is necessary to establish the full scale operating processing part.

4. There is no current cost estimate available for fuel processing. Any estimate would have to base on the fact that completed molten salt reactor studies included the fuel processing part as an integral part of the reactor and yet were competitive with other reactors. In the economical evaluations, it must be considered that no large scale fuel processing facility is necessary, and the risk associated with the integral facility is comparatively small.

Reactor or Accelerator System

1. As you noted correctly, in the MSR with integral processing, the plutonium stays in the reactor until completely consumed. Fuel cycle times and equilibrium fuel cycle have no meaning in a reactor with continuous processing.

2. In an MSR with continuous processing, there is no meaning to burnup or exposure. Also, there is no spent fuel. At present, it is not known if there will be a residue of plutonium in the waste. In any case, such a residue would be minuscule and non-significant, because it would be below the level of recoverable concentration since that would be the reason for leaving it in there. The plutonium in the fuel will remain there until completely consumed.

3. This question is not applicable to MSRs, there is no burnup in an MSR. All plutonium isotopes are treated "equally" they stay in the reactor until destroyed by fission or transmutation. No calculation has been made to estimate the fate and expected residence time of the different plutonium isotopes in the various possible MSRs.

4. Work scope and duration to bring an MSR on line depends on the immediate and long term goal for the project. The duration is funding dependent.

   A minimum program, based on completed development and with restricted processing can be accomplished in five years. This will be an operating molten salt reactor that can be fed small quantities of plutonium with processing that is limited to retention of the fuel in the reactor and, at least initially, only partial removal of waste. The project will provide proof of principle, demonstrate operability, consume some plutonium, provide experience, acceptance and input for the next steps, and produce power.

   A full development and optimization requires:

   - Update of the state-of-the-art and adjustment to today's requirements of laws, regulations, documentation, licensing, etc.

   - Analysis and design of desired options, optimization of parameters.

- For the thermal reactor option — development of the appropriate graphite or equivalent.
- For the epithermal reactor option — development and conceptual design.
- Upscaling, integration, and demonstration of the processing.
- Updating of the processing to today's state-of-the-art.
- Adjusting and optimizing the processing to today's needs.
- Including the plutonium in the processing scheme.
- Testing and demonstrating the processing.
- Optimization and handling of the waste stream.
- Complete safety analysis and NEPA.
- Demonstration and deployment.

The estimated duration of that development has such wide margins that any number or even range is meaningless. A lower end of ten years for concentrating on the more important aspects provided available adequate priority and funding. A higher estimate can be thirty years for more details and restricted parallel efforts, including perhaps a prototype.

5. At this time, there are no known feasibility questions remaining for the molten salt reactor. For the thermal option, using graphite as a moderator, there is limited lifetime for the graphite. The proposed solutions are either a limited power density, or a design that allows for the exchange of the graphite. Several material solutions were accomplished only at the last states of the molten salt reactor program and have not yet been demonstrated.

   The fuel processing part has been tried out as individual laboratory scale components only. A system integration and demonstration is needed. Much of the reactor design assures ability to exchange components. At the time of development of the MSR the remote and robotic technology were limited. The new available remote and robotic technology needs to be adapted for the MSR and demonstrated.

   No critical missing links are known. The information on plutonium processing is very limited.

   Licensing requirements are totally geared toward solid fuel reactors and many regulations account for LWRs only.

6. There are no current cost estimates for MSRs. Where attempts were made to provide more recent estimates, they were based on previous cost estimates. Cost estimates for MSRs have always been very favorable for the MSRs due to their relative simplicity and safety. These

factors may weigh even heavier today. In economical comparative studies of power supplies, MSRs were among the most economical. A recent publication suggested that the cost is within 5% of LWRs. In considering MSR economics, it must be remembered that the MSR with processing also closes the fuel cycle, essentially eliminates the spent fuel issue and makes an important contribution toward simplifying the waste issue. There is also the potential for actinide transmutation and breeding with the same basic design. These potentials can provide significant economical advantages and may contribute to the acceptance of the system and nuclear power in general. Essentially there is very little added cost for the plutonium burning option.

Waste Processing

1. It is expected that no additional waste disposal or conditioning is required for an MSR that is adapted for burning plutonium. There is no spent fuel. All waste conditioning and processing is included in the design and development of the processing. The head-end, dissolving the plutonium, is not expected to have waste streams. There is no fuel manufacturing.

2. There are no known technical issues that can impede the waste processing. The MSR processing was developed when waste disposal was not considered a significant issue. However, since the waste is already in a liquid form in a processing plant, it is very amenable to any changes that may be required.

3. The MSR, as currently planned, does not have a separate waste processing plant. The waste treatment is part of the integral fuel processing. No extra time or cost are required for a waste processing facility.

4. Development work is needed to optimize the waste processing and handling part of the fuel processing to current requirements and desires. This task is somewhat simplified by the absence of fuel in the waste. Development and demonstration are needed prior to integrating these steps in the plant.

Waste Disposal

1. The waste from an MSR is fuel free. The waste is in a chemical processing plant, therefore, it is expected that it can be readily optimized to the needs and requirements and possibly also the desires of any repository. Side stream wastes may contain chemicals that will require characterization. Again since the streams come from the chemical processing plant it is expected that they can be modified, classified and separated according to requirements.

2. The waste treatment development is part of the entire fuel processing development. No separate facilities are anticipated. One of the molten salt concepts that addresses safety and acceptance, envisions that the waste will be optimized in every respect such as: chemical composition, concentration, size, shape, the matrix and container, size of shipments or any other aspect. This is possible because the waste streams are encountered in a fluid form in

a chemical processing facility. The continuous treatment also limits them to small quantities at a time.

3. There are no known issues that are unique to the MSR that could impede waste disposal in a repository. The waste is fuel free and can be processed to the desired requirements of the repository. There may be some side streams that contain elements that require special treatment, such as fluorine and beryllium with radioactive materials included.

4. The MSR is not dependent on the opening of a repository for waste. The waste from a MSR is fuel free and relatively small in quantity because it is planned to recycle many of the usable components.

5. There are no cost estimates for the MSR's waste disposal system development nor for the disposal of the waste. The waste disposal system is part of the fuel processing. There are some new and additional steps that will be required to comply with current needs. These additional steps are not considered critical as they address a fuel free stream. This assumes that the recycle of plutonium has been accomplished under the fuel processing development. (In the defense fuel treatment such separation steps with fluorides are used.)

6. The MSR waste is the ultimate in "self protecting" it does not contain fuel in quantities that can be processed out of it. Furthermore, the waste can be concentrated to the extent desirable and is expected to be very active, containing mostly fission products.

If you need additional information or clarification, please contact me at (615) 574-0560.

Very truly yours,

Uri Gat

Uri Gat

UG:mw

cc: F. J. Homan
J. E. Jones Jr.
T. S. Kress

Appendix H

Particle Bed Reactor Sponsor Response

BROOKHAVEN NATIONAL LABORATORY
ASSOCIATED UNIVERSITIES, INC
Upton, Long Island, New York 11973
(516) 282-
FTS 666-2624

Department of Nuclear Energy

March 4, 1993

Dr. Duane J. Hanson
Idaho National Engineering Laboratory
Idaho Falls, ID

Dear Dr. Hanson:

The answers to your questions from the fax of February 22, 1993, are shown on the attached sheets.

Sincerely,

Hans Ludewig

HL:STC

Attachments

Fuel

1. The ground rules agreed to at the beginning of the study stipulated that the plutonium was pure ^{239}Pu. However, the process proposed for preparing particulate fuel suitable for a Particle Bed Reactor (PBR) is relatively insensitive to contaminant type and level. A detailed breakdown of the contaminants and their respective amounts would be required to give a definitive answer to these questions.

2. The process, being considered for the manufacturer of particulate fuel has only been demonstrated for uranium. No plutonium based fuel has been manufactured. However, it is felt based on the chemistry of actinides that a process similar to that demonstrated for uranium should have a high probability of success. Thus, the first step would be to repeat the steps carried out for uranium using plutonium feed stock. Following the successful demonstration of particle manufacturer, an irradiation program will be undertaken to demonstrate the ability to contain fission products at operating temperatures. Finally, the disposal of burned (95%-99% of Pu fissioned) particles will be demonstrated. In summary it is seen that a three step program is required to complete the development of the fuel particles.

At this stage an estimate of the duration of this program is not easy, but with sufficient funding the above three processes would be completed in approximately 5 years.

3 Having not manufactured any plutonium based fuel, it is essentially impossible to identify any outstanding issue which could impede fuel development. At this point in time none are envisioned, and the primary issue would be construction of a facility suitable to carry out the fuel development using plutonium feed stock. Furthermore, the irradiation tests would have to be carried out in an appropriate reactor, which also must be suitable to handle plutonium based fuel samples. Finally, the disposal experiment would have to be carried out as highly burned plutonium fuel particles, presumably also requiring a special facility. In summary cost and environmental issues are currently seen as the primary impediments to fuel development.

4. At this point in time no reasonable answers can be given to this question. All the environmental issues associated with a plutonium fuel manufacturing facility have essentially no historical references.

System

1. The initial plutonium loading in the bed is .2 gm/cc, and the reactor is assumed to operate at 5 M/ℓ. Under these conditions (see figure) it takes 20 days to reduce the initial inventory by 95%. From this curve it is estimated that 90% reduction is achieved after approximately 17 days. The burnup implied by these calculations are given on the following page.

Percentage Reduction	Burnup (MWD/MT)
90	425,000
95	500,000
99.75	1,000,000

It should be noted that in a particle bed reactor based system, the fuel particles will not be reprocessed but merely reloaded into different locations within the core.

Finally, it should be pointed out that the atom percent burnup in this fuel form is lower than 1% since the plutonium loading in each kernel is extremely low.

2. If chemical reprocessing is allowed and fuel particles are manufactured follow each reprocessing step, the results will be similar to those given above. In order to reduce the inventory by 99.75%, two 20-day irradiation periods would be required. Following three 20-day periods the original inventory should be reduced by 99.98%.

3. No analyses have been carried out in which all the plutonium isotopes have been tracked beyond a 20-day irradiation cycle. However, assuming that the behavior during subsequent cycles is similar to the behavior in the first cycle, it can be assumed that the total plutonium inventory is reduced by approximately 70% per cycle. Thus, the following results can be estimated:

Number of 20-Day Cycles	Exposure (MWD/MT)	Inventory Reduction (%)
1	500,000	72
2	1,000,000	91
3	1,500,000	97.3
4	2,000,000	99.2
5	2,500,000	99.75
6	3,000,000	99.93

It should be pointed out that it is not necessary to recycle the fuel through a reprocessing plant between each 20-day irradiation.

4. The program would be conducted in several phases, described briefly below:

Phase 1. The purpose of the first phase will be to carry out feasibility studies and identify go/no-go critical issues. The major milestones would consist of:

- Self-consistent conceptual design of reactor system
- Identification of go/no-go critical issues.
- Demonstration of fluid dynamic fuel refuel operations on full-scale mockup of fuel elements.
- Start fuel particle development.

This phase is expected to last two years and cost $8 M.

Phase 2: The second phase will consist of a preliminary system design and component development. Major efforts in this phase will be:

- Detailed preliminary design of reactor and process system.
- Fabrication and testing of prototype fuel particles.
- Design of high fluence test experiment for reactor materials compatibility.
- Electrically heated non-nuclear fluid dynamics and heat transfer experiments.

This phase will last three years and cost $50 M.

Phase 3: This phase will consist of an engineering design and component validation effort. The following major accomplishment will be expected:

- Detailed engineering design of reactor and process system
- Production type particles, fully tested.
- Nuclear test of PBR/Pu Burner fuel element
- High fluence tests of reactor materials.

This phase is expected to last 3 years and cost $150 M.

Phase 4: This phase consists of constructing a prototype PBR/Pu Burner which will serve as a demonstration unit. The duration of Phase 4 is expected to be 4 to 5 years and cost approximately $800 M-$1 B.

Phase 5: This is the operation phase. It consists of an overview of the demonstration operation, construction of six additional reactors, and disposition operation.

5. The primary technical issue which must be addressed concerns long term fuel element testing. Two issues need to be addressed in the development of the fuel element. First, the loading and unloading of particle fuel by hydraulic means needs to be confirmed. Second, fluid dynamics, heat transfer, and material compatibility experiments need to be carried out for a prototype fuel element. Initially the material compatibility tests will be carried out on test coupons in specially designed furnaces. The first set of fuel element tests can be carried out using electrical heating. Subsequent tests would be carried out in an appropriate reactor. (The final tests at target power densities will challenge existing test reactors, since substantial flux is required to drive a fuel element up to 10 M/L).

6. The costs are discussed below:

Capital Costs

Preliminary estimates have been made of capital costs for reactors and fuel fabrication/handling facilities. The estimates apply to the baseline configuration of three reactors and a fuel fabrication facility at each of two government-owned sites. This configuration will dispose of 50 Mg Pu in 16 years of 1.56 Mg/year at each site. Each reactor is rated at 630 MW_t. The fuel facility would have 10 parallel glovebox lines, each sized to handle 156 kg/year or 0.5 kg/day on a six-day work week.

The basic operations at the fuel facility involve dissolution of Pu and preparation of the particle fuel. After exposure in a PBR to high burnup, the spent fuel particles would be sealed in containers, stored for a period of time and sent to a federal waste repository. There is no reprocessing in the once-through fuel cycle. All the operations could take place under one roof. The capital investment for the fuel facility is estimated to be $1 B for two sites. The total capital investment, including reactors, is estimated to be $11.5 B for two sites.

Operating Cost

The operating cost of a PBR/Pu Burner site will consist primarily of wage expenses for operating and maintenance personnel. An estimate of the personnel for a three shift round-the-clock basis with a fourth shift for vacation, holidays, etc., is estimated to be 500 persons for one site. Based on these manpower estimates the total annual operating cost is estimated to be $50 M for one site and $100 M for two sites.

Total Cost

The total cost to dispose of 50 mg Pu with six 630 M_e reactors in 16 years is the sum of the above quantities, adjusted for a credit resulting from electricity sales. The development cost of $1 B is included as well as a credit for sale of electrical power (at $0.35/kWH). Power generation totaling of 3800 M_e at a 63% capacity factor would return $0.73 B/year and over 16 years would amount to $11.7 B. This revenue can be deducted from the total expenditure over 16 years of $14.1 B, resulting in a net cost for the entire campaign to dispose of 50 Mg Pu by PBR technology of $2.4 B.

Waste Processing

1. Since no waste reprocessing is envisioned for a PBR based concept, the waste conditioning step will be relatively simple. It is envisioned that the irradiated particles are suitably packaged and stored. Thus, the need for developing an elaborate flowsheet seems superfluous.

2. No technical issue can currently be identified to impede development of an irradiated fuel particle packaging scheme.

3. No design of such a processing and packaging facility has been carried out. However, it is not expected to be the item which controls the rate of construction.

4. No estimates have been made of a fuel particle packaging facility. However, the costs are not expected to be large compared to the reactor development, construction, and operating costs.

Waste Disposal

1. Since the waste packaging has not been explicitly defined, no precise answer can be given. However, the waste will consist of suitably packaged irradiated particulate fuel. Clearly, these packages could be adapted to conform to a geological repository environment.

2. The waste form will be irradiated particulate fuel for which a suitable packaging method needs to be defined. It is not expected to be technically impossible to find a suitable method of carrying out this step. Since no chemical reprocessing is required, it is expected to be a relatively small step compared to development and construction of the PBRs.

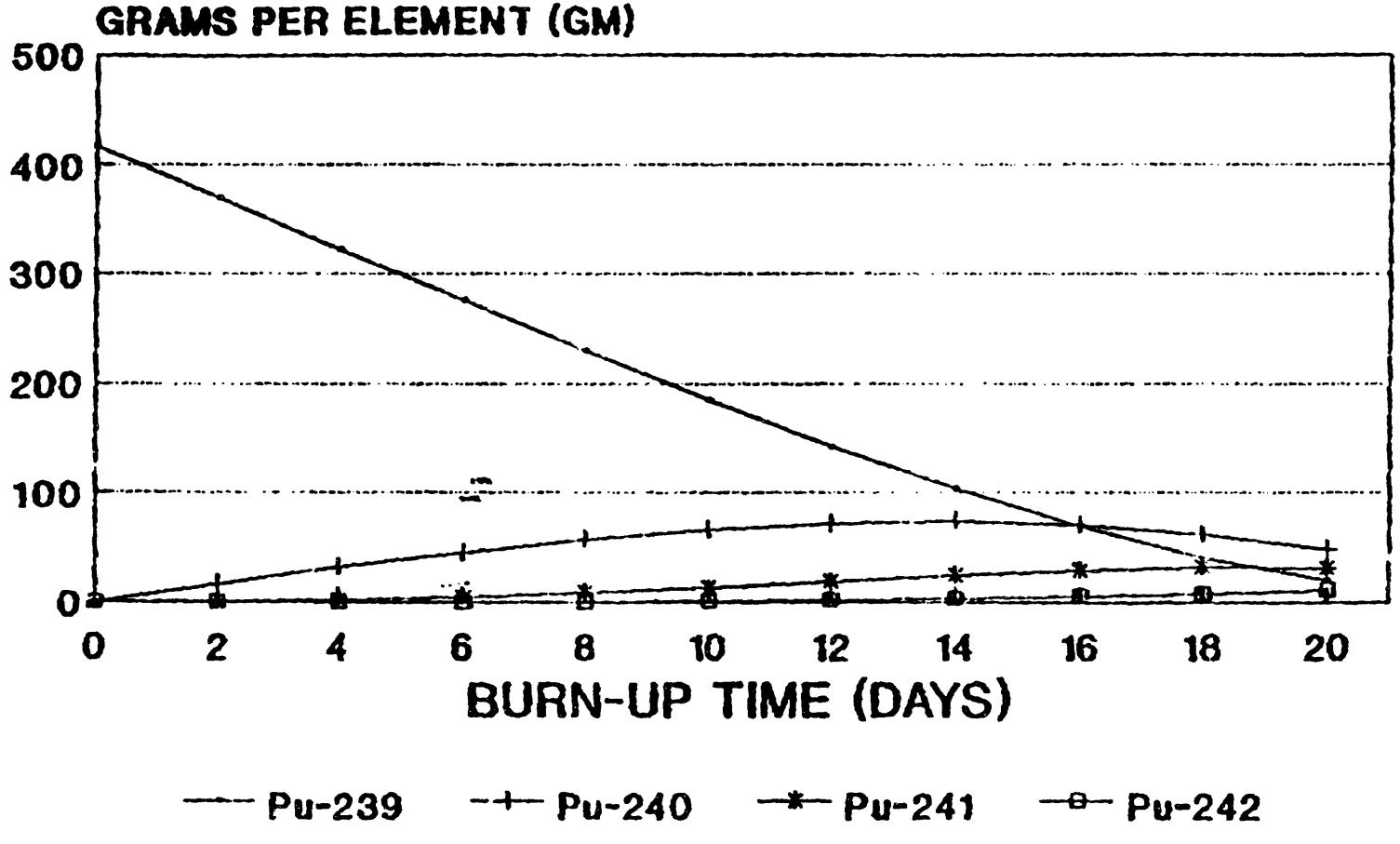
3. Currently no issue has been identified.

4. No estimate has been made of the time prior to opening a repository. Political issues could dominate this question, regardless of waste type or form.

5. No cost estimate has been made of a particle packaging facility. However, it will probably be small compared to the development and reconstruction of the PBRs.

6. Yes. No estimate has been made of the time which the package is self-protecting.

PU BURNER PERFORMANCE FOR AVERAGE ELEMENT @ 5 MW/LITER: ORIGEN2 RESULTS 127 ELEMENT CORE



Appendix I

Particle Bed Reactor Fuel Element

Appendix I

Particle Bed Reactor Fuel Element

Dr. C. S. Olsen

The Particle Bed Reactor (PBR) fuel element consists of three components: a cold frit, the particle bed fuel, and the hot frit. The coolant gas flows through the cold frit, the particle bed fuel, and out the cold frit (Figure I-1). The largest advantage for the PBR fuel element is the very high area to volume ratio in the bed which is very amenable to high heat transfer rates in the bed. This high heat transfer rate allows it to have high thrust to weight ratio for nuclear propulsion applications or high burnup rates in the case of plutonium recycling. This advantage can also be its Achilles heel in corrosion rates, flow instabilities, and fuel element integrity.

The coolant flow can be controlled by the pressure drop through the cold frit or through the fuel bed. In all reactors, flow is controlled by orificing. In the PBR fuel element, this orificing is done by the cold frit when the pressure drop is controlled through the cold frit. This has been the basis for the PBR fuel element design to date. The technical feasibility issues focus on fuel particle integrity, fuel element integrity, and power/flow matching. Each of these broad issues will be briefly discussed. Resolution of these issues would have to be performed through engineering design and fabrication and material process development.

Fuel Particle Integrity

The fuel particles for the PBR fuel element are approximately 0.5 mm in diameter and consist of a carbide fuel kernel, graphite coatings, and a carbide coating. Although the fuel tested to date has been a BISO fuel consisting of $UC_{1.7}$ kernel, a porous graphite layer, a dense graphite seal coating, and a zirconium carbide outer coating, TRISO-coated particles can be employed if the SiC layer is replaced by a ZrC layer. Currently, the NRC has only licensed TRISO fuel for commercial reactors.

The first issue for PBR fuel is the chemical compatibility of the fuel, coatings, fission products, and coolant with each other. For plutonium burning, because the coolant is assumed to be an inert gas or mixture of inert gases, the compatibility involves the remaining materials from which the fuel particle is fabricated. This material compatibility is manifested in the amoeba effect that occurs under a temperature gradient. This amoeba effect involves the diffusion of the fission products along the temperature gradient and the subsequent fission product chemical attack on the outer carbide coating. Plutonium diffusion may also occur along the temperature gradient.

Superimposed on the temperature gradient within a particle is the temperature gradient arising across the fuel bed annulus. This perturbation has not been evaluated or tested yet, but the magnitude of the gradient across the bed could be greater than that within an individual particle.

Another compatibility issue involves the fuel particle and the materials from which the cold and hot frits are constructed. These materials need to be chemically compatible for high temperatures and long times. In previous nuclear testing, either temperatures became high enough to melt the hot frit or the interaction between the fuel and the hot frit caused the hot frit to liquefy.

The diffusion of fission products in uranium carbide fuels appears to be governed by the diffusion coefficient of uranium in the uranium carbide. Because this diffusion coefficient is much less than that of uranium in UO_2, the fission products, particularly the gaseous ones, are retained. This retention leads to significant fuel swelling on the order of 20 to 30% with burnup when the fuel temperatures are greater than 1600 to 1700°C. This amount of swelling could easily lead to extensive fuel failure. Similar behavior would be

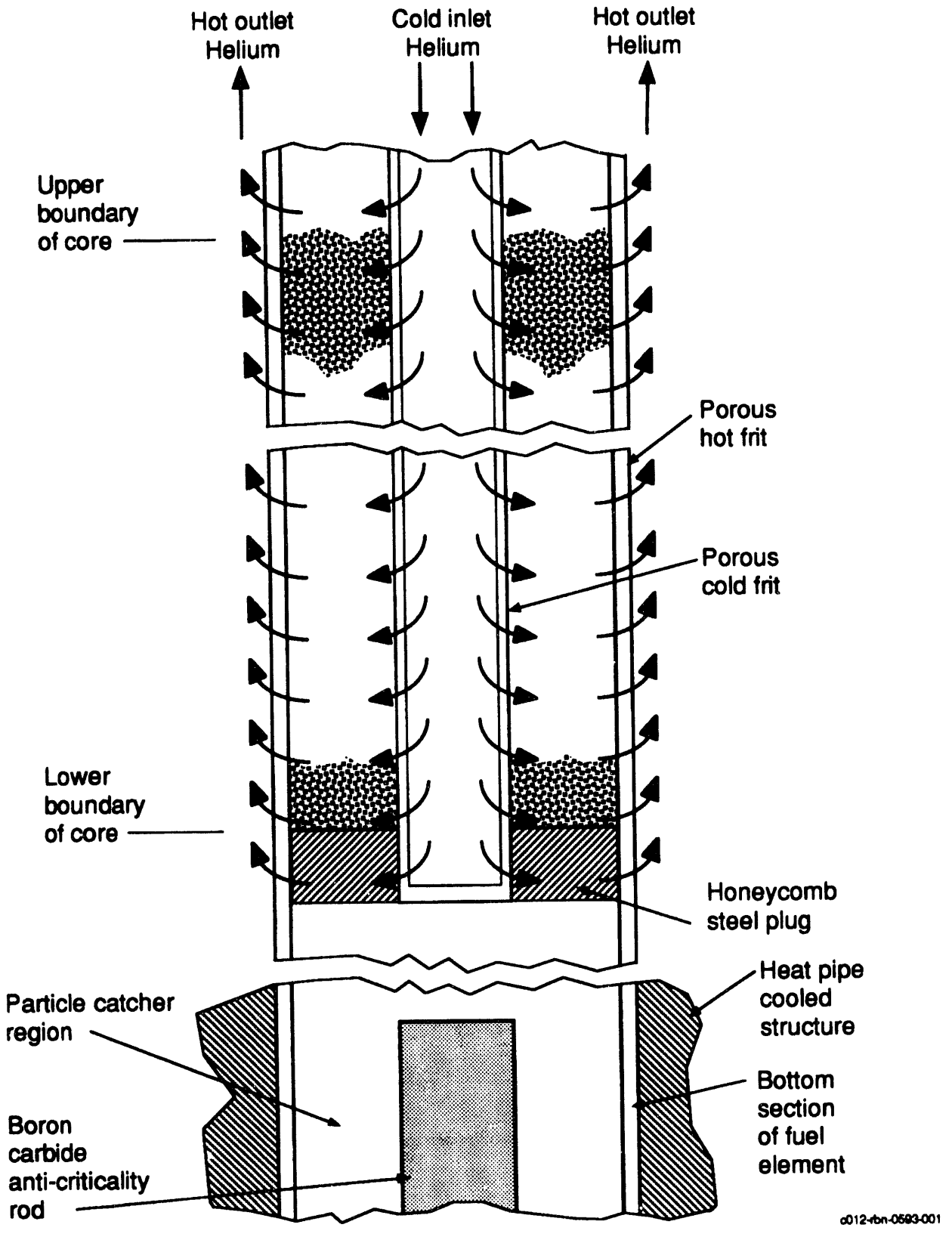


Figure I-1. Particle Bed Reactor fuel element.

expected in plutonium-based fuels. Fuel swelling has not been a problem for the HTGR fuels because the temperatures are lower than 1100°C.

Plutonium carbide fuels would have to be fabricated for the PBR elements with very low defect ratios. Some form of gelation process, internal or external, or a cryochemical process would have to be developed. Impurities in the plutonium may effect the development of these processes and the quality of the spherical particles produced. Extrapolation from the uranium fuels fabrication technology to plutonium is not warranted.

Fuel Element Integrity

The basic premise for the mechanical stability of the PBR fuel element is that the cold frit needs to act as an elastic sock and expand elastically when the PBR element is heated and contract elastically when it is cooled. The hot frit needs to be very rigid at operating temperature. This premise is compromised because the fuel particle bed in the annulus becomes very closely packed (locked) unless there is some provision made to prevent it. The hot frit will expand outward more than the cold frit placing the particle bed in compression and induce bed lockup. This bed lockup constrains the hot frit from expanding further axially, and additional temperature rise after the yield point is reached will cause plastic deformation of the hot frit. Upon cooling, the plastically deformed frit will contract. As a result the hot frit becomes shorter with cycling which could lead to fracture and loss of the fuel pellets. The number of cycles required for failure would depend on the mechanical properties of the material selected for the hot frit. Nuclear testing of two PBR fuel elements resulted in hot frit shortening and numerous cracks in one element, even though the initial material was very ductile. Good engineering design, appropriate material selection, and fabrication development may alleviate this problem, but currently this effect may limit the number of times the PBR fuel elements could be thermally cycled.

Power-Flow Matching

The cold frit had been fabricated from metal filters, which contain about 30% interlinked porosity and pores 5 to 10 microns in diameter. The flow passages in the particle fuel bed may be only slightly larger depending on the fuel particle size and its distribution. Bussard has questioned the thermal/hydraulic flow stability in these very small capillaries. Flow instability, if it occurs in a localized region, may propagate to adjacent regions because of changes in viscosity and density as a result of localized heating.

Analytical modeling of flow instability is extremely difficult, if not impossible, because of truncation and limits in the numerical analysis schemes handling the differential equations. Flow instabilities may only be determined by experimental measurements. Analysis of two particle bed experiments was performed by a graduate student at the Air Force Institute of Technology for a master's degree. The results of this analysis were inconclusive because some coefficients in the analytical model need to be determined experimentally. Experimental measurements of the flow resistance in cold frit materials resulted in values 2-1/2 times higher than that predicted theoretically.

Nuclear testing of two PBR fuel elements resulted in an apparent flow instability and loss of temperature control. The experiments experienced some experimental problems such as plugging the cold frits with graphite from blower motors and from closing the pores in the cold frit surfaces during element preparation to obtain a flat axial temperature profile. Even the thickness of the cold frit wall was varied to aid in shaping the axial temperature profile.

The Commonwealth of Independent States tested a PBR fuel element and concluded that the operating temperature needed to be limited to 2300°C and low power levels on the order of kilowatt or fractions of kilowatts per liter because of flow instabilities. They have engineered a different fuel form to achieve the high surface area to volume ratios.

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