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**Actinide Partitioning-Transmutation
Program Final Report.
I. Overall Assessment**

A. G. Croff
J. O. Blomeke
B. C. Finney



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NUCLEAR FUEL AND WASTE PROGRAMS

Waste Management Analysis for Nuclear Fuel Cycles
(Activity No. AP 05 25 10 0; FTP/A No. ONL-WHO1)

ACTINIDE PARTITIONING-TRANSMUTATION PROGRAM FINAL REPORT.

I. OVERALL ASSESSMENT

A. G. Croff
J. O. Blomeke
B. C. Finney

Date Published: June 1980

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GLOSSARY

actinides	in general, the group of elements heavier than and including actinium; in P-T discussions, often refers to only the transuranics (Np, Pu, Am, Cm, Bk, Cf)
AGNS	Allied-Gulf Nuclear Services' reprocessing plant in Barnwell, South Carolina
bidentate	dihexyl-N,N-diethylcarbonylmethylene phosphonate, a neutral organophosphorus extractant for actinide elements; diluted with diisopropylbenzene
CANDU	existing Canadian commercial reactor fueled with natural uranium and moderated with heavy water (D ₂ O)
CEC	cation exchange chromatography; used in actinide-lanthanide separations
CMP	dihexyl-N,N-diethylcarbonylmethylene phosphonate, a neutral organophosphorus extractant for actinide elements; diluted with diisopropylbenzene
consequence analysis	analysis of the risks from a fuel cycle operation or facility, assuming that all probabilities of occurrence are 100%
coprocessing	operating mode for a reprocessing plant in which only part of the uranium is separated from the plutonium (i.e., the plutonium is always diluted with uranium)
D ₂ O	heavy water; used as a moderator/coolant in reactors
2-EHOH	2-ethylhexanol
FFP	fuel fabrication plant; facility in which a mixture of uranium and plutonium oxides is fabricated into fuel assemblies
FRP	fuel reprocessing plant; facility in which spent fuel is dissolved and actinides are recovered
full-power year	indicates that the capacity factor was assumed to be 100% (i.e., the fuel was irradiated continuously for 365 days at full-rated power)
GeV	billion electron volts; used as a measure of particle energy/velocity
GWd	gigawatt-day = 10 ⁹ watt-days; used as a measure of the energy produced by nuclear fuel

GW(e)-year	gigawatt(electric)-year = 1000 MW(e)-year; in this report, the measure of the amount of reactor capacity [i.e., a 1000-MW(e) reactor operated for 1 year]. The amount of electricity actually produced by this capacity is less since a capacity factor must be included.
health effect	latent cancer or genetic damage
HEPA filter	high-efficiency particulate air filter
HLLW	high-level liquid waste; an aqueous solution from the first solvent extraction cycle in a fuel reprocessing plant which contains nitric acid, a small fraction of the uranium and plutonium, and virtually all of the other actinides and fission products
HLW	general term referring to HLLW or solidified HLLW
HM	heavy metal; usually refers to the total initial amounts of uranium and plutonium
HTGR	high-temperature gas-cooled reactor
K_d	parameter used as a measure of the degree to which nuclides are regarded as they migrate through the geosphere
kWhr(e)	kilowatt-hour (electric); a measure of the amount of electricity produced
lanthanides	rare-earth elements lanthanum through lutetium, which are generally chemically similar to the actinides and are produced as fission products
LMFBR	liquid-metal [-cooled] fast breeder reactor
LWR	light-water reactor
mill	10^{-3} dollar, or 0.1¢
MOX	mixed oxide; a mixture of uranium and plutonium oxides
MTHM, MTHM	metric tons of initial heavy metal in the unirradiated fuel
MWd	megawatt-day = 10^6 watt-days; used as a measure of the thermal energy produced by nuclear fuel
P-T	partitioning and transmutation; a process capable of reducing the amounts of certain long-lived, radiotoxic species (usually actinides but can also refer to fission products) normally present in radioactive wastes and converting them to shorter-lived or less toxic species

P-T cycle	a fuel cycle that includes provisions for P-T
partitioning	treatment designed to reduce the levels of elements having undesirable, long-lived isotopes in radioactive wastes to a greater extent than dictated by normal economic considerations
PWR	pressurized-water reactor
Purex	process for the extraction and purification of plutonium and uranium from an aqueous nitrate solution using TBP as the principal chemical reagent
reference cycle	a fuel cycle with no provisions for P-T, but with provisions for reprocessing and plutonium recycle
risk	average rate at which society is harmed
RO/CM	facility-operating philosophy in which the plant is operated remotely but maintenance is performed manually after decontamination
RO/RM	facility-operating philosophy in which the plant is both operated and maintained remotely
TBP	tributyl phosphate; a neutral organophosphorus compound used to recover actinides, primarily uranium, neptunium, and plutonium
transmutation	a process whereby long-lived nuclides are converted to shorter-lived or stable nuclides by bombardment with subatomic particles such as neutrons from nuclear power reactors
TRU	transuranic; indicates that the material is considered to contain sufficient amounts of certain actinides so that it is a "long-lived waste"
WTF	waste treatment facility; a facility containing actinide partitioning processes, which is colocated with another facility that generates wastes requiring partitioning

EXECUTIVE SUMMARY

This report is concerned with an overall assessment of the feasibility of and incentives for partitioning (recovering) long-lived nuclides from radioactive wastes and transmuting them to stable or shorter-lived nuclides for the purpose of reducing the long-term hazard of the waste. The actinides are the principal class of long-lived nuclides considered; however, a brief analysis of the partitioning and transmutation (P-T) of ^{99}Tc and ^{129}I is also given.

This assessment is based primarily on the results of a 3-year program directed by ORNL. The program included the following major aspects:

1. experimental and analytical studies of partitioning processes and flowsheets;
2. calculational studies of actinide, technetium, and iodine transmutation;
3. definition of reference (no P-T) and P-T fuel cycles and analysis of the incremental costs, risks, and benefits of P-T based on these cycles;
4. identification of other impacts of P-T;
5. an analysis of the feasibility of P-T;
6. a risk-cost-benefit analysis of the incentives for implementing P-T; and
7. an estimate of the time and costs required for the research, development, and demonstration that would be required to implement P-T.

Several organizations other than ORNL participated in the program because of their specialized experience and/or experimental facilities. The program considered conventional technology with a reasonable likelihood of near-term success. The reprocessing plant incorporated a coprocessing flowsheet, and the partitioning processes were based on achieving a workable, but not necessarily an optimum, process. LWRs were the primary transmutation devices considered, although other devices were investigated and evaluated.

The actinide partitioning flowsheets resulting from this program have two fundamental steps: (1) extraction of the actinides from the waste, and (2) recovery of the actinides. Processing of high-level waste to make the actinides extractable is unnecessary since most of the actinides contained therein are amenable to extraction. In the case of solid wastes such as dissolver solids and cladding, the actinides are solubilized into an extractable form with a nitric acid--hydrofluoric acid leach. The actinides present in HEPA filters and incinerator ashes are solubilized using a ceric nitrate--nitric acid solution to promote dissolution. Nonstrippable actinides are separated from salt wastes, such as solvent cleanup wastes, by acidification followed by contact with an alcohol [2-ethylhexanol (2-EHOH)]. The extractable actinides from these operations are then sent to actinide recovery, which consists of (1) recovering uranium, neptunium, and plutonium using Purex (TBP) extraction (not required for HLLW); (2) coextracting the remaining actinides plus lanthanides using a bidentate (a neutral organophosphorus compound) extractant; and (3) separating the actinides from the lanthanides using cation exchange chromatography.

The partitioning of actinides appears to be feasible based on the use of processing technology that has been experimentally verified at least at the laboratory level. Some of the partitioning processes, such as TBP extraction and acid leaching, have been demonstrated on a large scale in actual production operations. The partitioning of iodine also appears to be feasible using any of a variety of proven technologies. Although the partitioning of technetium was not specifically investigated, there is no known reason why it should not be feasible.

Calculational actinide transmutation studies indicate that actinide transmutation rates range from 5 to 11% per full-power year, with fast reactors having the higher rates. The effects of actinide recycle on uranium enrichment requirements and the fast reactor breeding ratio were small, except in the first two actinide recycles in the LWR. Power peaking is a consideration in the LWR, but not in the LMFBR. Overall, LMFBRs are better actinide transmutation devices, but not substantially so. The effect of higher-than-normal amounts of neptunium, americium,

and curium on fuel performance has not been determined. Technetium can be transmuted at the rate of 11% per full-power year in a PWR; iodine transmutation rates are about 3% per full-power year.

The transmutation of the actinides appears to be feasible in both thermal and fast reactors, subject to the acceptability of reactor fuels with higher-than-normal concentrations of the recycled actinides. Technetium transmutation also appears to be feasible, subject to the identification of a satisfactory form for incorporation into a rod (the metal may be satisfactory). Iodine transmutation is marginally feasible at best because of its low transmutation rate, the high volatility of iodine compounds at reactor operating temperatures, the corrosiveness of iodine and iodine compounds, and the production of xenon gas as a transmutation product.

The use of alternative transmutation devices (e.g., spallation devices or fusion reactors) appears to be feasible, assuming that the technology is feasible per se. These devices would have transmutation rates similar to those in LMFBRs. Therefore, their use would not offer any substantial advantage and would not affect the results of the assessment described here.

Actinide P-T would impact the fuel cycle by (1) requiring the casks that are used for shipment of fresh and spent fuel to be constructed of materials such as boron carbide, copper, and lithium hydride to provide biological shielding from the highly neutron active fuel, (2) making the disposition of existing waste and fuel inventories uncertain since irretrievable commitment of actinides to a repository might defeat the purposes of P-T and thus impede waste management until P-T was implemented, and (3) conflicting with existing U.S. nuclear policy by necessitating the use of reprocessing and plutonium recycle (i.e., the fuel cycle must be closed), both of which have been indefinitely deferred. The construction of a cask suitable for the shipment of P-T fuels appears to be possible in principle, subject to the fabricability of the unusual materials required for construction, although its payload is about two-thirds of that in a standard spent fuel shipping cask. With regard to the deferral of reprocessing and

plutonium recycle, it should be noted that this project represents the evaluation of an advanced waste management alternative and should not be construed as advocating or opposing the eventual implementation of these technologies.

The cost of implementing actinide P-T is about \$9.2 million (1979 dollars) per GW(e)-year [1.28 mills/kWhr(e)]. This cost principally results from the additional facilities and processing required to partition the actinides. It is estimated that the short-term radiological risk from the fuel cycle is increased by 0.003 health-effect/GW(e)-year and that the total short-term risk, including nonradiological risk, is increased by 0.57 health-effect/GW(e)-year. This is comparable to about 1.0 health-effect/GW(e)-year equivalent from natural background radiation. The radiological risk increases are predominantly due to routine effluents. The petroleum combustion products from the partitioning facility power plant boilers are the principal source of the nonradiological risk increase. The expected long-term benefit (i.e., incremental risk reduction) of P-T is 0.06 health-effect/GW(e)-year integrated over 1 million years. This is about 0.001% of the health effects to be expected from natural background radiation. The long-term risk results entirely from expulsive events such as volcanoes and meteorites since the actinides released in a leach incident do not emerge from the geosphere into the biosphere in 1 million years. It should be noted that this benefit is the difference between the long-term risk for the reference cycle, 5.16 health-effects/GW(e)-year, and that for the P-T cycle, 5.10 health-effects/GW(e)-year. The principal contributors to these long-term risks are ^{99}Tc (92%) and ^{129}I (8%). The actinides only account for a small fraction of the total, although it is the reduction in actinide content of the waste that gives the small reduction of 0.06 health-effect/GW(e)-year. It should be noted that the long-term benefits were based on a very conservative long-term risk analysis. This approach was used because it was an expeditious way to examine the incentives for implementing P-T. The values used for this project should not be considered appropriate for other studies of the same area or for other studies of other areas.

Combination of the costs, risks, and benefits yields a P-T cost of \$32,400 per person-rem saved if the short-term, nonradiological risk is excluded. If this risk is included, the short-term risks will exceed the long-term benefits integrated over 1 million years. Furthermore, sensitivity studies of the important parameters in the costs, risks, and benefits indicate that it is extremely unlikely that the cost of reducing the long-term risk by 1.0 person-rem could ever approach the often-used criterion of \$1000 per person-rem. This is principally because the calculation of the long-term risk is very conservative in favor of P-T. Thus, there are no cost or safety incentives for partitioning and transmuting the actinides for waste management purposes.

The incentives analysis is less clear in the case of technetium and iodine. If the conservative long-term risk analysis is used, there are incentives for technetium P-T (assuming that a partitioning process can be developed) and for iodine P-T (assuming that iodine transmutation is feasible). However, the incentives for technetium and iodine P-T are strongly dependent on the use of these very conservative long-term risk analyses; less conservative assumptions would make the benefits negligible as compared to the risks and costs.

Finally, it is estimated that the first commercial-sized partitioning facilities could be operable in about 20 years, assuming that a well-supported and orderly research, development, and demonstration (RD&D) program were instituted immediately and that no licensing delays occurred. The cost of the RD&D program would be about \$900 million (1979 dollars) for the entire P-T concept, most of which would be designated for a hot partitioning pilot plant. However, as a result of the lack of incentives for actinide P-T, further RD&D in support of P-T is not warranted unless a decision is made to proceed with P-T.

ACTINIDE PARTITIONING-TRANSMUTATION PROGRAM FINAL REPORT.I. OVERALL ASSESSMENT

A. G. Croff
J. O. Blomeke
B. C. Finney

ABSTRACT

This report is concerned with an overall assessment of the feasibility of and incentives for partitioning (recovering) long-lived nuclides from fuel reprocessing and fuel refabrication plant radioactive wastes and transmuting them to shorter-lived or stable nuclides by neutron irradiation. The principal class of nuclides considered is the actinides, although a brief analysis is given of the partitioning and transmutation (P-T) of ^{99}Tc and ^{129}I . The assessment is based primarily on a 3-year program directed by ORNL, with participation by several organizations having special expertise and facilities. The program included (1) experimental and analytical studies of partitioning flowsheets; (2) calculational studies and literature reviews of actinide, technetium, and iodine transmutation; (3) analysis of the incremental costs, risks, and benefits of P-T based on defined reference (no P-T) and P-T fuel cycles; (4) identification of other impacts of P-T on the fuel cycle; (5) analysis of the feasibility of P-T; (6) analysis of the incentives for P-T; and (7) an estimate of the time and costs required for the research, development, and demonstration (RD&D) that would be required to implement P-T on a commercial basis. The results obtained in this program permit us to make a comparison of the impacts of waste management with and without actinide recovery and transmutation.

Three major conclusions concerning technical feasibility can be drawn from the assessment: (1) actinide P-T is feasible, subject to the acceptability of fuels containing recycle actinides; (2) technetium P-T is feasible if satisfactory partitioning processes can be developed and satisfactory fuels identified (no studies have been made in this area); and (3) iodine P-T is marginally feasible at best because of the low transmutation rates, the high volatility, and the corrosiveness of iodine and iodine compounds. It was concluded on the basis of a very conservative repository risk analysis that there are no safety or cost incentives for actinide P-T. In fact, if nonradiological risks are included, the short-term risks of P-T exceed the long-term benefits integrated over a period of 1 million years. Incentives for technetium and iodine P-T exist only if extremely conservative long-term risk analyses are used.

It is estimated that P-T would take 20 years to implement on a commercial scale, assuming a well-supported development program and no licensing delays. The development program would cost about \$900 million (1979 dollars), with the largest fraction being required for a hot partitioning pilot plant. Further RD&D in support of P-T is not warranted.

1. INTRODUCTION, SUMMARY, AND CONCLUSIONS

1.1 Introduction

This report provides an overall assessment of the feasibility and incentives for operating the nuclear fuel cycle so that the most troublesome long-lived constituents of radioactive wastes are partitioned and transmuted. Partitioning, when conducted for waste management purposes, is defined as treatment designed to reduce the levels of chemical elements having undesirable, long-lived isotopes in radioactive wastes to a greater extent than that dictated by normal economic considerations. That is, partitioning involves separating the long-lived nuclides from the wastes and recovering them in a form suitable for further treatment. Transmutation is defined here as a process whereby long-lived nuclides are converted to shorter-lived or stable nuclides by bombardment with subatomic particles, such as neutrons from nuclear power reactors. Partitioning and transmutation (P-T), when taken together, form a waste management concept which would be capable of reducing the amounts of certain long-lived, radiotoxic species normally present in radioactive wastes and converting them to shorter-lived or less toxic species.

In this report, P-T is defined as a waste management option which could be implemented for waste management purposes if shown to be feasible and cost-effective. However, when all of the impacts of P-T on the nuclear fuel cycle are examined, it is clear that P-T is actually a new fuel cycle option since its implementation would affect most of the operations in the fuel cycle to varying degrees. To summarize, P-T would do the following:

1. require the installation of additional waste processing steps in the reprocessing plant,
2. require the installation of additional waste processing steps in the mixed-oxide (MOX) fuel fabrication plant,
3. alter the volume and composition of radioactive wastes being sent to a waste repository,

4. require requalification of reactor fuels and possibly new fuel fabrication techniques,
5. alter the neutronic behavior of nuclear reactors because of the presence of recycled actinides in the fuel, and
6. necessitate new shipping cask designs because of the increased amounts of neutron emitters present in nuclear materials containing recycled actinides.

Thus P-T should not be considered as a waste management operation affecting only the treatment and composition of radioactive wastes; instead, it is an overall fuel cycle concept that is significantly different from fuel cycles involving recycle of only the principal fissile and fertile values.

1.1.1 Background

Studies have been made of various selected aspects of P-T since the mid-1960s. The most common type of study involved actinide transmutation calculations followed by calculation of the toxicity index of the high-level waste with and without transmutation of the actinides. (Note: The toxicity index is the amount of water required to dilute all of the isotopes in a unit volume of waste to their Radionuclide Concentration Guide values given in 10 CFR 20.¹) The conclusion reached in most of these studies was that the toxicity, and therefore the risk, due to high-level wastes (HLW) in a repository could be reduced by factors of 100 to 200 for waste decay times greater than 1000 years.² However, these studies generally ignored partitioning, the more realistic impacts of transmutation on the transmutation device, and other fuel cycle impacts of P-T.

Limited studies of partitioning processes and technology were conducted during 1973-1975 (see refs. 3 and 4). The principal results of these studies were: (1) an evaluation of previous work and synthesis of this work into reprocessing plant flowsheets for partitioning actinides from the waste streams, and (2) recommendations concerning the approaches that should be used in future partitioning studies. Although laboratory investigations of the recommended processes were begun, they were terminated before significant results could be obtained.

Only one realistic study has been made to determine the long-term benefits of removing the actinides from high-level waste.⁵ The previously mentioned studies of long-term benefits that were based on the toxicity index are not realistic because the toxicity index assumes that the wastes are ingested directly with no change in composition. However, a more realistic assumption is that the nuclides might be leached from the waste in the repository in the distant future and then be slowly transported through the geosphere to the biosphere. During this transportation process, the chemical and physical interactions with the geosphere and the biological differentiation in the biosphere have the net effect of greatly retarding the release of radioactive isotopes and substantially changing the elemental and isotopic mixture ultimately ingested. The study cited above,⁵ which included these effects, concluded that ". . . for the situations investigated the incentives for a special effort to remove any elements, including the transuranics, from high-level waste are vanishingly small . . ." However, since the objectives of this study did not include consideration and comparison of the near-term risks and costs of removing the actinides from high-level waste to the calculated benefits, it is difficult to state conclusively that there are no incentives for actinide removal until the penalties incurred by the process are assessed.

No studies have been made of the other varied, but important, impacts of P-T on the nuclear fuel cycle. Examples of these impacts are the effects of the highly neutron-active transplutonium isotopes on fuel fabrication, transportation, and handling; the effects of neptunium, americium, and curium on in-reactor behavior, fabricability, and cladding compatibility of reactor fuels; and the disposition of actinides produced prior to the implementation of P-T.

Finally, and most importantly, no overall study has been made of the feasibility and incentives for implementing the P-T concept. This type of study, in which all short-term and long-term advantages and disadvantages would be included, is necessary if the incentives for P-T are to be realistically and believably evaluated.

In 1976, the U.S. Energy Research and Development Administration [now the U.S. Department of Energy (DOE)] asked the Oak Ridge National Laboratory (ORNL) to develop a program to establish the technical feasibility and incentives for partitioning elements having long-lived isotopes and transmuting them to shorter-lived or stable isotopes in power reactors. The program was broadly based, consisting of both experimental and computational activities that are required to develop a meaningful and defensible evaluation of the P-T concept. In addition to ORNL, several other organizations having specialized experience and experimental facilities also participated in the program. These organizations included:

1. Argonne National Laboratory;
2. Brookhaven National Laboratory;
3. Mound Laboratory;
4. Savannah River Laboratory;
5. Sandia Laboratory;
6. Rocky Flats Plant;
7. Idaho National Engineering Laboratory;
8. the Ralph M. Parsons Company;
9. Science Applications, Inc.; and
10. Los Alamos Technical Associates.

The program lasted approximately 3 years. The first seven organizations listed above, plus the ORNL Engineering Physics Division, conducted studies on specific aspects of P-T during the first 2 years of the program. The results of these individual studies were used by the staff of the ORNL Chemical Technology Division to develop fuel cycle material and facility descriptions, which were then subjected to risk and cost analyses by the last three organizations (see list above) during the third year of the program. In the final part of the program, all these results were evaluated, leading to an assessment of the feasibility and incentives for P-T and the specification of the research, development, and demonstration (RD&D) requirements needed to implement P-T. The information, conclusions, and recommendations developed by these efforts are expected to be used as the technical backup for DOE's decision as to whether P-T is to be implemented, subjected to additional study, or eliminated as a waste management option.

1.1.2 Scope and ground rules

This report summarizes the results of the ORNL-directed program described above. The procedures used and detailed data obtained in the individual studies conducted during the first 2 years of the program, and afforded by the risk and cost analyses conducted during the third year of the program, are contained in a series of programmatic progress reports⁶ and in topical reports⁷ prepared by each participating organization. This information will not be repeated except as it is required to support the assessment contained herein.

Many of the individual studies and risk and cost analyses may have more general applications than P-T. For example, some of the processes examined for partitioning purposes may also find use in reducing plutonium losses or reducing the volume of radioactive wastes from conventional (non-P-T) fuel cycles. In addition, the information on partitioning developed as a part of this program will be necessary to assess the feasibility and incentives of extraterrestrial disposal of potentially troublesome waste constituents, as is being considered in a separate program.⁸ Finally, the incentives analysis in this report represents the first known attempt to balance the short-term effects (risk and cost) of an advanced waste management option against the long-term benefits of that option. It should be noted that this study did not consider whether fuel cycle wastes could be sufficiently decontaminated from transuranic (TRU) nuclides to permit their disposal as non-TRU wastes.

The ground rules used to guide the P-T program were as follows:

1. Only conventional chemical processes with a reasonably high assurance of near-term success and availability were to be considered.
2. The reprocessing flowsheets considered in this program were to include coprocessing in accordance with ERDA (now DOE) guidance so that the resulting facilities would be consistent with the current Administration's nonproliferation objectives.
3. LWRs were to be considered as the primary transmutation devices in accordance with ERDA (now DOE) guidance; FBRs were examined as a parametric variation.

4. The major emphasis of the program was to obtain a meaningful and defensible analysis of the feasibility, incentives, and RD&D time and cost requirements of P-T; the objectives did not include the development of P-T technology.

Thus the analysis presented in this report is an overall, but not necessarily optimum, assessment of the P-T waste management concept. A sensitivity analysis (Sect. 1.2.4.3) was used to examine the effects of the nonoptimality. In many instances, it was necessary to make approximations or assumptions based on our best technical judgment due to budgetary or time constraints. The major approximations and assumptions that were used and a qualitative evaluation of their potential impact are given in Sect. 1.1.4.

The fuel cycle being examined must be "closed" in order to evaluate or even consider the implementation of P-T. That is, provisions must exist for reprocessing spent fuel to recover the principal fissile and fertile values and fabricating them into fresh fuel. If the fuel cycle is not closed, P-T is obviously impossible. However, it should be noted that this project represents the evaluation of an advanced waste management alternative and should not be construed as advocating or implying the eventual implementation of these technologies.

1.1.3 General approach

The general approach used in this program was to perform an incremental cost-risk-benefit analysis of the P-T concept. As a prerequisite, two closed LWR fuel cycles, a P-T cycle and a reference (no P-T) cycle, were defined. These fuel cycles are the same in all respects except that the reference fuel cycle involved the use of a coprocessing flowsheet for recovery of the economic values of uranium and plutonium and recycling the recovered uranium and plutonium, whereas the P-T fuel cycle used coprocessing plus additional partitioning processes to recover actinides from refabrication and reprocessing plant wastes and recycled these recovered actinide elements to the reactor.

The reference and P-T fuel cycles are depicted simultaneously in Fig. 1.1. The reference fuel cycle includes everything to the left of

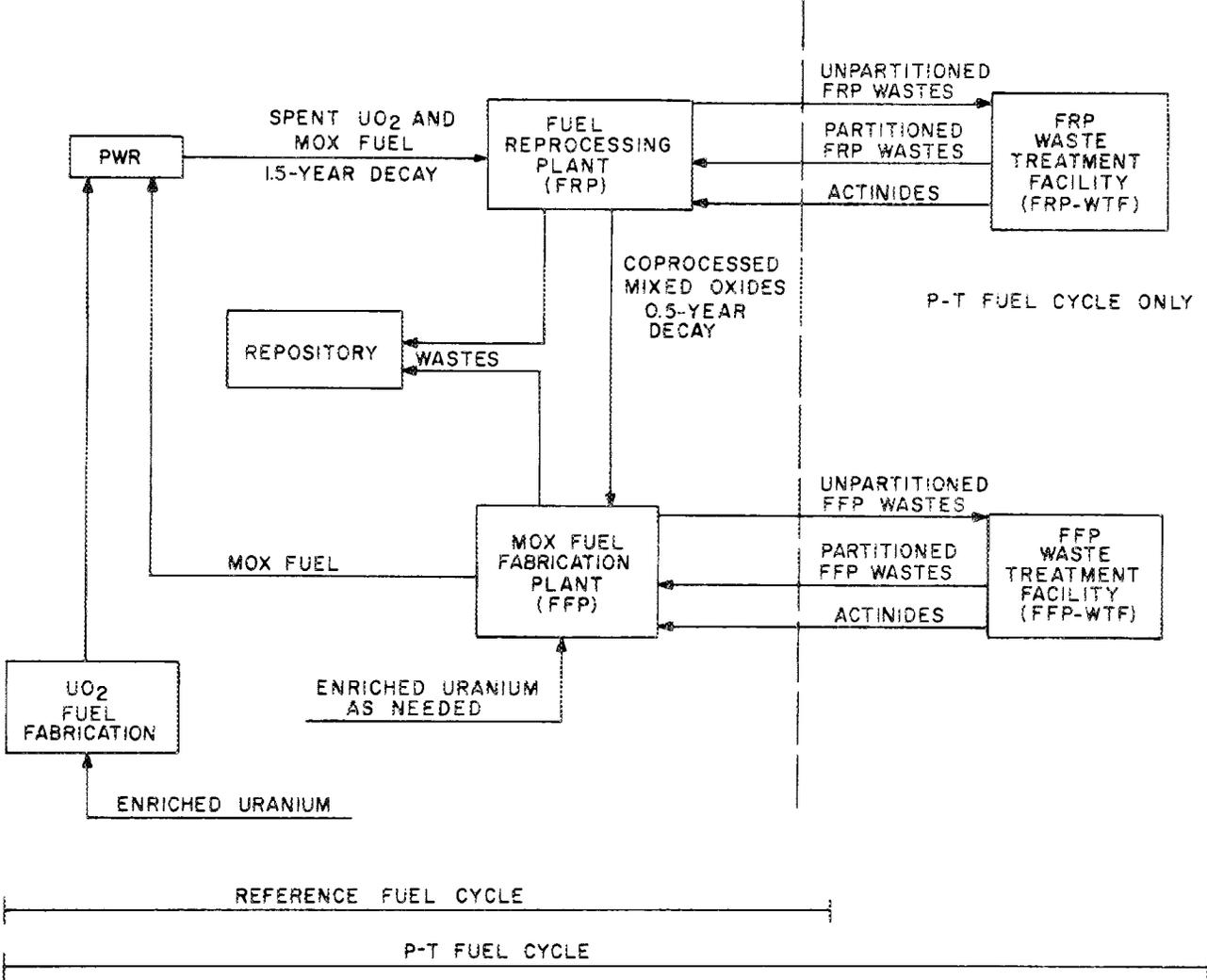


Fig. 1.1. Schematic diagram of reference and P-T fuel cycles.

the dashed, vertical line. The fuel charged to the PWR is assumed to be comprised of 33% MOX fuel and 67% enriched-uranium fuel. After an exposure of 33 GWd per metric ton of heavy metal (MTHM), which is achieved after 3 years in the reactor, the fuel is discharged, allowed to decay for 1.5 years, and then transported to the fuel reprocessing plant (FRP). The fuel is reprocessed using a coprocessing Purex flowsheet, which recovers and purifies uranium and plutonium with only a partial separation of the uranium from the plutonium. After an additional 0.5-year decay, the recovered uranium and plutonium, along with a substantial fraction of the neptunium, is then sent to the colocated MOX fuel fabrication plant (FFP), where 33% of the fuel for the next reactor reload is fabricated. At this point, some additional enriched uranium may be required to maintain the MOX fraction at 33% since the fissile plutonium content decreases as it is recycled. The wastes sent to the repository include about 4% each of the uranium and plutonium, 25% of the neptunium, and virtually all of the fission products, americium, curium, ^{14}C , and fuel assembly structural materials.

The P-T cycle is quite similar to the reference cycle except that the waste treatment facilities (WTFs) (see the right-hand side of the dashed line, Fig. 1.1) are brought into play. These facilities take the actinide-bearing wastes from the fuel reprocessing and MOX fuel fabrication plants and process them to reduce the actinide contents of the wastes. The streams returning to the main reprocessing and fabrication facilities are the actinide-depleted (partitioned) wastes and the actinides that were recovered by partitioning. The recovered actinides are then fabricated into MOX fuel, together with the uranium and plutonium recovered in the FRP. Based on experimental studies, the overall amount of all unrecovered actinides is estimated to be about 0.25% of the spent fuel in the P-T case. These unrecovered actinides are sent to the repository in combination with the various wastes.

The fuel cycle operations in which P-T is expected to have a significant impact are characterized according to three criteria: risks, costs, and benefits. In this report, risk is defined as the short-term routine or accidental, radiological or nonradiological impact of each facility.

This risk is developed by using source terms during routine operation and/or probabilistic accident frequencies and consequences with standard meteorological and biological models to determine the impact of each facility. The cost criterion, which includes only the monetary cost, is generated by developing conceptual plant designs and applying standard costing techniques. The benefits, which refer to the reduction in the long-term probabilistic dose from a waste repository, are determined by calculating the probabilistic accident consequences for the repository via computer codes and then using standard meteorological and biological models to determine the dose.

The differences in the risks, costs, and benefits of the operations in the two fuel cycles that are attributable to P-T can be calculated by examining separately each of the fuel cycle operations depicted in Fig. 1.1. These differences are designated as "incremental" risks, costs, and benefits. The advantage of using an incremental analysis is that those operations not affected by P-T will be identical in the two cycles (e.g., UO_2 fabrication) and need not be considered at all. This procedure significantly reduces the magnitude of the task.

The fuel cycle operations that are expected to be significantly affected by P-T, and thus have nonzero incremental changes in the risks, benefits, or costs, are as follows: (1) fuel reprocessing and associated waste treatment; (2) MOX fuel fabrication and associated waste treatment; (3) waste disposal; (4) transmutation; and (5) transportation of spent fuel, fresh fuel, and waste. However, by defining the scopes of the FRP and the MOX FFP appropriately, it is possible to neglect the differences in these two plants in the reference and P-T cycle (except for the short-term risk) and consider only their respective WIFs.

The final step in the incentives analysis is to compare the risks, costs, and benefits. This is done by comparing the individual values with other well known values such as natural background (for the risks and benefits) and the cost of electricity (for the economics of P-T). Then, the risks, costs, and benefits are combined to yield the total cost of reducing the long-term risk from the waste repository by 1.0 person-rem (i.e., \$/person-rem for the P-T concept). This value

can then be compared to an external criterion (e.g., the \$1000/person-rem value in 10 CFR 50, Appendix I⁹). The last part of the analysis is concerned with investigating the sensitivity of the comparisons to uncertainties in data and the effects of major assumptions.

The other two aspects of this assessment are feasibility and an estimate of the time and cost requirements for RD&D needed to implement P-T. The feasibility assessment is given as an integral part of the discussion of specific technical aspects of the P-T fuel cycle. A consolidated discussion of the time and cost requirements for RD&D is included at the end of the report after the feasibility and incentives of P-T have been presented.

1.1.4 Major assumptions and approximations

Three major assumptions were made in the P-T assessment program. The first was that a process capable of partitioning the actinides at a very low cost and impact (i.e., a "magical" process) does not exist. The result of this assumption is that, while partitioning processes superior to those described herein might be developed, they will not reduce the costs and impacts of P-T by more than a factor of 2.

The second assumption was that certain technological aspects of P-T are feasible, even though they presently have no firm experimental basis. Two important examples are fuel performance and the operation of partitioning facilities closely tied to FRPs and FFPs. Fuels containing high concentrations of neptunium, americium, and curium were assumed to have acceptable irradiation behavior based only on the acceptability of uranium-plutonium fuels. All of the processes included in the partitioning facility have been tested experimentally to some extent. However, no tests have been made of an integrated flowsheet, with its many recycle streams and possible impurities. Until such tests are performed, the effects of recycling these streams on the operability and performance of the facility are uncertain.

The third assumption was that a probabilistic risk analysis is an acceptable measure of the actual risks of operating fuel cycle facilities and waste repositories.

The single major approximation made in this analysis was that the impact of P-T on fuel cycle facilities other than reprocessing and refabrication plants is zero; therefore, such facilities do not need to be considered. In reality, there would probably be second- or third-order effects of a detectable, but probably insignificant, magnitude on the other facilities (e.g., the reactor, uranium enrichment). The accuracy of this approximation can be verified only by performing a detailed conceptual design of all fuel cycle facilities, an undertaking far beyond the scope of this program.

1.2 Summary

1.2.1 Partitioning

The ORNL program has developed conceptual partitioning flowsheets for both the FRP and the MOX FFP. The processes used in these flowsheets and the estimated degree to which they reduce the amounts of unrecovered actinides in wastes are described in Sect. 1.2.1.1. The possibility of separating two important long-lived fission products, ^{99}Tc and ^{129}I , is summarized in Sect. 1.2.1.2; the feasibility of the overall actinide partitioning process is addressed in Sect. 1.2.1.3. The possible impacts of advanced partitioning processes and the areas in which improvements might be expected are discussed in Sect. 1.2.1.4.

1.2.1.1 Actinide partitioning flowsheets. Partitioning of the wastes involves two generic steps: separation of the actinides from the waste by either leaching or breaking down strong organic-actinide chemical compounds, and recovery of the actinides. The second step is accomplished using the following standardized processing sequence:

1. The tetravalent and hexavalent actinides (uranium, plutonium, and neptunium) are recovered by TBP extraction, stripped from the solvent, and returned to the parent facility.
2. The trivalent actinides (americium and curium) and lanthanides are coextracted from the waste using a bidentate (CMP^{*}) extractant and stripped from the solvent for subsequent treatment using

* CMP = dihexyl-N,N-diethylcarbamylmethylene phosphonate.

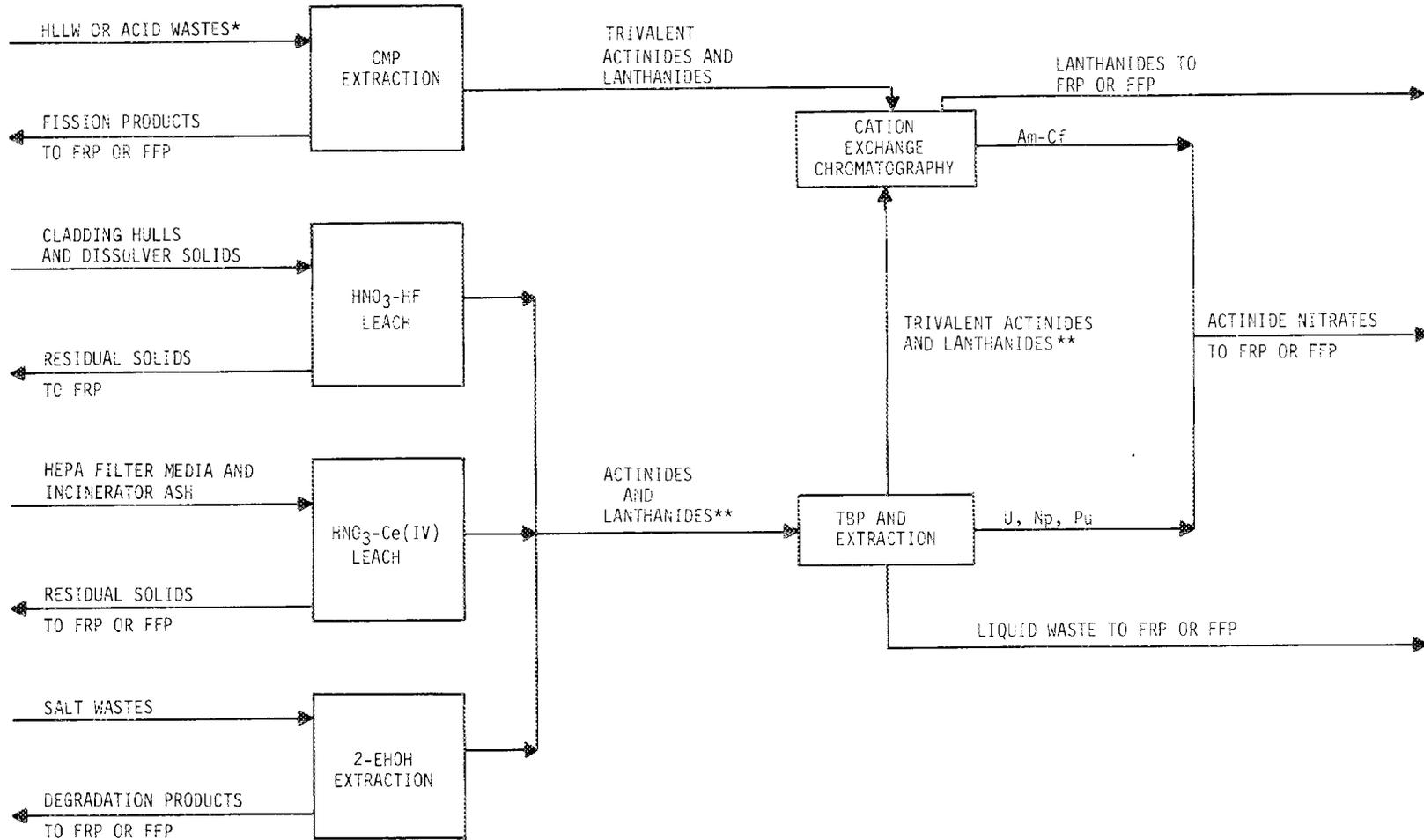
cation exchange chromatography (CEC). The residual waste is returned to the parent facility for solidification before disposal.

3. The actinides are separated from the lanthanides using CEC. The lanthanide fraction is returned to the parent facility and mixed with the treated HLLW prior to solidification. The recovered actinides (americium and curium) are returned to the parent facility for conversion to the oxides.

This sequence is depicted schematically in the generic partitioning flowsheet shown in Fig. 1.2.

The HLW, which is the raffinate from a TBP first-cycle solvent extraction in the FRP, is fed directly to the CMP extraction process for recovery of the trivalent lanthanides and actinides. These two groups of elements are subsequently separated using CEC.

The cladding hulls and dissolver solids, which have been previously leached with nitric acid, are subjected to a final leaching with HNO_3 -HF for removal of additional actinides. The fluoride ion catalyzes the dissolution of previously insoluble actinides. The HEPA filter and incinerator ash wastes from both the FRP and the FFP contain actinides that are also largely insoluble in nitric acid. Leaching the filter media with HNO_3 -HF results in an intractable, gooey mass that cannot be filtered or centrifuged and will result in complete dissolution of the incinerator ash. Leaching with a HNO_3 -Ce(IV) solution has been found to be a suitable alternative. This process allows the physical characteristics of the HEPA media to be retained and only partially dissolves the incinerator ash. The Ce(IV) is produced by the electrolytic oxidation of Ce(III); and when the leaching is complete, oxalic acid is added to the system to convert Ce(IV) to Ce(III) and thus reduce equipment corrosion rates. The salt wastes, principally the Na_2CO_3 solutions from solvent cleanup, contain a variety of actinides, many of which are bound in nonstrippable actinide-organic complexes. These complexes are destroyed and removed by extraction with 2-ethylhexanol (2-EHOH). The actinides from these three wastes are then separated and recovered, first by TBP



*RAFFINATES FROM TBP EXTRACTION IN FRP OR FFP
 **PREDOMINANTLY STABLE CERIUM AND GADOLINIUM

Fig. 1.2. Generic actinide partitioning flowsheet.

(for uranium, neptunium, and plutonium) and then by CMP (for trivalent actinides and lanthanides) extraction, followed by CEC (to separate the trivalent actinides and lanthanides).

All of the actinide-depleted wastes are returned to the parent facility for final treatment (e.g., vitrification, concretion) and packaging for disposal. In general, the increases in process waste volumes are held to a minimum because of the use of chemicals that can be recovered and recycled. The concreted wastes increase significantly in volume (about 50%), and this can be attributed to the wastes produced by the CEC process and the additional solvent cleanup chemicals used in the WTFs. Failed equipment waste increases by 100% at the FRP site and 25% at the FFP site due to the WTFs.

The estimated total amounts of actinides reporting to the fuel cycle wastes for both the reference case and the P-T case are summarized in Table 1.1. These values are based on results from experimental partitioning studies. As is evident, the total amounts of unrecovered actinides have been decreased by a factor of 16 for uranium and plutonium, by a factor of 100 for neptunium, and by a factor of 400 for the transplutonium actinides. Only 25% of the neptunium reports to wastes in the reference cycle because it follows the coprocessed product in the reprocessing plant under the assumed operating conditions.

1.2.1.2 Partitioning of technetium and iodine. While procedures for partitioning technetium from the reprocessing plant streams and dissolver solids in which it occurs were not identified in these studies, there is no known reason why acceptable techniques could not be designed within the framework of the actinide partitioning flowsheets that have been developed.

Iodine removal is, in effect, already incorporated in FRP flowsheets. Its removal from FRP streams by techniques such as caustic scrubbing, mercuric nitrate scrubbing, zeolite adsorption, charcoal adsorption, and the Iodox process has been demonstrated.

Table 1.1. Total actinides reporting to wastes as a percentage of facility feed without and with partitioning

Element	Fuel reprocessing plant		MOX fuel fabrication plant	
	Reference cycle	P-T cycle	Reference cycle	P-T cycle
Uranium	2.0	0.1	2.0	0.15
Neptunium	21.4	0.1	4.5	0.15
Plutonium	2.0	0.1	2.0	0.15
Americium ^a	100.0	0.1	20.5	0.15
Curium ^a	100.0	0.1	20.5	0.15

^aThe same values are assumed for berkelium and californium.

1.2.1.3 Feasibility of actinide partitioning. Six processes used in the partitioning facilities are described in Sect. 1.2.1.1: Purex (tributyl phosphate, TBP) extraction, CMP extraction, CEC, 2-EHOH extraction, nitric acid--hydrofluoric acid leaching, and nitric acid--ceric nitrate leaching. Based on the engineering evaluations and experimental studies conducted as a part of this program, we believe that all of these processes are feasible and consequently that partitioning, as a whole, is feasible. Both TBP extraction and nitric acid--hydrofluoric acid leaching are well demonstrated technologies that have been used for years within the nuclear defense program. CEC has also been demonstrated, although less extensively than the Purex process. The CMP extraction process has been tested under radioactive conditions on small samples of discharged LWR fuel and on Idaho Chemical Processing Plant waste. Finally, both the 2-EHOH extraction and the nitric acid--ceric nitrate leach processes have been demonstrated on a laboratory scale with tracer levels of radioactivity.

The principal caveat concerning the partitioning flowsheets developed in the course of this program is that they are very highly integrated (i.e., considerable internal recycle of plant streams is required) to prevent actinides from reporting to the wastes. As a result, it is possible that one or more chemical species having a detrimental effect

on plant performance could form and continue to build up within the plant, causing unacceptably high amounts of actinides to report to the waste or forcing plant shutdown due to operating difficulties.

Additionally, the CEC process, although judged to be feasible, has some serious operational drawbacks arising from the severe radiation damage to the resins by the $^{242,244}\text{Cm}$. This damage results in a possible safety hazard from the production of explosive degradation products, and leads to increased costs and operational complexity because of the need to discard the resin after a single use.

1.2.1.4 Advanced partitioning processes. With one exception, the flowsheets described in this report probably constitute a near-optimal partitioning technology since (1) the reagents are relatively radiation-stable, (2) the wide variety of wastes to be treated will require multiple processes in any case, and (3) the processes generally involve the use of liquid or dissolved reagents that require relatively small but reliable equipment and result in modest increases in final waste volumes. The exception is, of course, the CEC process, where (as discussed in the previous section) considerable improvement is desirable. A second possible advancement of partitioning technology would be the integration of partitioning directly into the parent facilities (i.e., the FRP or FFP), thus allowing overall optimization. Based on the CEC cost and our engineering judgment, we believe that the combination of an improved actinide-lanthanide separation process and an optimized, integrated reprocessing-partitioning flowsheet might reduce the cost of partitioning by as much as a factor of 2.

1.2.2 Transmutation

1.2.2.1 Actinide transmutation. The basic scenario for transmutation is that the actinides are recovered from the wastes using partitioning processes, as described previously. They are then fabricated into fuel in a MOX fabrication plant with provisions for partitioning those wastes as well. Following this, they are inserted into the transmutation device and irradiated. Upon discharge, they are allowed to decay for a predetermined period of time and then reprocessed to recover the untransmuted

actinides for reirradiation, while the transmutation products are sent to waste disposal.

At this point, it should be noted that much of the available information on actinide transmutation is quantitatively useless because of flaws in the calculational assumptions or methodology. The two principal problems with these sources are that (1) many of the calculations allow unfissioned actinides to be removed from the system during recycle and thus are not "closed;" and (2) incorrect criteria are used to determine the enrichment of the fuel, which results in substantial errors in the composition and transmutation rate of the recycled actinides. Thus much of the discussion concerning transmutation must be based on a qualitative evaluation of existing studies with considerable interpretation.

The primary transmutation device considered here is a PWR operating in the self-generated, coprocessed plutonium recycle mode. Recycle of the actinides in this reactor is possible by any of several methods: homogeneous dispersal in enriched uranium, dispersal in fuel enriched with plutonium, concentration in target rods in a normal fuel assembly, and concentration in an assembly composed totally of target rods. Based on transmutation rate, fabrication cost, and reactor operational considerations, dispersal in the plutonium-enriched fuel appears to be the preferred mode.

Information derived from the transmutation studies that were conducted as a part of this program, as well as other studies, allows us to make the following statements concerning actinide transmutation in PWRs:

1. Actinide transmutation rates range from 5 to 7% per full-power year (2.5 to 3.5% per calendar year).
2. The enrichment penalty resulting from actinide recycle varies from cycle to cycle but declines from a penalty, initially, to zero at about the fifth recycle. Thereafter, the recycled actinides result in an enrichment decrease (i.e., benefit).

3. Recycle of the actinides results in increases in the amounts of ^{238}Pu and the transplutonium actinides in the fuel, particularly the troublesome $^{242,244}\text{Cm}$ and ^{252}Cf nuclides.
4. Presence of the actinides in the PWR may cause significant power peaking, which in turn may require more extensive enrichment grading within fuel assemblies.

Finally, and most significantly, subject to the acceptability of the actinide recycle fuels (see Sect. 1.2.3.2), the transmutation of the actinides appears to be entirely feasible and the impacts on the reactor do not appear to be unduly large. However, operational changes would probably be required as a result of the radiation and neutronic characteristics of the P-T fuels.

1.2.2.2 Technetium and iodine transmutation. Only brief, survey calculations have been made to determine the feasibility of the transmutation of technetium and iodine. If present as the metal (melting point ~ 2400 K), ^{99}Tc can be transmuted at the rate of about 11% per full-power year. The products of the irradiation are mostly stable isotopes of ruthenium. Thus the transmutation of ^{99}Tc appears to be feasible, subject to the identification of an acceptable fuel form and to the completion of a study of the impact of the small amounts of long-lived ^{98}Tc formed by irradiation of ^{99}Tc .

Assuming that iodine is present as sodium iodide, its transmutation rate would only be about 3% per full-power year, which is relatively low. In addition, most iodine compounds are volatile or unstable at reactor operating temperatures, iodine is very corrosive toward virtually all metals, and the transmutation product is xenon gas, which could cause fuel rod pressurization problems. For these reasons, the transmutation of iodine in power reactors should probably be considered to be marginally feasible at best.

1.2.2.3 Impact of alternative transmutation devices. The use of thermal power reactors other than PWRs has been investigated to some extent, and the results show that, in general, their performance is approximately equivalent to that of the PWR when operating with their own self-generated actinides. Thus consideration of alternative thermal power reactors would not significantly impact the statements in Sect. 1.2.2.1 concerning PWR transmutation or the incentives analysis given in Sect. 1.3.

The use of high-flux thermal reactors as transmutation devices would offer higher transmutation rates. However, these reactors are very expensive to operate since (1) they typically require highly enriched uranium cores, (2) they normally operate at temperatures near the normal boiling point of water and hence do not produce power, and (3) they are generally small due to heat removal considerations. For these reasons, high-flux thermal reactors should not be considered useful for the P-T application.

The use of an LMFBR as the transmutation device is the most widely studied actinide transmutation option. The following statements can be made about actinide transmutation in LMFBRs:

1. The actinide transmutation rate ranges from 5 to 11% per full-power year, depending on the actinide composition and the LMFBR design.
2. Actinide recycle has no significant impact on the fissile requirements, breeding performance, or power peaking in the LMFBR.
3. LMFBRs produce smaller amounts of the principal neutron-active nuclides, particularly ^{252}Cf .
4. Actinide transmutation is feasible in LMFBRs, subject to the acceptability of the P-T fuels.

Based on the available information, it would appear that given a choice between an existing LMFBR and an existing PWR as a transmutation device, the LMFBR has a significant advantage. However, transmutation could be readily implemented using only thermal reactors, subject to the availability of fissile material.

The use of fusion reactors as transmutation devices has also been extensively studied. The results of recent, more sophisticated calculations indicate that, while actinide transmutation is presumably feasible in the fusion reactor blanket, the transmutation rates will probably be about the same as those in an LMFBR due to material radiation damage and heat transfer limitations.

The use of accelerator-driven spallation neutron transmutation devices appears to be an extremely inefficient method for producing neutrons in light of the energy lost in producing electricity from heat and in converting the electricity to high-energy charged particles. Furthermore, these devices are limited by the same materials damage and heat transfer considerations as described for LMFBRs. Thus it is likely that spallation reactors would be at a substantial cost disadvantage with respect to LMFBRs and LWRs with no overriding benefits to compensate for this.

Nuclear explosives have also been proposed as a means for transmutation. However, the large number of explosions required, the expected negative social and political reaction, the irretrievable entombment of the residuals at the explosion site, and treaty restrictions on the use of nuclear explosives that may be involved in the future make this option very unattractive.

The particular device selected for transmutation has little impact on the costs, risks, and benefits of actinide P-T because (1) the effect of the actinides on the transmutation device is almost negligible, and (2) the actinide transmutation rate has only a small effect on the operations accounting for the costs, risks, and benefits (e.g., FRP, FFP, WTFs, repository). Thus the incentives for actinide P-T are nearly independent of the transmutation device considered.

1.2.3 Other impacts of P-T

Partitioning-transmutation has four other impacts that do not fall within either the partitioning or transmutation areas. These involve:

1. the design of casks for shipping fresh and spent fuels,
2. fuel fabrication and the suitability of fuels containing recycled actinides,
3. the disposition of fuel and/or waste produced before the implementation of partitioning, and
4. relationship of this concept to current U.S. nuclear policy.

1.2.3.1 Transportation impacts. The design of a shipping cask for both fresh and spent P-T fuels is markedly different from that for ordinary MOX fuels because of the very high neutron activity of the $^{242,244}\text{Cm}$ and ^{252}Cf present in P-T fuels. A shipping cask that appears to meet all applicable requirements was conceptually designed as a part of the ORNL P-T program. In contrast to ordinary casks, which are typically fabricated of lead, steel, or uranium, the P-T cask would contain major amounts of boron carbide, copper, and lithium hydride; stainless steel would be used for structural integrity. These unconventional materials are necessary to reduce the external dose rate from neutrons to acceptable levels while maintaining the accident resistance required of the casks. Another important aspect of the cask is that its payload is only two-thirds of that for existing spent fuel casks because of weight and size limitations.

1.2.3.2 Fabrication impacts. The presence of the neutron emitters in the P-T fuels also affects the design philosophy of the MOX FFP. In the reference case, it is assumed that the fuel will be fabricated remotely but that the FFP can be sufficiently decontaminated to be contact-maintained. In the P-T case, on the other hand, MOX fuel is considerably more radioactive and increased shielding thickness must be provided in the FFP; in addition, sufficient decontamination to permit contact maintenance does not appear to be possible. Thus the P-T MOX FFP is assumed to be both remotely operated and remotely maintained. This leads to a substantial cost penalty (see Sect. 1.2.4.1) in the fabrication process and additional

feasibility questions since a facility of this type has never been built or operated.

A second possible impact of P-T on fabrication results from the increased concentrations of neptunium, americium, and curium in the fuel. The effect of this increase is to call into question the suitability of the fuels with respect to their irradiation behavior (cracking, swelling, etc.) and their interaction with the cladding. This will probably not be a problem in the recycle modes where these actinides are relatively dilute in either all of the fuel or in the MOX fuel; however, if target rods or assemblies with their higher actinide concentrations are used, significant effects may occur.

1.2.3.3 Impacts on existing inventories. The possible future implementation of P-T raises the question of the disposition of spent fuel and/or waste during the interval prior to implementation. If too much of this material is committed to a repository, the overall effect of P-T will be small because of the large amounts of actinides already in the repository. On the other hand, holding this material on the surface is both more expensive and more risky. The decisive argument here would appear to be that, since the actinides are shown to have a very small impact on the risk from the repository in most cases, the quantity committed to a repository is not important from a radiological risk standpoint.

1.2.3.4 Policy implications. As is probably evident, the implementation of P-T would conflict with current U.S. policy concerning nuclear power. Specifically, the implementation of P-T would require that spent fuel be reprocessed and that all actinides, including plutonium, be recycled. In fact, P-T requires considerably more processing and actinide recycle than a standard uranium-plutonium fuel cycle. (Current policy, of course, states that both reprocessing and the recycle of plutonium have been indefinitely deferred.)

A second policy conflict would occur if LMFBRs were to be selected as the transmutation devices since the use of these reactors has also been deferred. It is important to recognize that implementation of P-T would eventually require the use of LMFBRs as transmutation devices

because of long-term limitations in the supply of fissile materials for thermal reactors.

1.2.4 Analysis of the incentives for P-T

1.2.4.1 Costs, risks, and benefits of P-T. The costs of P-T are taken to be the increase in the cost of nuclear electricity that would result from the implementation of P-T. The risks of P-T are the increases in short-term (contemporary) radiological and nonradiological risks to the public resulting from the additional processing steps required, the increased amounts of toxic actinides in the fuel cycle, and the larger numbers of fresh and spent fuel shipments required. The benefit of P-T is derived from the reduction in long-term risk from the repository that contains actinide-depleted wastes.

Costs. The incremental cost of implementing P-T is estimated to be 1.28 mills/kWhr(e). This cost is attributed to various fuel cycle functions as follows: reprocessing WTF, 0.50 mill/kWhr(e) (39%); MOX fuel fabrication WTF, 0.30 mill/kWhr(e) (23%); P-T MOX fuel fabrication penalty, 0.30 mill/kWhr(e) (23%); transportation, 0.16 mill/kWhr(e) (13%); waste management, 0.01 mill/kWhr(e) (1%); and fissile carrying charges, 0.01 mill/kWhr(e) (1%).

The two WTFs are the principal partitioning facilities in the P-T fuel cycle. The MOX fabrication penalty results from the increased shielding thickness and the remote maintenance requirements. The incremental transportation and waste management costs are attributed to the larger waste volume and the smaller capacity of the cask needed to carry the fresh and spent P-T fuel. The fissile cost arises from the interest charges on the larger amount of plutonium in the P-T cycle.

The 1.28 mills/kWhr(e) cost given above is equivalent to (1) \$340 per kg of heavy metal charged to the reactor, (2) 2.9% of the average 1978 price of delivered residential electricity, and (3) \$9.2 million, in 1979 dollars, per GW(e)-year.

Risks. The increase in short-term risk to the general public resulting from the implementation of P-T is equivalent to 0.57 health-effect/GW(e)-year. This figure is comprised of (1) 0.003 health-effect/GW(e)-year from increases in routine (continuous) radiological releases due to the increased processing and increased concentrations of toxic actinides; and (2) 0.57 health-effect/GW(e)-year due to increases in nonradiological risks, principally petroleum combustion products from the generation of steam and heat at the WTFs and from physical damage during transportation. As a basis of comparison, natural background causes about 1.0 health-effect/GW(e)-year.

Benefits. The benefit of P-T is taken to be the reduction in the expected long-term dose from the geologic repository, which is assumed in this analysis to be located in bedded salt at the site of the Waste Isolation Pilot Plant (WIPP). The measure of the long-term risk used in this study is the number of health effects expected from the repository over 1 million years per unit of electrical capacity represented by the 7000 GW(e)-year of waste in the repository. Risk values of 5.16 and 5.10 have been determined for the reference and P-T cases, respectively; thus the P-T case gives a benefit of 0.06 health-effect/GW(e)-year. The first two values are each about 0.08% of the health effects due to natural background. The benefit, which only amounts to about 0.001% of the effects of natural background radiation, is principally derived from the reduction in the amount of ^{226}Ra precursors in the repository. It is small because the nuclides that control the expected risk from the repository are ^{99}Tc and ^{129}I , which constitute 92% and 8% of the integrated 1-million-year risk respectively. The benefit includes contributions from a slow-leach incident, a volcano growing through the repository, and the impact of a very large meteorite. It should be noted that the latter two events account for the small contribution of the actinides to the total repository risk. This contribution is small because the probability of either the volcano or the meteorite impact is small (about once every 100 billion years). In the more probable leach incident (about once every 7 million years), the actinides are retarded during their migration through the geosphere to such an extent that they do not emerge into the biosphere in significant quantities within a million years.

Summary. Table 1.2 gives a summary of the costs, risks, and benefits of P-T, expressed on a GW(e)-year basis. Additionally, the risk values in terms of health effects have been converted to person-rem by using a conversion factor of 2×10^{-4} health-effect/person-rem.

1.2.4.2 Incentives for actinide P-T. One method used in this study to determine whether there are any incentives for implementing actinide P-T is to calculate the cost of reducing the expected long-term risk by 1 person-rem and compare it to the \$1000/person-rem criterion that has been prescribed for use in determining whether additional effluent control systems on reactor plants are justified.⁹ If the cost of P-T does not meet this criterion, there are presumably alternative investments for the money that would save more lives than P-T and, hence, no justification for implementing P-T.

From Table 1.2, it is evident that the cost/person-rem figure can be generated in more than one way, principally due to the treatment of the short-term risk values. Three values were developed for the purposes of this analysis. The first is the cost of reducing the expected long-term risk irrespective of the short-term risk, which is \$9,200,000/300 person-rem, or \$31,000/person-rem. The second value, which is based on the net, radiological risk reduction, is found by subtracting the short-term risk from the long-term risk, that is, \$9,200,000/(300 - 16) person-rem = \$32,400/person-rem. The final value is based on the overall risk reduction, including the short-term, nonradiological risks, which have been converted to equivalent radiological units for the purposes of this analysis using a conversion factor of 5000 person-rem/health-effect. This value is \$9,200,000/(300 - 2850), or -\$3600/person-rem. The fact that it is negative should be interpreted to mean that the short-term risks of P-T exceed the long-term benefits and that we must pay \$3600 to increase the overall risk by 1 person-rem. Even if a nuclear plant were used to generate the process heat for the partitioning facilities, this value would still be negative because of the transportation accidents.

In summary, all three of the measures fail by a wide margin to meet the \$1000/person-rem criterion that would justify their use; thus there are no apparent incentives for implementing P-T. The justification for

Table 1.2. Summary of the costs, risks, and benefits of P-T per GW(e)-year

	Reference case	P-T case	Incremental
		<u>Costs</u>	
Fuel cycle costs, $\$10^6$	181.9	191.1	9.2
		<u>Short-Term Risk</u>	
Radiological dose, person-rem	4	20	16
Total risk, ^a person-rem	1700	4550	2850
Natural background, person-rem (for comparison)	5000	5000	0
		<u>Long-Term Benefit</u>	
Radiological dose, person-rem ^b	25,800	25,500	300
Natural background, person-rem ^b (for comparison)	33.5×10^6	33.5×10^6	0

^aIncludes nonradiological risks expressed as equivalent radiological impact using a conversion factor of 5000 person-rem/health effect.

^bExpected dose integrated over 1 million years.

rejecting radiological protection options on this basis is supported by the 1972 BEIR report:¹⁰

The public must be protected from radiation but not to the extent that the degree of protection provided results in the substitution of a worse hazard for the radiation avoided. Additionally, there should not be attempted the reduction of small risks even further at the cost of large sums of money that spent otherwise would clearly produce a greater benefit.

1.2.4.3 Sensitivity analysis. Many questions will probably be asked concerning the conclusion that there are no incentives for implementing actinide P-T. A logical one is: How sensitive is this conclusion to the assumptions and uncertainties in the analysis? The uncertainties in the cost analysis and the short-term risk analysis are estimated to be about a factor of 2 in each case. The uncertainty in the cost analysis represents the possible error in cost estimation and possible partitioning improvements. The uncertainty in the short-term risk, which is small relative to most risk analyses, results from uncertainties in the impact models used in the analysis. The uncertainties are smaller here than in most risk studies because the routine effluents, which have a probability of 1.0 by definition, are the major contributors to the risk. By using these maximum uncertainties in a manner most favorable to P-T, one can calculate that the expected long-term benefit of actinide P-T must be about 4600 person-rem to attain the \$1000/person-rem value, about a factor of 15 larger than the calculated value of 300 person-rem/GW(e)-year. The question is now reduced to considering the uncertainty of the long-term risk analysis.

The long-term risk analysis that resulted in the benefit of 300 person-rem/GW(e)-year has a number of large, known conservatisms built into it. Principal among these are: (1) the time horizon is extremely long, that is, 1 million years; (2) the release fractions from the volcano and meteorite, which are a factor of 10 to 15 greater than the generic environmental impact statement (GEIS) value for a meteorite,¹¹ directly affect the long-term benefits; (3) the nuclides that are released are not allowed to be removed from the zones surrounding the repository by wind, river flow, etc.; (4) the population is

assumed to remain in place both during and after the volcano and meteorite events; and (5) a conservative (small) retardation factor is used for ^{237}Np , the actinide most likely to emerge first into the biosphere following a leach incident. All of these factors are conservative in that they favor the implementation of P-T more than the actual values would. This approach was used because it was an expeditious way to examine the incentives for implementing P-T. The values used should not be construed as being appropriate for other studies of the same area or for studies of other areas.

It must be pointed out, however, that there are other principal parameters in which changes might significantly increase the benefits of P-T. First, changes in either the probability or the release fraction from the volcano or meteorite, which are responsible for the calculated benefits of P-T, would directly affect the magnitude of the benefit. As noted above, the release fraction is thought to be conservative by a factor of 10 to 15. The probabilities of these events are approximately the same as those used in the GEIS¹¹ for the meteorite. A second class of parameters that might increase the calculated benefits of P-T is related to the leach incident. The 300 person-rem/GW(e) benefit from actinide P-T does not include any contribution from the leach incident since the actinides are retarded to such an extent that they do not emerge into the biosphere in significant amounts within 1 million years. However, by allowing changes in one or more of the leach parameters, the migration rate of the actinides can be accelerated to the point that they do emerge and contribute to the calculated leach incident consequences. Based on the sensitivity analyses conducted for this program and on the conservative nature of the actinide retardation parameters mentioned above, the only parameter that could accomplish this is an increase in the water migration velocity through the geosphere. A considerable increase in this value is required before any changes in the benefits would be observed, but thereafter the benefits would increase linearly with parameter increases. (That is, there is a threshold below which the leach incident is insignificant.) The water velocity assumed for this analysis is 1.46 m/year. Measurements made in the area of the WIPP site indicate that the velocities

range from essentially zero to a maximum of 4.6 m/year. Increasing the water velocity to 4.6 m/year would be expected to increase the benefits of P-T by a factor of 10 or less. However, a compensating factor is that the probability of a leach incident used in the analysis is conservative (i.e., high) by a factor of about 1 million based on comparison with the GEIS value.¹¹ This means that the leach incident consequences are high by about the same factor. Thus the known conservatisms in the leach incident calculation would compensate for any possible uncertainties and unknown nonconservatisms in the analysis. Increasing or decreasing the leach rate or surface area of the waste form has little effect on the leach incident risk unless the waste form maintains its integrity and the leach rate is less than about 10^{-7} g cm⁻² day⁻¹.

1.2.4.4 Incentives for technetium and iodine P-T. Using the conservative assumptions described previously, the benefits of technetium and iodine P-T are found to be about 100 times that of actinide P-T, or about 30,000 person-rem/GW(e)-year. In addition, the costs and short-term risks of partitioning are probably much smaller since most of these elements occur only in a few specific locations in the fuel cycle (HLLW and dissolver solids for the technetium and the dissolver off-gas for the iodine). Therefore, we would expect the cost-risk/benefit value for technetium and iodine P-T to be less than the \$1000/person-rem criterion if conservative assumptions are used in the long-term risk analysis. This would indicate that, under the very conservative conditions used in the long-term risk analysis, there are incentives for technetium P-T; if iodine transmutation is feasible, there are also incentives for iodine P-T. This conclusion is a direct result of the conservative value used for the leach incident probability (i.e., a factor of 1 million greater than that in the GEIS¹¹). Even if the probability were reduced by only a factor of 100, the incentives for both technetium and iodine P-T would probably be eliminated.

1.2.5 Research, development, and demonstration required to implement actinide P-T

As might be expected, considerable RD&D would be required before actinide P-T could be commercially implemented. Partitioning studies are required in a variety of areas; principal among these are testing the integrated flowsheet and searching for an alternative to the CEC process for separating actinides and lanthanides. Cross-section measurements and actinide transmutation studies are required to identify more precisely the best recycle modes and to determine their impacts. Studies in the fabrication area, including test irradiations, are needed to demonstrate the acceptability of fuels containing higher-than-normal concentrations of neptunium, americium, and curium. A shipping cask must be built of the relatively unusual materials (required to ship the neutron-active P-T fuels) to show that they can be fabricated and that the cask meets applicable safety regulations. Finally, overall studies should be made to establish the best methods for recycling the actinides.

It is estimated that this RD&D would take about 15 years and that the first partitioning facilities should be on-line in 20 years, assuming that the program is well-supported and no licensing delays occur. The limiting aspects in this entire process are the RD&D, design, and construction for the partitioning facilities.

The estimated cost of the RD&D, excluding the costs of the commercial-sized plants and RD&D conducted under the auspices of other programs, is about \$900 million (1979 dollars). The largest portion of this money would be designated for a hot (radioactive) pilot plant to test the integrated partitioning flowsheet.

1.3 Conclusions

1.3.1 Feasibility

The conclusions with respect to P-T feasibility are as follows:

1. The partitioning of actinides appears to be feasible based on the use of currently identified technology, all of which has been experimentally verified at the laboratory level and much of which has been verified at the hot, production-scale level.
2. Although the partitioning of technetium has not been adequately investigated, there is presently no reason to believe that it is not feasible.
3. Iodine partitioning is feasible using existing, demonstrated technology.
4. The transmutation of actinides appears to be feasible in thermal, fast, and fusion reactors, subject to the acceptability of fuels containing higher-than-normal concentrations of neptunium, americium, and curium.
5. The transmutation of technetium appears to be feasible, subject to the identification of an acceptable fuel form.
6. The transmutation of iodine is marginally feasible at best because of low transmutation rates, the volatility of iodine compounds, the production of xenon gas as a transmutation product, and the corrosiveness of iodine and its compounds.
7. The transportation of highly neutron-active P-T fuels appears to be feasible at a reasonable cost.
8. Implementation of P-T, as a whole, is not possible since current U.S. nuclear policy defers both reprocessing and plutonium recycle.

1.3.2 Incentives

The conclusions concerning the incentives for P-T are as follows:

1. The costs of actinide partitioning are relatively high, \$9.2 million/GW(e)-year, because of the variety of wastes that must be partitioned.
2. The short-term (contemporary) risks from P-T are substantial if the nonradiological impacts are taken into account, amounting to 0.57 health-effect/GW(e)-year. The short-term radiological risks are small, amounting to 0.003 health-effect/GW(e)-year.
3. The long-term benefits (i.e., risk reduction) of P-T, using very conservative assumptions, are small, amounting to only 0.06 health-effect/GW(e)-year, or about 0.001% of the effects of natural background radiation.
4. There are no incentives for actinide P-T, even if very conservative assumptions are used in the analysis. The cost of the actinide P-T benefits is \$32,400/person-rem if the nonradiological risks are ignored; if the nonradiological risks are included, the short-term risks *exceed* the long-term benefits integrated over 1 million years.
5. Incentives may exist for technetium P-T if very conservative long-term risk analysis assumptions continue to be used and if partitioning processes can be developed.
6. Incentives may exist for iodine P-T if very conservative long-term risk analysis assumptions continue to be used and a feasible method for transmuting iodine can be identified.
7. Sensitivity analyses indicate that the above conclusions concerning the incentives for P-T are valid for a wide range of input assumptions and parameters.
8. The incentives for P-T are virtually independent of the transmutation device used. Thus the existence of advanced devices would not alter the incentives.

1.3.3 Research, development, and demonstration required

RD&D requirements for the implementation of P-T are as follows:

1. It is estimated that approximately 20 years would be required to bring the first commercial-sized partitioning facilities on-line, assuming a well-supported (but not crash) program and no licensing delays.
2. Approximately 15 years of intensive RD&D would be required at the cost of about \$900 million (1979 dollars). The partitioning RD&D requires the majority of the monies and is the limiting aspect of the schedule.
3. Because of the lack of incentives for actinide P-T, additional RD&D in support of P-T is not warranted unless a decision is made to proceed with P-T.

1.3.4 Other

Some of the partitioning technology identified in this program may find application in the cleanup of wastes and reduction of waste volumes in normal fuel cycles involving plutonium recycle. Primary candidates are the ceric nitrate--nitric acid leaching process and the 2-EHOH process for treatment of solvent cleanup wastes.

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2. PARTITIONING

Partitioning is the enhanced separation and recovery of long-lived radioactive elements from nuclear wastes. The net result of this process is a waste product with a greatly reduced concentration of the long-lived elements; thus, the amounts of these elements that might be released from a repository in the distant future are reduced. The long-lived nuclides of interest are primarily the actinides, because of their relatively high concentrations and toxicity. However, ^{99}Tc (half-life = 213,000 years) and ^{129}I (half-life = 15.9 million years) are also important and will be considered briefly. Other fission products are much smaller contributors to the long-term waste toxicity and therefore are not considered.

Partitioning is generally achieved via the application of chemical and/or physical processes, such as leaching, solvent extraction, and ion exchange, to the waste streams containing the long-lived nuclides. Thus, the first consideration is to describe the wastes in which the long-lived nuclides arise and the facilities generating them. Then, based on the experimental and analytical studies available in the literature and those conducted as a part of the ORNL P-T project,^{1,2} partitioning processes and flowsheets can be identified. The description of the wastes, facilities, and partitioning processes is contained in Sect. 2.1 for a reference fuel cycle which has no provisions for P-T (see Sect. 1.1.3). A similar description of a P-T cycle is contained in Sect. 2.2. These descriptions are for actinide partitioning only and are based on work discussed in detail in refs. 1 and 2.

As noted above, there are two long-lived fission products (technetium and iodine) of interest to partitioning. The technology that might be used to separate and recover them is discussed in Sect. 2.3.

After the partitioning processes have been described in summary fashion, the feasibility, limitations, and incentives or penalties of partitioning are evaluated. The feasibility of partitioning both the actinides

and the fission products is discussed in Sect. 2.4. The incentives or penalties of implementing partitioning are deferred until Sect. 5 so that a complete picture concerning the costs, risks, and benefits of P-T can be presented.

A final aspect of partitioning that must be addressed is the effect of advanced partitioning technology on the conclusions of this report. This subject will be considered briefly in Sect. 2.5.

2.1 Description of Reference Fuel Cycle

2.1.1 Overall description

Both the reference and P-T fuel cycles are depicted schematically in Fig. 2.1. These fuel cycles are based on a single 1250-MW(e) PWR operating in the self-generated plutonium recycle mode. The FRP is assumed to be operating in a coprocessing mode (i.e., the uranium, plutonium, and any other actinides are intimately mixed). The fuel is assumed to remain in the reactor for three calendar years. Out-of-reactor decay times are 1.5 years between reactor discharge and reprocessing and 0.5 year between reprocessing and refabrication.

The reference cycle begins with the insertion of a reload of fuel into the reactor (refer to the left and center portions of Fig. 2.1). The mass of the reload is 34.19 MTHM, which is comprised of two-thirds 3.2 wt % enriched UO_2 and one-third coprocessed MOX (i.e., plutonium-enriched uranium, including some neptunium). After this fuel has been irradiated to a burnup of 33,000 MWd/MTHM, it is discharged and allowed to decay for 1.5 years. Both the UO_2 and MOX fuels are then reprocessed in combination to yield a coprocessed MOX product, which is sent to the refabrication plant after a 0.5-year delay, and TRU wastes, which are transferred to a repository. The MOX product, which is in powder form, is refabricated after a sufficient amount of enriched uranium has been added to achieve the desired end-of-cycle reactivity. TRU wastes from the refabrication of MOX fuels are sent to the repository. Simultaneously with MOX fuel refabrication, the 3.2 wt % enriched UO_2 fuel is fabricated

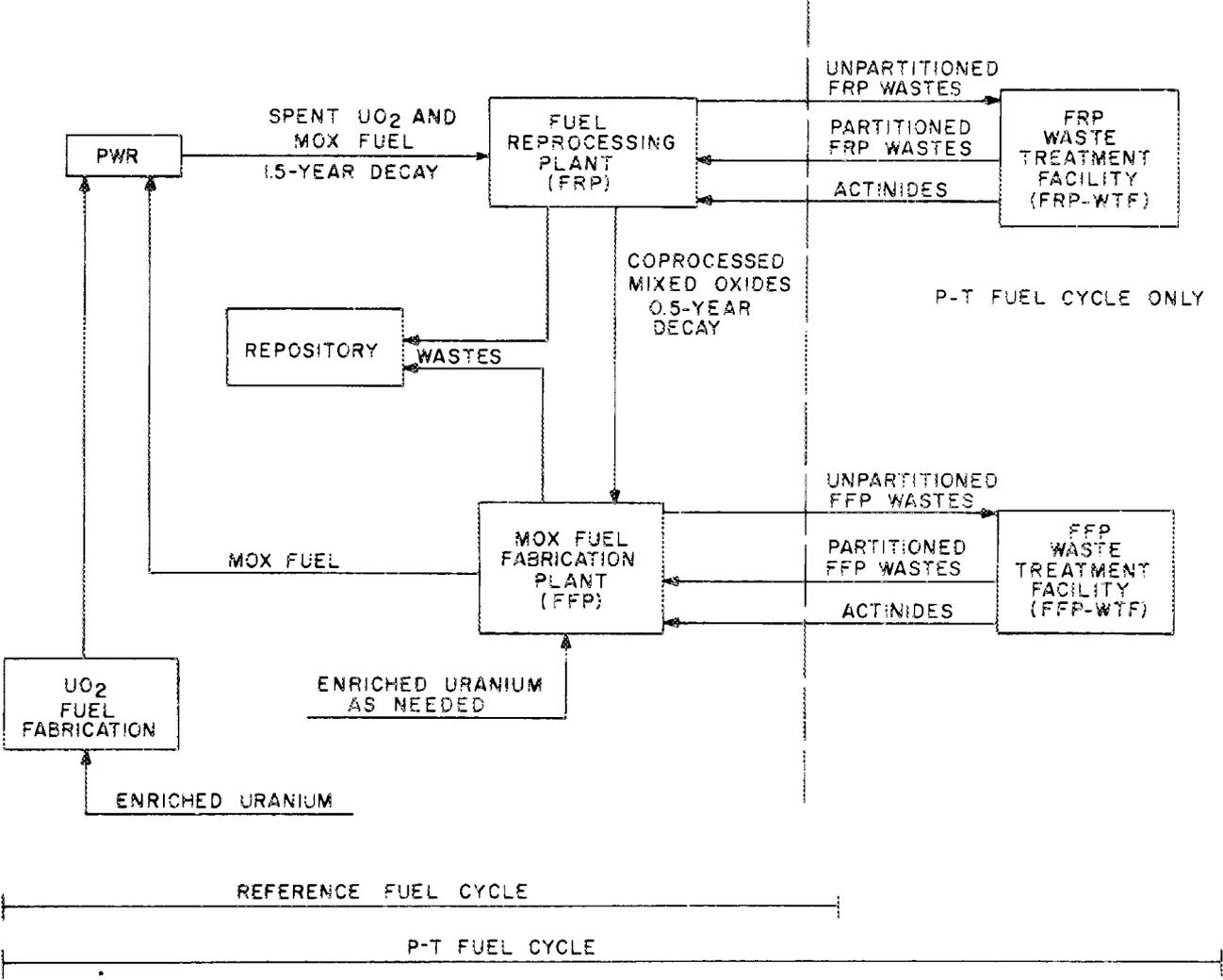


Fig. 2.1. Schematic diagram of reference and P-T fuel cycles.

in a separate facility. The cycle is completed with the insertion of the refabricated fuels into the reactor.

2.1.2. Description of facilities for processing long-lived actinides

Two fuel cycle facilities routinely process long-lived actinides (other than uranium): the FRP, and the MOX FFP. The FRP (assumed nominal capacity, 2000 MTHM per year) uses a Purex flowsheet which has been modified so that the product is an intimate mixture of uranium and plutonium dioxides in power form. This type of product is obtained by separating only part of the uranium initially present with the plutonium and then using a process which simultaneously converts the plutonium-uranium nitrate mixture to a solid oxide powder. Based on the conditions in the coprocessing solvent extraction steps, it was assumed that 80% of the neptunium in the spent fuel would be present in the MOX powder product.

The FRP was assumed to include provisions for reducing the routine releases of noble gases (i.e., ^{85}Kr), carbon (^{14}C), and iodine; tritium was assumed to be released quantitatively to the atmosphere. Otherwise, the general plant design philosophy and technology were assumed to be similar to those used in Allied-Gulf Nuclear Services' reprocessing plant located in Barnwell, South Carolina.

The MOX FFP blends the coprocessed $(\text{Pu,U})\text{O}_2$ powder produced by the FRP with a sufficient amount of natural uranium to achieve the desired final enrichment (about 6.3 wt % plutonium). Standard techniques are then used to fabricate the powder into pellets. The plant is assumed to be operated remotely because of the relatively high radioactivity of the plutonium as compared with enriched uranium. However, it is also assumed that the equipment can be decontaminated sufficiently so that contact maintenance is possible. The capacity of the plant is 660 MTHM per year, which is an appropriate size to handle the output of the FRP (i.e., one-third of the reactor makeup fuel is MOX).

2.1.3 Wastes containing long-lived actinides

Four principal types of waste in the FRP and FFP contain significant

amounts of actinides: the high-level liquid waste (HLLW), cladding waste and dissolver solids, HEPA filters and incinerator ash wastes, and salt wastes. These waste types are grouped in a manner related to their chemical and/or physical similarity, which will become important when the partitioning of these wastes is discussed (Sect. 2.2).

The HLLW is an aqueous, acid waste generated by the first solvent extraction cycle in the FRP. It contains virtually all of the nonvolatile fission products and transplutonium actinides plus part of the uranium, neptunium, and plutonium in the FRP feed material. The HLLW is 3.5 M in nitric acid, has a volume of about 5100 ℓ/MTIHM, and emits large amounts of gamma rays and heat as a result of its high concentration of fission products. Gadolinium is also present in the HLLW since it is used as a soluble neutron poison in the FRP.

The cladding waste, which is generated in the FRP, consists of the Zircaloy cladding plus grid spacers and end pieces that comprise the structure of the fuel assembly. The Zircaloy cladding is typically contaminated with transuranics because some of the fuel is physically trapped in the crevices and pinched ends of the sheared rod segments, or because some of the fuel material is too refractory to dissolve in the nitric acid. (The latter problem is particularly common in MOX fuels.) Dissolver solids, which are also generated in the FRP only, consist of noble metals (i.e., Pd, Rh, Ru, Tc, Mo) that are not soluble in the nitric acid in the fuel dissolver. Thus, a mixture of these metals remains as a residue in the dissolver after dissolution is complete. Significant amounts of actinides are present along with these solids as a result of the carrier effect of the noble metals. The solids, which are recovered via centrifugation, amount to about 1.0 ℓ per metric ton of initial heavy metal (MTIHM).

HEPA filter wastes are produced by both the FRP and the FFP. They result from the contamination of the filters with particulate TRU elements, primarily from the actinide nitrate-to-oxide conversion in the FRP and the pellet grinding activities in the MOX FFP. The HEPA filters, which are produced at the rate of about 400 ℓ/MTIHM, are not combustible

(except possibly for the frames). Incinerator ash wastes are produced at the rate of about 70 ℓ /MTIHM in both the FRP and the FFP via the combustion of wastes such as spent solvent, paper trash, plastics, and rubber.

Salt wastes are produced at a combined rate of about 26,000 ℓ /MTIHM in both the FRP and the FFP. They consist of a mixture of a wide variety of alkaline and acidic salt wastes arising from various sources, including solvent cleanup scrubs, incinerator off-gas scrubs, analytical laboratory wastes, decontamination solutions, and laundry wastes. These wastes generally contain a diverse assortment of sodium salts, such as the carbonate, bicarbonate, nitrate, sulfate, and chloride. The solvent cleanup scrubs are the most actinide-contaminated of these wastes.

2.1.4 Waste products and characteristics

The function of the previous section was to determine the wastes that would be treated, identify their sources, and discuss some of their characteristics. This section describes the final wastes that are produced by the FRP and FFP. It should be noted that there is not a one-to-one correspondence between the wastes to be treated (see Sect. 2.1.3) and the final waste products described here.

The reference fuel cycle produces four different waste products containing TRU actinides: high-level solidified waste, cladding waste, non-immobilized waste, and concreted waste. The first two of these are produced only at the FRP, whereas the latter two are produced at both the FRP and the FFP.

The combined volumes of these reference cycle wastes are summarized in Table 2.1. The amounts of actinides assumed to report to these wastes, which are based essentially on the use of existing technology, are summarized in Table 2.2. These tables also contain comparable values for the P-T cycle.

Table 2.1. Volumes of final actinide wastes

Type of waste	Waste volume (ℓ/MTIHM)	
	Reference cycle	P-T cycle
High-level solidified waste	63	63
Cladding waste	212	212
Nonimmobilized waste	977	977
Concreted waste	827	1200

Table 2.2. Percentage of actinides reporting to final actinide wastes

Waste product	Element	Fuel reprocessing plant ^a		Fuel fabrication plant ^b	
		Reference cycle	P-T cycle	Reference cycle	P-T cycle
High-level solidified waste	U	0.5	0.01		
	Np	20.0	0.01		
	Pu	0.5	0.01		
	Am	99.9	0.01		
	Cm ^c	99.9	0.01		
Cladding waste	U	0.1	0.01		
	Np	0.1	0.01		
	Pu	0.1	0.01		
	Am	0.1	0.01		
	Cm ^c	0.1	0.01		
Nonimmobilized waste	U	0.4	0.02	0.5	0.05
	Np	0.4	0.02	0.5	0.05
	Pu	0.4	0.02	0.5	0.05
	Am _c	0.0	0.02	0.5	0.05
	Cm ^c	0.0	0.02	0.5	0.05
Concreted waste	U	1.0	0.06	1.5	0.1
	Np	1.0	0.06	4.0	0.1
	Pu	1.0	0.06	1.5	0.1
	Am _c	0.0	0.06	20.0	0.1
	Cm	0.0	0.06	20.0	0.1
Total	U	2.0	0.1	2.0	0.15
	Np	21.4	0.1	4.5	0.15
	Pu	2.0	0.1	2.0	0.15
	Am _c	100	0.1	20.5	0.15
	Cm ^c	100	0.1	20.5	0.15

^aBased on feed to FRP.

^bBased on feed to FFP.

^cValues for Bk and Cf are assumed to be the same as those for Cm.

2.2 Description of P-T Fuel Cycle

The general outline of the P-T cycle, which is depicted schematically in Fig. 2.1, is the same as that for the reference cycle; that is, it employs a PWR operating on self-generated plutonium recycle, an FRP operating in the coprocessing mode, and decay times of 1.5 and 0.5 years before reprocessing and MOX fabrication, respectively. However, the reference and P-T fuel cycles exhibit substantial differences that will be identified in the subsections that follow.

2.2.1 Overview

The P-T fuel cycle is basically the same as the reference fuel cycle except that partitioning facilities (see the rightmost portion of Fig. 2.1) have been added. As before, the cycle begins with the insertion of a fuel reload (mass, 34.19 MTHM) into the reactor. This reload is comprised of two-thirds 3.2 wt % enriched UO_2 and one-third coprocessed MOX containing all of the recovered waste actinides (neptunium and the transplutonics) homogeneously dispersed throughout. After the reload has been irradiated to a burnup of 33,000 MWd/MTHM, it is discharged and allowed to decay for 1.5 years. The UO_2 and MOX fuels are then reprocessed in combination. The TRU wastes are sent to the FRP-WTF for partitioning. Streams of recovered actinides and actinide-depleted wastes are returned to the FRP. The recovered actinides are combined with the coprocessed MOX and routed to the FFP after a 0.5-year delay. The actinide-depleted wastes are transferred to the repository. The MOX product (including the waste actinides), which is in powder form, is refabricated into fuel after the addition of a sufficient amount of enriched uranium to achieve the desired end-of-cycle reactivity. The TRU wastes from refabrication are then sent to the FFP-WTF for partitioning. A stream of recovered actinides and a stream of actinide-depleted wastes are returned to the fabrication plant. The recovered actinides are incorporated into MOX recycle streams within the facility. The actinide-depleted wastes are sent to the repository. Simultaneously, the 3.2 wt % enriched UO_2 fuel is being fabricated in a separate facility. The cycle is completed when the

refabricated fuels are inserted into the reactor. The details of the WTF process and plant design are given in refs. 1 and 2.

2.2.2 Description of facilities for processing actinides

As is evident from the preceding description, four facilities in the P-T cycle process the actinides: the FRP, the FFP, the FRP-WTF, and the FFP-WTF. The design and operation of the FRP and the FFP are essentially as described in Sect. 2.1.2 and will not be repeated here.

The purpose of the two WTFs in the P-T cycle is to recover actinides that would ordinarily report to the actinide wastes. This is accomplished by (1) routing the TRU wastes to either the FRP-WTF or the FFP-WTF, (2) using various chemical processes to separate and recover the actinides from the wastes, and (3) returning the recovered actinides and the actinide-depleted wastes to the parent facility. In the parent facility, the recovered actinides (as a liquid nitrate solution) are combined with the main actinide nitrate solution (in the FRP) or with an internal nitrate recycle stream (in the FFP) and thus are included in the final, solid product which is oxide powder for the FRP and fabricated fuel for the FFP. The actinide-depleted wastes are treated and/or packaged in the parent facility to provide the appropriate form for final disposal.

2.2.3 Wastes containing actinides

The wastes containing the actinides in the P-T case being considered here are the same as those in the reference case (viz., HLLW, cladding waste and dissolver solids, HEPA filters and incinerator ash, and salt wastes). However, in the P-T cycle, the wastes described in Sect. 2.1.3 are subjected to partitioning to decrease their actinide contents before being treated for disposal as described in Sect. 2.1.4. Sections 2.2.3.1 - 2.2.3.4 describe the treatment of each of these four waste types.

The partitioning of the wastes involves two generic steps: separation of the actinides from the waste by leaching, solvent extraction, or breaking down strong organic-actinide chemical compounds; and recovery

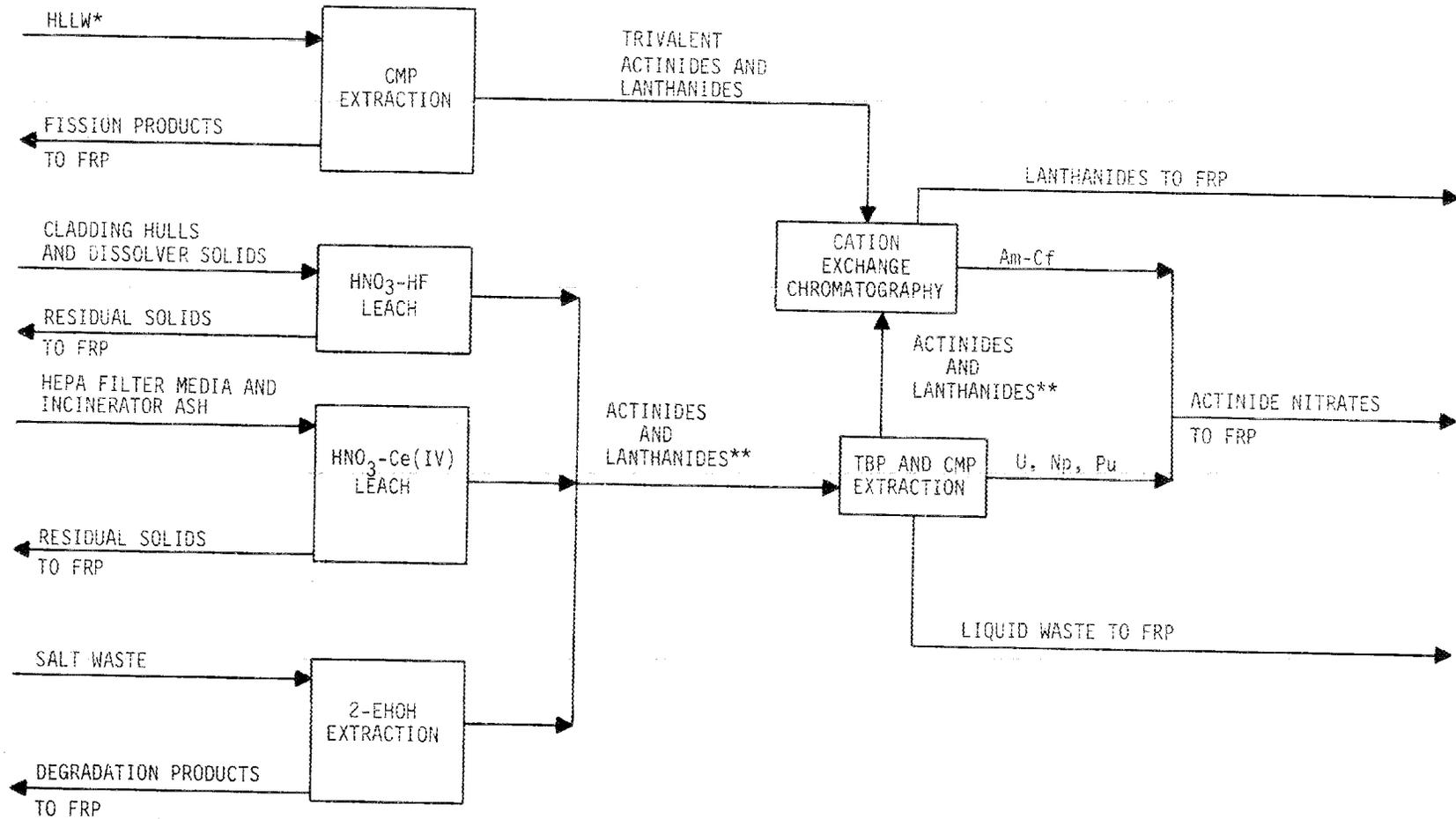
of the actinides. The second step is accomplished using a standardized processing sequence for all of the waste types. This sequence is as follows:

1. The tetravalent and hexavalent actinides (uranium and plutonium) are recovered via Purex (TBP) extraction. The recovered actinides are then stripped from the solvent and returned to the parent facility.
2. The trivalent actinides (primarily americium and curium) and lanthanides are coextracted using a bidentate extractant (i.e., CMP). The actinide-depleted wastes are returned to the parent facility for final treatment before disposal. The trivalent actinides and lanthanides are stripped from the solvent and subjected to CEC.
3. The actinides are separated from the lanthanides using CEC (ion exchange). The lanthanide fraction is returned to the parent facility for final treatment before disposal. The recovered actinides are returned to the parent facility and incorporated with the main stream of aqueous actinide nitrates.

This sequence is depicted schematically in Figs. 2.2 and 2.3 for the FRP-WTF and the FFP-WTF, respectively.

2.2.3.1 High-level liquid waste. The first partitioning step for the HLLW, Purex extraction, is accomplished in the FRP. As a result of this, no further TBP extraction is required, and the HLLW is sent directly to CMP extraction and CEC for recovery of the trivalent actinides. After CMP extraction, the actinide-depleted HLLW is returned to the FRP for final treatment.

2.2.3.2 Cladding waste and dissolver solids. The cladding waste and dissolver solids have previously been leached with nitric acid during the dissolution process in the FRP. Thus stronger measures are required to solubilize the remaining actinides. The approach used in the FRP-WTF is to contact these solid wastes with a mixture of nitric and hydrofluoric acids. (The hydrofluoric acid catalyzes the dissolution of the remaining insolubles.) Hydrofluoric acid is not generally favored for use in the



*TBP EXTRACTION GENERATING HLLW IS IN THE FRP
**PREDOMINANTLY STABLE CERIUM AND GADOLINIUM

Fig. 2.2. Schematic actinide partitioning flowsheet for the waste treatment facility associated with the fuel reprocessing plant (FRP-WTF).

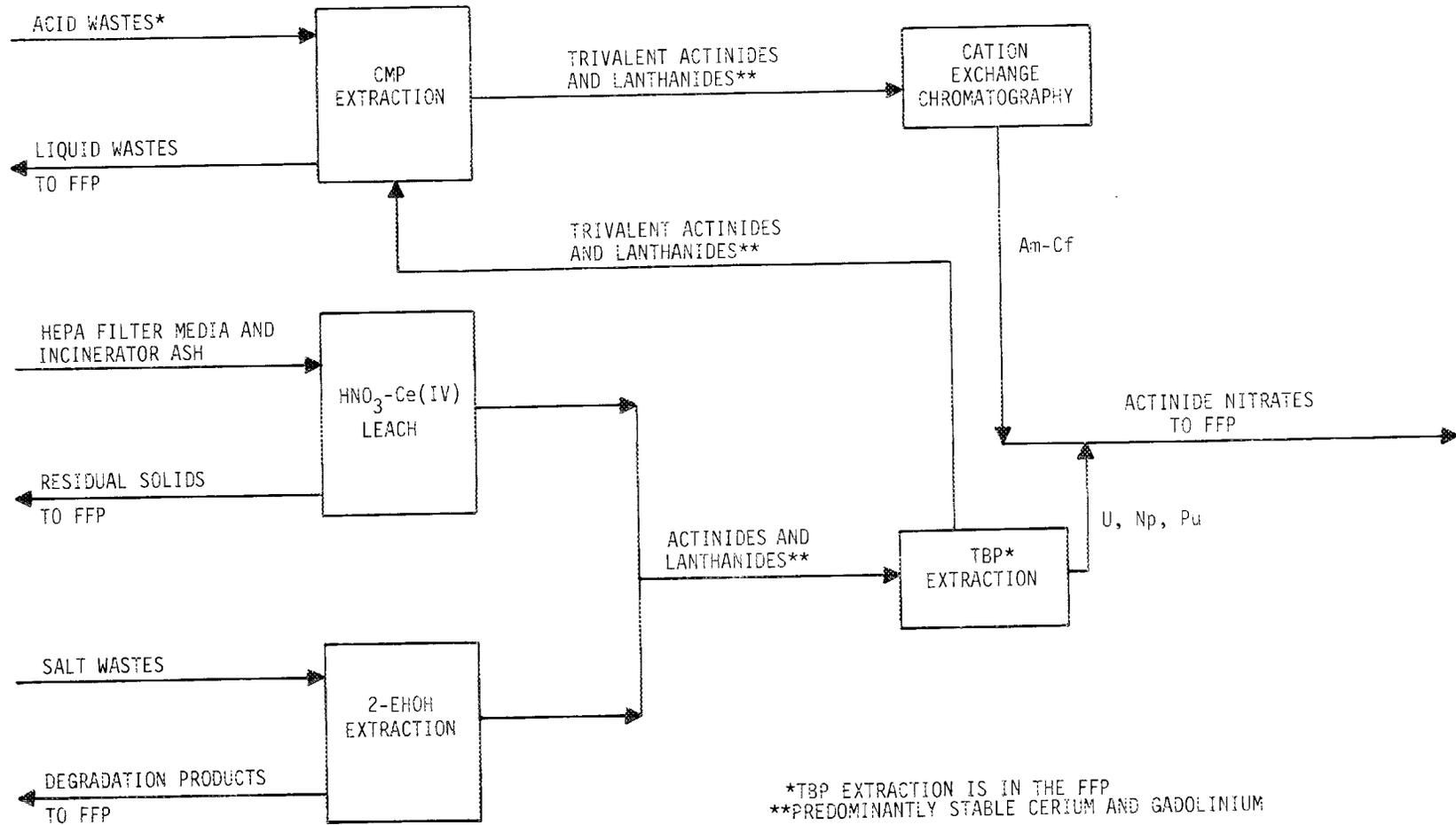


Fig. 2.3. Schematic actinide partitioning flowsheet for the waste treatment facility associated with the MOX fuel fabrication plant (FFP-WTF).

parent facility because it complicates the calcination and vitrification of the HLLW and its corrosiveness requires special materials of construction, which are much more expensive than those normally used. After being leached, the actinide-depleted wastes are returned to the parent facility for final treatment; the nitric-hydrofluoric acid solution containing the actinides is sent to the TBP extraction, CMP extraction, and CEC processes for actinide recovery. It should be noted that, in the case of the FFP (Fig. 2.3), the TBP extraction is contained in the parent facility. After TBP extraction in the FFP, the stream containing the trivalent actinides is returned to the FFP-WTF for CMP extraction and CEC.

2.2.3.3 HEPA filters and incinerator ashes. The problems encountered with partitioning the HEPA filters and incinerator ashes are very similar to those for the cladding waste and dissolver solids. The major difficulty involves the insolubility of the actinides in nitric acid because of their refractory nature. However, in this instance, an additional complication is present -- that is, both the HEPA filters and the incinerator ashes contain substantial amounts of silica (SiO_2). If the nitric-hydrofluoric acid mixture described in Sect. 2.2.3.2 were to be used in this case, the result would be a soft, goeey mass because of the action of the hydrofluoric acid on the silica. The product would be virtually impossible to handle in a radioactive system. The solution to this problem was to select an alternative chemical, Ce(IV) in the form of ceric nitrate [$\text{Ce}(\text{NO}_3)_4$], for solubilizing these wastes. Ceric nitrate acts on the insoluble actinides by oxidizing them to more soluble oxidation states while being reduced to Ce(III). The Ce(III) is subsequently reoxidized to Ce(IV) electrolytically. This technique allows recycle of the cerium within the WTFs. The actinide-depleted wastes are returned to the parent facility for final treatment and the nitric acid--cerous nitrate solution containing the solubilized actinides is sent to the TBP extraction--CMP extraction--CEC recovery sequence. As previously, the TBP extraction process in the fabrication facility is contained in the FFP, but not in the FFP-WTF.

2.2.3.4 Salt wastes. The salt wastes present a somewhat different type of problem than either the HLLW or the solid wastes. In this case,

the actinides are already in solution but are combined with organic phosphates that effectively prevent them from being stripped from extractants such as TBP and CMP. The most common organic phosphates causing this stripping problem are dibutyl phosphate (DBP) and monobutyl phosphate (MBP), both of which are degradation products of the TBP used in the Purex process. These nonstrippable species are destroyed, and the actinides are made extractable by contacting the acidified salt wastes with 2-EHOH, which selectively extracts the degradation products away from the actinides. The resulting solution can then be sent to the TBP extraction, CMP extraction, and CEC actinide recovery processes. After TBP extraction, the actinide-depleted salt wastes are returned to the parent facility for final treatment and disposal. As with the solid wastes discussed above, the TBP extraction process is carried out in the FFP, and not the FFP-WTF.

2.2.4 Waste products and characteristics

The waste products from the P-T cycle are the same as those from the reference cycle (viz., high-level solid waste, cladding waste, nonimmobilized waste, and concreted waste; see Sect. 2.1.4). The types of waste included in each of these products are also the same as those in the reference cycle.

2.2.4.1 Waste volumes. The volumes of the final waste products from the P-T cycle are summarized in Table 2.1, along with the volumes cited previously for the reference cycle. As is evident from the data presented here, P-T has no effect on the final volume of high-level solid waste and cladding waste. This is to be expected because the solvent extraction of the HLLW does not add any significant amounts of inert salts to the stream that would increase the volume; on the other hand, the density of the vitrified product is controlled by the fission product heating, which is also not affected. The volume of the cladding waste remains unchanged since the surface effects of the nitric acid--hydrofluoric acid leach have virtually no influence on the volume or mass of the cladding.

As shown in Table 2.1, the volume of concreted waste produced in the P-T cycle is estimated to increase by 50% over that for the reference cycle. This increase reflects the wastes produced by the CEC process

(zinc barrier ions) and the additional solvent cleanups (TBP, CMP, and 2-EHOH extractions).

Considerable uncertainty is involved in the estimation of waste quantities unless the latter are based on actual operating experience (which is not available in the case at hand). Therefore, the values given in Table 2.1 should be viewed as rough estimates to be used only for comparison purposes.

2.2.4.2 Actinides reporting to wastes. The amounts of actinides reporting to the four actinide-contaminated waste products from the P-T cycle are summarized in Table 2.2. As is evident, P-T substantially reduces the amounts of unrecovered actinides. Overall, only 0.25% of all of the actinides in the spent fuel end up in the combined FRP and FFP wastes; the remainder are recovered for recycle. Compared to standard practice in the past, the amounts of americium and curium in the wastes have been reduced by a factor of 400, the amount of neptunium by a factor of 100, and the amounts of plutonium and uranium by a factor of 16. These factors are, of course, based on conceptual studies and limited experimental studies, and thus are only rough estimates. The limitations of these values are discussed in more detail in Sect. 2.4.

It is interesting to note that the HLW is not the principal source of unrecovered uranium and plutonium in either the reference cycle or the P-T cycle. The HLW contains about 25% of the unrecovered actinides in the reference cycle, and only about 4% in the P-T cycle. This serves to re-emphasize the fact that the non-high-level wastes are extremely important when considering the partitioning of actinides as a means of reducing the overall waste toxicity.

2.3 Partitioning of Iodine and Technetium

This section is intended to provide background with regard to the partitioning of the two predominant long-lived fission products: ^{129}I and ^{99}Tc . Since no studies of partitioning processes were conducted for these nuclides under the ORNL P-T project, this discussion is based on previously existing technology.

The partitioning of iodine should be relatively straightforward since virtually all of the iodine occurs in a single FRP stream, the dissolver off-gas. Technology for iodine partitioning is also well known since a variety of effluent control processes have been developed and demonstrated to remove iodine from gas streams. These processes include caustic scrubbing, mercuric nitrate scrubbing, silver zeolite adsorption, charcoal adsorption, and Iodox (scrubbing with extremely concentrated nitric acid). It would be impossible to determine which process is most effective until the desired chemical form of the iodine product was specified and a comparative engineering evaluation of the processes was performed in this context.

The partitioning of technetium is more complicated than that of iodine since technetium occurs in both the HLLW and the dissolver solids. To date, little attention has been given to flowsheets for recovering technetium. However, there is presently no known limitation that would preclude the recovery of technetium.

2.4 Feasibility, Limitations, and Other Applications of Partitioning Processes

2.4.1 Actinide partitioning

2.4.1.1 Feasibility. Six identifiable processes are used in the partitioning plants described in Sect. 2.2: Purex extraction, CMP extraction, CEC, 2-EHOH extraction, nitric acid--hydrofluoric acid leaching, and nitric acid--cerium nitrate leaching. Based on the engineering evaluations and experimental studies conducted as a part of this project, we believe that all of these processes are feasible and in turn that partitioning, as a whole, is feasible. Both TBP extraction and nitric acid--hydrofluoric acid leaching are well demonstrated technologies as a result of nuclear defense activities over the past two or three decades. CEC has also been demonstrated on a plant scale, although less extensively than the Purex process. The CMP extraction process has been tested under radioactive conditions on small samples of discharged LWR fuel and on Idaho Chemical Processing Plant waste. Finally, both the

2-EHOH extraction and the nitric acid--ceric nitrate leach processes have been investigated on a laboratory scale.¹

The principal caveat concerning the partitioning flowsheets developed in these studies is that they are very highly integrated (i.e., there is considerable internal recycle of plant streams) to prevent actinides reporting to the wastes. Therefore, it is possible that one or more chemical species having a detrimental effect on plant performance could form and could continue to build up within the plant because of the recycle, ultimately causing either unacceptably high amounts of actinides to report to waste streams or forcing plant shutdown because of process inoperability. Since the integrated flowsheets were not tested in the ORNL P-T project, the likelihood of this occurrence is presently unknown.

2.4.1.2 Limitations. Despite the feasibility of the six partitioning processes listed above, some of them have serious limitations. The process with the most constraints is the CEC process, which is used to separate the trivalent actinides (americium and curium) from the trivalent lanthanides. The materials being handled in this process, specifically $^{242,244}\text{Cm}$, are relatively short-lived alpha emitters that cause severe damage to the ion exchange resin. This damage, which is enhanced by the relatively high concentrations of curium in the recycle scenario, makes it necessary to discard the resin after a single processing sequence. This results in the requirement for large numbers of ion exchange columns and large quantities of ion exchange resin, as well as the associated expense and difficulty of frequent resin replacement. Further, radiation-damaged resins are susceptible to explosion if they are overheated and/or overexposed. However, CEC appears to be the best process currently available for performing the very difficult separation between chemically similar elements.

A second potential limitation is the effect of the extensive internal recycle of plant streams on plant feasibility and operability (see the previous section). Although this effect has not actually been established as a limitation, it would have to be carefully investigated before a demonstration plant could be operated.

A third potential limitation is the availability of sufficient amounts of pure CMP extractant for a large nuclear economy. During the course of this project, much of the effort in investigating the CMP extraction process was directed toward development of methods for removing detrimental impurities from the commercially available product. While this was finally accomplished by using liquid chromatography, the product was still only 80 to 85% pure. Consequently, there is some question as to whether this extractant can be procured in relatively large quantities and in relatively pure form.

A final limitation involves the performance of the partitioning processes, particularly those which have not been operated on a large scale. For the purpose of this project, engineering analyses and experimental studies were performed to determine the degree of actinide decontamination that could be achieved with the partitioning processes described previously. The "best estimates" of the actinides reporting to wastes are reflected in Table 2.2. However, these estimates are bounded by relatively broad uncertainties ranging from essentially total actinide recovery to only minor improvements over existing technology in some cases. Since the benefits to be achieved from P-T are a direct function of the degree to which the unrecovered actinides can be reduced, the actual process performance is important, particularly if it is lower than expected (i.e., the amounts of actinides reporting to wastes are larger than expected).

2.4.2 Technetium and iodine partitioning

Based on the brief discussion given in Sect. 2.3, the only defensible conclusion concerning the feasibility of partitioning technetium is that it is an unknown quantity since it has not been investigated. Processes for recovering technetium have not been studied or identified, although there is no known reason why such processes could not be developed.

The preceding, brief discussion concerning iodine partitioning, the well known nature of iodine chemistry, and the wealth of experience in recovering iodine from gas streams lead us to believe that the partitioning of iodine would be feasible using existing technology.

2.4.3 Other applications of actinide partitioning technology

Even if partitioning were never implemented, the technology developed in support of it could prove to be useful in an ordinary fuel cycle that includes reprocessing. The two processes that have the most potential are the 2-EHOH extraction and the nitric acid--ceric nitrate leaching.

Extraction with 2-EHOH may represent a simple, workable process to drastically reduce the amounts of valuable fissile and fertile materials reporting to the solvent cleanup waste. As is evident from Table 2.2, the concreted waste, which is largely comprised of salt wastes, contains about 50% of the unrecovered uranium and plutonium in the reference fuel cycle. The value of the plutonium thus recovered (based on the cost of equivalent UO_2 fuel minus a fabrication penalty) would be about \$3 million per year for an FRP handling spent UO_2 fuel and as much as \$15 million per year for a facility handling MOX fuels.

The nitric acid--ceric nitrate leaching process holds promise as a method for solubilizing refractory actinide oxides from a wide variety of solid wastes such as dissolver solids, incinerator ashes, and HEPA filters. This might result in recovered plutonium values roughly half of those given for the 2-EHOH process above.

2.5 Effects of Alternative Partitioning Technology

Taken in their entirety, and assuming 100% feasibility, the processes discussed in this section probably constitute a near-optimal partitioning technology for the following reasons:

1. The extractants used (subject to the capability for producing enough pure CMP) perform satisfactorily, are radiation resistant, are relatively insoluble, and do not require the addition of nonvolatile chemicals. Since a liquid system is preferable for achieving high reliability, it is difficult to see how these extraction processes could be improved to any significant extent.
2. The nitric acid--ceric nitrate leaching process represents a relatively low-cost method for solubilizing actinides in wastes

that were previously not amenable to treatment without the addition of large quantities of nonrecyclable reagents.

3. The wide variety of wastes that must be partitioned and their diverse characteristics (e.g., solid and liquid, acid and alkaline) necessarily requires that multiple processes be used.
4. The principal cost of partitioning is not the processes or process equipment, but the process building and the utilities supporting the processes. Since the development of alternative processes in most instances will require equivalent space and utility support, costs would remain substantially unchanged.

An exception to these statements must be made in the case of the CEC process for separating the actinides from the trivalent lanthanides. This process, which is very bulky and unwieldy, is unfortunately the most effective method for achieving the required separation. However, it alone accounts for about 25% of the cost of the WTFs. Thus, if partitioning were implemented, the overall partitioning scheme would probably benefit substantially from the development of an alternative, liquid-phase process.

The principal point to be emphasized in this section is that the development of alternative partitioning processes, with the exception of CEC, is not expected to substantially change the costs or difficulties inherent in partitioning actinides from a variety of fuel cycle wastes. The combination of an advanced process to replace CEC and totally optimized process flowsheets might optimistically result in a 50% reduction in the size and costs of the WTFs while improving operability to some extent; however, a benefit of this magnitude does not appear likely and should be regarded as an upper bound on the cost reduction.

2.6 References for Section 2

1. D. W. Tedder, B. C. Finney, and J. O. Blomeke, Actinide Partitioning-Transmutation Program Final Report. II. Partitioning Processes for LWR Fuel Reprocessing and Refabrication Plant Wastes, ORNL/TM-6982 (June 1980).
2. A. E. Smith and D. F. Davis (R. M. Parsons Co.), Actinide Partitioning-Transmutation Program Final Report. V. Preconceptual Designs and Costs of Partitioning Facilities and Shipping Casks (Summary Report), ORNL/TM-6985 (January 1980).

3. TRANSMUTATION

Transmutation can be defined as the conversion of partitioned long-lived nuclides to short-lived or stable nuclides by irradiation with neutrons. That is, the nuclides are recovered, fabricated, irradiated, and reprocessed — a cycle which is repeated indefinitely. Each time through, some of the long-lived species are destroyed. Transmutation complements partitioning in that it provides the mechanism for eliminating the long-lived nuclides recovered from the wastes by the partitioning processes. If P-T is persistently pursued, the result will be to eliminate essentially all of the potentially troublesome nuclides from radioactive wastes via conversion to nuclides that will decay to innocuous levels within 1 thousand years.

As in partitioning, the principal candidates for transmutation are the transuranic actinides, ^{99}Tc and ^{129}I . Virtually all the other intermediate-lived and long-lived nuclides are not amenable to transmutation because of their small capture cross sections and/or because isotopic separation would be required before transmutation. Examples of nuclides in this category are ^{14}C , ^{90}Sr , ^{137}Cs , and ^{85}Kr .

Transmutation is generally accomplished by (1) separating and recovering (partitioning) the nuclides of interest, (2) fabricating them into an appropriate fuel form, (3) irradiating them in some type of neutron-producing device, and (4) reprocessing the irradiated fuel to recover the untransmuted nuclides. The (short-lived) transmutation products are sent to waste disposal. Thus, as long as the long-lived nuclides are being generated in the fuel cycle, the recycle of these nuclides for transmutation purposes must continue.

This section constitutes a summary description of the results of the transmutation studies that have been conducted to date, both by ORNL (under the auspices of the P-T project) and by independent investigators. This discussion will be limited to a consideration of the overall trends and effects observed in transmuted long-lived nuclides; relatively little emphasis will be given to specific compositions. The readers who are

interested in more details are directed to ref. 1. A compilation of the more recent transmutation studies is also contained in ref. 1; a compilation of older studies is given in ref. 2.

In the discussion given here, only existing and projected commercial nuclear power reactors (both fission and fusion) are considered in detail as the source of transmutation neutrons. The reasons for this restriction are that (1) the relatively large amounts of material that would be irradiated in any realistic P-T scenario could only be accommodated by the large number of existing and projected power reactors, (2) special transmutation devices are generally expensive since their total cost must be charged to transmutation and not to power production, and (3) nuclear power reactors are relatively well characterized. However, special transmutation reactors, spallation devices, and nuclear explosives will be discussed briefly.

A brief analysis of the transmutation literature is given in Sect. 3.1. Section 3.2 discusses actinide transmutation in LWRs, the existing type of power reactor; the transmutation of ^{99}Tc and ^{129}I in LWRs is considered separately in Sect. 3.3. Transmutation of actinides in other devices is discussed in Sect. 3.4. Section 3.5 summarizes the feasibility and cost impacts of transmuting the actinides, iodine, and technetium. The discussions in Sects. 3.1 - 3.5 are presented on a generic basis without reference to a specific transmutation scenario. Section 3.6 characterizes the fuel cycle materials (fresh and spent fuel and wastes) for both a reference cycle and a P-T cycle so that the merits of P-T can be further evaluated.

3.1 Analysis of the Transmutation Literature

In contrast to partitioning, where virtually all of the U.S. contributions have been made under the auspices of the ORNL P-T project, studies of transmutation have been performed by a wide variety of investigators.^{1,2} These studies have concentrated on actinide transmutation, although several have briefly investigated the use of high-flux devices for transmuting intermediate-lived fission products such as ^{90}Sr and ^{137}Cs .

Within the area of actinide transmutation, the most heavily studied transmutation devices are LMFBRs, followed successively by LWRs and other neutron-producing devices.

In an overall sense, many of the transmutation studies documented in the open literature are uncoordinated and simply repeat work already performed by others. Furthermore, virtually all of them suffer from a single major deficiency in that the transmutation fuel cycle under consideration is not "closed" with respect to the actinides being transmuted. For example, in many studies the transmutation efficiency is measured by determining the difference between the initial and the final masses of waste actinides (i.e., neptunium, americium, and curium). However, this approach is not valid since some of the waste actinide nuclides produce either uranium or plutonium via neutron capture or decay. In such cases, the waste actinides are not fissioned but yet are not counted in the residual waste actinide mass.

The fuel cycle is also not closed in instances where the plutonium is stored or treated as a waste and is not recycled for transmutation. Reporting and analysis of transmutation results on this basis yield spurious transmutation rates that are not valid for comparison with the results of other studies. These deficiencies generally make it necessary to describe and evaluate transmutation only in combination with experienced judgment and to report the results qualitatively or in terms of trends.

3.2 Actinide Transmutation in LWRs

LWRs will form the basis of the nuclear power reactor system in the United States for the immediate future. Studies of transmutation in LWRs have been limited in the past because (1) the necessity of accounting for cross-section and neutron spectrum changes made the calculations onerous, and (2) most investigators intuitively concluded that the hard (high-energy) neutron spectrum in the FBRs would yield higher transmutation rates and thus concentrated on them. However, in light of the substantial number of LWRs currently in existence and those expected to be built in the future, it is desirable that actinide transmutation in these systems be characterized.

Unfortunately, along with the absence of a closed system with respect to the actinides (see Sect. 3.1), virtually all LWR transmutation studies suffer from another deficiency that has been identified only recently. That is, an incorrect criterion was used as the basis for establishing the fuel enrichment in fuels containing recycled actinides. Most of the studies used the beginning-of-irradiation (BOI) fuel conditions to establish the enrichment, whereas the end-of-irradiation (EOI) conditions are more correct, although much more tedious and expensive to calculate. The differences can be substantial. For example, if the BOI conditions are used, about 4% ^{235}U is required in the fifth actinide recycle in addition to the plutonium being recycled. However, if the EOI conditions are used, the required ^{235}U enrichment is only 0.7%. This difference substantially affects the neutron flux level (i.e., the transmutation rate) and the recycled actinide composition through the production rate of ^{237}Np . However, as noted previously, accurate qualitative information concerning actinide transmutation in thermal reactors can be obtained from these results by exercising caution and applying experienced judgment.

3.2.1 Actinide recycle modes

Four principal methods are considered for recycling actinides in LWRs:

1. dispersing the actinides homogeneously throughout the entire fuel reload,
2. dispersing the actinides homogeneously in only the MOX fuel,
3. concentrating the recycled waste actinides in target rods within an otherwise ordinary fuel assembly, and
4. concentrating the recycled waste actinides in target rods that are subsequently used to make up a target assembly.

In the first two methods, the actinides include all of the plutonium generated in the reactor. In the last two methods, plutonium is excluded from the targets but is recycled in a normal manner (i.e., as MOX fuel). The advantages and disadvantages of each of these recycle modes are summarized in Table 3.1.

Table 3.1. Advantages and disadvantages of various LWR actinide recycle modes

Actinide recycle mode	Advantages	Disadvantages
Homogeneous dispersal in all fuel	<ol style="list-style-type: none"> 1. Maximum transmutation rate 2. Maximum actinide dilution; therefore, minimum impact on fuel behavior and minimum radiation 	<ol style="list-style-type: none"> 1. Remote fabrication of fuel required because of the high radiation level
Homogeneous dispersal in MOX fuel	<ol style="list-style-type: none"> 1. Minimized fuel requiring remote fabrication 2. Acceptable transmutation rate 3. Actinides still relatively dilute 	<ol style="list-style-type: none"> 1. Lower flux in the MOX assembly than in the first recycle mode
Target rods in an ordinary assembly	<ol style="list-style-type: none"> 1. Allows most fuel to be handled normally 2. Allows for special processing for target rods 3. Acceptable transmutation rate 	<ol style="list-style-type: none"> 1. Lower transmutation rate than the above modes 2. Special processing facilities needed 3. Fuel behavior uncertainties 4. Rods must be removed from assembly before reprocessing and reinserted before irradiation
Target rods in a target assembly	<ol style="list-style-type: none"> 1. Does not require target rod insertion and removal 2. Allows for special processing 	<ol style="list-style-type: none"> 1. Possibly unacceptably low transmutation rates 2. Power peaking problems within assembly 3. Extremely high radiation levels 4. Fuel behavior uncertainties

Based on a preliminary qualitative evaluation, it would appear that the second recycle mode is preferred over the others. The first mode does not have a transmutation rate that is sufficiently greater than that of the preferred option to justify the extra fabrication costs involved. The third recycle mode (target rods) requires insertion and removal of the rods both before and after irradiation and special, small-scale processing facilities; in addition, it has a much greater potential for fuel behavior problems. The fourth option also requires the special processing facilities and is subject to the same types of potential fuel behavior problems. Further, it has problems with power peaking and a markedly lower transmutation rate.

3.2.2 Fuel enrichment requirements

As noted at the beginning of Sect. 3.2, virtually all of the LWR actinide transmutation studies conducted to date are deficient in that they used the initial fuel conditions to determine fuel enrichment instead of the final fuel conditions. This makes a substantial difference in the enrichment requirements in the P-T cycle since the amount of fissile material in the fuel assemblies containing TRU actinides does not decline as rapidly as that in either a normal UO_2 or MOX fuel assembly. Thus, because it is desirable to have all fuel assemblies subcritical to the same extent at the EOI, the initial degree of criticality can be less in the P-T assembly than in the MOX assembly.

Based on limited analyses, it appears that the extra enrichment requirements of a P-T fuel assembly may be somewhat greater or less than that of a reference MOX assembly, depending principally on the cycle considered. For example, in a case studied by Gorrell,³ the ^{235}U enrichment in the P-T assembly was 2.9% in the first recycle (as compared with the reference value of 0.7%) but had dropped to 0.7% by the fifth recycle and showed signs of continuing to decrease. The lower enrichment requirements result from the enhanced buildup of more fissile species, principally plutonium, during later recycles. The overall conclusion is that the enrichment penalty from transmuting actinides in a PWR is not large and can be either positive or negative, depending on the recycle examined.

In light of this, it would seem appropriate to consider this penalty to be zero.

3.2.3 Changes in fuel composition

This section is intended to briefly summarize the fuel composition changes resulting from the recycle of the waste actinides. The recycle of the neptunium, along with the use of recycled uranium (containing ^{236}U), has one predominant effect: it increases the ^{238}Pu (half-life = 88 years) content of the plutonium from about 2% in the reference case to 8% in the P-T case (including the effects of ^{242}Cm mentioned below).

The recycle of americium has relatively few direct effects except that it produces curium and is a significant neutron poison.

The recycle of curium has multiple effects. One of these is to increase the amounts of ^{242}Cm (half-life = 163 days) in the reactor, which in turn increases the amounts of ^{238}Pu , its decay progeny. A second effect is to increase the concentrations of $^{242,244}\text{Cm}$ in the fuel. Both of these nuclides are major contributors to decay heat and also dominate neutron activities during the first few recycles. A third effect is to allow the buildup of highly fissile ^{245}Cm , a significant contributor to the destruction of the transplutonic waste actinides. The final effect of curium recycle is to produce various isotopes of californium, particularly ^{252}Cf . This isotope, which only begins to be significant after a few recycles, is an intense neutron emitter which can cause considerable difficulty in handling P-T fuel materials.

In an elemental sense, recycle of the waste actinide tends to reduce the neptunium-americium fraction of the recycled actinides and increase the plutonium-curium-californium fraction. Calculations made by tracking the waste actinides and their progeny independently of "fuel" uranium and plutonium show that, after ten recycles, the plutonium ($\sim 50\%$ ^{238}Pu) content has increased to 40%. This is the reason, as mentioned earlier, that the actinide recycle scenario must be closed with respect to uranium and plutonium if meaningful results are to be obtained.

3.2.4 Transmutation rates

This section summarizes briefly the transmutation rates that can be expected from actinide recycle in a PWR. The expected transmutation rate is about 5 to 7% per full-power year of irradiation. If we assume a 75% reactor capacity factor and 3 years (calendar) in the reactor followed by 2 years out-of-reactor (for decay, reprocessing/partitioning, and re-fabrication/partitioning), this amounts to about 2.5 to 3.5% per calendar year. These values imply that, at steady-state self-generated actinide recycle (where actinides are being fissioned at the same rate that they are being produced), about 1 to 2% of the P-T MOX fuel assemblies will be comprised of waste actinides. This rate will, of course, depend on the input composition of the actinides.

3.3 Transmutation of Iodine and Technetium

Results of the incentives analyses conducted as a part of the ORNL P-T program indicate that ^{99}Tc and ^{129}I are the two major contributors to the risk from a geologic repository. Although these nuclides are not the principal concerns of this program, it seems appropriate to briefly describe the results of studies concerning their transmutation.

In general, the transmutation of fission products is most effectively accomplished in a thermal reactor since the flux — cross-section product (i.e., reaction rate) is larger in a thermal reactor than in a fast reactor and the fission-to-capture ratio is irrelevant. Evidence of this is readily apparent in the form of the much larger poisoning effect of the fission products in a thermal reactor than in a fast reactor. Accordingly, the iodine and technetium transmutation studies described here will only consider thermal reactors, specifically uranium-enriched PWRs.

3.3.1 Transmutation of ^{99}Tc

Technetium-99, which is produced as a result of the fissioning of actinides, is the only technetium isotope that is both present in significant quantities (about 770 g/MTIHM) and is long-lived (half-life =

213,000 years). In the nuclear fuel cycle, ^{99}Tc appears predominantly in the HLLW and dissolver solids, from which it would be recovered and purified in preparation for transmutation (Sect. 2.3). The form of the technetium during irradiation has not been investigated, although the metal is a likely candidate since its melting point is about 2400 K.

Calculations yield a transmutation rate of 11% per full-power year. The discharged technetium target rod has the following composition per initial kilogram of ^{99}Tc : 728 g of ^{99}Tc , 270 g of ^{100}Ru , 2.5 g of ^{101}Ru , and 0.006 g of ^{102}Ru .

3.3.2 Transmutation of ^{129}I

Iodine-129, which has a half-life of 15.9 million years, is produced in the amount of 194 g/MTIHM by fissioning actinides. It is accompanied by about 61 g of stable ^{127}I per MTIHM. In the nuclear fuel cycle, ^{129}I can be made to report almost quantitatively to the dissolver off-gas and can be removed with a variety of techniques (see Sect. 2.3). The form of iodine appropriate for insertion into a reactor is not clear due to the proclivity of iodine compounds to volatilize or decompose and their corrosiveness toward virtually all metals at elevated temperatures. For calculational purposes, the sodium iodide (NaI) form has been assumed.

Transmutation calculations yield a transmutation rate of 3% per full-power year. The discharged iodine target rod had the following composition per initial kilogram of iodine: 221 g of ^{127}I , 704 g of ^{129}I , 1.1 g of ^{128}Te , 18.1 g of ^{128}Xe , and 56.2 g of ^{130}Xe . The initial iodine composition was 240 g of ^{127}I and 760 g of ^{129}I .

In summary, the following problems are associated with the transmutation of ^{129}I :

1. iodine and iodine compounds are corrosive toward potential cladding materials;
2. most iodine compounds are decomposed or volatilized at LWR operating temperatures;
3. the xenon gas produced by the transmutation will put considerable stress on the cladding in a corrosive environment, and
4. the transmutation rate is very low.

3.4 Other Actinide Transmutation Reactors

Three additional classes of reactors might be considered as actinide transmutation devices: (1) thermal reactors other than LWRs; (2) fast reactors, such as LMFBRs; and (3) fusion reactors.

3.4.1 Other thermal reactors

Two other types of thermal actinide transmutation reactors, D_2O -moderated reactors and HTGRs, have been considered in brief studies.

3.4.1.1 D_2O -moderated reactors. Two heavy water reactors have been briefly examined: a Savannah River Plant (SRP) production reactor, and a CANDU* reactor. The SRP production reactor is a specially designed, high-power-density, high-flux reactor operating at atmospheric pressure. As a result of the high flux, its transmutation rate is quite high as compared with other thermal reactors. However, the use of this type of reactor as a transmutation device entails several substantial penalties not present in commercial reactors. These penalties can be summarized as follows:

1. SRP production reactors do not produce usable power or heat. Thus the entire cost of building and operating the reactor must be allocated to transmutation.
2. Attainment of the high fluxes requires the use of highly enriched uranium as the driver fuel.
3. If the waste actinide loadings are not kept small, the high neutron flux will be seriously degraded.

Such penalties make the D_2O -moderated reactor unattractive, especially when insertion of the actinides into commercial thermal reactors does not cause any substantial difficulties. However, this conclusion might be altered if it were possible to build a high-flux reactor that produced enough usable power to pay for itself.

*An existing Canadian commercial reactor fueled with natural uranium and moderated with heavy water.

A plutonium-recycle CANDU reactor has been studied briefly as an actinide transmutation device. Overall, its performance appears to be about the same as that of LWRs when operating on the same actinide composition and mass. However, when operating in the self-generated actinide recycle mode, it appears to be somewhat better than LWRs because of the lower actinide production rate resulting from the low fuel burnup.

3.4.1.2 High-temperature gas-cooled reactors. HTGRs have been studied in a very preliminary fashion for use as transmutation devices. The actinide transmutation rates achievable in an HTGR are somewhat higher than those in a LWR, principally because of the somewhat higher flux levels in the former.

3.4.2 Fast-fission reactors (LMFBRs)

The LMFBR is unquestionably the most intensively studied transmutation device. This emphasis was principally a result of the intuitive judgment that the high actinide fission-to-capture ratio resulting from the high-energy neutron spectrum would transmute (i.e., fission) the actinides more quickly with less impact than would LWRs. These contentions have generally been borne out, although the differences are not as great as had been hoped.

A qualitative comparison of actinide transmutation in LMFBRs with that in LWRs is given in Table 3.2 for the most important characteristics of a transmutation system. In summary, the breeding ratio and fissile makeup attributes are not significant for either reactor. The LWRs are clearly favored with regard to availability due to their widespread deployment at present and their likely continued deployment. All of the other characteristics favor the LMFBR, although none is sufficiently detrimental to the LWR to preclude its use. The LMFBR transmutation rate, although generally larger than in the LWR, does not justify an LMFBR economy on this basis alone. However, it is equally clear that if LMFBRs were to exist at the time when P-T was being implemented, they would be preferred as transmutation devices.

Table 3.2. Qualitative comparison of actinide recycle in an LWR and an LMFBR

Characteristic	LWR	LMFBR
Transmutation rate	About 5 to 7% per full-power year of irradiation; depends on actinide composition and recycle mode	About 5 to 11% per full-power year; usually higher than LWR; depends on actinide composition and reactor design
Fissile makeup requirements	Extra fissile makeup required during first few recycles; small fissile benefit in later cycles	Small fissile benefit
Breeding/conversion ratio effects	Very small effect	Very small effect
Neutron activity of recycled actinides	10^{13} to 10^{14} neutrons per second per metric ton of recycled actinides	Factor of 10 to 100 lower than the LWR actinides
Power peaking	Significant in all except homogeneous dispersal in all fuel; enrichment grading probably required; most difficult in target assembly	Virtually nonexistent if a proper amount of diluent (e.g., UO_2) is used for the recycled actinides
Long-term viability	Limited due to the continuous need for fissile material (i.e., ^{235}U)	Very good; limited only by the availability of ^{238}U or thorium

Finally, a major disadvantage of the LWR is its limited long-term potential as a transmutation device due to the limited availability of naturally occurring fissile materials required for fuel. Since implementation of P-T would not be expected until after the year 2000, the the remaining lifetime of the LWR with respect to transmutation would be restricted to perhaps five or six cycles (at 5 years per cycle) before the number of LWRs would begin to decline.

3.4.3 Other transmutation devices

Several neutron-generating devices other than fission reactors have also been proposed and briefly studied as transmutation devices. These devices fall into three categories: (1) fusion reactors, (2) particle accelerator devices, and (3) nuclear explosives.

3.4.3.1 Fusion reactors. The use of fusion reactors as transmutation devices has been studied by several organizations during the past few years. Virtually all of these studies have assumed the use of a tokamak-type fusion reactor with the actinides arranged in annular blankets around the torus. As a result of its very high neutron fluxes, the fusion reactor initially appeared to be very attractive as a transmutation device. However, recent work using more sophisticated calculational methods showed that the actual transmutation rates were not nearly as high as indicated previously. This effect appears to be a result of the sharp decline of the neutron flux as it passes through the actinide blanket because the blanket is subcritical. The portions of the blanket nearest the torus were highly irradiated, whereas those farthest from the torus were virtually untouched. The conclusions were that actinide transmutation in fusion reactors with wall loadings on the order of 1 MW/m^2 was marginally acceptable (transmutation rates appear to be about 8% per full-power year) but that wall loadings on the order of 10 MW/m^2 were necessary to make actinide transmutation in fusion reactors attractive. However, at these higher wall loadings, heat removal and radiation damage considerations may require that the actinide density in the blanket be sharply reduced to maintain an acceptable power density. Thus, under these conditions, more fusion reactors would be required for a given mass of recycled actinides. In summary, actinide transmutation in fusion re-

actors appears to be feasible and at least as efficient as LMFBRs; however, a substantial number of engineering problems must be solved. These devices do not now appear to be overwhelmingly superior to other transmutation devices as they once did.

3.4.3.2 Particle accelerators. Another class of transmutation devices that have been heavily investigated is based on particle accelerators. The most popular concept appears to be that of using a particle accelerator to make a beam of protons with energies on the order of 1 to 10 GeV. These protons are impacted on a molten, heavy-metal target such as lead, bismuth, or uranium; the target is molten for heat removal purposes. The high-energy protons disintegrate the heavy nucleus in a spallation reaction, producing 5 to 50 neutrons per interaction. These neutrons are then used to transmute the actinides that are present in a blanket surrounding the spallation target. The large numbers of neutrons produced by each proton result in a very intense neutron source; however, this concept has several serious drawbacks, such as:

1. The neutron source is effectively a point source, and the neutron flux declines very rapidly in the subcritical actinide blanket in a manner similar to that in the fusion reactor.
2. A method for recovering the heat from the spallation device must be devised; otherwise, the entire cost of the operation must be charged to the transmutation operation, a very expensive proposition.
3. This concept would require the design and construction of high-energy particle accelerators that are much larger than any built thus far.
4. It would appear to be inefficient to use nuclear fission heat to make electricity, which in turn is used to generate high-energy protons (and more energy losses from inefficiency). The protons are then used to produce neutrons, which finally accomplish the transmutation. Actinide transmutation in the fission reactor seems much more straightforward and eliminates the unavoidable intermediate energy losses.

5. The handling of molten metals in a high-intensity, high-energy neutron flux at high temperatures would appear to be a formidable engineering obstacle.

Thus the engineering and theoretical limitations of this concept would appear to greatly overshadow any possible advantages.

3.4.3.3 Nuclear explosives. The final alternative transmutation process that has been proposed involves the use of nuclear explosives as the source of transmutation neutrons. The detonation of a nuclear device produces large quantities of high-energy neutrons, which would transmute the actinides almost instantaneously. Estimates indicate that about three to four 100-kiloton (kt) explosions would be required annually per 1000-MW(e) reactor at a cost estimated to be about 17% of the cost of the electricity. If we ignore the relatively high dollar cost of this concept, three fundamental problems are involved:

1. The residual nuclides would be irretrievably encased in rock at the (underground) explosion site in a manner that may not be suitable for long-term isolation.
2. At present, there are limitations on the testing of nuclear weapons, and further limitations are expected in the future.
3. The detonation of three to four 100-kt nuclear devices annually for each 1000-MW(e) nuclear reactor for waste management purposes would certainly be socially and politically unacceptable.

Thus, although it appears to be technically feasible, the political realities, high cost, and logistical difficulties make this concept unacceptable.

3.5 Assessment of Transmutation

Although much of the material presented in Sects. 3.5.1-3.5.4 may already be evident from the preceding portions of Sect. 3, it is useful to briefly recapitulate the most important results of the transmutation assessment.

3.5.1 Transmutation feasibility

The results of this study concerning transmutation feasibility are:

1. Actinide transmutation is feasible in thermal and fast reactors, subject to the development of satisfactory fuel forms (see Sect. 4.2). Transmutation rates range from about 6% per full-power year in a PWR to about 9% per full-power year in an advanced-oxide LMFBR.
2. The transmutation of ^{99}Tc appears to be feasible in thermal reactors at the rate of about 11% per full-power year, subject to the development of a satisfactory fuel form and investigation of the production rate of long-lived ^{98}Tc .
3. The transmutation of ^{129}I is marginally feasible at best because of the low transmutation rate, the volatility and corrosiveness of iodine and iodine compounds, and the production of large amounts of xenon gas from the transmutation reaction.

3.5.2 Transmutation impacts

An assessment of impacts of transmutation on the reactors is as follows:

1. Actinide transmutation has a significant impact on a thermal reactor, mostly in the areas of power peaking and enrichment changes. In most cases, this impact can probably be accommodated by using existing technology.
2. The transmutation of iodine or technetium would also have a significant impact on a thermal reactor, principally in the form of enrichment penalties. These impacts can probably be accommodated using existing technology.
3. Actinide transmutation in an LMFBR has virtually no impact on the reactor.

3.5.3 Transmutation comparisons

The results concerning the comparison of the transmutation reactors are:

1. Under present conditions, the superiority of either the LMFBR or the LWR as a transmutation device depends on the transmutation scenario postulated. On the one hand, LWRs exist at the present time and actinide transmutation is feasible. On the other hand, LMFBRs suffer less impact from the actinides than do thermal reactors, they transmute the actinides more quickly, and they are viable over the long term when P-T might be implemented.
2. Under conditions where both LWRs and LMFBRs exist, the LMFBR would be preferred over the LWR as a transmutation device.
3. Alternative thermal power reactors do not offer either significant advantages or disadvantages as compared with LWRs.
4. Fusion reactors (assuming their existence) would be acceptable actinide transmutation devices but probably no better than LMFBRs.
5. Special, high-flux transmutation devices (SRP production reactors or spallation devices) do not appear to offer significant transmutation advantages and would probably be economically disadvantageous.
6. Actinide transmutation using nuclear explosives is not acceptable from sociopolitical and economic viewpoints.

3.5.4 Other assessments

Most of the transmutation studies conducted to date are not comparable or correct and thus are marginally useful because of methodological (i.e., the recycle scenario is not "closed") or calculational (i.e., incorrect criterion for determining enrichment) flaws.

3.6 Characterization of Fuel Cycle Materials

The purpose of this section is to briefly summarize the composition and characteristics of the fuel materials in both a reference (no P-T) cycle and a P-T cycle. The specific fuel cycles considered are those described in Sects. 2.1.1 and 2.2.1, and the actinide losses are based on those given in Sect. 2.1.4. The details of the fuel irradiation calculations are given in ref. 1. The composition and characteristics of the reference and P-T fuels, both fresh and spent, are given in Table 3.3.

The recycle of the waste actinides results in a substantial concentration of the heavier actinides in the P-T fuels, as expected. Of particular note are the higher levels of ^{244}Cm and ^{252}Cf , both of which are spontaneous-fission neutron emitters. These higher levels are reflected in the second portion of the table, which gives the neutron activities of the fuel materials. The recycle of the actinides increases the neutron activity by about a factor of 10 for the spent fuels and by over a factor of 100 for the fresh fuels. In designing fuel cycle facilities, these increases must be taken into account by increasing the thicknesses of neutron shielding (e.g., concrete). Another example of the impact of the neutrons will be discussed in Sect. 4.1, which is concerned with the design of fuel transportation casks.

The dose rates from unshielded fuel assemblies are given in the third portion of Table 3.3. Again, the impact of the increased neutron activity is evidenced by the much larger dose rate from the fresh P-T fuel as compared with the fresh reference fuel. The dose rates for unshielded spent fuel are identical for both reference and P-T fuels because they are controlled by the fission products, which are the same in both cases.

The final portion of Table 3.3 gives decay heat values for the various fuels, including fission products in the case of the spent fuels. As is evident, the presence of the recycled actinides, particularly ^{244}Cm , increases the decay heat from the fresh P-T fuel so that it is almost identical with the decay heat from a spent reference fuel assembly. The decay heat from the spent P-T fuel assembly is nearly twice that of the

Table 3.3. Composition and characteristics of fresh and spent reference and P-T fuels from the fifth recycle

	Fresh MOX fuels ^a		Spent MOX fuels ^b	
	Reference	P-T	Reference	P-T
Composition, g/MTHM				
²³⁵ U	6938	6768	3846	3517
²³⁶ U	6853	6829	6409	5846
²³⁸ U	918,200	907,600	900,200	883,800
²³⁷ Np	4284	2576	3599	2827
²³⁸ Pu	5241	4442	5473	5260
²³⁹ Pu	24,230	25,670	13,830	17,600
²⁴⁰ Pu	16,650	17,880	13,360	14,870
²⁴¹ Pu	9530	9203	7675	7925
²⁴² Pu	7286	10,400	7071	11,070
²⁴¹ Am	463	1535	1197	1388
²⁴³ Am	0	2749	2408	2923
²⁴² Cm	0	28	15	26
²⁴⁴ Cm	0	3478	941	4526
²⁴⁵ Cm	0	484	54	813
²⁴⁶ Cm	0	215	24	332
²⁴⁷ Cm	0	13	0.14	25
²⁴⁸ Cm	0	2.8	3.4-03	6.2
²⁴⁹ Bk	0	6.9-04	8.3-06	6.7-02
²⁴⁹ Cf	0	6.2-03	2.3-05	0.21
²⁵⁰ Cf	0	1.5-03	4.2-06	6.8-02
²⁵¹ Cf	0	2.2-04	3.7-07	1.0-02
²⁵² Cf	0	2.7-04	1.4-07	2.1-02
Neutron activity, neutrons sec ⁻¹ MTHM ⁻¹	1.84+08	4.16+10	1.19+10	1.05+11
Dose rate from one assembly, rem/hr ^c	0.17	11.2	11,900	11,900
Decay heat, W/MTHM	3250	12,440	12,970	23,170

^aDecay of 0.5 year assumed.

^bDecay of 1.5 years assumed.

^cDose is 1.0 m from the assembly midplane.

spent reference assembly. The magnitude of these numbers clearly indicates that (1) the fresh P-T fuel will have to be handled exactly as spent fuel is now, and (2) the handling of the spent P-T fuel will require special design considerations.

3.7 References for Section 3

1. J. W. Wachter and A. G. Croff, Actinide Partitioning-Transmutation Program Final Report. III. Transmutation Studies, ORNL/TM-6983 (in press).
2. A. G. Croff, D. W. Tedder, J. P. Drago, J. O. Blomeke, and J. J. Perona, A Preliminary Assessment of Partitioning and Transmutation as a Radioactive Waste Management Concept, ORNL/TM-5808 (September 1977).
3. T. C. Gorrell, Transmutation of Waste Actinides in Light-Water Reactors, DP-1518 (April 1979).

4. OTHER FUEL CYCLE IMPACTS OF PARTITIONING-TRANSMUTATION

The purpose of this section is to discuss four aspects of P-T that are important considerations in any complete evaluation of this option but do not readily fit within the scope of either the previous partitioning or transmutation sections. These aspects, which are widely varied and only related in that each of them bears on P-T, are as follows:

1. the design of a special shipping cask for transporting reactor fuels that have high decay heat levels and intense neutron radiation;
2. consideration of the impacts of P-T on fuel fabrication and the suitability of reactor fuels containing substantial amounts of neptunium, americium, and curium;
3. consideration of the disposition of currently existing inventories of spent fuel and wastes until such time that P-T could be implemented; and
4. consideration of the implications of P-T for current U.S. policy concerning nuclear power and its fuel cycle.

The subject matter discussed in this section is based on more detailed work presented in ref. 1.

4.1 P-T Fuel Shipping Cask

As noted in Sect. 3.6, both fresh and spent P-T fuels are intensely neutron active. Although the dose rates given in Sect. 3.6 for the spent P-T fuel do not indicate that the neutrons are significant in this case, they still dominate the dose in the case where shielding is present since they are much more penetrating than the gamma rays. In stationary facilities, such as the reprocessing and MOX fuel fabrication plants, the increased shielding would simply take the form of somewhat thicker concrete walls. However, a serious problem arises with a fuel shipping cask since it must conform to definite weight and size limitations and the normal materials of construction are extremely poor attenuators of neutrons.

Thus the shipping cask must be redesigned to accommodate P-T fuels. This task was undertaken as part of the ORNL P-T project, and the resulting conceptual (but not necessarily optimal) design is described below. It should be noted that the fresh P-T fuel must be shipped in exactly the same manner as the spent P-T fuel because of its high neutron activity and thermal power.

The design-basis neutron activity for the shipping cask was taken to be 1.1×10^{12} neutrons/sec. This somewhat conservative value, as compared with the values in Sect. 3.6, was used to ensure that fuels from further recycles could be accommodated and to account for some simplifying assumptions and burnup variability. The design-basis thermal power of the fuel was assumed to be that in Sect. 3.6 for the spent P-T fuel (viz., 23,170 W/MTIHM). The maximum allowable weight of the cask was assumed to be 100 MT. A cask of this size could only be transported by rail.

Calculations for both the fresh fuel and the spent fuel have indicated that the shielded dose from the neutrons is more limiting than that from the gamma rays. The general approach, then, is to construct the cask of materials that successively thermalize (slow down) the neutrons and absorb (capture) them. The thermalization ability of a medium declines in proportion to the square of its atomic weight. Therefore, the best medium for neutron thermalization is hydrogen, followed by deuterium and the other light elements. Neutron absorption does not follow any simple pattern; however, materials such as boron, cadmium, and gadolinium are good neutron absorbers. In combination with the ability to thermalize and absorb neutrons, the cask must also be able to conduct heat reasonably well since the P-T fuels have considerable thermal power. Unfortunately, this requirement tends to conflict with the neutron capture requirement since hydrogenous materials generally have poor thermal conductivities. However, a suitable balance of these requirements was ultimately achieved. A cross section of the resulting conceptual P-T cask design is shown in Fig. 4.1. The capacity of the cask is six PWR fuel assemblies, which is significantly less than that of a normal shipping cask (about ten assemblies). The capacity was reduced to keep the weight and size within the specified limits.

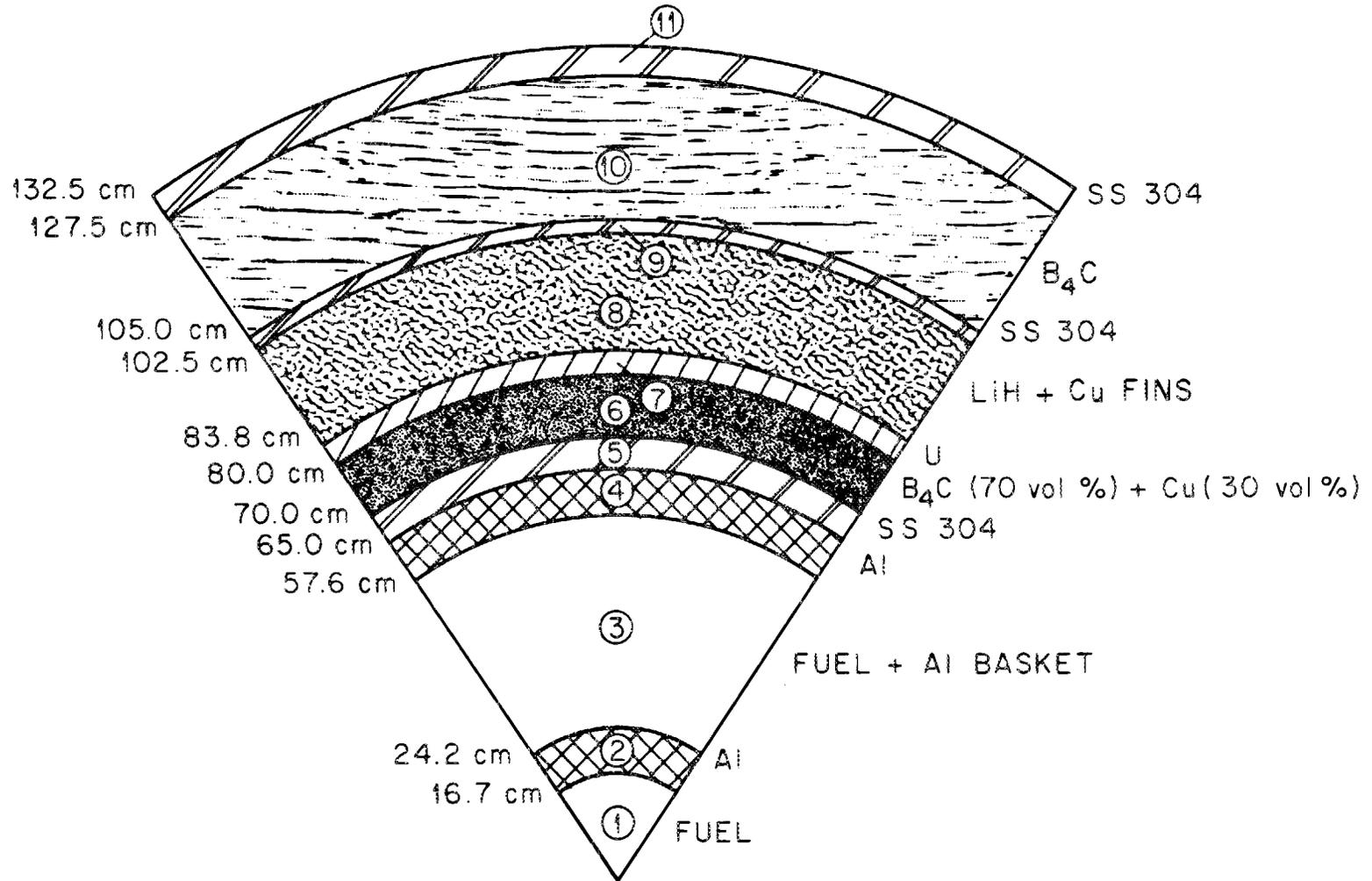


Fig. 4.1. Conceptual design of cask for shipping P-T fuel.

The first four zones of the cask comprise the fuel material and the aluminum structure required to hold it. The three stainless steel zones (zones 5, 9, and 11) and the uranium zone (7) serve the dual purpose of shielding from the primary and secondary (neutron-induced) gamma rays and providing structural integrity. The two boron carbide (B_4C) zones plus the lithium hydride (LiH) zone are the principal neutron thermalization and absorption areas. These materials are entirely comprised of light elements to maximize thermalization. Additionally, boron is an excellent neutron absorber, while lithium is fairly good. Copper fins have been included both in the inner boron carbide zone and in the lithium hydride zone to accommodate the heat load from the fuel. External aluminum fins are included to increase heat removal. The composition of the cask is as follows: 30.8 wt % stainless steel, 29.2 wt % B_4C/Cu , 18.2 wt % uranium, 11.1 wt % aluminum, 5.9% LiH/Cu, and 4.8% fuel. The weight of the cask (excluding the rail car) is about 100 MT. Its outside diameter, including fins, is 285 cm (about 9.4 ft). An initial evaluation indicates that the P-T cask would probably meet the stringent requirements for spent fuel shipping casks. During the hypothetical accident, the lithium hydride layer would most likely be lost since it reacts with water. However, dose rates during such accidents would still be within limits since the remaining shielding materials are stable at elevated temperatures and in an aqueous environment. The consideration of elevated temperatures eliminated materials such as borated water from consideration as a shielding medium.

4.2 Impacts of P-T on Fuel Fabrication

4.2.1 Fabrication plant design

4.2.1.1 Design philosophy. The fabrication plants in the P-T assessment program were assumed to be remotely operated and contact-maintained (RO/CM) in the reference cycle and remotely operated and remotely maintained (RO/RM) in the P-T cycle, with both using powder-to-pellet technology. This difference occurs because the neutron activity of the P-T fuel is higher by a factor of 200 (see Sect. 3.6), and it was felt that the equipment could not be decontaminated sufficiently to allow contact

maintenance as is possible with the reference fuel. However, there are two cases in which this difference may not occur. In the first instance, the reference MOX fuel would have a sufficiently high activity to require RO/RM fabrication. The effect of this would be to substantially reduce the fabrication penalty incurred by the P-T fuel cycle since both plants would be RO/RM, although some differential would remain due to the greater shielding (i.e., wall) thicknesses required in the P-T cycle.

In the second instance, a fabrication process that is more amenable to decontamination than the powder-to-pellet process is used, thereby permitting contact maintenance in both the reference and P-T cycles. A process that uses principally liquids, such as a sol-gel process, might fall in this category.

In any event, we do not presently know the "correct" answer to the dilemma concerning the appropriate design philosophy to be used. Resolution of this problem will require considerably more detail concerning the P-T fuels to be fabricated, process performance and cleanup characteristics, and occupational dose limits.

4.2.1.2 Fabrication process feasibility. A second question related to fabrication plant designs concerns the feasibility (or, more precisely, the operability) of an RO/RM facility. To date, fuel fabrication using a powder-to-pellet process under an RO/RM philosophy has not been demonstrated, even on a small scale. Furthermore, very little conceptual work has been done in this area. Consequently, there is considerable uncertainty as to whether a commercial-size (660-MTHM/year) facility can be operated with a reasonable capacity factor under these conditions. Perhaps, after some investigation, we may conclude that the only feasible method of fabricating these materials is to use another process (e.g., a liquid process) that facilitates remote operation and maintenance.

4.2.2 P-T in-reactor fuel performance

4.2.2.1 Actinides. The in-reactor performance of P-T fuels is subject to question since their composition is, in some cases, quite different from that of standard LWR fuels. The performance characteristics

of concern include swelling behavior, resistance to cracking, fuel-cladding interaction, cladding corrosion, and amenability to dissolution after irradiation.

In the cases where the waste actinides (neptunium and the transplutronics) are diluted in a large fraction or all of the fuel, relatively few problems may be experienced since the behavior should be controlled by the uranium and plutonium. However, in the fuel cycles that involve target element or assemblies, the concentrations of actinides may range from a few percent to 40%. In these cases, the fuel performance can only be determined via actual irradiation tests. For the purposes of assessing P-T, it was assumed that satisfactory fuels could be developed for all actinide mixtures.

4.2.2.2 Iodine and technetium. In this study, brief consideration was given to the transmutation of iodine and technetium. Since these types of materials have never been irradiated in any form similar to that anticipated if they were to be recycled (i.e., concentrated in target rods), some discussion of their performance is probably in order.

Technetium, as the metal, has a relatively high melting point (about 2400 K). Because it is to be recycled in a target rod, the metal may well be an acceptable form since no damage will occur from fission fragments and its predominant transmutation products are chemically similar ruthenium isotopes.

Iodine, on the other hand, presents formidable problems with respect to the identification of a satisfactory form. The four principal problems are:

1. Iodine and its compounds tend to be corrosive at reactor operating temperatures (at least 700°F).
2. Most iodine compounds volatilize or decompose at reactor operating temperatures.
3. The transmutation daughter of iodine is xenon, which would tend to disrupt the physical integrity of the iodine form and pressurize (and stress) the fuel pin.

4. Most iodine compounds are fairly soluble in water.

In view of these difficulties, the feasibility of iodine transmutation appears marginal based on materials considerations alone.

4.3 Disposition of Long-Lived Materials Prior to P-T

As a result of the extensive amount of RD&D that would be required, it is improbable that P-T could be implemented before the year 2000. This conclusion leads to some difficulty concerning the disposition of the existing and to-be-produced spent fuel and/or long-lived wastes during the interim.

The options available for handling the spent fuel until P-T could be implemented* are:

1. retain all of the spent fuel in storage until it could be partitioned,
2. reprocess the fuel and retain the long-lived wastes until P-T was available, or
3. continue with the disposal of spent fuel and/or high-level waste until P-T becomes available.

The first option, while feasible, would probably entail the surface storage of the spent fuel. Based on the projections given in ref. 2 and an away-from-reactor (AFR) storage facility with a capacity of 5000 MTHM, approximately 18 AFRs would be required in the year 2000 and the construction rate would be more than one per year. Additionally, holding this large amount of spent fuel on the surface is more risky than placing it in a geologic repository. This consideration is not trivial when it is realized that the spent fuel inventory in the year 2000 would be generating 342 MW(t) and contain about 86 GCi of activity.

The second option has many of the same disadvantages as the first. Large storage facilities would be required to contain the liquid HLW, all of the TRU wastes from reprocessing and refabrication, and the iodine-containing waste (if iodine is being recovered). This option is somewhat

*This discussion assumes that P-T will be implemented.

more risky than the previous one because the HLW has the same activity and thermal power as the spent fuel but is more mobile because it is a liquid. Further, it is not presently clear whether the recovery of actinides from aged HLW is feasible because of the postulated difficulty in recovering the actinides from insoluble precipitates that form over a period of time. Thus, although this option allows for the recovery and recycle of the fissile and fertile values in the near term, it appears to be less advantageous than the first.

The third option would be to continue treating and disposing of long-lived wastes until P-T could be implemented. This approach minimizes the storage costs and risks mentioned above by removing all wastes to the repository except a working inventory. However, it is disadvantageous in that the long-lived nuclides which are sent to the repository are not available for recovery after P-T is implemented. If a realistic, probabilistic risk analysis is used, the risk from disposing of the long-lived wastes is extremely small since the long-term risk from the repository is so small. On the other hand, if the analysis is based on perceived risk (which is probably proportional to the long-term toxicity of the waste), only a small fraction of the wastes can be committed to the repository and still reduce the toxicity relative to spent fuel by only a factor of 50. For example, if unpartitioned wastes are sent to a repository for 25 years, and then partitioned wastes are sent for the next 25 years, the maximum reduction in the waste actinide content of the repository (assuming perfect partitioning) would be a factor of 2.

The most reasonable and technically defensible approach would be to base the decision on a realistic risk analysis and to dispose of unpartitioned fuel and/or wastes in a repository until such time that P-T is available and a decision in favor of implementation has been made by responsible officials.

4.4 Policy Implications of Partitioning-Transmutation

The implementation of P-T holds certain implications for existing U.S. nuclear policies. Specifically, three aspects of P-T will impact existing policies:

1. the use of reprocessing and actinide recycle in P-T,
2. the possible use of fast breeder reactors as transmutation devices, and
3. the effects of P-T on geologic disposal requirements.

The following sections will describe the impacts of these aspects on existing policies and discuss the reasons why the impacts occur.

4.4.1 Implications of reprocessing and actinide recycle

By definition, the implementation of P-T would require that spent fuel be reprocessed (to recover the actinides) and that actinides be recycled (to fission them). In fact, P-T involves considerably more processing activities than a normal fuel cycle with plutonium recycle since additional processes are required to reduce the actinide contents of the wastes to low levels. The use of reprocessing and actinide recycle conflicts with existing U.S. policy, which was stated by President Carter on April 7, 1977, in the following manner: ". . . we will defer indefinitely the commercial reprocessing and recycling of the plutonium produced in the U.S. nuclear power programs."³ Thus it would be impossible to implement P-T under this policy.

Current U.S. policy, as stated above, would have to be altered to allow reprocessing and recycle before P-T could be implemented. Furthermore, in light of the substantial amounts of RD&D needed on P-T (see Sect. 6), it is clear that the policy would have to be altered so that reprocessing and recycle received vigorous governmental support and were not simply given permission to proceed.

4.4.2 Implications of using LMFBRs as transmutation devices

The use of LMFBRs for transmuting actinides is not mandatory since transmutation in thermal reactors appears to be feasible. However, comparison of the transmutation characteristics of thermal reactors with those of LMFBRs (see Sect. 3.4.2) indicates that existing LMFBRs are superior actinide transmutation devices, particularly when resource considerations are accounted for. Thus, it might be highly desirable

from both economic and safety standpoints to use LMFBRs as the transmutation devices if P-T were to be implemented.

Current U.S. policy on the introduction and use of LMFBRs is as follows:³ ". . . we will restructure the U.S. breeder program . . . to defer the date when breeder reactors would be put into commercial use." This policy, while not prohibiting the use of LMFBRs, would postpone an LMFBR economy of the type needed for transmutation purposes until well into the twenty-first century. In light of the tremendous spent fuel or waste backlogs that might be present at that time and the possible perceived ineffectiveness of implementing P-T after substantial amounts of unpartitioned wastes have been disposed of (see Sect. 4.3), it is clear that current LMFBR policy might conflict with the implementation of P-T over the long term.

4.4.3 Implications of P-T for waste disposal

The waste disposal operations being conducted in a fuel cycle in which spent fuel is being reprocessed and plutonium is being recycled (i.e., the reference cycle) would consist of geologic isolation of HLW and other TRU-contaminated wastes. The density (per unit of electricity) of the HLW in the repository would be established by using a criterion based on the maximum allowable temperature of the waste. The density of the non-HLW (i.e., non-heat-generating waste) depends on the volume of the waste.

In the P-T fuel cycle, HLW and TRU-contaminated wastes would still be produced. The quantities and the heat generation rate of the HLW would be about the same in both the reference cycle and the P-T cycle, primarily because the fission product content is the same in each case. The amounts of non-high-level, TRU-contaminated wastes would be somewhat larger in the P-T cycle than in the reference cycle by a significant, but not an overwhelming, amount.

In summary, the implementation of P-T would have two principal implications for waste disposal activities:

1. waste disposal facilities would still be needed, and

2. the design or required size of a geologic repository would not be significantly affected.

Thus there is no major impact of P-T on waste disposal except as the P-T facilities (or the lack thereof) might affect licensing.

4.5 Summary of Other Impacts of P-T

This subsection provides a brief recapitulation of the principal points made in the preceding portions of Sect. 4. These points can be summarized as follows:

1. Casks designed to handle fresh and spent P-T fuels are substantially different from those for fresh and spent normal fuels because of the predominance of the neutrons in the P-T cycle. However, it appears that these fuels can be transported in a cask made of presently known materials, although (a) the cask capacity is reduced, (b) the cask can only be transported by rail, and (c) somewhat exotic and expensive materials (B_4C and LiH) are required.
2. Based on current knowledge, it appears that the P-T MOX fuel fabrication plant will have to be both remotely operated and remotely maintained, whereas the reference plant could be remotely operated with contact maintenance. This difference is due to the improbability of being able to adequately decontaminate the P-T facility handling the more neutron-active fuels.
3. The irradiation performance of actinide, iodine, and technetium fuels is unknown. It would appear that the actinide fuels containing relatively dilute neptunium, americium, and curium (less than about 1%) would be satisfactory, based on past operating experience.
4. There is some question about the disposition of the spent fuel and/or wastes that would be produced prior to the implementation of P-T. Holding either unprocessed spent fuel or liquid wastes in surface facilities is expensive and more risky than geologic disposal. Long-term risk analyses indicate that disposal of the

unpartitioned materials would not alter the benefits of P-T significantly since they are very small, even if all wastes are partitioned. However, if perceived risk (in the form of waste toxicity) were used as the long-term risk measure, very little unpartitioned material could be sent to the repository before partitioning would become ineffective in an overall sense.

5. Implementation of P-T requires the substantial use of reprocessing and actinide recycle, and may involve using LMFBRs as transmutation devices. Present U.S. policy has indefinitely deferred both of these options.
6. Even if P-T were to be implemented, a waste repository of about the same type and size as that for a normal fuel cycle (in which only plutonium was being recycled) would still be required.

4.6 References for Section 4

1. C. W. Alexander and A. G. Croff, Actinide Partitioning-Transmutation Program Final Report. IV. Miscellaneous Aspects, ORNL/TM-6984 (in press).
2. C. W. Alexander, C. W. Kee, A. G. Croff, and J. O. Blomeke, Projections of Spent Fuel to Be Discharged by the U.S. Nuclear Power Industry, ORNL/TM-6008 (October 1977).
3. Office of the White House Press Secretary, "Statement by the President on Nuclear Policy," Apr. 7, 1977.

5. ANALYSIS OF THE INCENTIVES FOR IMPLEMENTING PARTITIONING--TRANSMUTATION

Three major factors must be considered when making a determination concerning the incentives for implementing P-T:

1. the incremental short-term risks resulting from the extra processing required to implement P-T,
2. the incremental cost resulting from the additional facilities and operations required to implement P-T, and
3. the incremental reduction in the long-term risk from the repository resulting from implementation of P-T.

An analysis of each of these three factors has been performed by analyzing these aspects of the reference and P-T fuel cycles as described in Sects. 2.1.1, 2.2.1, and 3.6. The incremental impacts were then determined by taking differences between the appropriate values in each of the two fuel cycles. Summary descriptions of these analyses are given in Sects. 5.1 through 5.3, respectively.

Section 5.4 relates the three factors to each other to yield a meaningful cost-risk/benefit analysis for the actinide P-T concept. A brief discussion for the fission products technetium and iodine is given in Sect. 5.5.

Section 5.6 examines the sensitivity of the results in Sect. 5.4 to various assumptions made in the underlying analyses -- the final step in the analysis of the incentives for actinide P-T.

5.1 Short-Term Risk Analysis of P-T

The implementation of P-T would result in increased risk from the fuel cycle because of (1) the increased processing or transportation, (2) the increased toxicity resulting from actinide recycle, (3) the expenditure of resources required to operate the facilities (i.e., utilities), and (4) the larger number of workers in the facilities.

The approach taken was to perform a probabilistic (fault-tree) risk analysis for both the reference cycle and the P-T cycle, which were described earlier. This analysis included only the facilities that would be significantly affected by the implementation of P-T since the desired end result is the incremental risk. Two general populations, the public and occupational personnel, must be considered for the case at hand. There are two general types of risk, radiological and nonradiological, for each of these groups. Radiological risk, which results from exposure to radiation, can be either routine or accidental. In the analysis described herein, two sources of routine radiological dose were considered: (1) off-gas releases from the fuel reprocessing plant, fuel fabrication plant, and waste treatment facility off-gases; and (2) the dose to bystanders along transportation routes. Standard environmental transport and biological models were used to calculate the dose rates and resulting incidence of health effects (cancers and genetic effects). The sources of accidental radiologic risk were accidents in the FRP, FFP, WTFs, or transportation that breach containment and result in the unplanned dispersal of radioactivity. Nonradiological risk results from activities associated with the fuel cycle other than those caused by radiation. As in the case of the radiological risk, nonradiological risk is comprised of routine and accidental components. The routine component considered was the emission of petroleum combustion products into the atmosphere. Petroleum is assumed to be used in firing the boilers that supply heat to the fuel cycle facilities and for fueling transportation vehicles. The sources of accidental, nonradiological risks are collisions that occur during transportation (physical damage) and industrial accidents. Each of the two risk analyses (reference cycle and P-T cycle) calculated all of the risk components listed above. The details of this analysis are contained in ref. 1; thus they will only be summarized here.

5.1.1 Results of short-term risk analysis

Table 5.1 gives the results of the short-term risk analysis for both the reference and P-T cases. The following paragraphs will discuss the significant aspects of these results.

Table 5.1 is comprised of three principal portions: public risk, occupational risk, and overall (total) risk. Each portion contains absolute risk values, incremental risks, and risk ratios for accidental and routine radiological risk and nonradiological risk in both the reference cycle and the P-T cycle. The first important aspect of these results is evident from the last column, which shows that the short-term risk from the P-T cycle is higher than that from the reference cycle, generally by a factor of 2 to 3. This is a result of the more toxic nature of the material in the P-T fuel cycle, which contains such nuclides as ^{241}Am and $^{242,244}\text{Cm}$.

The second significant aspect of these results is that the accidental radiological risks are much lower than the routine radiological risks, and hence can generally be ignored. The small values for the accidental risk result from the improbability of serious accidents and the difficulty in dispersing significant portions of the facility contents.

A third significant aspect of Table 5.1 is that the occupational radiological dose is about the same as the public radiological dose. This similarity raises serious questions concerning whether the occupational risk can be directly included in a cost-risk-benefit comparison. This point is considered further in Sect. 5.4.

The fourth, and final, important aspect of Table 5.1 is the overriding magnitude of the nonradiological risk in both the reference cycle and the P-T cycle. The principal component of the nonradiological risk is the number of health effects resulting from the combustion products (mainly CO and SO₂) of petroleum used to make the process heat and steam used at the FRP, FFP, and the WTFs. It is interesting to note that if electricity were used both directly and to make steam, and if it were assumed that this electricity came from nuclear plants, the nonradiological

Table 5.1. Summary of the impact of P-T on short-term risk

Risk source	Risk [health effects/CW(e)-year]			
	Reference cycle	P-T cycle	Difference [(P-T) - reference]	P-T Reference
General Public				
Radiological				
Accident	5×10^{-7}	7×10^{-7}		1.4
Routine	8×10^{-4}	4×10^{-3}		5.0
Subtotal	8×10^{-4}	4×10^{-3}	3×10^{-3}	5.0
Nonradiological	0.34	0.91	0.57	2.9
Total	0.34	0.91	0.57	2.7
Occupational				
Radiological				
Accident	7×10^{-8}	1×10^{-7}		1.4
Routine	1.2×10^{-3}	2×10^{-3}		1.7
Subtotal	1.2×10^{-3}	2×10^{-3}	8×10^{-4}	1.7
Nonradiological	5×10^{-4}	1.1×10^{-3}	6×10^{-4}	2.2
Total	1.7×10^{-3}	3.1×10^{-3}	1.4×10^{-3}	1.6
Overall				
Radiological				
Accident	6×10^{-7}	8×10^{-7}		1.3
Routine	2×10^{-3}	6×10^{-3}		3.0
Subtotal	2×10^{-3}	6×10^{-3}	4×10^{-3}	3.0
Nonradiological	0.34	0.91	0.57	2.7
Total	0.34	0.91	0.57	2.7
Natural background radiation at 125 mrem/person- year	1.0	1.0	0	1.0

risk would decrease substantially. A secondary, but significant, portion of the nonradiological risk is due to physical damage occurring in transportation accidents. The value for the P-T cycle is higher because of the greater number of fuel (reduced-capacity fuel cask; see Sect. 4.1) and waste shipments required.

5.1.2 Perspectives on short-term risk analysis

Placing the risks given in Table 5.1 into perspective requires a basis of comparison, which is difficult because of differences in populations and ground rules in other studies. However, one comparison that is readily available is natural background radiation. Assuming an annual dose of 125 mrem/person, the population used in this study would incur about 1.0 health effect from natural background per GW(e)-year of fuel reprocessed, transported, and refabricated. Thus the number of natural background health effects is a factor of 250 greater than the increase in radiological impact ($= 4 \times 10^{-3}$) caused by the implementation of P-T. In an absolute sense, then, the short-term radiological impact of implementing P-T is very small.

If the nonradiological effects are compared with the natural-background radiological effects, it is clear that the impact of P-T is nearly equal to the background radiological effects.

5.1.3 Sensitivity analysis

Based on the error analysis given in ref. 1, we believe that the overall risks summarized in Table 5.1 are accurate to within a factor of 2. The principal reason for this relatively high degree of accuracy is that the controlling aspects of the short-term risk are the routinely released radiological and nonradiological effluents, which have a release probability of 1. Therefore, the uncertainty associated with the accident probabilities, which would normally make the overall error very large, is not a factor in this case due to the insignificance of the accidental risk and the certainty of the routine release probabilities (defined to be 1.0).

5.2 Cost of P-T

The cost of P-T is the cost of building and operating the P-T fuel cycle minus the cost of the reference fuel cycle. Thus a calculation of the cost of nuclear-produced electricity from the reference and P-T fuel cycles is required. This cost, in turn, requires knowledge of (1) the unit costs of the various fuel cycle operations, which include both capital amortization and operating costs; (2) the amount of material being handled by each of the fuel cycle operations; and (3) specification of other economic parameters such as the discount rate and the times at which the various cash flows occur.

5.2.1 Fuel cycle cost data

The fuel cycle costs and other economic parameters used in this study were based heavily on the currently ongoing programs to evaluate alternative reactors and fuel cycles: the Alternative Fuel Cycle Evaluation Program (AFCEP), and the Nonproliferation Alternative Systems Assessment Program (NASAP). Since these sources give a range of values for many of the parameters, the numbers used in this document simply represent "typical" values and will be used without specific reference.

In the case of the operations unique to P-T (viz., the two WTFs and the specially designed fuel shipping cask), the costs were based on estimates by R. M. Parsons,² the firm that prepared the preconceptual design of the WTFs. The capital costs of the FRP-WTF and FFP-WTF are \$1035 million and \$436 million, respectively, in mid-1979 dollars. The annual operating costs are \$71.5 million for the FRP-WTF and \$25.6 million for the FFP-WTF. Assuming that interest during construction is 36.6% of the direct capital cost and the fixed charge rate on capital is 22.59%/year ("typical industrial" values in the AFCEP and NASAP), the resulting unit costs are \$195/kg HM for the FRP-WTF and \$240/kg HM for the FFP-WTF. The cost of transporting fresh and spent P-T fuels is \$90/kg HM, or three times the cost of transporting the spent reference fuel.

The amounts of heavy metal being handled by each facility are relatively straightforward. In general, the operations handling only the MOX

fuel have one-third the throughput of the reactor and reprocessing plant. The fissile and fertile credits are based on the fuel compositions given in Sect. 3.6 and well-known mass balance techniques for the uranium enrichment facility. In the interests of simplicity, 100% actinide recovery was assumed in both cycles.

The discounting procedure used in the economic analysis was also simplified in the interests of clarity and lucidity. The costs of all preirradiation activities as well as the cost of the reactor itself were assumed to occur at the beginning of irradiation, which was defined as time zero for discounting purposes. All postirradiation activities were assumed to occur 4 years after the beginning of irradiation. A discount rate of 10% was used.

5.2.2 Cost analysis of P-T

A set of summary fuel cycle costs can be developed for both the reference cycle and the P-T cycle by using the costs, mass flows, and other economic assumptions described above. This summary cost analysis, which is given in Table 5.2, is described in detail in ref. 3. The first column in Table 5.2 presents each of the fuel cycle cost components considered in the analysis. The second column lists the measurement units for these components; the third column lists the unit costs. The fourth column provides the discounted unit costs based on the assumptions described above. The fifth column summarizes the mass flows for the reference cycle. The sixth and seventh columns give the contributions of the individual fuel cycle components to the total, bus-bar cost of nuclear-produced electricity in thousands of dollars per metric ton of heavy metal (= \$/kg HM) charged to the reactor and in mills per kilowatt-hour of electricity produced [kWhr(e)], assuming a fuel burnup of 33,000 MWd/MTHM and a 33% thermal efficiency. The eighth column summarizes the contribution of each component as a percentage of the total. Columns 9 through 12 contain information on the P-T cycle similar to that shown in columns 5 through 8 for the reference cycle. The last column in Table 5.2 summarizes the cost increases (decreases) resulting from the implementation of P-T.

Table 5.2. Summary of impact of P-T on nuclear bus-bar electricity costs

Fuel cycle cost component	Units	Unit cost (\$)	Discounted unit cost (\$)	Units per MTHM	Reference cycle Cost			P-T cycle Cost			Difference, (P-T) - reference [mills/kWhr(e)]	
					\$10 ³ /MTHM	mills/kWhr(e)	% of total	Units per MTHM	\$10 ³ /MTHM	mills/kWhr(e)		% of total
Uranium	lb U ₃ O ₈	60	60	11,030	662	2.51	9.8	11,030	662	2.51	9.4	0
Conversion	kg U	4	4	3,930	16	0.06	0.2	3,930	16	0.06	0.2	0
Enrichment	kg SWU	100	100	3,180	318	1.21	4.7	3,180	318	1.21	4.5	0
Plutonium	g fissile	30	30	11,330	340	1.29	5.1	11,700	351	1.33	5.0	0.04
UO ₂ fabrication	kg U	110	110	670	74	0.28	1.1	670	74	0.28	1.0	0
MOX fabrication												
Reference	kg HM	430	430	330	142	0.54	2.1	0	0	0	0	0.30
P-T	kg HM	670	670	0	0	0	0	330	221	0.84	3.1	
MOX fabrication WTF	kg HM	240	240	0	0	0	0	330	79	0.30	1.1	0.30
Reactor	kg HM	5,150 ^a	5,150	1,000	5,150	19.52	76.4	1,000	5,150	19.52	72.8	0
Reprocessing	kg HM	400	273	1,000	273	1.03	4.0	1,000	273	1.03	3.8	0
Reprocessing WTF	kg HM	195	133	0	0	0	0	1,000	133	0.50	1.9	0.30
Waste management												
Reference	kg HM	50	34	1,000	34	0.13	0.5	0	0	0	0	0.01
P-T	kg HM	55	38	0	0	0	0	1,000	38	0.14	0.5	
Transportation												
Spent, standard MOX	kg HM	30	20	330	7	0.03	0.1	0	0	0	0	(0.03)
All other	kg HM	70	48	1,000	48	0.18	0.7	1,000	48	0.18	0.7	0
Fresh, P-T MOX	kg HM	90	90	0	0	0	0	330	30	0.11	0.4	0.11
Spent, P-T MOX	kg HM	90	61	0	0	0	0	330	20	0.08	0.3	0.08
Uranium credit	kg U	(155) ^b	(106)	950	(101)	(0.38)	(1.5)	950	(101)	(0.38)	(1.4)	0
Plutonium credit	g fissile Pu	(30)	(20)	11,330	(227)	(0.86)	(3.4)	11,700	(234)	(0.89)	(3.3)	(0.03)
Total					6,736	25.54			7,078	26.82		1.28

Note: It is assumed that one-third of the fuel is MOX and two-thirds UO₂.

^aEquivalent to \$800/kWhr(e) at a fixed charge rate of 16%/year plus \$14/kW(e)-year for operating costs.

^bThe enrichment of the discharged uranium is about 0.7 wt % ²³⁵U; this unit cost is equivalent to \$60/lb U₃O₈.

Examination of the last column of Table 5.2 reveals that only a few of the many fuel cycle operations account for the cost differential between the reference cycle and the P-T fuel cycle. The net cost differential of the plutonium (preirradiation cost minus postirradiation credit) is only 0.01 mill/kWhr(e), and this results from the carrying charges on the plutonium during its residence in the reactor.

A significant cost differential of 0.3 mill/kWhr(e) is attributed to the fabrication of the MOX fuel. This differential arises because the reference fuel is fabricated in a less expensive RO/CM plant, whereas the P-T fuel requires the RO/RM plant (see Sect. 4.2).

Two other obvious cost differentials result from the presence of the WTFs in the P-T cycle, but not the reference cycle. These costs are 0.30 and 0.50 mill/kWhr(e) for the FFP-WTF and the FRP-WTF, respectively.

The cost of waste management is also increased slightly [0.01 mill/kWhr(e)] in the P-T cycle because of the larger volume of concreted TRU wastes requiring disposal.

The net cost differential from transportation operations, 0.16 mill/kWhr(e), arises from two sources: the much higher unit cost of transporting the spent P-T MOX fuel (\$90/kg HM vs \$30/kg HM in the reference cycle), and the fact that the fresh P-T MOX fuel must also be shipped in the same cask.

Thus the overall cost differential due to the implementation of P-T is calculated to be 1.28 mills per kWhr(e) of nuclear-produced electricity. This cost is equivalent to an increase of \$342 per kg HM charged to the reactor. Other bases of comparison will be discussed in Sect. 5.2.4.

5.2.3 Sensitivity analysis

The total cost of P-T could be significantly altered by the discovery of some superior process(es) for the treatment of the wastes that would substantially reduce the size of the WTFs. Because of the variety of wastes that must be treated (and hence the variety of processes required), it is unlikely that the costs of the WTFs will ever be reduced to insignificant levels (see Sect. 2.5). However, there is a substantial opportunity for reducing the plant cost by replacing the CEC process, which is used to

separate actinides from the chemically similar lanthanides, with a process that is less space- and materials-intensive. If the replacement process is assumed to have negligible costs compared with the rest of the WTF (a very conservative assumption), this would reduce the 0.50 mill/kWhr(e) cost of the reprocessing WTF by about 25%, or by 0.12 mill/kWhr(e).

In addition, it might be possible to substantially reduce the fuel fabrication cost differential by the means described in Sect. 4.2 (i.e., use of the same maintenance philosophy in both fuel cycles). In this case, the maximum additional reduction in the P-T cost differential would be 0.3 mill/kWhr(e), ignoring the increased wall thicknesses required because of the higher neutron activity of the P-T fuels.

Finally, it is possible that errors were made in estimating the capital and operating costs of the WTFs. According to ref. 2, there is a 20% chance that the actual costs would exceed the stated WTF costs and an 80% chance that they would be less. It is not possible to further quantify this source of error without additional R&D on the partitioning processes and a more detailed conceptual design and cost analysis.

In summary, a reasonable lower bound for the cost penalty associated with P-T appears to be about 50% of the costs stated in Sect. 5.2.2, or 0.64 mill/kWhr(e).

5.2.4 Comparisons

A wide variety of cost comparisons and methods of expressing the P-T cost differential are available. Some of these are listed below.

The cost differential of 1.28 mills/kWhr(e) attributable to P-T is equivalent to the following:

1. 4.8% of the bus-bar (i.e., no distribution costs) cost of nuclear electricity,
2. 2.9% of the average 1978 price of delivered residential electricity in the United States [= 44 mills/kWhr(e)],
3. \$342/kg HM,

4. \$380 million/year for the entire United States based on the production of nuclear electricity in 1978,
5. \$1.50/year for each person in the United States based on the production of nuclear electricity in 1978, and
6. \$9.2 million/GW(e)-year.

Items 4 and 5 will increase in direct proportion to the fraction of electricity that is generated with nuclear power.

In summary, the cost of P-T is significant and should clearly be a key parameter in any decision concerning the implementation of P-T. However, it is also clear that the cost of P-T is not sufficiently large to preclude its implementation based on cost considerations alone.

5.3 Long-Term Risk Analysis of P-T

The purpose of the long-term risk analysis is to examine the impact of P-T on the risk from the geologic repository, which is assumed to be situated in bedded salt. The approach taken to perform this analysis was to calculate the amounts of radionuclides expected to be released from the repository over the long term, beginning with the time the repository is sealed. Four basic steps are involved in the analysis: (1) calculation of the repository nuclide inventory as a function of time; (2) calculation of the expected fraction of the inventory released as a function of time; (3) determination of the rate at which the repository releases migrate through the geosphere to the biosphere; and (4) determination of the uptake of the radionuclides by humans, which in turn gives the dose rates to individual organs and thus the health effects resulting from the repository. The last three steps in the analysis were held constant for both the reference cycle and the P-T cycle, and the repository nuclide inventory was changed to correspond to a high-actinide-content waste in the reference case or a low-actinide-content waste in the P-T case. The results of the analysis were then compared to determine what effect the implementation of P-T had on the repository. Of course, the implementation of P-T is expected to have a beneficial effect by reducing the amounts of actinides released over the long term.

The results of the repository risk analysis and the data used in its calculation are described in detail in ref. 4. This analysis considers the repository risk over a 1-million-year time span. Both the reference repository and the P-T repository were assumed to be located in bedded salt at the proposed WIPP site in Los Medanos, New Mexico. This selection was made because of the large amount of data available for the site.

The repository contains the wastes from 17,000 GWyr(t) of energy from a PWR. Based on an 80% capacity factor and 33% thermal efficiency, this is equivalent to $17,000 \times 0.33/0.80 = 7000$ GW(e)-year of reactor capacity. This value is used to convert the impacts from a "repository" basis to a "unit-reactor-capacity" basis.

It should be noted that the long-term benefits were based on a very conservative long-term risk analysis. This approach was used because it was an expeditious way to examine the incentives for implementing P-T. The values used for this project should not be construed as being appropriate for other studies of the same area or for studies of other areas.

The risk analysis considers two principal types of repository accidents that result in the release of nuclides: a slow leach incident, and expulsive events (e.g., a volcano or a meteorite). In the slow leach incident, the repository is assumed to be breached, and water enters and dissolves the salt while simultaneously leaching the radionuclides from the waste at a specified rate. The water then flows to an aquifer, which in turn flows underground at the rate of 1.46 m/year to a river about 20 km from the repository. The retardation effects of the geosphere on the migrating radionuclides are accounted for. (These effects are particularly important for the actinides, which are strongly sorbed.) When the water finally reaches the river, it is diluted and people are assumed to use it for swimming, drinking, irrigating, and livestock watering. An environmental pathways model accounts for the buildup of nuclides in the biosphere and the dose rates to several organs, such as total body, bone, and thyroid. The probability of such a leach event occurring was estimated to be 1.4×10^{-7} /year.

The explosive event release of part of the contents of the repository involves a violent phenomenon which physically disgorges the radionuclides onto the ground surface, where they are available for consumption, resuspension by the wind and inhalation, and contamination of water supplies. This type of accident, of course, bypasses the salt dissolution, waste leaching, and nuclide migration phases of the slow-leach incident. The nuclides are deposited directly in the biosphere, and the doses are calculated from the environmental pathways model. As might be expected, a very large explosive event is required to disgorge even part of the contents of a repository located 800 m underground. The explosive events considered are meteorite impact and volcanism. Since only large volcanoes or meteorites can affect the repository, the probability of this type of accident occurring is very small, on the order of 10^{-11} /year to 10^{-13} /year.

The accident analysis described above can be performed in two basic modes: probabilistic analysis, and consequence analysis. In the probabilistic mode, the analysis yields the expected dose rate as a function of time. For example, if the probability of an accident was 1.4×10^{-7} /year and we were considering a 1000-year time increment, then $1.4 \times 10^{-7} \times 10^3$ ($= 1.4 \times 10^{-4}$) of the accident would be assumed to occur during that time increment. If the consequences of the accident were a dose rate of 100 mrem/year, then the consequences on a probabilistic basis would be $100 \times 1.4 \times 10^{-4} = 0.14$ mrem/year. Although this results in a lower dose rate than would actually be incurred by the accident, the dose rate is assumed to occur during every time increment. This mode of operation is the most meaningful for evaluating risk since both the probability and the consequences of an accident are accounted for. The second mode of operation ignores probability and considers only the consequences of an accident. In the above example, the consequences would be 100 mrem/year. The consequence analysis is realistic in that it tells what the dose would be if the accident occurred, but is unrealistic because it does not distinguish between accidents that are likely and those that are effectively impossible because of their low probabilities.

After nuclide release, the transport and concentration of nuclides are calculated for each environmental receptor (e.g., water, land surface)

at defined geographical zones surrounding the repository. Then the environmental receptor concentrations are translated to radiation dose commitments to the various body organs using the applicable environment-to-man pathways. These body organ dose commitments are then combined with affected zonal and nonspecific populations and pertinent incident rates of health effects to calculate overall health effects. These computations provide the number of health effects for any desired time increment during the risk assessment period of 1 million years. The total zonal population (i.e., the population in the immediate area of the repository) was assumed to remain constant at 1.9 million.

5.3.1 Results of long-term risk analysis

The results of the long-term risk analysis are summarized in Table 5.3 for the reference and P-T fuel cycles. The upper portion of the table gives the principal contributors to the long-term probabilistic (expected) risk on a GW(e)-year basis. The middle portion of the table summarizes the consequences of a slow-leach incident assumed to be initiated after the repository has been closed for 1000 years. The last portion of the table summarizes the consequences of a volcano growing through the repository 100,000 years after closure.

The effects of the slow leach incident on a probabilistic basis are measured using the probabilistic risk from 1 GW(e)-year of waste from the repository integrated over 1 million years. This is the number of deaths that would be expected from the waste during the entire period, based on the assumptions used in the calculation. (These will be considered further in Sect. 5.3.3.) As is evident, the long-term risk is controlled to a very large extent by the contributions from ^{99}Tc and ^{129}I , which constitute over 99% of the integrated risk and result in 5.16 and 5.10 health effects for the reference and P-T cycles, respectively. These values are for 1 GW(e)-year; similar values for the entire repository are 36,300 and 35,900 health effects over 1 million years for the reference and P-T cycles, respectively. Technetium and iodine control the risk for two reasons: (1) the slow leach incident dominates the long-term probabilistic risk because it was assumed to have a much

Table 5.3. Summary of the impact of P-T on long-term repository risk

	Reference fuel cycle	P-T fuel cycle
Integrated Probabilistic Risk [health effects/GW(e)-year]		
^{99}Tc	4.7	^{99}Tc 4.7
^{129}I	0.4	^{129}I 0.4
^{226}Ra	0.05	^{226}Ra 0.002
^{229}Th	0.007	^{229}Th 9×10^{-5}
^{237}Np	0.001	^{126}Sn 6×10^{-5}
^{225}Ra	0.001	^{242}Pu 6×10^{-5}
^{242}Pu	5×10^{-4}	^{230}Th 3×10^{-5}
^{230}Th	5×10^{-4}	^{237}Np 1×10^{-5}
^{240}Pu	3×10^{-4}	^{225}Ra 1×10^{-5}
Total	5.16	5.10
% of natural background	0.077	0.076
Actinide total	0.0633	0.0023
Consequence Analysis - Leach Incident at 1000 years		
Maximum health effects		
Rate, health effects/year	6.6	6.6
Average health effects		
Rate, health effects/year	2.7	2.7
Principal contributors	^{99}Tc (92%) ^{129}I (8%)	^{99}Tc (92%) ^{129}I (8%)
Total health effects/GW(e)-year	385	385
% of natural background	5.6	5.6

Table 5.3 (continued)

	Reference fuel cycle	P-T fuel cycle
Maximum dose rate, mrem/year in zone 8		
Total body	8	8
Bone	12	12
Thyroid	3,300	3,300
Time at which maximum occurs, years	300,000	300,000
Consequence Analysis - Volcano Incident at 100,000 years		
Maximum health effects		
Rate, health effects/year	363	16
Average health effects		
Rate, health effects/year	14	0.6
Principal contributors	^{226}Ra (72%)	^{226}Ra (79%)
	^{239}Pu (21%)	^{239}Pu (8%)
	^{229}Th (3%)	^{126}Sn (7%)
	^{237}Np (2%)	^{242}Pu (2%)
Total health effects/ GW(e)-year	2,000	90
% of natural background	29	1
Maximum dose rate, mrem/year in zone 2		
Total body	7,200	360
Bone	16,300	640
Thyroid	40	30
Time at which maximum occurs, years	100,000	100,000
Natural background, health effects/year	48	48

higher probability of occurrence; and (2) only those nuclides that sorb poorly or not at all (i.e., technetium, iodine, neptunium, carbon) migrate through the geosphere rapidly enough to reach the biosphere within 1 million years. Because of the dominance of the leach incident and the low actinide migration rate, P-T has very little effect on the integrated risk, reducing it to only 0.06 health effect over the 1-million-year time horizon. It should be noted that this calculated benefit of 0.06 health-effect/GW(e)-year only occurs because of the effect of the expulsive events, which cause the actinides to be physically disgorged from the repository and to circumvent the slow geospheric migration. If these expulsive events (which are extremely improbable) were not considered, the benefits of P-T would be immeasurably small. It should further be noted that this is effectively the maximum expected benefit of P-T since actinide levels are negligible in the P-T wastes and additional actinide reductions would be insignificant.

The leach incident consequence analysis reflects the calculated effects of the incident, assuming that it occurred 1000 years after repository closure with a probability of 1.0. The result is a maximum of 6.6 health effects/year from the entire repository [not from a GW(e)-year], with corresponding maximum dose rates of 8, 12, and 3300 mrem/year to the total body, bone, and thyroid, respectively. These maxima occur about 300,000 years after repository closure. The leach incident consequence analysis is useful in that it measures the effect of the incident under the assumed ground rules, if it occurs, and also shows the total dominance of the ^{99}Tc and ^{129}I as in the probabilistic case. As noted above, the contribution of the actinides is negligible in this incident. It should be remembered, however, that the probability of the occurrence of this incident is estimated to be only 1.4×10^{-7} /year, or about once every 7 million years.

The results of the volcano incident consequence analysis have markedly different characteristics from those of the leach incident. In the former incident, the volcano is assumed to grow through the repository, erupt, and physically eject part of the repository contents into the air (suspension) and onto the ground and water surface (surface

deposition). Accordingly, the effects of retardation during geospheric migration are no longer applicable and the nuclide mix ejected into the biosphere is the same as that in the repository at the time of the incident. The result is that the more toxic actinides dominate the maximum dose, particularly ^{226}Ra . The maximum effect from the entire repository is 363 health effects/year for the reference cycle and 16 health effects/year for the P-T cycle, with the difference reflecting the benefits of P-T. Both of these maxima occur at the time of the incident (viz., 100,000 years after repository closure). The maximum dose rates for total body, bone, and thyroid are 7200 (360), 16,300 (640), and 40 (30) mrem/year for the reference and (P-T) cycles, respectively. Although the consequences of this accident are much larger than those from the slow leach incident, their contribution to the probabilistic risk are small because the probability of the volcano incident is only about 10^{-11} /year (once every 100 billion years). To put this number in perspective, the age of the universe is estimated to be 9 to 18 billion years.

5.3.2 Perspectives on long-term risk analysis

A total population of about 1.9 million was assumed to be living within about 150 km of the repository site in the year 2020, based on a "high" projection. It was further assumed that (1) this population would remain constant and in-place during all accident scenarios, (2) the natural background radiation dose was 125 mrem/year, and (3) the health effect rate was 2×10^{-4} /person-rem. The integrated effects over 1 million years from the entire repository relate to the natural background as follows: probabilistic (expected) - 0.077% (reference cycle), 0.076% (P-T cycle); leach incident consequences - 5.6% (reference and P-T cycle); and volcano incident consequences - 29% (reference cycle), 1.3% (P-T cycle). As is evident, the long-term impact of the repository is small compared with natural background, even if it is assumed that extremely unlikely incidents actually occur. As is evident, the only instance in which P-T would have a substantial value is if the expected effects are ignored and the volcano incident, which has a probability of only 1 in 100 billion, actually occurs. Even then, the benefits of P-T are small when compared with natural background.

5.3.3 Sensitivity analysis

As might be expected, many of the parameters used in the long-term risk analysis could be questioned, principally because of the long times over which the parameters are assumed to be known. The known conservatisms used in the long-term risk analysis are identified in the first part of this section. Next, the parameters where midrange values were used and/or where the resultant long-term risk is sensitive to the parameter are listed and discussed. Finally, the overall possible changes in the benefits of P-T are described briefly.

As noted in the previous section, the benefits of P-T under the base-case assumptions result entirely from the expulsive events (volcanoes and meteorites) since retardation by the geosphere does not allow significant amounts of actinides to be released during the million-year time horizon. Thus, for the benefits of P-T to be significantly changed, either the risk from the expulsive events must be altered or the parameters controlling the migration of the actinides from the leaching incident must be modified so that migration is more rapid and becomes significant. As will be evident, many of the parameters in the study do not affect either of these.

5.3.3.1 Conservatisms. The major known conservatisms in the long-term risk analysis are as follows:

1. The probability of the (overwhelmingly dominant) leach incident was assumed to be 1.4×10^{-7} /year in this analysis. However, the draft GEIS on commercial waste management⁵ uses a value of 2×10^{-13} /year. This value (or any larger value up to 2×10^{-10}) would lower the probabilistic effects and the benefits of P-T by about a factor of 1000 until the expulsive events (e.g., the volcano) were controlling.
2. The fraction of the repository inventory released as a result of a volcano was assumed to be 15%, and the portion released as a result of a large meteorite impact was assumed to be 10%. The draft waste management GEIS⁵ assumed a maximum release of 1% for a large meteorite. This assumption would result in an

overprediction of the consequences of the explosive events by a factor of 10 to 15 in the calculations reported here.

3. The nuclides released from the repository are not allowed to move between zones or to be transported out of the 150-km-radius circle being considered. This constraint substantially increases the concentrations of the nuclides on the surface and in the water.
4. The population is assumed to remain in place and to behave in an unchanged manner even after catastrophic explosive events or in the event that water becomes unpotable because of salt contamination.
5. The time horizon was assumed to be 1 million years.
6. Both the local (circular zone with a 150-km radius) and the nonspecific (nonlocal) risks are accounted for.
7. Neptunium was assumed to be in its most mobile oxidation state; therefore, a K_d value of 8.1 was used as the basis for its migration rate. In many types of groundwater commonly found in geospheric media, neptunium will be in a less mobile oxidation state and thus will have a substantially lower migration rate.
8. Technetium is assumed to have a K_d of 0 and to be in a mobile oxidation state, whereas it will probably have a higher K_d .
9. Nuclides were assumed to be removed from the environment with a half-life of 30,000 years; however, the actual value is much lower, indicating a more rapid removal.

5.3.3.2 Midrange parameters. Those parameters in the long-term risk analysis that have some likelihood of significance and were midrange (or expected) values are discussed in the following paragraphs.

Leach rate. The leach rates used in the long-term risk analysis were $6 \times 10^{-5} \text{ g cm}^{-2} \text{ day}^{-1}$ for cesium and strontium and $6 \times 10^{-6} \text{ g cm}^{-2} \text{ day}^{-1}$ for all other isotopes. The surface-to-volume ratio for the HLW and concreted wastes, which are the principal contributors to the risk, were taken to be 0.2 cm^{-1} , a value representative of a monolithic waste form fractured into a few large pieces. There are two possible areas of

uncertainty here: the leach rate itself, and the surface area (i.e., degree of fragmentation) of the waste form.

The uncertainty of the leach rate was studied by recalculating three sensitivity cases using assumptions different from those of the base cases (reference and P-T). In the first case, the leach rate was increased by a factor of 1000. The resulting dose rates showed no changes from the base cases. This is because the rate at which the salt dissolves and is carried out of the repository during the leach incident (0.1 cm/year) is limiting (i.e., is slower than the rate at which the waste form is leaching) for both the base and increased leach rate values. Therefore, the water does not contact or remove the nuclides any more quickly, and the effects remain the same as in the base case.

In the second sensitivity case, the leach rate was decreased by a factor of 1000 from the base-case values. As a result, the leach rate became more limiting than the salt dissolution rate and the long-term risk from the leach incident was decreased by a factor of 100. The benefits of P-T were not changed since they are controlled by the explosive incidents.

"Realistic, variable" leach rates were used in the third sensitivity case. Here the leach rate was initially a factor of 800 higher than in the base cases to reflect the higher initial temperatures of the waste and declined nonlinearly to the base-case values after 200 years. Between 200 and 1000 years, the leach rate was assumed to be constant at the base-case values cited above. After 1000 years, the leach rate per se continued to remain constant at the base-case values; however, the waste form was assumed to disintegrate linearly with time such that it was in the form of 1-mm spheres (surface area increased by a factor of 300) at 1 million years. This scenario had no impact on the long-term risk since, as in the first sensitivity case described above, the salt dissolution rate is controlling at all times.

Salt dissolution. As is evident from the above results, changes in the salt dissolution rate are more important than changes in the leach rate or surface area of the waste form for the waste forms of interest.

Thus a sensitivity case was evaluated in which the leach rates were increased by a factor of 1000 from the base-case values and the salt dissolution rate was increased by a factor of 10. This increased the long-term risk by about a factor of 10 for both the reference and P-T cases; however, as before, the benefits of P-T remain unchanged because they are controlled by the expulsive events and not by the slow migration of the actinides. It should be noted that the base-case value of 0.1 cm/year is derived from geologic evidence, and not from theoretical calculations. Furthermore, the salt removal rate is limited by its solubility and the rate at which water enters the repository. Consequently, there is no reason to believe that the higher value assumed in the sensitivity study is appropriate or correct.

Nuclide retardation. The degree to which a nuclide is retarded by the geosphere in this region is roughly proportional to the inverse of 15 times the K_d value; that is, a nuclide with a K_d of 10 will migrate about 1/150 as fast as water through the geosphere. The result of considering the path lengths and a 1-million-year time horizon in the base cases is that a nuclide with a K_d greater than 10 will not emerge into the biosphere within 1 million years, and nuclides with K_d values greater than 7 or 8 will not generally be significant. Most nuclides have K_d values that are much greater than 10 and therefore do not appear (e.g., the majority of the actinides).

As noted earlier, the assumed K_d values for ^{237}Np (8.1) and ^{99}Tc (0.0) are probably conservative as compared with their actual values. Of the other nuclides that are sufficiently long-lived to migrate to the biosphere, only radium has a K_d low enough (= 20) so that it might emerge if this value were substantially in error. Fortunately, the migration rates of radium have been intensively studied since radium is present in natural uranium ores, and there appears to be little chance that its K_d would be substantially lower than 20.

Water velocity. The water velocity parameter refers to the rate at which water migrates through the geosphere to the biosphere, which was assumed to be 1.46 m/year in the base cases. A sensitivity case in which the water velocity was increased by a factor of 10 was also examined.

The results showed that the dose rates in the reference and P-T cases were increased by factors of 1000 and 100 respectively. These effects were principally due to the fact that ^{226}Ra ($K_d = 20$) would be released to the biosphere before the end of the 1-million-year time horizon; therefore, this type of parameter change would affect the benefits of P-T. However, the range of water velocities for this site is only 0.02 to 4.6 m/year. A water velocity of 6 m/year is required in order for the radium peak concentration to reach the biosphere. For this reason, the sensitivity calculation value is outside the expected range of water velocities and is overly conservative. In fact, even if the maximum velocity ($= 4.6$ m/year) were used, the radium would only be about 25% of the distance to the biosphere. It should also be noted that the high dose rates in this case are also inflated because of the lack of transport of the radium out of the near-repository zones (item 3, Sect. 5.3.3.1). Additionally, the probability of the leach incident is very conservative, and this conservatism alone negates any possible effects of a variable water velocity.

Discount rate. The propriety of discounting future risks and benefits to put them into perspective vis-à-vis present-day risks and costs has been a subject of continuing discussion for many years. In the long-term calculations described thus far, the discount rate has been assumed to be zero (i.e., no discounting); however, a persuasive case can be made for using a positive discount rate.³ The basic thrust of this argument is that the use of discounting is necessary to ensure that limited resources (i.e., money) are spent in a manner that maximizes the total number of lives saved throughout time. The actual procedure used in discounting is to allocate a sum of money spent at the present to each of the lives it would save in the future. Mathematically, this is equivalent to discounting the lives saved by that money. Despite the logical and moral arguments that favor discounting, the method may be unacceptable to some. Therefore, the overriding thrust of this report concerns only the zero discount rate case. However, for comparative purposes, a case with a discount rate of 7% per year will be described below.

The effects of the 7%-per-year discounting are to render events occurring beyond 1000 years inconsequential. In the probabilistic case, then, the most important nuclide becomes ^{90}Sr , followed by ^{137}Cs , which is about a factor of 1 million lower. In both of the consequence cases (leach and volcano), the effect is to make all nuclides insignificant since the doses do not begin to occur for about 10,000 years in the leach incident and at 100,000 years in the volcano incident. This means that the resources used to mitigate any detrimental impacts of the repository in the distant future should be invested in the economy and allowed to grow so that they can be applied to the repository if and when it becomes a problem.

Probability of expulsive events. As noted earlier, the benefits of P-T in the base case depend directly on the expulsive event risks (i.e., the probabilities and consequences). As noted in Sect. 5.3.3.1, the consequences appear to be conservative. The probabilities are very similar to those used by others in long-term risk studies. There is no reason to expect them to be erroneous, other than the fact that they are based on extrapolation of more likely events (e.g., smaller meteorites) or assume random volcanic action. Very little can be said about the accuracy of these values except that their associated events are extremely improbable.

5.3.3.3 Sensitivity discussion. With respect to the benefits of actinide P-T, the following statements can be made based on the preceding analysis:

1. The methods for calculating the dose from the assumed releases are conservative because nuclides cannot be removed from the impact site, the time horizon is very long, a zero discount rate is used, and both local and nonlocal doses are taken into account.
2. The expulsive event risk, which controls the benefits in the base case, is comprised of a consequence (release fraction) that is conservative by a factor of 15 to 20 and a probability about the same as that used elsewhere,⁵ but of unknown conservatism.

3. The leach event, which might begin to affect the benefits significantly if the actinide migration were to be increased considerably, can have this effect only if either the geospheric water velocity or the K_d values of certain nuclides, such as radium, are substantially in error. However, even the maximum water velocity that has been observed at the repository site would not have a large effect, and the K_d value of one "fast migrator" actinide, ^{237}Np , would likely be conservative while that of radium is well known. Even if these unfavorable events did occur, the probability of the leach incident is so grossly conservative that, if a somewhat less conservative value were used, the effects of these changes would be negated.

In summary, there appears to be no reasonable mechanism for increasing the benefits; on the other hand, there seems to be ample justification for saying that they are higher than would actually be expected.

It is clear that the overall risk from the repository is totally controlled by the leach incident, and that this value is very conservative because of the conservative probability of the incident, the conservative method of calculating the doses and health effects, and the conservative K_d for technetium (the principal contributor to the risk).

5.4 Analysis of the Incentives for Actinide P-T

Sections 5.1-5.3 have described the development of a cost-risk-benefit ratio, which will be the focal point of the actinide P-T incentives analysis. Before developing this ratio, however, it will be useful to briefly recapitulate the previous results and to convert them into more useful units. This is done in summary form in Table 5.4.

The basis selected for the comparison is a unit of nuclear electricity production capacity, namely, 1 GW(e)-year. The units of the incremental facility cost are given in dollars, as expected. However, units of the short-term and long-term risk, which have generally been expressed in health effects heretofore, have been converted to person-rem for the

Table 5.4. Summary of incremental costs, risks, and benefits of P-T

	Health effects/GW(e)-year	Person-rem/GW(e)-year ^a
Incremental Short-Term Risk to the General Public		
Radiological	0.003	16
Total	0.57	2850
Incremental Long-Term Risk to the General Public		
Probabilistic	0.06	300
Leach incident	0	0
Volcano incident	1900	9.5×10^6
Incremental Cost		
\$9.2 million per GW(e)-year		

^aBased on 2×10^{-4} health effect/person-rem.

purposes of the cost-risk-benefit analysis. The conversion factor used, 2×10^{-4} health effect/person-rem, is based on data given in the 1972 BEIR report.⁶ It should be noted that the total, incremental short-term risk to the general public includes a substantial contribution from the nonradiological risk. The health effects from this source have been converted to "equivalent" person-rem using the same conversion factor to facilitate comparisons. As is evident, the short-term risk used in the incentives analysis is the risk to the public. This is because occupational risk is voluntary and compensated for by a salary (see ref. 3).

The principal basis of comparison for the cost-risk-benefit values to be calculated from the data presented in Table 5.4 will be the \$1000/person-rem criterion given in 10 CFR 50, Appendix I.⁷ This means that, if the cost of reducing the dose commitment to some population is greater than \$1000/person-rem or is negative, justification for spending the money for P-T does not exist. Alternative projects (e.g., highway safety or air traffic control improvements) would provide greater benefits (i.e., save more lives) for society as a whole. Thus the \$1000/person-rem criterion is simply a method for ensuring a more optimal expenditure of limited resources.

The next task is to develop comparable values in the form of dollars spent per person-rem eliminated. Unfortunately, a variety of methods are available for combining the data in Table 5.4 to arrive at such a value for actinide P-T, but none of these is generally accepted. Therefore, the approach that will be taken is to calculate a range of these values and compare the range to the criterion. A summary of these calculations and results is given in Table 5.5.

The most appropriate basis for evaluating the actinide P-T concept is the expected (probabilistic) benefits since using the consequence analyses as a basis ignores the extreme differences in the likelihood of the various incidents. For instance, it is unreasonable to base the analysis on an incident which would not be expected to occur during the entire lifetime of the universe, as is the case with volcanoes (and meteorites). Nevertheless, such a value has been included to indicate the effect of such an assumption.

Table 5.5. Calculation of the cost of reducing the long-term impact of a waste repository by actinide P-T

-
1. Cost of reducing expected long-term risk irrespective of short-term risk:

$$R_1 = \frac{\$9,200,000}{300} = \$31,000/\text{person-rem}$$

2. Cost of reducing *net* expected radiological (long-term minus short-term) risk:

$$R_2 = \frac{\$9,200,000}{300 - 16} = \$32,400/\text{person-rem}$$

3. Cost of reducing *net* expected total (including nonradiological) risk:

$$R_3 = \frac{\$9,200,000}{300 - 2850} = -\$3600/\text{person-rem}$$

4. Cost of reducing long-term consequences of leach incident irrespective of short-term risk:

$$R_4 = \frac{\$9,200,000}{0} = \infty$$

5. Cost of reducing consequences of volcano incident irrespective of short-term risk:

$$R_5 = \frac{\$9,200,000}{9.5 \times 10^6} = \$1.0/\text{person-rem}$$

The first three ratios (R_1 - R_3) in Table 5.5 are based on the expected long-term risk (i.e., benefits) from actinide P-T. The first ratio ignores the short-term risks completely, the second ratio includes only the short-term radiological risk, and the third ratio includes all short-term risks (i.e., includes nonradiological). As shown, each of these fails to meet the \$1000/person-rem criterion by a wide margin. In fact, the correct interpretation of the negative R_3 is that we would be paying \$3600 to increase the net risk by 1 person-rem.

The last two ratios, R_4 and R_5 , consider the cost of reducing the consequences of the repository incidents irrespective of the short-term risks (assuming that they actually occur). In the case of the leach incident, the cost is infinite since a significant amount of actinides is not released within the time horizon. In the case of the volcano incident, the cost per person-rem falls well within the \$1000/person-rem criterion at \$1.0/person-rem. However, as noted above, this type of incident is so unlikely that it cannot be considered as a reasonable design or decision basis.

On balance, the most appropriate measure of the incentives for actinides P-T is ratio R_3 , which accounts for the total incremental cost, the total short-term risk, and the total long-term benefits. The failure of this ratio to meet the \$1000/person-rem criterion, along with the net expected risk increase for actinide P-T, strongly indicates that there are no technical incentives for actinide P-T. Further support for rejecting technical options on this basis is provided in the 1972 BEIR report:⁶

The public must be protected from radiation but not to the extent that the degree of protection provided results in the substitution of a worse hazard for the radiation avoided. Additionally, there should not be attempted the reduction of small risks even further at the cost of large sums of money that spent otherwise, would clearly produce a greater benefit.

5.5 Analysis of the Incentives for Iodine and Technetium P-T

This section briefly discusses the incentives for partitioning and transmuting technetium and iodine. Since relatively little work has been done in these areas, the discussion is somewhat speculative.

5.5.1 Technetium

As noted in Sect. 2.3, no processes are currently available for recovering technetium from fuel cycle wastes, principally the HLLW and dissolver solids. As a result of this, both the costs of technetium P-T and the short-term risks are unknown. However, as is evident from an inspection of Table 5.3, the possible benefits of removing the technetium from the waste are much larger than those of the actinides [4.7 health-effects/GW(e)-year, or 23,000 person-rem/GW(e)-year]. Thus, there may be incentives for the recovery and transmutation of technetium *if*

1. the conservative base-case assumptions are used,
2. adequate technetium partitioning processes can be developed for a few million dollars per GW(e)-year or less, and
3. the short-term risks of technetium P-T are at least as small as those for actinide P-T.

For example, if the costs are \$3 million/GW(e)-year and the short-term risks are 3000 person-rem/GW(e)-year, the cost of the net risk reduction is $\$3,000,000 / (23,000 - 3,000)$, or \$150/person-rem. However, it should be reemphasized that this conclusion is based on a very conservative long-term risk analysis (see Sect. 5.3.3.1) that provides calculated benefits which are substantially larger than those anticipated from an actual repository.

5.5.2 Iodine

The situation for iodine is very similar to that for technetium. The benefits are about a factor of 10 lower than for technetium, but much of the P-T cost can be considered to be sunk (i.e., should not be

attributed to P-T) since iodine is routinely removed from off-gas streams without P-T. If all of the assumptions listed above for technetium P-T are also true for iodine P-T and, in addition, iodine transmutation is indeed feasible (see Sect. 3.3.2), then there are probably incentives for iodine P-T. As also noted above, this conclusion would be valid only if the conservative long-term risk analysis results were used.

5.6 Analysis of the Sensitivity of the Incentives for Actinide P-T

The purpose of this section is to briefly examine the sensitivity of the "there are no incentives for actinide P-T" conclusion to the possible variation in the three parameters used in the incentives analysis. Based on the discussion given in Sects. 5.1.3 and 5.2.3, it will be assumed that the smallest possible value for the cost of P-T is 50% of the calculated cost (i.e., \$4.6 million) and that the short-term risk is 50% of the calculated short-term risk (8.0 and 1425 person-rem for radiological and total, respectively). For the purposes of this discussion, then, we are interested only in effects that might reduce the absolute values of the ratios in Table 5.5. By using the \$1000/person-rem criterion, we can back-calculate that the long-term benefit must be 4600 person-rem/GW(e)-year if only the short-term radiological risk is considered, and about 7000 person-rem/GW(e)-year if the total short-term risk is considered. Consequently, the long-term probabilistic benefits must be about a factor of 15 larger to justify actinide P-T. However, in light of the known conservatisms in the long-term risk analysis (see Sect. 5.3.3), the chance that circumstances would provide incentives for actinide P-T seems very remote.

5.7 References for Section 5

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6. RESEARCH, DEVELOPMENT, AND DEMONSTRATION REQUIREMENTS FOR IMPLEMENTATION OF PARTITIONING-TRANSMUTATION

The following sections summarize the RD&D requirements (areas of study, time, and cost) for five different aspects of P-T: partitioning, transmutation, fabrication, transportation, and miscellaneous.

6.1 Partitioning

The following listing describes the most important areas of study that would be required if partitioning were to be implemented:

1. Exhaustive extraction of HLLW - The maximum extent to which the actinides can be removed from HLLW, including the effects of fluoride on the decontamination factors, should be determined more precisely.
2. Alternatives to CEC - The CEC process is most probably a workable, but cumbersome, process for separating the trivalent actinides from the trivalent lanthanides. Alternative processes for performing this separation and still achieving the goals of partitioning are very desirable.
3. Ceric nitrate--nitric acid leaching - The precise mechanism by which the ceric nitrate--nitric acid system solubilizes the actinides is not well understood and requires further investigation. Additionally, work is needed on the equipment design for this leaching process, including studies of the electrolytic regeneration of the ceric nitrate.
4. Testing of an integrated flowsheet - As noted earlier, each of the processes selected for the partitioning facilities is believed to be feasible based on separate laboratory testing. However, it is crucial that the entire flowsheet, including all internal recycle streams, be tested to ensure that products that will interfere with operation, safety, or performance do not build up inside the plant.

5. Process integration studies - Further analytical and experimental studies would be required to achieve an optimum flowsheet for the main plant and its associated WTF. By combining these two operations into a single facility, some simplifications, such as eliminating one or more of the major processes (e.g., Purex extraction), might be possible.

Seven general steps are required to fully develop and demonstrate partitioning: (1) process research and development, (2) cold pilot plant, (3) hot (radioactive) pilot plant, (4) plant equipment design and testing, (5) plant design and licensing, (6) plant construction and testing, and (7) operation. Since these steps generally overlap, it is not possible to specify a duration for each and to sum them to obtain an overall time. Generally speaking, the first two steps are expected to require about 7 years. The second two steps will begin during this period and extend about 4 years beyond. The fifth step will begin during the previous two steps and extend about 4 years beyond that (assuming a favorable licensing climate); the construction and testing phase will require an additional 4 years. Thus it is estimated that a direct and orderly program to develop P-T and prepare the first plant for operation will take about 20 years. A crash program might accomplish this goal in only 10 to 15 years, but the costs would be higher and the risk of plant failure due to unforeseen process difficulties would be much greater.

The cost of RD&D (but not the first commercial-size plant) is estimated to be about \$700 million (1979 dollars). Most of this cost is attributable to the design and construction of the hot pilot plant.

6.2 Transmutation

If actinide transmutation were to be implemented, three principal areas would have to be investigated:

1. Cross-section measurements - Cross sections of many of the actinides are not sufficiently well known for actinide transmutation purposes. The most important isotopes are ^{237}Np , ^{242}Pu ,

$^{241-243}\text{Am}$, and $^{243-245}\text{Cm}$. Measurements for nuclides heavier than ^{245}Cm would be desirable, but only in the longer term.

2. Calculational studies - Realistic calculational reactor physics studies would be required for both thermal and fast reactors under a variety of recycle modes for comparative evaluation purposes.
3. Irradiation experiments - A few test assemblies should be irradiated to full burnup using the preferred recycle modes to verify the calculations and identify any unanticipated effects.

Considering the substantial existing capabilities in all of these areas, these studies should require about 15 years and \$20 million (1979 dollars).

6.3 Fabrication

One of the major uncertainties in both actinide and fission product transmutation is the performance of the various fuels during irradiation. Immediate testing of fuels with high concentrations of neptunium, americium, curium, or combinations thereof is required to determine whether these fuels behave in an acceptable fashion and are compatible with standard cladding materials.

With respect to the fission products, technetium also requires verification testing as mentioned above. If iodine is to be transmuted, considerable development will probably be required to identify the best fuel form and a cladding material that is compatible with the corrosive iodine.

Testing of the actinide and technetium fuels should be relatively straightforward and require about 15 years and \$100 million (1979 dollars). Iodine could add significantly to this cost and take as long as the RD&D for the actinides, depending on the difficulty encountered in resolving the unknowns.

6.4 Transportation

A prototype shipping cask fabricated of the relatively unusual materials proposed for the P-T cask would have to be designed, constructed, and

tested. Specific aspects requiring attention are:

1. techniques for fabricating B_4C/Cu and LiH ,
2. investigation of the ability of the cask to conduct the heat from the fuel contents, and
3. the effect of the unusual construction materials on transportation safety.

Since this is principally a question of applying unusual materials to a well-defined situation, about 10 years should be required at a cost of \$70 million (1979 dollars).

6.5 Miscellaneous Aspects

The most important miscellaneous impact that would require attention if P-T were to be implemented is the need to continue overall studies in order to define the preferred methods of operating the P-T fuel cycle, as well as the impacts and benefits of this operation. This should be an ongoing activity throughout all other RD&D phases with an estimated cost of \$10 million (1979 dollars).

6.6 Need for RD&D

Inasmuch as there are no incentives for implementation, further studies in any of the above areas in support of P-T are not warranted until such time that a decision to implement P-T is made by the responsible federal agencies.

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